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第29回IAEA/IWGFR定例年会報告

1996年6月

動力炉・核燃料開発事業団

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第29回IAEA / IWGFR定例年会報告要旨

平成8年5月27日

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平成8年5月14日～17日、カザフスタン共和国のアクタウ市（旧シェフチェンコ）で開催された第29回IAEA / IWGFR定例年会に、日本委員である動力炉開発推進本部伊藤和元次長の代理として出席した。出席国（人数）は、ブラジル（1）、中国（1）、フランス（2）、ドイツ（1）、インド（1）、イタリア（1）、日本（2）、カザフスタン（5）、ロシア（2）、スイス（1）、イギリス（1）、アメリカ（1）の12ヶ国及びIAEA（2）であった。このほかBN-350プラントの技術者も参加した。

本年会では、高速炉に関する1995年の活動のレビュー、1996～1997年の活動計画の審議・調整を行うとともに、各国における高速炉開発状況について報告・討論を行った。日本から高速増殖炉開発の概況をレビューするとともに、「もんじゅ」ナトリウム漏えい事故の原因調査状況等について報告した。「もんじゅ」の早期立ち上げを望むとの声が多く聞かれた。

各国の高速炉開発状況は、アメリカ、イギリス、ドイツの先行国はアクティビティが極めて少なくなってきているものの、実プラントを保有し運転している日本、フランス、ロシア、カザフスタン、インド、並びに具体的な建設計画を持つ中国では、それぞれの国情に応じて着実に開発計画が進められている。

BN-350プラントは、本年会が開かれたアクタウ市のマンギシュラク原子力エネルギーコンビナート内にあり、1973年に運転を開始した。1993年に設計寿命に達したため、その後は1年毎に運転許可を取得しながら運転を続けている。公称1000MWtの熱出力は、発電（150MWe）と海水淡水化（200MWe相当）に使われるが、現在、炉出力は約50％に制限されている。

このプラントは、約18万人のアクタウ市民への電力、熱水及び上水の供給源として、地域に密着した極めて重要な施設であることを実感した。

次回、第30回定例年会は、1997年5月13日～16日、中国で開催されることとなった。

第29回IAEA/IWGFR定例年会報告

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第29回IAEA / IWGFR定例年会報告

1. 開催期間：1996年5月14日～17日

2. 開催場所：カザフスタン共和国、アクタウ市、マンギシュラク原子力エネルギーコンビナート（MAEK:BN-350サイト）

3. 出席国（人数）：ブラジル（1）、中国（1）、フランス（2）、ドイツ（1）、インド（1）、イタリア（1）、日本（2）、カザフスタン（5）、ロシア（2）、スイス（1）、イギリス（1）、アメリカ（1）の12ヶ国及びIAEA（2）

4. 議事概要

会議開催に当たり、KAEKのチーフエンジニアのI. Nazarenko、カザフスタン原子力庁（KAEA）次官のS.Krechetov 及びIAEA原子力部長のP.E.Juhnから、高速炉として23年間に及ぶ運転実績を持つBN-350サイトで本会議を開くことの意義と、高速炉開発の進展に向けて本会議が成功裏に終わることを望む旨の挨拶が述べられた。

議長にはイギリス代表のC.Picker（AEA Technology）が互選された。

4.1 会議メンバーの紹介

昨年、中国、韓国、カザフスタンが正式メンバー国となったのに続いて、今回からブラジルがメンバー国として正式にIAEAから承認された。

なお、スイスとアメリカはオブザーバー国として扱われている。

4.2 1995年におけるIWGFR活動成果のレビュー

(1) 1995年に開催された技術会議

・「新型高速炉の設計概念」（1995年10月3～6日：インド、カルパッカム）

・「吸収材料、制御棒並びに燃焼炉心・非増殖炉心のバックアップ炉停止システム設計」（1995年7月3日～7日：ロシア、オブニンスク）

これらの技術会議は、各国における増殖炉または燃焼炉の設計概念の選択根拠の理解を促進し、設計概念の比較、討論を行う重要な機会となったと評価された。

(2) コンサルタント作業会

・「BN-800ゼロボイド炉心のシビアアクシデントに関するIAEA/CEC共同ベンチマーク」

作業会は1995年7月26～28日、ブラッセル及び1995年12月11～13日、ウィーンの2回が開かれ、定常状態の炉心特性の解析結果のレビューが行われた。

(3) 共同研究

・「高速炉耐震解析コードの相互比較研究」

最終会合が1995年5月30日～6月2日、イタリアで開かれた。実験データと解析の比較に基づきコードの妥当性が取りまとめられ、本共同研究は終了した。

得られた成果は高速炉以外の炉にも有効と評価された。

・「沸騰又はナトリウム－水反応検出に向けた音響信号処理研究」

最終会合が1995年7月11日～14日、ロシアオブニンスクで開かれた。現実的なデータを基に種々の信号処理技術が評価され、本共同研究は終了した。音響検出システムの設計者にとって貴重な情報をもたらすものと評価された。

4.3 1996年のIWGFR活動計画

(1) 1996年に開催予定の技術会議準備状況

・「高速炉の設計に使用するクリープ疲労損傷規定」に関する技術会議

1996年6月11日～13日、イギリス、マンチェスターで開催される。日本、フランス、インド、韓国、ロシア、イギリスから出席する予定。

・「放射性物質放出と移行の評価」に関する技術会議

1996年11月11日～13日、大洗工学センターでの開催を要請されていた。今回、「もんじゅ」のナトリウム漏えい事故の発生に鑑み、日本からスコープとして「ナトリウム火災」を追加すること、技術会議タイトルを「放射性物質の放出とナトリウム火災の評価」と修正すること、開催期間を4日間とすることを提案し、承認された。

日本として開催を受け入れる旨、IAEAに6月20日までに公式ルートで文書回答するよう要請された。

・「高速炉燃料開発と性能」に関する技術会議

本技術会議は、IAEAのIWGNF（核燃料国際ワーキンググループ）が1996年12月9日～12日、ウィーンで開催を提案している。

IWGFRとしては、1997年にロシアで開催することが決定された「炉心構造材料の高線量照射の影響」との重複を避けるため、スコープを「高速炉燃料のフロント・バックエンド解決と関連トピックス」に変更するよう提案した。

(2) コンサルタント作業会

・「BN-800ゼロボイド炉心のシビアアクシデントに関するIAEA/CEC共同ベンチマーク」

次回の作業会は、計算結果の討論とベンチマーク第1フェイズの結論を導くとを目的として1996年6月25日～27日、ブラッセルにおいて開かれる。

(3) 共同研究

・「原子力用免震構造の解析手法の相互比較」

1996年から1999年まで4年計画で、インド、イタリア、韓国、ロシア、イギリス、アメリカが参加して活動する。第1回会合は、1996年5月27日～31日、ロシアのサンクトペテルブルクで開かれる。

・「高速炉用構造解析コードと熱流動解析コードの実験データによる調和と検証」

1996年から1998年までの3年計画で活動する予定。提案国のフランスから「2次系のT継ぎ手」のベンチマーク問題を扱ってはどうかとの提案があった。

4.4 1997年のIWGFR活動計画

(1) 1997年に開催する技術会議テーマの選定

各国の投票により次の2テーマ（提案国が開催国となる）が選定された。

・「炉心構造材料挙動の高線量照射による影響」

1997年6月、ロシアにおいて開催予定。

・「通常運転時及び廃止時におけるナトリウムの除去と処分」

1997年11月、フランスにおいて開催予定。

なお、日本からは「破損燃料ピンの運転性」（1985年に提案）及び「高速炉信頼性データの評価」（1991年に提案）の2件を技術テーマとして提案しているが、後者については近年の各国の取り組み状況から判断して取り下げたい旨表明し承認された。

最近はロシアからの提案が多く目立っている。日本からも新たなテーマを積極的に提案していく必要がある。

(2) IAEA主催のシンポジウム計画

・「原子力エネルギーによる海水淡水化」に関するシンポジウム

1997年5月26日～30日、韓国の大田で開く予定で、既に9ヶ国代表からなる運営委員会を設け準備を進めており、暫定プログラムを計画している。

シンポジウムのスコープは、各国の開発状況、淡水化技術の経験、原子力によ

る淡水化プラントの設計と安全、経済性、市場、技術移転、国際協力等としており、約150人の出席者を見込んでいる。韓国出身のIAEA原子力部長P.E.Juhnが熱心に説明していた。

・「原子燃料サイクルと炉の戦略」に関するシンポジウム

1997年6月2日～6日、ウィーンでOECD/NEAとの共催で開く計画。

シンポジウムの目的として、核兵器保有国やPuを扱う国に論議の場を提供する、Pu利用の国際的共通認識の形成、種々の処分に対するセーフガード、Pu管理と処分の透明性強化、Puの製造、貯蔵、使用、処分に関する将来の国際協力等をあげている。

現在、運営委員会と6つのワーキンググループを設けて、1996年末までに主な論点を整理する予定。日本からは核物質管理センターの栗原専務理事が運営委員会に参加している模様。

4.5 その他の活動

(1) 「高速炉データベース」に関する技術レポート

世界の高速炉プラントの主要設計パラメータを整理すべく、1993年から各国の協力で作業を進めてきた。IAEA技術レポートとして完成・発行され、メンバー国に配布された。

(2) 「高速炉技術の現状」に関する技術レポート

1994年から作業してきたが、各国分担の原稿が全て集まり、IAEAレポートとして発行準備中。ドラフトが6月中に送付され、8月中にコメント若しくは最新情報を提供することとなった。

「もんじゅ」については、2年前にまとめたもの（臨界達成まで）であり、その後の進捗を追加するよう要請された。

5. 各国における高速炉開発状況

【ブラジル】

現在６２.５万のWH製軽水炉を運転中、さらにSiemens製130万プラント建設を1999年運開目標に進めている。ＦＢＲ開発は１９６９年に開始していたが、１９９２年に２５～３０年後の高速実験炉の運転開始を目指した長期研究開発計画が策定され、現在開発体制の整備や国内の合意形成、燃料、熱流動などの各分野において基礎研究が行われている。この実験炉の規模は出力６０ＭＷｔになると見られ、ＩＦＲ概念を参考に設計パラメータが決定され、１次主冷却系統の設計だけが完了している。

【中国】

軽水炉の運転状況および今後の建設計画、実験炉ＣＥＦＲの基準設計案および開発状況についての報告がなされた。ＣＥＦＲ（２０ＭＷｅ）プロジェクトは１９９５年末国家計画委員会（ＳＰＣ）の承認が得られた。現在、設計、立地、安全評価、環境影響評価、コスト評価など審査を受けており、これをパスすればＳＰＣの予算で建設が進められる。１９９７年敷地造成、２００２年初臨界達成を現在のマイルストーンとしている。関連研究として、イタリアより譲渡されたナトリウムループ施設を建設中であり、燃料集合体の熱衝撃試験等を計画している。その他にゼロ出力試験装置や、ナトリウムループ試験装置を使用した計装品の試験等を実施している。

【フランス】

フェニックス、スーパーフェニックスの運転状況、欧州統一炉ＥＦＲ開発計画（設計および関連Ｒ＆Ｄ）、ＣＡＰＲＡ（Ｐｕ燃焼）計画、ＳＰＩＮ（マイナーアクチニド分離・燃焼）計画についての報告があった。

フェニックスは１９９５年４月に出力３５０ＭＷthで第４９サイクルの運転を支障無く終えたのち、１０年間の寿命延長措置を進めている。既にＩＨＸ３器の発注を終え、１９９７年春には搬入される予定。現在は原子炉を停止している。

スーパーフェニックスは、１９９５年上期にＩＨＸアルゴンガス管リーク箇所の補修を終えたものの、その後発生した幾つかの小さな不具合のため、再起動は１９９５年末となった。現在は段階的に出力を上昇しながら運転中であり、６月には６０％に出力を上昇し、その後９０％出力上昇の申請をしていく予定とのこと。なお、１９９７年始めにはブランケット１層を削除する計画である。

【ドイツ】

欧州諸国および日本とのＣＤＡ解析コード（SIMMER-III、SAS4A）共同開発、ＣＡＰＲＡ計画、ＳＰＩＮ計画、ＢＮ－８００過渡事象解析作業等へ参画している。その他ＨＦＲにおける照射試験の状況についての報告があった。

【イタリア】

米国、欧州各国との静的安全炉（ＬＷＲ）共同設計、ＥＮＥＬとＧＥとの協定によるＰＲＩＳＭ炉心の過渡解析、ＩＡＥＡ／ＥＣによるＢＮ－８００炉の過渡事象解析などに参画している。また米国、欧州各国共同による耐震研究への参画等の活動状況が報告された。イタリアとしては、少なくとも２０００年まではＳＰＸのパートナーシップを堅持するとの姿勢であった。

【インド】

高速増殖実験炉ＦＢＴＲの運転経験、特に１９９４～９５年にかけて２回発生した異常な出力上昇事象（２回目は４５０ｋＷ／分）の原因検討および調査などについて報告があった。しかしながら原因は未だ特定されておらず、また以後同様の現象の再発は見られていない。その他に、原型炉ＰＦＢＲの概念設計がコスト低減を主眼に進められ、主要仕様を決定した（２ループ、５４７℃、ＭＯＸ燃料、燃焼度１０万、サイトはカルパッカム）。ナトリウムポンプ試験施設の建設、５ＭＷＳＧ施設の建設計画などが示された。

【日本】

１９９４年改訂の原子力長計に基づく高速炉開発計画、「常陽」および「もんじゅ」の建設・運転経過、関連する研究開発の状況などを報告した。また、トピックスとして「もんじゅ」の２次系ﾅﾄﾘｳﾑ漏えい事故の経過について報告を行った。

諸外国とも一様に「もんじゅ」事故に対しては高い関心を示しており、早期の復旧・再起動に期待する声が数多く聞かれた一方で、『停止期間が長すぎる』、『（２次系配管の）中心温度を計る必要があるのか』といった意見が聞かれた。

【カザフスタン】

ＢＮ－３５０のプラント概要、過去に蒸発器のナトリウム－水反応が発生した際に蒸発器の胴が破損し、２次ナトリウムが漏えいした事例が報告された。

１９９３年には２０年の設計寿命に達したため、ロシアの協力を得ながら、安全システムの補強、ポンプ予備品の製作、安全性評価、崩壊熱除去系の追加等の寿命延長策を講じながら運転を継続したいとしている。

【ロシア】

高速増殖実証炉ＢＮ－６００、実験炉ＢＯＲ－６０、ＢＲ－１０プラントの運転経験、ＢＮ－８００モデル炉心の臨界集合体試験、プルトニウム、マイナーアクチニドの利用促進研究の状況が報告された。

１９９５年のＢＮ－６００プラントの運転状況は、計６件の冷却機能部分喪失事象が発生しており、その原因には小規模ながらナトリウム漏洩も含まれている。昨年の原子炉停止は３回（内１回が計画外）であり、負荷率は７０．３１％であった。

【スイス】

フランスＣＥＡとの共同研究でＣＡＰＲＡプログラム、ＣＩＲＡＮＯプロジェクトに参加し、Ｐｕ燃焼炉心臨界集合体試験およびコード評価等を実施している。、また崩壊熱除去時におけるストラティフィケーションに関してナトリウム流の混合現象に関する研究結果などが示された。スイスは原子力依存が３７％である。

【英国】

ＰＦＲの閉鎖・解体状況の他、欧州統一炉ＥＦＲの開発、ＣＡＰＲＡ計画への参画状況等について報告があった。ＰＦＲの解体は、１９９５年度に炉心の撤去と模擬炉心の構築、２次冷却系、崩壊熱除去系のドレン作業が完了している。今後は、不要なサービス系の隔離などが行われる予定である。

【アメリカ】

米国は政府としては完全に高速炉分野から撤退している。一部の民間企業（ＧＥ、ロックウエルなど）においてメンテナンス、研究開発を手掛けている。ＥＢＲ－ＩＩについては１９９５年に炉心燃料の取り出しを開始し、１９９８年には完了させてSafety Strage Modeにもっていく計画とのこと。その他アルゴンヌ国立研究所での燃料コンディショニング施設建設の話題等が報告された。

6.　ＢＮ－３５０プラントの施設見学

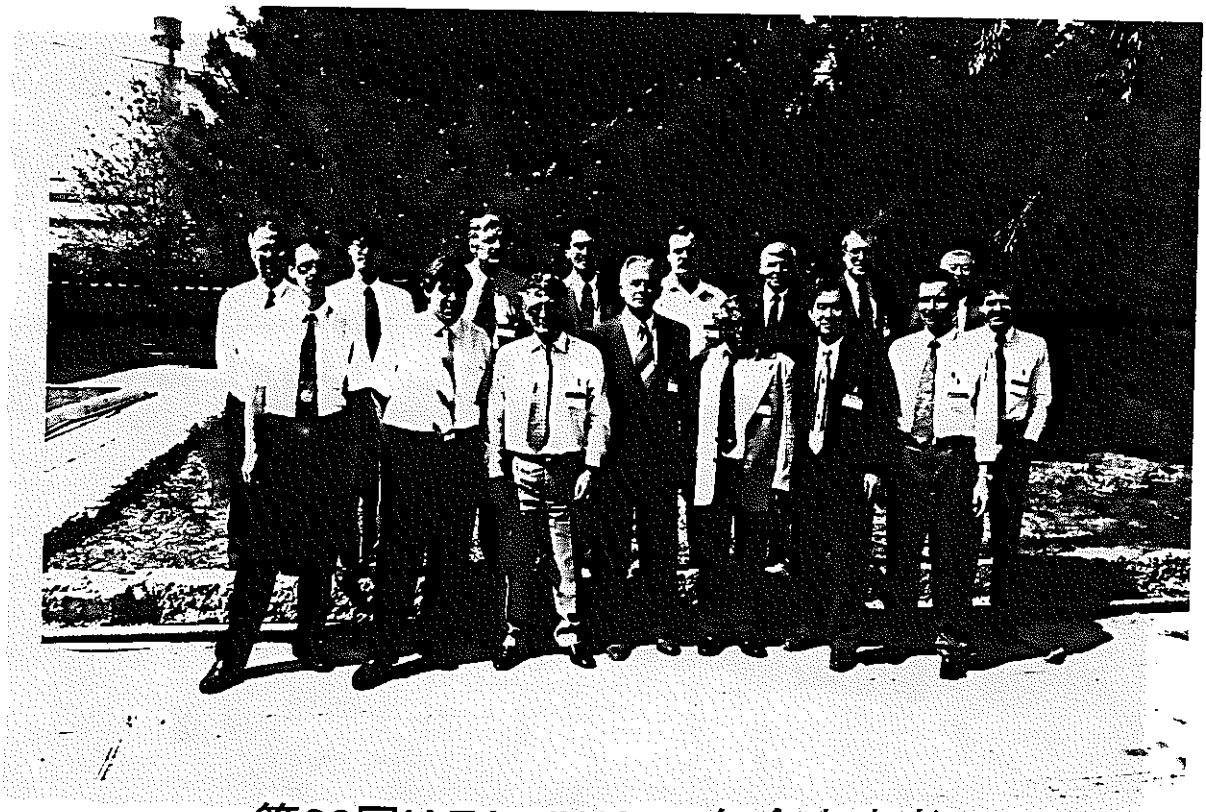
今回の会議日程の第３日（５月１６日）には、原型炉ＢＮ－３５０施設の見学の機会を得た。公開されたのは原子炉建物内の中央制御室、運転床、および原子炉上部室の３箇所ならびに同一敷地内の海水淡水化施設である。

ＢＮ－３５０原子炉建物内には、原子炉および１次主冷却系（６系統）が納められているが、格納容器の様な密閉型格納構造は設けられていなかった。また汚染管理・個人被ばく管理等の面で、西側諸国の施設と比べてあまり厳格でない様に思われた。

施設の運転は、中央制御室および隣接する燃料交換設備制御室において、原子炉からタービン発電機までの範囲を運転し、付属する海水淡水化プラントの運転監視は、原子炉から離れた別棟において実施していた。ＢＮ－３５０の運転体制は通常時は当直長＋運転員３名の計４名体制であり、異常発生時には支援要員を直ちに呼集出来る様にしているとのことである。

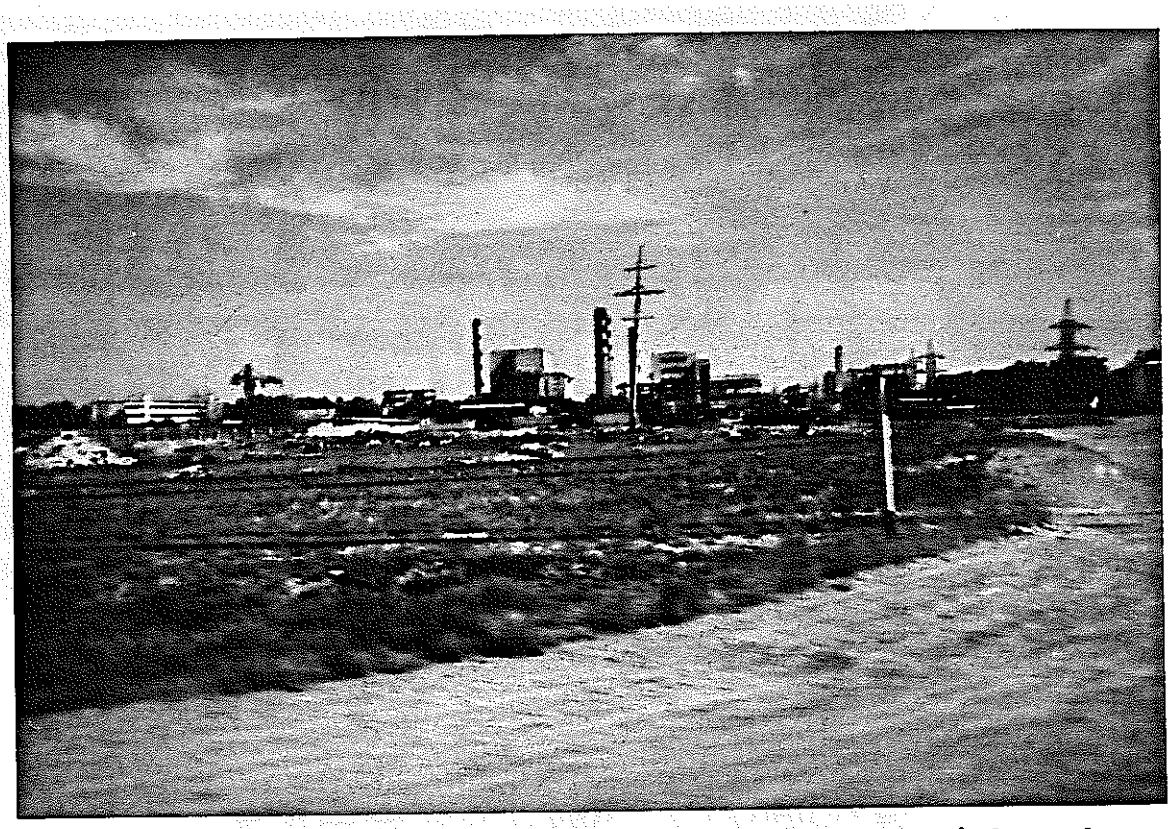
海水淡水化プラントは、原子炉からの発生蒸気を熱源として蒸留法により淡水を精製する施設であり、多段カスケード型の蒸留器を使用している。

原子炉は保守のため計画停止中であったため、海水淡水化施設は補助熱源により駆動していた。



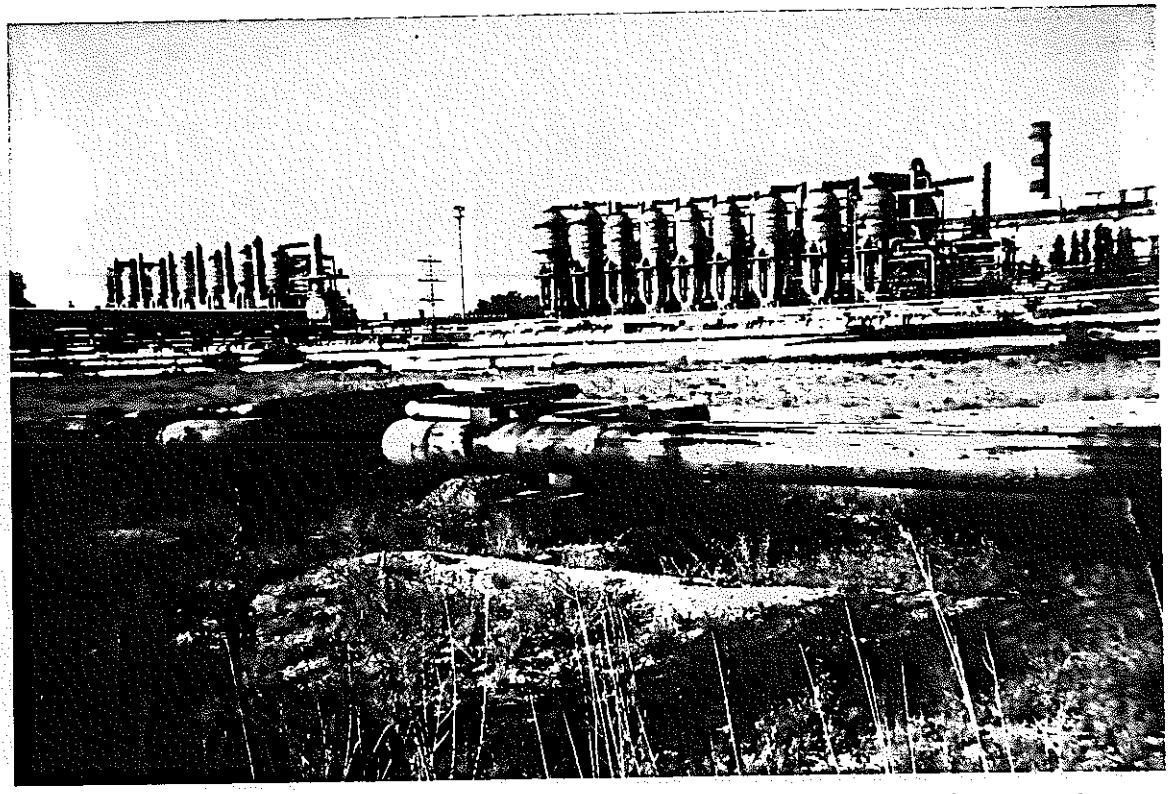
第29回IAEA / IWGFR年会出席者

第29回IAEA / IWGFR年会状況



マンギシュラク原子力エネルギーコンビナート

（MAEK、中央左がBN-350プラント）



マンギシュラク原子力エネルギーコンビナート

（海水淡水化のユニット、１１ユニットある）

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29th Annual Meeting of the International Working Group on Fast Reactors, Aktau, Kazakhstan, 14-17 May 1996

AGENDA

1. Welcome address MAEK, 9.30 am

2. Opening of the meeting

3. Introduction of IWGFR Members and Observers

4. Approval of the Agenda

5. Report regarding the activities of the Working Group

   5.1. Technical Committee Meetings

   a) Absorber materials, control rods and design of backup reactivity shut down systems for breakeven cores and burner cores for reducing plutonium stockpiles, Obninsk, Russia, 3-7 July 1995
   b) Conceptual design of advanced fast power reactors, Kalpakkam, India, 3-6 October 1995

   5.2. Consultancies

   a) IAEA/EC comparative calculations for hypothetical severe accident (ULOF) in a BN-800 reactor with near zero void core design, Brussels, Belgium, 26-28 July 1995, and in Vienna, Austria, 11-13 December 1995
   b) Intercomparison of analysis methods for seismically isolated nuclear structures, Vienna, Austria, 14-16 November 1995
   c) Harmonization and validation of fast reactor thermomechanical and thermohydraulic codes using experimental data, Vienna, Austria, 1-2 February 1996

   5.3. Consideration of joint research activities

   a) Co-ordinated Research Programme on Intercomparison of LMFR's Core Seismic Analysis Codes,
   b) Co-ordinated Research Programme on Acoustic Signal Processing for the Detection of Boiling or Sodium/Water Reaction,
   c) Benchmark on evaluation of the BN-800 NPP safety characteristics,

   d) New CRPs:
      1) Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures
      2) Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes Using Experimental Data

6. Consideration of conferences

IAEA's symposia:

a) "Nuclear Fuel Cycles and Reactor Strategy: Adjusting to New Realities", Vienna, Austria, 2-6 June 1997 (tentative).
b) "Desalination of Seawater with Nuclear Energy", Taejon, the Republic of Korea, 26-30 May 1997.

7. Consideration of arrangement for meetings in 1996.
   a) Technical Committee (TC) on "Creep fatigue damage rules to be used in fast reactor design", Manchester, UK, 11-13 June 1996.
   b) TC on "Evaluation of radioactive materials release and transport", O-arai, Japan, 11-13 November 1996.
   c) TC on "Fast reactor fuel development and performance" (Division of NENF), Vienna, Austria, 2-5 December 1996.

8. Selection of topics for meeting to be held in 1997.

9. Consideration of activities
   a) Status Report of LMFR
   b) Fast Reactor Data Base

10. National Reports on Status of Fast Reactor Development.

11. Other business (if any).

12. The date and venue of the 30th Annual Meeting of the IWGFR.

** ** ** ** ** **

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rt\16 April 1996

Technical Committee Meeting on "Progress of FBR Developments"
(29th Annual Meeting of the IWGFR)
Aktau, Kazakhstan, 14-17 May 1996

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INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors (IWGFR)

29th Annual Meeting of the International Working Group on Fast Reactors, Aktau, Kazakstan, 14-17 May 1996

Summary Report
(Draft)

1. INTRODUCTION

The 29th Annual Meeting of the IAEA International Working Group on Fast Reactors (IWGFRs) was held in Aktau, Kazakstan from 14-17 May 1996. The meeting was attended by the Members and some substitute members of the group from IWGFR Member States, Brazil, China, France, Germany, India, Italy, Japan, Kazakstan, Republic of Korea, the Russian Federation, as well as by observers from Switzerland and the United States of America. A list of participants is given as Annex I.

The meeting was welcomed by Mr. Piotr I. Nazarenko, Chief Engineer, Mangishlak Atomic Energy Complex & by Mr. Sergey Krechetov, Deputy Director General KAEA

The meeting was opened by Mr. P.-E. Juhn, Director, Division of Nuclear Power of the International Atomic Energy Agency and was chaired by Mr. Ch. Picker, of the United Kingdom.

During the Meeting a technical tour to LMFR BN-350 and desalination complex took place.

2. INTRODUCTION OF IWGFR PARTICIPANTS

The secretary informed the IWGFR about changes in its membership.

The Director General of the IAEA has approved Brazil's Membership in the IAEA's International Working Group on Fast Reactors with the wish of the Brazil Government and has designated Mr. Arthur da Cunha Menezes, Project Manager, as a Member of the IWGFR.

Participants introduced themselves and described briefly their activities and interests as relevant to the Working Group.

3. APPROVAL OF THE AGENDA

The Agenda of the Meeting (Annex 2) was accepted.

4. REPORT OF THE SCIENTIFIC SECRETARY

Reports of the IWGFR meetings for the period since the 28th Annual Meeting of the IWGFR had been distributed in advance to the participants of the Group. The Scientific Secretary summarized the main points of the IWGFR activities.

4.1. Technical Committee Meeting

Advanced liquid metal cooled fast reactors designs are currently under development in several countries. There was a need to identify the common trends to improve economics, reliability and safety and to evaluate the flexibility of fast reactor designs for managing plutonium stockpiles and to transmute minor actinides. To this end two TCMs (on Conceptual designs of advanced fast power reactors; and on Absorber materials, control rods and designs of backup reactivity shutdown system for break-even cores and burner cores for reducing plutonium stockpiles) were held in India and Russia respectively. These provided an important opportunity to discuss, review and compare different design approaches for ALMFRs with the purpose of further improving the common understanding of the respective rationale for selected design features and main characteristics for breeder and burner reactors.

4.2. Consultancies

A joint IAEA/EC benchmark exercise for hypothetical severe accident (unprotected loss of flow, ULOF) in a BN-800 reactor with near zero void core has been continued in 1995 and 1996. Two consultancies were held: One in July 1995 in Brussels and other in December 1995 in Vienna, to review and agree the results provided by organizations at the steady state core characterizations. The secretary informed the participants that the next consultancy meeting will be held in Brussels in June 1996 to discuss the results of calculations and formulate general conclusions about the first phase of benchmark.

4.3. Consideration of joint research activities

A CRP on "Intercomparison of LMFR Seismic Analysis Codes" aimed to validate and improve the codes used for reactor core seismic analysis through benchmark exercises.

The final RCM, which was held in Italy, in May 1995, summarized the validation of the codes on the basis of intercomparison of experimental and analytical data. It was noted that the studies carried out within the framework of this CRP have a general scientific value, since the results obtained can be employed in different types of reactors. A CRP on "Acoustic Signal Processing for the Detection of Boiling or Sodium Water Reaction in LMFRs" was initiated by IAEA in 1989. The final Research Coordination Meeting (RCM), held in July 1995 in Russia, concluded that the CRP has evaluated different signal processing techniques with real field data and hence the results could provide invaluable information to the designers of Acoustic Detection Systems. In the long run, this would lead to enhancing the economics of LMFRs through better surveillance of reactor core and steam generators. Signal processing techniques developed in this CRP for anomaly (boiling or leak) detection can be applied for detecting other types of anomalies in thermomechanical equipment (like cavitation, rattling etc.) in fast and thermal reactors.

The Scientific Secretary informed the participants that according to proposals of the 28th IWGFR meeting the Director General of the IAEA approved the following new CRPs. These had been preceded by consultancies in 1995 and 1996.

1) Intercomparison of analysis methods for seismically isolated nuclear structures (1996-1999),
2) Harmonization and validation of fast reactor thermomechanical and thermohydraulic codes using experimental data (1996-1998).

Mr. Rinejski noted that all necessary formalities for participation of: India, Italy, Republic of Korea, Russian Federation, UK and USA in CRP on "Intercomparison of analysis methods for seismically isolated nuclear structures" has been completed. The first RCM will be held in St. Petersburg, Russian Federation, from 27-31 May 1996.

In the frame of CRP on "Harmonization and validation of fast reactor thermomechanical and thermohydraulic codes using experimental data". France proposed a benchmark exercise on "Tee junction of LMFR secondary circuit". On behalf of the IAEA Mr. Rinejski thanked the French

representatives for the support and help in receiving the input data in "Tee Junction of LMFR Phenix secondary circuit" under a consultancy.

A question was raised with regard to the requirements for, and differences between AGM, TCM, CRP and consultancies. The Secretariat agreed to distribute a note outlining these.

5. CONSIDERATION OF IAEA'S SYMPOSIA

A Symposium "Desalination of seawater with nuclear energy" will be held in Taejon, the Republic of Korea from 26 to 30 May 1997. The background and scope of the symposium was indicated by Mr. P.-E Juhn. He pointed out that, considering the growing international interest in nuclear desalination, as well as ongoing programs in some Member States, the Agency has decided to hold an international symposium on the subject in 1997. The Republic of Korea has expressed willingness to host this symposium. For detailed preparations of the Symposium, a Steering Committee, consisting of representatives from nine Member States and relevant international organizations, has been established. At its first two meetings, the tentative program schedule has been defined. The official announcement shall be made shortly.

The objective of the Symposium will be to provide an international forum for the review and discussion of the status and prospects for seawater desalination processes using nuclear energy. The scope of the symposium includes the following items: (1) National programs; (2) Experience in desalination technologies; (3) Design and safety aspects of nuclear desalination complexes; (4) Economic aspects; (5) Market potential; (6) Technology transfer, and (7) Options for international co-operation.

About 150 participants (senior officials from government organizations, regulatory organizations, utilities, industry, research establishments and international organizations) from Member States with own nuclear desalination activities and Member States with interest in future deployment of nuclear desalination and interested international organizations are expected to attend.

A Symposium on "Nuclear Fuel Cycle and Reactor Strategy: Adjusting to New Realities" to be held at the VIC, Vienna, from 2 to 6 June 1997 in co-operation with OECD/NEA.

The objectives of the symposium is to provide a forum: (i) to prepare for decision-makers (including in nuclear-weapon states in relation to excess weapons material), and the public, scientifically assessed fuel cycle and reactor strategies for dealing with plutonium; (ii) to examine the policy options and explore the scope for international common understanding on these options; (iii) to examine the safeguard aspects of various disposition options; (iv) to enhance the transparency of the management and disposition of plutonium, and (v) to examine its scope for future international collaboration, in matters relating to the production, storage, use and disposal of plutonium.

At present time the Steering Group and six Working Groups have been formatted. Each Working Group will prepare a key issue paper, which will be printed as proceedings before the symposium. Deadline of key issue papers - end of 1996.

The IWGFR expressed concern about the representation on the Steering Group and Working Groups. It was believed that there was insufficient input on fast reactors. which have on important part to play in this topic. The Secretariat agreed to circulate a list of participants of the Steering Group and Working Groups. Members are requested to recommend alternative/additional representation to reflect the fast reactor views.

6. CONSIDERATION OF ARRANGEMENT FOR MEETINGS IN 1996

(a) TCM on "Creep fatigue damage rules to be used in fast reactor design", Manchester, UK, 11-13 June 1996.

An official agreement of the United Kingdom to host the meeting has been received at the IAEA at the beginning of January 1996. Official invitation letters with information sheet and call for papers were sent out to the Member States at the end of January 1996. The IAEA has been informed that the representatives from France, India, Japan, the Republic of Korea, the Russian Federation and UK will participate at the TCM.

(b) TCM on "Evaluation of radioactive materials release and transport in fast reactors"

An official agreement that the Government of Japan is, in principle, willing to host this meeting has been received the IAEA on 11 April 1996. The second host Government letter to ask Japan to host the meeting in question had been sent at the end of April 1996. Mr. Yamashita agreed the Japanese Mission will reply officially by 20 June. He proposed that the meeting should take place in Japan, on 11-14 November, will a change in title to read "Evaluation of Radioactive Material Release and Sodium Fires in Fast Reactor". His proposal was accepted at the meeting. The Secretary asked the members to speed up the nomination of the participants to the meeting.

(c) TCM on Fast reactor fuel development and performance, 9-12 December 1996, VIC, Vienna.

The secretary reminded Members that, after discussion during the last IWGFR meeting in the field of fuel technology of LMFR's, it was agreed that a meeting on the same topic could be organized by another group (IWGNF) in close collaboration with IWGFR. The participants asked that the IWGFR should be kept informed and recommendations should be requested from IWGFR in due course. Mr. P.-E Juhn informed the Working Group of the objectives and scope of the TC on Fast Reactor Fuel Development and Performance.

After discussion the Working Group agreed that this TCM will be organized in 1996, but suggested that the scope be concentrated on fuel aspects in order to avoid potential duplication in scope with 7.1 below. The Working Group suggested that the scope be changed to "Fast reactor fuel front and back-end solutions and related topics",. If written feedback is forwarded to the IAEA from members by the end June to ensure that the back-end topic can be included.

7. SELECTION OF TOPICS FOR MEETING TO BE HELD IN 1997

As a result of voting it was recommended that the IAEA in the frame of the IWGFR should organize the following TCMs:

1. Influence of high dose irradiation on core structural materials behavior, Obninsk, Russian Federation
2. Sodium removal and disposal from LMFRs in normal operation and in the framework of decommissioning, Aix-en-Provence, France.

The Russian and French delegates confirmed their willingness to host the meeting and proposed the following dates:

In Russia - June 1997
In France - November 1997

8. CONSIDERATION OF ACTIVITIES

Fast Reactor Data Base: The Secretary informed the IWGFR that the reference document had been published and distributed to member states.

Fast Reactor Technology - Status Report: The Secretary summarized the contents and noted that all contributions have already been received. Mr. Deitrich will advise the Secretariat by end May, whether he can provide new material on the safety philosophy of IFR for chapter 5. The document is under preparation for publication. The Secretary will distribute a draft for comment in June; comments and updates are requested by end August

9. PRESENTATION AND DISCUSSION ON NATIONAL PROGRAMME OF FAST REACTORS

Presentation of fast reactor development were made by Members of the Group of Brazil, China, France, Germany, India, Italy, Japan, Kazakstan, Russian Federation, United Kingdom, and Switzerland. Presentation were followed by lively discussions. Dr. Heusener proposed that the Fast Reactor countries should launch an international project similar to that for fusion and asked the IAEA to consider this. The Working Group considers this to be an interesting proposal. It emphasizes that these might be other alternatives such as broadening existing international co-operatin The Working Group recommends that the Agency considers this proposal and initiates appropriate steps in

order to have a deeper discussion on this proposal at the next meeting.

10. THE DATE AND VENUE OF THE 30TH ANNUAL MEETING OF THE IWGFR

It was agreed that the 30th Annual Meeting of the IWGFR will be held in China from 13-16 May 1997. It was proposed that the meeting be chaired by the Chinese representative, Prof. Xu Mi.

29am-sum.ar\rt\8 May'96

INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors (IWGFR)

REPORTS OF THE IWGFR MEETINGS

FOR THE PERIOD SINCE 28TH ANNUAL MEETING

(MAY 1995 - MAY 1996)

MARCH 1996

CONTENTS

1. 28th Annual Meeting of the IWGFR (Vienna, Austria, 9-11 May 1995).

2. Research Coordination Meeting of IAEA CRP on "Intercomparison of LMFR Seismic Analysis Codes", (Bologna, Italy, 30 May - 2 June 1995).

3. Technical Committee Meeting on "Absorber Materials, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles" (Obninsk, Russia, 3-7 July 1995).

4. Research Coordination Meeting on "Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium/Water Reaction in LMFR" (Obninsk, Russia, 11-14 July 1995).

5. Technical Committee Meeting on "Conceptual Designs of Advanced Fast Power Reactors", (Kalpakkam, IGCAR, India, 3-6 October 1995).

6. 2nd Consultancy on "IAEA/EC Comparative Calculations for ULOF in the BN-800 Reactor", (EC, Brussels, Belgium, 26-28 July 1995).

7. Consultancy on "Intercomparison of Analysis Methods for Seismically Isolation Nuclear Structures", (Vienna, Austria, 14-16 November 1995).

8. 3rd Consultancy on "IAEA/EC Comparative Calculations of ULOF in the BN-800 Reactor", (Vienna, Austria, 11-13 December 1995).

9. Consultancy on "Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes using Experimental Data", (Vienna, Austria, 1-2 February 1996).

INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors (IWGFR)

28th Annual Meeting of the

International Working Group on Fast Reactors,

Vienna, Austria

9 - 11 May 1995.

SUMMARY REPORT

1. Introduction

The 28th Annual Meeting of the IAEA International Working Group on Fast Reactors was held in Vienna (IAEA headquarters) from 9 to 11 May 1995. The meeting was attended by the Members and some alternate members of the Group from IWGFR Member States - China, France, Germany, India, Italy, Japan, Kazakhstan, Korea Republic of, the Russian Federation, the United Kingdom, as well as an observer from Switzerland. The list of participants is given as attachment I.

The meeting was opened by Mr. B. A. Semenov, DDG, Head of the Department of Nuclear Energy and Safety, who welcomed the participants of the IWGFR on behalf of the IAEA and gave a briefing on the status of nuclear energy and fast reactor development. He noted the additional possible important mission of LMFRs to transmute the long-lived nuclear waste.

The meeting was chaired by Mr. S.B. Bhoje from India.

2. Introduction of IWGFR members

Mr. A. Rinejski informed the IWGFR about changes in its membership.

The Director General of the IAEA has approved People's of Republic of China's full membership in the IAEA's International Working Group on Fast Reactors and has designated Mr. Xu Mi, Professor of China Atomic Energy Institute, as Chinese representative to the IWGFR.

The Director General of the IAEA has approved the Republic of Kazakhstan's membership in the IAEA's International Working Group on Fast Reactors with the wish of the Kazakhstan Atomic Energy Authority and has designated Mr. Levitin, Director of the fast reactor BN-350 Power Station, to participate in this meeting.

The Director General of the IAEA has approved the Republic of Korea's Membership in the IAEA's International Working Group on Fast Reactors in accordance with the wish of the Government of Republic of Korea and has designated Mr. Cho Mann, Manager of LMFR Development Project as a Member of the IWGFR.

The UK mission informed the Agency that Mr. Charles Picker (AEA Technology) and Mr. Ken Ainsworth (BNFL) are to replace Mr. J.I. Bramman as United Kingdom Members of IWGFR. Mr. Rinejski, Scientific Secretary of the IWGFR, mentioned that this would be John Bramman's last attendance at the IWGFR, and thanked him on behalf of the Agency and the other Member States' delegates for all his work over a long period as the UK member in the IWGFR.

3. Approval of the Agenda

The Agenda of the Meeting (attachment II) was approved by the participants.

4. Report of the Scientific Secretary regarding the activities of the Working Group

Reports of the IWGFR meetings for the period since the 27th Annual Meeting of the IWGFR had been distributed in advance to the participants of the Group. The Scientific Secretary summarized the main points of the IWGFR activities.

4.1. A TCM on "Evaluation of Material Coolant Interaction and Material Movement and Relocation in LMFR" was held at the O-arai Engineering Center, PNC, Japan, from 6-9 June 1994. A total of 34 contributed papers on experimental and analytical work was presented in eight technical sessions. The participants reviewed and discussed the recent progress in in-pile and out-of-pile experimental data (fuel and coolant interaction) and material movement and relocation during the whole spectrum of Core Disruptive Accidents (CDAs) postulated for Sodium Cooled Fast Reactors: (1) Sodium boiling and molten fuel-coolant interactions; (2) Molten material movement and relocation in fuel bundles; (3) Melt penetration and freezing and (4) Molten pool dynamics. The dynamics of material motion and interaction are especially important in the evaluation of core disruptive accidents. This is because the reactivity effects of material motion are crucial in determining the accident sequences and their consequences. The recriticality and resulting energetics potential is primarily determined by the dynamics of material motion. The dynamics of material motion also determine the final material distribution inside the reactor vessel, and this is related to the post-accident heat removal. The TCM covered the entire scope of aspects of material motions which may occur during CDAs. However, this meeting was not restricted to whole-core CDAs but also considered fuel-pin dynamics since this is an important concern in the event of inadvertent control-rod withdrawal. The general consensus of the participants was that the future direction of safety research should also consider some ways to harmonize designs, economical needs and reliable operation of future LMFRs.

4.2. Specialists's Meeting on "Correlation between material properties and thermohydraulics conditions in LMFRs", Aix-en-Provence, France, November 22-24, 1994. The meeting was devoted to thermomechanical aspects of temperature fluctuations in LMFR such as mixing jet phenomena, temperature gradient fluctuations and transfer of fluctuations from the fluid to the structure materials wall. One key objective of the meeting was to identify common trends in the interpretation of experimental and analytical work and the influence on design codes. A total of 20 contributed papers were presented in 7 technical sessions. The meeting noted that great advances have been made in the last few years to improve calculation methods for predicting fluid temperature fluctuation amplitudes and frequencies, involving simulation and direct solution of the Noveis-Stokes equation. Within a short time, when improved computing capacity currently planned has become available, it should be possible to provide information on fluid temperature at all times and at all important locations. When this level of detail in theromohydraulic calculations capability is reached, all the requirements for calculating stresses in structural materials will have been met. Two problems are outstanding, however. Firstly, the question of appropriate boundary conditions between more approximate fluid mechanics codes used to predict overall flow properties and codes to predict the fine detail of thermal fluctuation in critical regions has to be solved. Secondly, the adequate experimental validation including measurements of suitable detail and precision has to be provided. Significant progress has been made in substantiating the phenomenological basics of the material design codes available to reactor designers. Nevertheless, in spite of this improvement, the design codes are still very

restrictive in terms of allowable temperature amplitudes, because they still include a "safety factor" of about 2. It is clear that progress will be made only when the basis of this safety factor is scrutinized and a firm experimental justification for reducing it is provided. The meeting agreed that the best way forward for the IWGFR would be to organize benchmark calculation comparisons. Two possible benchmark problems were suggested: a square channel with transverse jet, and the Phenix expansion tank crack. The former would test thermohydraulic calculation and experiments, while the latter would test thermohydraulic and mechanical calculations.

4.3. IAEA/CEC Comparative calculations for a severe accident (unprotected loss of flow, ULOF) in BN-800 reactor with near zero void effect. At the Annual Meeting of the IWGFR in May 1994, it was proposed to begin an evaluation of the BN-800 reactor transient characteristics. After discussion with contributions of all member countries this proposal was unanimously supported. Mr. Balz of CEC indicated that the Commission of the European Communities (CEC) would support this benchmark. A joint IAEA/CEC benchmark exercise for a severe accident (ULOF) in a BN-800 with near zero void core has been endorsed by all participants. France, Germany, India, Italy, Japan (PNC and Hitachi), the UK and the Russian Federation have informed the IAEA that they will participate in the benchmark activities. The Scientific Secretary of the IWGFR has been in contact with Russian experts (Institute of Physics and Power Engineering, IPPE) and with CEC Whole Core Accident (WAC) Group to prepare the exercise. The main aim of the comparative calculation is: (1) to establish a basis for evaluating the pros and cons of a BN-800 type reactor with a near zero sodium void reactivity core design under hypothetical severe accident conditions, as far as an energetic ULOF is concerned, and (2) to analyze the conditions which allow avoidance of prompt fuel and steel melting in the improved fast reactor core under ULOF-type and other severe accident conditions. It was proposed to anlayze two cases: (1) fully unprotected loss of flow (all safety rods are out of operation); and (2) unprotected loss of flow (all active safety rods are out of operation but all passive safety rods keep their ability to work normally). The IAEA convened a Consultancy (at VIC, 5-6 December 1994) to review the general objectives and scope of the comparative calculations, to discuss and agree on the input data and to establish a work plan and work methods for the first year of the exercise.

4.4. A joint NENP and NENF TCM on "Unconventional Options for Pu Disposition" was held at Obninsk, the Russian Federation, 7-11 November 1994, to review and discuss unconventional methods and approaches for elimination of stockpiles and disposition of plutonium. The scope of the TCM was broad, covering all advanced reactor systems (light and heavy water, liquid metal-cooled, gas-cooled and molten salt reactors), their fuel cycles, and their capability for plutonium consumption. The participants noted that plutonium can be efficiently and effectively consumed in LMFRs, primarily by removing the uranium "Blanket". This capability has already been demonstrated. Elimination of Pu stockpiles and consumption can be further increased in a MOX core, by decreasing the fuel volume fraction, increasing core leakage, or by adding neuron absorbing materials. In addition, it may be possible to remove the Pu producing isotope U-238 from the fuel, replacing the core U-238 by inert matrix carriers, so that the net Pu consumption increases significantly. It was noted that the possibilities of Pu consumption and fissioning of minor actinides in LMFRs are of particular interest today. The participants concluded that the conventional options (MOX fuel in LWRs, LMFRs, and ATRs), that are already applied industrially, are able to cope efficiently with the disposition of separated Pu stockpiles, provided that the

manufacturing capacity for MOX fuel in increased adequately. Some of the other objectives and criteria are only partially met, or not met. A rather long list of future work was agreed upon.

5. Consideration of Conferences

The Scientific Secretary informed the IWGFR that a joint NENP and NENF Symposium on "Nuclear Fuel Cycles and Nuclear Power Reactors of the 21st Century-implications of various approaches" (preliminary title) is planned, in spring 1997, at the VIC, Vienna. The preliminary background and scope of the symposium were indicated by Mr. Rinejski. He pointed out that the nuclear community faces new realities which were not foreseen two decades ago. The rate of introduction of nuclear power in most countries is lower than earlier forecasts, and a closed commercial fuel cycle has not been fully realized. Conversely, there is an accumulation of plutonium from civilian programmes and dismantled nuclear warheads. Disposition of this large amount of plutonium is of serious international concern. Because of these developments the nuclear community should re-examine the future nuclear power reactor and fuel cycle options to understand the impact of various approaches. The Symposium will work towards attaining an understanding of the implications of different approaches, rather than achieving a consensus. The Symposium will examine two of the most important issues that have emerged: (1) various reactor and fuel cycle options from the standpoints of technology, economics, resource utilization, safety, safeguards and environmental impact, and (2) the methods of disposition of plutonium and of limiting future stockpiles deriving from nuclear power production.

After discussions with contributions of all members countries this symposium was unanimously supported. The members pointed out that the subject of this symposium is important to the fundamental strategic justification of the fast reactor which is founded on the long term capability of the system to breed and recycle plutonium. It is essential that in pursuing "short term" roles for the fast reactor, e.g. to reduce existing plutonium stockpiles, that this strategic justification is not put into doubt. The symposium needs to ensure that a balanced view is put forward from the fast reactor utilities, designers and fuel cycle companies in addition to the R&D organizations. This issue could be addressed firstly in the formulation of the symposium steering committee.

6. Consideration of joint research activities

6.1. RCM of the IAEA CRP on "Intercomparison of LMFR Seismic Analysis Codes", 26-28 September 1994, at PNC, Japan: The 2nd meeting was held to discuss and compare the experimental and analytical results obtained by various organizations for the analysis of French, Japanese, and Italian tests of Rapsodie, Monju and PEC mock-ups of validating LMFR structural codes. The discussions on the presentations covered wide ranging topics regarding the core seismic behaviors of the LMFRs.

The following points were readily appreciated by the participants. With the present state of the art computer power, it is very difficult to perform a whole core seismic analysis. It was suggested that the central row seismic analysis gives a conservative displacement and shock force. The effect of fuel pins on the natural frequency of the wrapper, which has so far been neglected, might be given a more closer look. This may become all the more

important in the context of irradiated sub-assemblies (S/A), wherein the pins touch the wrapper tube due to swelling. In the context of poor matching of displacement responses for the neutron shield elements between the experiment and the calculations, for the Rapsodie data, the need to look into the boundary condition details of the neutron shielding elements was stressed. There is a need for digitized experimental data, without which the comparison of results becomes very difficult. For the case of Monju data the Japan expressed their inability to provide the digital data, which are not presently available. There is a need for some experimental results for the Rapsodie in water data. These activities are consistent with the conclusion that detailed core seismic analysis is considered an important item to ensure reactor safety during an earthquake.

6.2. RCM of CRP on Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium/water Reaction in LMFR, held at Indira Gandhi Centre for Atomic Research, Kalpakkam, India, 1-3 November 1994. The work on this CRP on development of diagnostics systems which can enhance nuclear power plant surveillance is of interest to the operators of NPP. The first stage concerned acoustic monitoring of the reactor core by detecting sodium boiling which produce acoustic effects; the research programme has shown that there is a range of techniques available to detect boiling. The second stage of the CRP covered the acoustic detection of leaks in steam generators units (SGU). In February 1994, the participants were provided with data on background noise from 4 transducers on the SGU, of PFR in UK, leak noise from 4 transducers on the ASB loop in Germany, and mixed files of background and leak noise with various signal to noise ratios. The objectives were to determine the leak start time and duration, to assess the advantage of multichannel analysis over single channel processing for acoustic leak detection, to assess the reliability and false trip rate of the techniques employed and to evaluate, if possible, the location of the leak. The purpose of this meeting was to review and discuss the results of investigations carried out by the participating countries, to discuss and finalize the preparation of test data from experiments carried out on PFR - SGU for the year 1995 and to make recommendations for the future. The RCM has contributed significantly to the progress of the acoustic leak detection in test data made from mixing of background noises measured on power plant SGUs and of leave noise measured on loops. The possibility to detect leaks between 0.1 and 1 g/s in a few seconds was demonstrated. The RCM has made significant contributions and the successful completion of the CRP in 1995 would assist to the development of a reliable diagnostic system for enhancing the safety surveillance of SGU and LMFR.

The Scientific Secretary noted that participants agreed to conclude both CRP activities within the first half of 1995 (RCMs will be held in June-July 1995 in Bologna, Italy, and Obninsk, Russia).

6.3. New Research Programmes

Scientific Secretary pointed out that Coordinated Research Programmes (CRPs) are tools which are effectively used in Agency activities to promote exchange of scientific and technical information and assist in research and development in nuclear power reactor technology. CRPs are typically 3 to 6 years in duration and often involve experimental activities in selected technology areas of mutual interest to the participating countries. CRPs allow a sharing of efforts on an international basis and benefit from the experience and expertise of researchers from the participating institutes. Institutes in developing countries

participate in Coordinated Research Programmes through Research Contracts with the Agency, and institutes in industrialized countries participate through Research Agreements with the Agency. Institutes holding Research Contracts may receive small financial contributions from the Agency to carry out tasks within the objective and scope of the CRPs.

Mr. Rinejski reminded the participants that the IWGFR at its 26th meeting (4-7 May 1993) expressed a wish for the Agency to continue to support two CRPs. The Secretary informed the participants that Italy proposed a CRP on "Intercomparsion of Analysis Methods for Seismically Isolated Nuclear Structures" (attachment III). The IWGFR Specialist Meeting on "Correlation between Material Properties and Thermohydraulics Conditions in LMFR", (France, November 1994) proposed a CRP on "Verifications and Validations of the Thermomechanical Codes using Experimental Data" (Attachment IV). The Russian Federation proposed a CRP on "Harmonization of Thermal Hydraulic Relations for Advanced Fast Breeder Reactors" (Attachment V).

As a result of discussion, it was recommended that the IAEA should organize the following Co-ordinated Research Programmes: (1) "Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures", and (2) "Verification and Validation of the Thermomechanical Codes using Experimental Data".

It was asked to Mr. Le Rigoleur to examine the possibility that CEA provides the detailed experimental data for benchmark problems of the CRP on "Verification and Validation of the Thermomechanical Codes using Experimental Data" of the square channel with transverse jets obtained during the fluctuation temperature thermohydraulic experiments performed in Grenoble and the detailed data observed in the expansion tanks of Phenix which were presented during the TCM in Aix-en-Provence. An answer will be given during October 1995.

7. Consideration of arrangement for meetings in 1995

(a) TC on "Absorber Material, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles", at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia from 3-7 July 1995.

An official agreement of Russia to host the meeting has been received at the IAEA by the end of 1994. Official invitation letters with information and sheet and call for papers to the Member States were sent out by the Middle of February 1995. The Scientific Secretary asked the Members to speed up the nomination of the participants.

(b) TC on "Conceptual Design of Advanced Fast Power Reactors" at the Indira Gandhi Center for Atomic Research, Kalpakkam, India, 25-28 September 1995.

Some members requested that the date be changed to first week of October since an important meeting is taking place in last week of September 1995. The scope of the TCM may also be reviewed considering the time available and already published information on operating experience and R&D. This was agreed.

8. Selection of topics for the meeting to be held in 1996

A majority of members of the group agreed that two TCMs should be held in 1996. As a result of voting, it was recommended that the IAEA should organize the following TCMs:

- Creep fatigue damage rules to be used in fast reactor design (in UK)
- Evaluation of radioactive materials release and transport (in Japan).

The UK and Japanese delegates confirmed their willingness to host the meetings and proposed the following dates:

- In UK, June 1996,
- In Japan, November 1996.

9. Consideration of Activities

Fast Reactor Data Base: The reference document contains detailed data on a number of LMFRs - specifically plant parameters and design details. Each LMFR power plant is characterized by about 400 parameters, by design data and by relevant pictorial material. The document includes a total of 30 fast reactors that provides basic information to those interested in fast reactor technology. New additional data from China, France, India, Japan, Russia and the UK have been incorporated in the 1994 draft, which had been circulated to Members of the IWGFR in August 1994. Work is continuing on the drafting of eight appendices giving compositional analyses of structural materials used in constructing major components of fast reactors: much of the data for these appendices has already been supplied by IWGFR members but there are omissions. Scientific Secretary said that these eight appendices will be distributed to IWGFR with specific request for the supply of missing information.

Updating of "Status of Liquid Metal Cooled fast Reactors", IAEA, TRS No: 246. The Scientific Secretary informed the Group of the following contributions that the Agency received from members or had been prepared by Agency's staff:

- experience in construction and pre-operational tests of prototype LMFR Monju (Japan)
- LMFR Plant PFBR-500 (India)
- Experimental LMFR CEFR (China)
- US ALMFR and integral fast reactor (IFR) fuel cycle prepared by Agency.

The Scientific Secretary summarized the contents and noted that almost all contributions have already been received. He kindly requested Mr. Le. Rigoleur to assist with the Agency to receive the material of Chapter 5 "LMFR Physics and Core Design" (subscriber Mr. Fink of Cadarache), as soon as possible. It was agreed that both TECDOCs should prepared for publication in 1995.

10. Report regarding the activities of the nuclear materials and fuel cycle technology section (NENF Division)

Mr. Sukhanov summarised the main points of the Nuclear Materials and Fuel Cycle Technology Section's activities (attachment VI). After discussion it was agreed that meetings in the field of fuel technology of LMFR's should be organized under the aegis of the IWGFR or NENF in close collaboration.

11. Evaluation and recommendation for IAEA activities on LMFR (Project A.3.03 Liquid Metal Cooled reactors) for 1995 and 1996

Mr. J. Kupitz, Head-Nuclear Power Technology Development Section, presented tasks planned for 1995-96 for project A.3.03 Liquid Metal-cooled Reactors. He said that the IAEA wishes to evaluate the programme performance for 1995. The purpose of the evaluations is to provide sound decision making information for use in finalizing the 1997-98 programme and budget request.

After discussion it was unanimously agreed that the objective of the Project A.3.03: "To encourage, within the framework of the IWGFR, the international exchange of information on the improvement of fast reactor systems which are useful for achieving more effective utilization of fuel materials" is appropriate and of highest priority now.

With regard to the future of the IWGFR the Group observed that whilst some advanced fast reactor countries are experiencing reduced programmes and delay in the commercial deployment of FRs, an increasing number of other countries are planning for FRs and embarking on their own development programmes. In these circumstances the international exchange of information and experience becomes of increasing importance to ensure that no country is left to develop the technology in isolation and thereby to ensure safe practice in FR design and operation. The group therefore concluded that it was important that the activities of the IWGFR continue to provide what is the only forum in the world for such international exchange.

The experience gained from research and development programmes as well as the operation and construction of fast reactors has been reviewed periodically by the International Working Group on Fast Reactors (IWGFR) which is composed of leading specialists in national programmes from 10 Member States and 2 international organizations. International co-operation on fast reactors in the framework of the IWGFR has helped to attain a consensus on design principles and general features of fast reactors in Member States, although specific designs and national projects are different. Milestones of LMFR development are the restart of the 1200 MWe Superphenix and the 250 MWe Phenix in France, the commercial operation for 15 years of the 600 MWe BN-600 in Russia, the criticality of the 280 MWe fast reactor Monju in Japan in last year, and initiation of a fast reactor programme in the People's Republic of China and Korea, Rep.of. Other important milestones have been the successful operation of the 750 MWt fast reactor BN-350 in Kazakhstan for seawater desalination and the start of the CAPRA project for enhanced plutonium consumption in fast reactors (France with UK, Germany, Italy, Switzerland and Russia).

Members of the IWGFR noted that in the present situation the international co-operation relating to the development of fast reactors is highly desirable. Advice from the IWG allows the Agency's activities to be adjusted to the real needs and therefore assure their continuous relevance. The work of the Agency thus contributes to the development of LMR technology which is an option for optimal utilization of fuel materials and radwaste burning. All actions for the completion of 1995 tasks had been discussed and agreed at the IWGFR meeting. All planned project work activities for future years are consistent with recommendations made by Member States of the IWGFR.

In conclusion, the IWGFR asked the administration of the IAEA to continue the annual IWGFR meeting and recommended the topics of two TCMs and two CRPs.

12. Presentation and discussion on national programmes of fast reactors

Presentations on fast reactor development were made by the Members of the Group as well as by observer. Presentations were followed by lively discussions. Scientific Secretary informed the participants that the 28th Annual Meeting has received the paper presentation title "Review of Fast Reactor Activities in Brazil" from Mr. Artur Menezes, Instituto de Estudos Avançados/Centro Téchnico Aeroespecial. As introduced by the paper Brazil has a moderate programme to develop fast reactor technology. Mr. Xu Mi, representative of China suggested in the meeting, that the Agency might consider accepting Brazil as a Member State if Brazil make an official application to the Agency, in order that it could share the achievements of LMR technology development among Member States. This suggestion has been supported by all representatives of Member States.

13. Any other business

Mr. Pedersen of IAEA informed the IWGFR that the NPTD Section has initiated work on a TECDOC on "Definitions of Terms Describing Features of Advanced Reactors". The aims are to improve the understanding of widely used technical terms for characterizing new nuclear power plant designs, and clarifications regarding proper usage of these terms, as well as similar related terms. He said that a draft TECDOC has been prepared by two Consultants' Meetings, a copy of this draft was included in the handout material to the IWG participants and within a short it will be distributed to the other established IAEA IWGs for review and comments. The participants after discussion have agreed to this proposals. Comments were requested before the end of 1995.

14. The date and venue of the 29th Annual Meeting of the IWGFR

It was agreed that the 29th Annual Meeting of the IWGFR should be held in Shevchenko (Aktau), Kazakhstan, from 14 - 17 May 1996. It was proposed that the meeting be chaired by the Kazakhstan representative, Mr. Levitin.

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Technical Committee Meeting on "Progress of FBR Research and Developments"
(28th Annual Meeting of the IWGFR)
Vienna, Austria, 9-11 May 1995

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28th Annual Meeting of the International Working Group on Fast Reactors, Vienna, Austria, 9-11 May 1995, Conference Room: C07IV

AGENDA

1. Opening of the meeting - 09.00 am

2. Introduction of IWGFR Members and Observers

3. Approval of the Agenda

4. Report of the Scientific Secretary regarding the activities of the Working Group

5. Consideration of conferences
   The IAEA's symposium "Nuclear Fuel Cycles and Nuclear Power Reactors of the 21st century - Implications of various approaches" (Preliminary title), Vienna, Austria, Spring 1997.

6. Consideration of joint research activities
   a) Co-ordinated Research Programme on Intercomparison of LMFR's Core Seismic Analysis Codes,
   b) Co-ordinated Research Programme on Acoustic Signal Processing for the Detection of Boiling of Sodium/Water Reaction,
   c) Benchmark on evaluation of the BN-800 NPP transient characteristics,
   d) New CRP: - Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures (Proposed by Italy).
      - Verification and Validation of the Thermomechanical Codes using Experimental Data (Proposed during the SPM on Correlation between Material Properties and Thermohydraulics Conditions in LMFR- Aix-en-Provence, France, November 1994).
      - Harmonization of Thermal-hydraulic Relations for Advanced Fast Breeder Reactor (proposed by Russia).

7. Consideration of arrangement for meeting in 1995.
   a) Technical Committee (TC) on "Absorber Material, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for reducing Plutonium Stockpiles", Obninsk, Russia, 3-7 July 1995.
   b) TC on "Conceptual Design of Advanced Fast Power Reactors", India, 25-28 September 1995.

8. Selection of topics for meeting to be held in 1996.

9. Consideration of activities
   a) Status Report of LMFR
   b) Fast Reactor Data Base

10. Presentations and discussions on national programmes on LMFR development.

11. Any other business.
    Terms for describing new, advanced nuclear power plants (IAEA-TECDOC).

12. The date and venue of the 29th Annual Meeting of the IWGFR.

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CO-ORDINATED RESEARCH PROGRAMME ON "INTERCOMPARISON OF ANALYSIS METHODS FOR SEISMICALLY ISOLATED NUCLEAR STRUCTURES"

BACKGROUND

At the IAEA/IWGFR Specialists' Meeting on "Seismic Isolation Technology", held in San Jose (California, USA) in 1992 it was stressed that the seismic analysis and design of seismically isolated nuclear structures (reactors, spent fuel storage pools, etc.) requires that particular attention is paid to correct numerical modelling of the dynamic behaviour of the isolation devices. The reason is that the demonstration of seismic safety of isolated nuclear structures relies on the demonstration of the adequate performance of the isolation system and good predictability of such a performance, and that the latter cannot be achieved simply by experiments. In fact, on the one hand the capability of available facilities for testing large isolators to the necessarily very high seismic excitation levels is limited, and on the other hand experiments, which are rather costly, cannot cover the whole wide range of parameters which affects the behaviour of the isolators.

In particular, it is necessary that reliable numerical methods (i.e. validated based on experimental data) are made available for both the detailed evaluation of dynamic behaviour of isolation devices and for the assessment of simplified isolators' models to be used in the analysis of the isolated structure. The latter is necessary to also allow for the correct calculation of the seismic response of the superstructure itself and inner equipment.

Various isolation systems have been developed for nuclear applications: in addition to the neoprene bearings and their combination with sliding materials (which were adopted several years ago in the Cruas and Koeberg PWRs, and the La Hague spent fuel storage pools), advanced systems consisting of high damping rubber bearings, lead rubber bearings, rubber bearings or three-dimensional pneumocord devices combined with separate steel or viscous dampers have been considered for new nuclear projects in Europe, Japan, Korea, the P.R. China, Russia and the USA (ALMR, EFR, DFBR, NPR, VVER-500, spent fuel storage pools). The question of which among the aforesaid systems is the best for isolated nuclear reactors is still unsolved.

The definition of both detailed and simplified models of isolation devices is not an easy task, because these in general have a quite non-linear behaviour and because models shall apply to quite large deformations: the stiffness values of the isolation systems depend on displacement and other parameters, and for many isolator types damping is mostly hysteretic in nature (thus, again displacement-dependent). In addition, the high damping rubber, for instance, has a hyperelastic behaviour, hardening at large deformations, and requires careful evaluation of creep, temperature, radiation and aging effects; also, the plastic behaviour (under large deformations) of the inner reinforcement steel shims of rubber bearings shall be accounted for, and analysis of failure modes due to both deponding or rubber rupture shall be feasible, together with detailed evaluation of the effects of various possible initial defects.

SCOPE

Numerical models which enable both the detailed analysis of isolation devices and the assessment of simple models to be used in the analysis of isolated structures have been developed in several countries. These models have been implemented in existing codes or in special purpose codes, specifically developed for seismic isolation analysis (for instance, in Japan).

Validation of the models is in progress in each country mainly based on national experimental data. However, due to the previously mentioned complexity of the dynamic behaviour of isolation devices, depending on several parameters, the high cost of experiments on such devices, and the non-negligible number of isolation systems which have excellent potential for nuclear applications, it would be of great interest for each country to extend their validation studies to the analysis of experimental data obtained in other countries, so as to both check the reliability of the systems considered in national experiments in a more complete and less costly way, and to permit the evaluation of performance of other systems on which no national test experience is available. The comparison of different analysis methods and results, based on the use of different models and codes in different countries (for test cases for which experimental data are available to confirm adequacy of the numerical studies) would be very beneficial, in that it will enhance the quality and extent of validation of models and codes, and will allow for a better understanding of various physical phenomena.

PROPOSAL

According to the above-mentioned remarks, it is proposed that an adequate set of experimental data related to different isolation devices and different isolated structures is made available to all the countries participating in the CRP, and that these countries apply their models and codes to analyse such data.

As regards isolation devices, it is proposed that test data provided by each country should concern all the systems which are considered there as the most promising for nuclear applications. The data supplied should be adequate to allow performing both detailed analysis and analysis of isolated structures using such systems (geometry, material characterization, exact testing conditions at various excitation levels, measured results).

As regards the isolated structures, each country should provide data concerning mock-ups or real (civil) structures, isolated by the aforesaid systems, on which detailed measurement of the response was recorded at various elevations under dynamic tests or (possibly) in actual earthquakes (exact description of the structure and isolators, time-histories of the motion applied to the foundations, time-histories or recorded data).

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CO-ORDINATED RESEARCH PROGRAMME ON VERIFICATION AND VALIDATION OF THE THERMOMECHANICAL CODES USING EXPERIMENTAL DATA

BACKGROUND AND OBJECTIVES

In the coolant circuits of an LMFR, where there is turbulent mixing of coolant streams of different temperatures, there are temperature fluctuations in the fluid. If an item of the reactor structure is immersed in this fluid it will, because of the good heat transfer from the flowing liquid metal, experience surface temperature fluctuations which will induce dynamic surface strains.

In 1992 extensive cracking was found in a control rod guide tube that had been removed from the core of the UK sodium-cooled Prototype Fast Reactor (PFR). The cracks penetrated the full thickness of the guide tube.

In 1993, during the course of routine inspection, extensive surface cracking (leading in one case to a through crack) was found in the secondary sodium pump vessels on Phenix. Repair of the damaged vessels interrupted the reactor operating programme.

The IWGFR recommended to the IAEA to organize a Specialists Meeting on Correlations between Material Properties and Thermohydraulics Conditions in LMFR (Fluid temperature fluctuations and induced thermomechanical damage) to identify common trends in the interpretation of experimental and analytical work and the influence on design features.

The meeting noted that various computer codes for thermomechanical analyses have been and are being developed in the different countries. Great advances have been made in the last few years to improve calculation methods for predicting fluid temperature fluctuation amplitudes and frequencies. Significant progress has been made in substantiating the phenomenological basis of the material design codes available to reactor designers.

Two problems remain to be solved, however. Firstly, the question of appropriate boundary conditions between more approximate fluid mechanics codes used to predict overall flow properties and codes for predicting the fine details of thermal fluctuations in critical regions has to be solved.

Secondly, the adequate experimental validation including measurements of suitable detail and precision has to be provided.

The above mentioned meeting stressed that the best way to improve calculation codes would be organize benchmark calculation comparison. Two possible benchmark problems were suggested: the square channel with transverse jet (French experiment) and the Phenix expansion tank crack. The former would test thermohydraulics calculations and experiments, while the latter would test thermohydraulic and mechanical calculations.

At the IWGFR Specialist' Meeting on "Correlations between Material Properties and Thermohydraulics Conditions in LMFR", (Aix-en-Provence, France, 22-24 November 1994) it was proposed to perform benchmark analysis in order to verify and validate the thermomechanical codes using the above mentioned French experimental data.

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COORDINATED RESEARCH PROGRAMME ON HARMONIZATION OF THERMAL HYDRAULIC RELATIONS FOR ADVANCED FAST BREEDER REACTOR

1. BACKGROUND

Development of advanced fast reactors - power ones, breeders, utilizators of radiowastes and weapon plutonium - starts from design and technological studies, the essential features required of that are improved safety potential and acceptable economics. It becomes necessary to justify an engineering approaches to the reactor design and primarily thermal hydraulic aspects to meet all these requirements. This calls for more accurate and valid calculations of thermal hydraulics and more extended range of the processes investigated. Also, this bring to the fore the following topics:

- complex investigation of transient behavior in reactor equipment, taking into account important effects;
- multi-dimensional simulation of the processes under operating conditions being maximum approximated to real ones and based on the recent theoretical advances.

Now, in different countries there are a dozens of thermal hydraulic models and codes which are considered in reactor design. They all was developed at different times, based on a various prerequisites, as a rule, are imperfectly verified and allow the separate effects to be considered without regard for completeness of the phenomena. Only recent attention has been focussed in some countries on the development of combined codes which permit a thermal hydraulic consideration of a whole reactor loop.

Considerable advances have been made in the area of two-phase (multi-phase) thermal hydraulics and as applied to Pressurized Water reactors (PWR).

Thus, in this connection it becomes common that computer codes used for Water-cooled reactors are extended to liquid metal fast breeder reactors (LMFBR) without regard for heat and mass transfer in liquid metal under low pressure.

Points mentioned above indicate that international cooperation in studies remain to be carried out on garmonization of fast reactor thermal hydraulic relations derived in different countries approved in a wide range of parameters (nominal, non-nominal, transients, accidents).

2. OBJECTIVES

(i) To systematically list and evaluate requirements for thermal hydraulic relations in support of LMFBR systems during different conditions. To recommend a consistent set of thermal hydraulic relation based on analysis and estimation of models, constants, data available and their verification.

(ii) To draw up proposals to coordinated research program of experimental and theoretical studies of fast reactor thermal hydraulic relatins.

3. AREAS REQUIRED TO BE COVERED

3.1. Geometries, factors

3.1.1. Core

- pin bundle periphery
- distorted pin bundle
- non-uniform power release
- wire wrap
- lateral heat and mass transfer
- depressurization of elements
- blockage
- coolant boiling

3.1.2. Upper plenum

- coolant jets above fuel pins
- stratification
- construction above core
- temperature pulsation
- free surface of coolant
- heat transfer between coolant mirror and reactor cover

3.1.3. Intermediate heat exchanger

- lateral inlet and exit
- size of inlet and exit windows
- nonuniform temperature profile at inlet
- gas within flow

3.1.4. Bottom plenum

- symmetric and asymmetric transients

3.2. Situations

3.2.1. Stationary

- up to 10% of nominal power load
- natural circulation (reactor shut-down)

3.2.2. Transient
- fast in reactivity insertion
- slow:
  - power rise
  - flow drop
  - failure cooling

3.2.3. Steady conditions and transients, coolant boiling within core, depressurization of fuel pin, local coolant/fuel interaction

3.2.4. Complex heat transfer in upper plenum

3.2.5. Gas entrainment from coolant free surface due to eddies

3.2.6. Critical heat flux under depressurization or boiling behind blockage

3.3. Methods and parameters

3.3.1. Methods
- 3-D model for pool flow
- porous body and subchannel for pin systems
- homogeneous, drift, two-liquid for two- and multicomponent system
- k-e model of turbulent flow

3.3.2. Liquid metal relations (single- and two-phase)
- friction
- interphase heat and mass transfer
- mixing
- heat transfer, including boiling and condensation

4. METHODOLOGY

(i) Carefully examinate and compare fast reactor thermal hydraulic approaches and relations.

(ii) Draw up recomendations on relations to be used for various fast reactor behaviors.

(iii) Based on this selected data, validate and if required update the proposed thermal hydraulic relations for conditions.

(iv) Work out proposals to Coordinated Research Program of further experimental and analytical studies.

5. SCHEDULE

The estimated duration of this project is about 48 months. Thee first Research Coordinating Meeting (RCM) will be held in 1995. Subsequent RCMs will be held at approximately 1 year intervals, with final meeting in 1999.

6. COUNTRIES - PARTICIPATORS

The following countries are possible candidates for participation in RCMs:

- Russia (IPPE)
- Korea (KAERI)
- Germany (KfK)
- Japan (JAERI, EPDC)
- China (BIAE)
- Czech Republic (UJV Rez)
- India (Kalpakham)
- France (CEA, EdF)
- USA (LANL)
- Italy (CISE)
- UK (AEA, Nuclear Electric)

7. DELIVERABLE

The final report will contain the following proposed chapters:

1. Introduction.
2. Overwiew of
   - calculation approaches,
   - needed constant,
   - generalized relations for separate processes.
3. Recommendations on garmonization of thermal hydraulic relations used in fast reactor desigh.
4. Proposals to Coordinated Program for further investigations.

Summary of activities in advanced fuel technology

Nuclear Materials and Fuel Cycle Technology Section
(Division of Nuclear Fuel Cycle and Waste Management)

During the last years the Nuclear Fuel Cycle and Waste Management Division of the Agency has been giving much attention to the collection and the exchange of information in the field of reactor fuel technology. Nearly all activities in the area of Water Reactor Fuel is being conducted in the framework of the International Working Group on Water Reactor Fuel Performance and Technology (IWGFPT).

In addition to these activities, the Agency organized a number of meetings in the field of nuclear fuel technology, such as:

- Utilization of particle fuel in different reactor concepts (1983)
- Advanced fuel performance and technology (1985 and 1990)
- Thorium-based nuclear fuel (1987)
- Advanced fuel for fast reactors (1988)
- Unconventional options for Pu disposition (1994).
- Problems concerning the accumulation of separated plutonium (1994).
- Safe handling, transport and storage of plutonium (1994).

It is worth mentioning that many advanced nuclear fuel technology problems were analyzed at the KfK Conferences on Characterization and Quality Control of Nuclear Fuels (1987 and 1990) which were co-sponsored by the Agency.

Two Advisory Group Meetings were organized in the area of advanced fuel technology in 1984 and 1989. Participants from 14 countries reviewed experience in the subject area and gave advice and assisted the Agency in formulating its programme. The participants acknowledged the importance of holding meetings in these areas, and recommended to convene them at least every 4-5 years.

The third Advisory Group Meeting (entitled "Fuel Technology and Performance of Non-Water Cooled Reactor") was to review progress since 1989 on non-water reactor fuel technology and performance and to report on the current state of the art. An important objective of the meeting was to reach a consensus view on the most important topics in this area and, where necessary, recommend to the IAEA areas where further work might usefully be initiated or strengthened.

The summary report of the meeting was prepared by consensus during the meeting and is intended for publication as a TECDOC. The main part of the summary report consists of reviews of progress in the areas of High Temperature Gas Reactors (HTGRs), Fast Reactors (FRs), thorium fuels and the fuel fabrication technologies relating to all three areas.

It is worth mentioning that all meetings (which are planning in IAEA in the subject area) have to be organized in close collaboration with the Agency's International Working Groups on Fast Reactors and Gas-Cooled Reactors.

Summary Report of the RCM on "Intercomparison of LMFR Seismic Analysis Codes"

1. INTRODUCTION

The final Research Co-ordination Meeting was held in Bologna, Italy on 30 May - 2 June 1995.

The meeting was attended by participants from France, India, Italy, Japan and the Russian Federation. A list of participants is attached to this report (Appendix I).

The purpose of the meeting was to review the benchmark results obtained by various organizations for the analysis of Italian. French and Japanese tests on PEC, RAPSODIE and MONJU mock-ups. The key objective of the meeting was to make an overall review of the results of calculations with respect to experiments, prepare the final report of the CRP and make recommendations for the future work.

The meeting was chaired by Mr. Martelli of ENEA, Italy. The welcome address was made by Mr. Martelli on behalf of ENEA was followed by the opening remarks by Mr. Rinejski. The agenda proposed by Mr. Rinejski was approved (Appendix II).

This was followed by presentations of the representatives of Japan, India, Russian Federation, Italy, and France.

2. ANALYSIS PERFORMED BY ORGANIZATIONS

Japan: As the final report of the CRP, Mr. Morishita presented an overview of PNC's analysis work by the FINAS code on the PEC, MONJU, and RAPSODIE core mock-up experiments. Although these experiments varied in size, number of subassemblies, core configurations (free-standing or restrained), and excitation conditions, the results by FINAS in general were in good accordance to experimental data with a reasonable accuracy. From these results, he concluded that the present analysis method with the FINAS code was judged to be capable for practical use. In addition he also pointed out that there was a room for further improvement in the shock calculations (especially for MONJU, in which the shock stiffness is very high) and the fluid-structure interaction modelling (especially for RAPSODIE, in which the annulus between the core and the vessel is very narrow). He made some intercomparison of the analysis results obtained by the participants' codes on some of the typical cases. For the PEC and MONJU experiments, they gave fairly close results to each other and to the experiments. For example, the variation in the displacement results was within 20% for the PEC experiments. However, the variation among the codes was relatively large for the RAPSODIE experiment, which he attributed to the largest number of subassemblies in the mock-up. Finally, he identified some issues for further study from the viewpoint of more reliable seismic qualification of FBR core, which included fluid-structure interaction, response to 2D horizontal excitations, response to base isolation conditions, response to vertical excitations, and the effect of pin bundles.

Mr. Morishita also presented, on behalf of Mr. Kobayashi, Toshiba Corp., the final report on the results by the SALCON code. For the PEC and MONJU experiments, good agreements were seen between the analysis and the experiments in terms of displacement

responses. As for the RAPSOIDE experiment in air, the calculated displacement also agreed with the experiment. On the other hand, he made several analysis runs for the test in water, with changing the parameter regarding the fluid-structure interaction effect. Although it was possible to obtain good results, he (Mr. Kobayashi) pointed out the need for additional information to confirm the validity of the selected calculation condition. He also studied the effects of the direction of the input seismic load (flat-to-flat or corner-to-corner) by comparing the analysis results on PEC experiments. He opinioned from the study that the effect of input direction is negligible and that it was justified to use single row models with flat-to-flat input for seismic design purposes.

Mr. Horiuchi made presentations for himself and on behalf of Mr. Itoh, Mitsubishi Heavy Industry, Japan. In the first presentation, Mr. Horiuchi showed his calculation results mainly on Monju experiments. He found that it was difficult to get good results in the collision test with the provided stiffness. therefore, he used a modified stiffness in the seismic calculations. In the comparisons of displacements and impact forces, he obtained close values to the measured in the calculations. However. the calculated strains were smaller than the measured. He considered that this was caused by impulsive strain made by impact, which cannot be considered in the calculation model. His conclusion obtained from these calculations was that the program used in the calculations, SAFA, was verified and can be utilized for practical use. His recommendation for future research was to obtain precise vibration characteristics based on experiments and clarity the design criteria for determination targets of program improvements. Also, he noted that research on the method to consider fluid force is important.

In the second presentation, Mr. Horiuchi showed reviews of calculations made by Mr. Itoh using Mitsubishi-original code, FINDS. Although, there were some errors in the calculated results which may be caused by uncertainty of input parameters, Mr. Itoh concluded that FINDS can simulate the response of core structures well. In addition, the results on Monju experiment using not FINDS but ANSYS, a general-purpose FEM program, was shown. They were good in both displacements and impact forces. Mr. Itoh also made some recommendations for future research including the clarification of design criteria, such as control-rod-insertion time and structural integrity of subassemblies.

India: Mr. Vaze made a presentation of the final reports on the analyses performed at i) Bhabha Atomic Research Centre (BARC), Bombay, and ii) Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam.

A) In BARC, analyses were performed using a general purpose code COSMOS.

Following observations were made.

- The natural frequencies could be predicted well in most cases.

In PEC tests, this necessitated adjustment of spike stiffness;

- The displacement responses compared well with experiment. The calculated values showed good agreement with observed values both in magnitude and frequency content.

- The accelerations were over protected, but as pointed out during discussions this could be due to the filtering of data in experiments resulting from rather large time intervals between data acquisition;

B) In IGCAR, Kalpakkam, the analyses were performed using the in-house code CORE-SEIS.

The observations were similar to those made by BARC.

- For the MONJU tests, the high local stiffness posed numerical difficulties necessitating some adjustment.
- The RAPSODIE test was analysed as a row model and good results were obtained for fuel assemblies. But the results for neutron shield elements showed larger discrepancy.

The overall conclusions were as follows:

- It's necessary to perform simple tests on single assembly and on small clusters. Based on these tests, the parameters of numerical model such as structural damping, shock stiffness, shock damping, etc. need to be tuned to get the measured frequencies, impact forces.
- Using the numerical models based on limited test data, the computer codes can calculate the seismic response of LMFBR cores with a good accuracy.

It was recommended that the studies be extended to

i) design issues such as reactivity changes including effect of vertical ground motion, control rod insertability etc.;
ii) backward integration into computer codes for calculating the response of Reactor Block. It was suggested that calculations could be performed for a reference pool type LMFBR. Seismic isolation can also be incorporated in these studies.

Russia: As the final report of the CRP, Mr. Silaev presented an overview of OKBM's analysis work by the DINARA code on the PEC, MONJU and RAPSODIE core mock-up experiments. Tests were performed in air an in water. He reviewed a state-of-the-art of seismic experimental and analytical activities of OKBM, described the core seismic analysis computer codes, summarized the work has being done during 1990-1995. Mr. Silaev also described the results of validation and verification of the DINARA code using the reactor core mock-ups experiments. His conclusion obtained from calculations is that the code DINARA used in calculations was verified and can be used for practical purposes.

Italy: Mr. Martelli presented the analysis performed by ENEA of the results of shake table test which had been carried out by CEA in air and water on the Rapsodie core mock-up.

The features of the analysis had been explained in detail in Vol. 2 of the proceedings of CRP activities. The analysed tests concerned the entire mock-up, which contains a large

number of core elements, is characterized by rather reduced tank wall-to-core diameter ratio, and had been tested with and without restraint ring.

The study of ENEA (which was carried out using both the 1D code CORALIE and the 2D code CLASH) aimed at fully assessing the fluid-structure interaction (FSI) model which is applicable to the seismic analysis of restrained Liquid Metal Reactor (LMR) cores. The first purpose was to check the adequacy of the model that had been previously developed based on the results of shake table tests of groups of 7 and 19 PEC core elements (see IAEA-TECDOC-798).

The second purpose was to improve this model so as to account for the effects of the relatively small thickness of the liquid layer which surrounds the core. Both purposes were fully achieved, in spite of some uncertainties related to the core geometry (clearances at neutron element top), shock stiffness and element restraint conditions (non perfect encastre for the neutron shield elements).

Indeed, the analysis showed that:

1) the two-concentric shell FSI model which had been developed for the PEC vessel-core coupling, based on experiments on groups of PEC core elements in full scale, is also applicable to other LMR core geometries (in the case that these are restrained, at least), provided that the effects of the relatively thin liquid layer which is present between core and tank are correctly taken into account (the tank-core distance was quite large in the PEC experiments).
2) to account for these effects, the calculations of the related correction factor of the diagonal and non-diagonal added mass terms must be based on the radius of the circular section which has the same area as the core cross-section.
3) the "line" analysis based on the single core diameter is not adequate for cores like the Rapsodie mock-up (the interactions with the adjacent element rows are important, probably more than 2D effects). This may be attributed to the fact that the fuel and neutron shield elements of this mock-up are characterized by quite different natural frequency values.
4) shock stiffness (K sh) have a great effect on the results. Satisfactory agreement between calculations and measurements has been obtained using large K sh values, which are consistent with the assumption of simultaneous shocks on more than two wrapper tube faces and that of no effects of the modes neglected in the dynamic analysis (these assumptions were justified by the results of previous studies).
5) the nominal values of some parameters used for the dynamic analysis seemed not correct. This was especially the case, as mentioned, of the restraint conditions of neutron shield elements, which do not behave as encastres at high excitations. Also the gaps at the top of these elements seemed overestimated.

France: Mr. Fontaine, in his presentation concerned the Rapsodie core mock-up. Most of results have already been presented on the last CRP meeting. The main object of the presentation was some new calculations about in water tests. After a brief description of the subassembly models, the results about in-air tests were commented. It can be seen that the profile of displacement obtained from calculation was very close to the experimental one, despite some discrepancies concerning neutronic shield elements. Afterwards, the in-water

results were presented. An important point was made on the determinations of the lessening of the seismic input, due to the coupling between assemblies and vessel: without this lessening, the calculations over-estimated very much the displacements of assemblies.

3. DISCUSSION OF CALCULATION RESULTS

Each presentation was followed by a discussion. The highlights of the discussions are given below.

- In relation to the shock stiffness, Mr. Martelli observed that the shock stiffness changes depending on how many faces of the hexcan are loaded. The hexcan is very stiff when all the six faces are loaded and more flexible when only two opposite faces are loaded. An intermediate value is used in the analysis.
- One study compared the face-to-face excitation results with corner-to-corner excitation. It was shown that there is not much difference. However, a comment was made by participants that this may not be always true, because the gaps in the two directions are different.
- The shock damper is used in a manner such that it is effective in approaching and receding phases of the contact. Mr. Martelli observed that the damper may not be effective in the receding phase.
- It was pointed out that the acceleration of the assembly is a local phenomenon and is very sensitive to rate of data acquisition. In view of relatively slow rate, the comparison with theory should be made only after passing through a low-pass filter.
- The analyses which resulted in higher responses (by Mr. Silaev) ignored the reduction of input excitation due to off-diagonal mass terms. Inclusion of this factor will produce better comparison.
- Mr. Martelli pointed out that in view of the uncertainties in many of the parameters, these were better determined based on the results of experiments. Some of the parameters so determined are shock stiffness, damping, reduction factor for input excitation etc.
- It was proposed that the uncertainties in the values of gaps, frequencies, shock stiffness could be taken into account in a stochastic way. Mr. Martelli brought to attention, an earlier study performed by him using "response surface" method.
- As regards single row models, as opposed to cluster models, Mr. Martelli advised the use of cluster models which take into account the 2-D effect. There is also a difficulty in calculating reactivity change. He also said that 3-row model may be even worse than single row model.
- The importance of determining design criteria was brought out. Reactivity changes due to horizontal motion of fuel elements, insertability of control rods in a deformed core, and proper stress limits to preclude damage are some of criteria to be used in seismic design.
- Mr. Rinejski observed that the study conducted so far considers only the horizontal ground motion. An important effect of vertical excitation is the possibility of ejection of assembly from the diagrid because the difference between the downward acting weight and uplift due to pressure is not much. The participants agreed that this also needs to be considered.
- While discussing the recommendation for studying seismic isolation, Mr. Martelli pointed out that the isolation system is designed for a rather large earthquake. Due

to the nonlinear nature of isolator stiffness, it may not be very effective for a smaller earthquake.

4. MAIN RESULTS OF THE CRP AND RECOMMENDATIONS

Participants were representatives of nine organizations from five countries, e.g. CEA (France), ENEA (Italy), IGCAR and BARC (India), OKBM (Russia), and HITACHI, MITSUBISHI, PNC, and TOSHIBA (Japan). Three of them, CEA, ENEA and PNC provided their experimental data for the benchmark study of the reactor core:

- France (CEA): RAPSODIE mock-up (291 elements' cluster);
- Italy (ENEA): PEC mock-ups (to 19 elements)
- Japan (PNC): MONJU mock-ups (29 elements in a line, and 37 elements' cluster).

Meetings were held at IAEA, Vienna (Austria) in 1990, at ISMES, Bergamo (Italy) in 1992, again at IAEA Vienna (Austria) in 1993, at PNC, O-arai Engineering Center (Japan) in 1994, and at ENEA, Bologna (Italy) in 1995.

Data concerned tests in air and water, and included measured results for displacements, accelerations and shock forces. These were analysed using different computer codes, namely:

- CASTEM - 2000 (CEA);
- CORALIE and CLASH (ENEA)
- FINAS (PNC);
- FINDS (MITSUBISHI);
- SAFA (HITACHI);
- SALCON (TOSHIBA).
- CORE-SEIS (IGCAR);
- COSMOS (BARC);
- DINARA (OKBM).

The main results of the analyses can be summarized as follows:

1) All codes gave consistent results, provided that the same data were used, in spite of some differences in the methodologies applied (modal superposition or direct integration, etc).
2) In some cases, dynamic line analysis of the core (e.g. limited to the elements in the core diameter) may be sufficiently accurate; however, this feature should be carefully checked in each single case.
3) As regards the fluid effects, an accurate definition of the values of both diagonal and non-diagonal terms of the added mass matrix is necessary, in order to correctly evaluate both natural frequencies in the liquid and the effects on seismic load.
4) Some analysis of test data seems to indicate that fluid effects on damping are limited.
5) The effects of possible clearances at the spike of core elements have to be carefully taken with into account, depending on the excitation level.
6) The shock model to be used in the non-linear core analysis should be carefully determined as regards shock stiffness and damping and integration time step. The values of shock parameters shall take into account the possibility, if any, of simultaneous shocks on different hexcan faces.

7) The presence of some uncertainties related to some input data (e.g. restraint conditions, gaps, shock stiffness, seismic input, natural frequency values) somewhat complicated the analysis, although it did not hinder clear conclusions to be reached on the main topics.

The lessons learned from the study allow for the following recommendations:

1) The seismic analysis of the core shall be based on tests where the data (geometry, natural frequencies in air and liquid, restraint conditions, stiffness of hexcan section at shock locations etc.) shall be very well controlled and known (through specific measurements, if necessary).
2) Experimental validation and calibration of numerical models shall be based on comparisons concerning not only maximum response values, but also time-histories, RMS values and results in the frequency domain.
3) The indication that additional damping in the core elements due to the liquid is negligible, requires some more experimental evidence.
4) More detailed evaluations are also necessary as regards shock parameters, namely:

   - to assess shock damper coefficients before and after the maximum interference is reached between impacting elements;
   - to determine fluid effects, if any, on shock forces.

5) The use of some kind of probabilistic approaches is advisable to account for the effects on the response values, of various uncertainties in the input parameters. The limits of validity of such methods (e.g. Response Surface Methodology) shall be clearly kept in mind.
6) The two-dimensional effects on core response shall be carefully investigated. To this aim it may be appropriate to also perform two-directional simultaneous horizontal shake table tests on core cluster mock-ups in air and water, and to perform detailed non-linear analysis of the test data.
7) The need for detailed core analysis still remains for the isolated reactors, in order to avoid tuning and check the effects of minor earthquakes.
8) For some reactors, detailed analysis of the vertical excitation effects on core elements (uplift, reactivity changes, etc.) might be necessary. However different numerical tools shall be used to this purpose, with respect to those considered in this CRP.
9) As regards the use of the results of seismic analyses for the evaluation of structural integrity, scramability and reactivity effects, an international discussion would be very useful.

The participants pointed out that the studies performed in this framework were very useful for the verification and improvement of the seismic analysis methodologies of the reactor core.

The Italian representative described the scope and objectives of a new CRP on "Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures" (Appendix A). After discussion the meeting recommended to the Agency to approve the above-mentioned CRP and asked the Agency to convene a Consultancy Meeting within 1995.

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Summary Report of the TCM on "Absorber Materials, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles"

1. INTRODUCTION

The IAEA Technical Committee Meeting (TCM) on "Absorber Materials, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles", was hosted by the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia, from 3 to 7 July 1995. This was the third IAEA meeting held on the subject of absorber materials and control rods for Liquid Metal Fast Reactor (LMFR). The other two Meetings were held at Dimitrovgrad, Russia (1973) and Obninsk, Russia (1983).

The countries participating in the TCM were France, Germany, India, Japan, Republic of Kazakhsan, Russia and Republic of Georgia (Observer). Thirty-five specialists participated in the TCM. The participants were welcomed by Mr. V. Poplavsky, Head of Fast Reactor Department, IPPE, MINATOM. The meeting was opened by the Scientific Secretary, A. Rinejski of the International Atomic Energy Agency, and was chaired by V. Matveev, Director, Division of IPPE, and Chairman of the Local Organizing Committee. Messrs. R. Voznesensky and A. Ivanov also served as scientific secretaries from the host organization.

The meeting had seven sessions. The sessions were chaired by B. Kryger (France), R. Babu (India), M. Edelman (Germany), K. Okada (Japan), T. Kaito (Japan), D. Favet (France), V. Risovanny (Russia), and J.P. Truffert (France).

The main topics of discussions were:

- status of designs of control rods for fast reactors and experience with their operation,
- properties and behaviour of absorber materials for control rods; results of post-irradiation examination of absorber materials, and mechanisms affecting their properties and behaviour,
- design of a backup reactivity shutdown system utilizing passive mechanisms:
  - curie point electormagnetic mechanism,
  - enhancement of thermal expansion of absorber rod drive lines,
  - hydraulically suspended control rod,
  - gas expansion modules in the core; and
- the possibility of optimizing the reactivity coefficients and the efficiency of Pu burning by using absorber and moderator materials in the core.

A total of 21 papers were presented, and a technical tour of the IPPE during the TCM also took place.

2. TECHNICAL BACKGROUND

A safe, reliable and economic operation of nuclear reactors depends upon the reliable operation of reactor shutdown and reactor regulation systems. In liquid metal cooled fast reactors, these consist of control assemblies, drive mechanisms, guide tubes and other structural components. The main element of the control assembly is the control rod, which

consists of neutron absorbing material and is (usually) clad in stainless steel. Neutron absorbing materials in development include the ceramic boron carbide and europium oxide. Current designs of control assemblies have worked well, but the latest development has shown that their lifetime in the core and reliability could be increased by improvements in design.

Fast reactor R&D programmes of some countries reflect the effort to enhance the flexibility of fast reactors to manage the plutonium stockpiles. Studies performed recently have demonstrated that using natural boron in combination with $^{11}B_4C$ could improve the safety of fast reactors in the plutonium burner mode of operation.

Due to improvements in the reliability of operation of reactivity shutdown and decay heat removal systems, core melting accidents may be ignored as highly improbable. The first step to prevent core melting is to be able to shutdown the reactor under all imaginable circumstances. For advanced fast reactors there are two independent reactor shutdown systems: a primary conventional "active" system and a backup passive system. At present, there are many proposals to cause the passive control rods to be inserted into the core.

The IWGFR proposed that the IAEA organize a TCM on "Absorber materials, control rods and designs of backup reactivity shutdown systems for breakeven cores and burner cores for reducing plutonium stockpiles", to review:

- the operational experience with absorber elements and control rods in fast reactors;
- the effect of this experience on control rods design;
- the properties and behaviour of absorber materials under irradiation; and
- the development of advanced passive backup shutdown systems.

3. DISCUSSION OF PAPERS, CONCLUSIONS AND RECOMMENDATIONS OF THE TCM

3.1. Status of control rod designs for fast reactors and experience with their operation

Significant experience, accumulated on absorber materials for fast reactor control rods, was presented at the meeting. Boron carbide of high enrichment (45-90% of ^{10}B) and vented absorber pins (with sodium bonding) have been successfully used in the Phenix, BN-350, BN-600 and Superphenix (SPX) reactors. The control rods' lifetime for the BN-600 reactor is 500 effective days (e.d.); the lifetime predicted for the SPX reactor (based on the tests in the Phenix reactor), is 640 e.d. Boron carbide is the favoured absorber material. Europium oxide (Eu_2O_3) appeared to be an alternative to boron carbide of natural enrichment.

Investigation of properties and behaviour of B_4C have been carried out in BN-350, BN-600, BOP-60, Phenix, and JOYO reactors for a broad range of conditions affecting its endurance. Data from these studies defining the behaviour of irradiated absorbing materials, (both pins and the whole rods), and the results provide the basis for improved control rod designs and extended control rod life times.

Swelling of the irradiated boron carbide is one of the main factors limiting the life time of the rods. The dependence of boron carbide swelling rate ($\Delta d/d\%/B\%$) on the burn-up was determined from post-radiation studies of control rods from the BN-600 reactor (fig.1). After about 500 effective days the boron burn-up was ~ 18%. A minimum swelling rate of ~ 0.1% for enriched boron carbide and 0.2-0.4% for natural boron carbide was observed at a temperature of 500°C. At a working temperatures of about 800°C, the swelling rate amount is ~ 1.0% per percent burn-up for natural boron carbide and 0.6% per per cent burnup of enriched boron. Some reserve remained, since the gap between the absorber and cladding was not filled completely. Under these conditions, mechanical interaction with the cladding was non-observed, and the structural material retained a sufficient reserve strength.

A series of tests were carried out in the "Phenix" reactor irradiating an enriched boron carbide rod for 240 e.d. These experiments showed that the in-pile residence time of the pin is limited by the mechanical interaction of boron carbide with the cladding, due to fragments entering the gap between the absorber and the cladding.

Irradiated boron carbide data were also obtained in Japan in tests in the experimental fast reactor JOYO. Burn-up of about 23% ($230*10^{26}$ capt/m^2) was achieved at maximum temperature of 1400°C. The dependence of pellet swelling and gas release on burn-up was measured (fig. 2). Japanese and French results are in good agreement but differ in some cases from Russian results. It appears that the differences can be explained by differences in methods of obtaining enriched boron carbide. However, the results obtained are not contradictory, and they improve the understanding of the swelling and gas release processes.

It was noted, that to achieve control rods lifetimes of 1000 e.d. will require new absorbing materials and new pin designs. Some measures to increase the control rods lifetime for existing reactors were proposed by the French. The first proposal was to reduce the capture rate in boron carbide by lowering the ^{10}B enrichment of the B_4C pellets in the lower part of the control rods pins. The second proposal was to prevent the fragment relocation by providing a thin stainless steel shroud to enclose the pellet stack.

Considerable progress has been made in Russia in the design and manufacture of rods containing absorbing (boron carbide) and moderating (zirconium) hybrid materials. Using this design with its life-time of 450 e.d. would result in considerable saving of high cost enriched boron.

The TCM participants agreed that the most efficient way to utilize high enrichment boron is to separate it from irradiated materials. The development of technology for reprocessing irradiated boron carbide, which provides complete removal of radionuclides from irradiated materials was presented at this meeting. This permits repeated use B^{10} enriched B_4C in fast reactors.

It became clear after discussion that the French control rod calculation code REGAIN can analyze both steady state and transient pin behaviour and covers thermal, mechanical and some chemical characteristics. For boron carbide materials the code can deal with both sodium bonded and helium bonded design concepts. It was agreed that REGAIN which runs on a classical work station, is an easy code to use.

The meeting participants concluded that the problems of reliable and effective control rods for sodium cooled fast reactors are being solved.

3.2. Optimizing core safety parameters for a plutonium burner reactor using absorber and moderator materials

Only France presented a paper on this topic. It was noted that two approaches to optimize a fast reactor core for plutonium (or minor actinide) burning have been considered in France: the "dilution approach" (about two third of pins are fissile pins, the remaining one third are fuel free) and "poisoning approach" (the absorber B_4C is introduced in the diluent subassemblies). At present, "the dilution approach" is preferred since it allows better core safety parameters (mainly the Doppler constant). Data presented indicated that the introduction of the moderator in the fuel bundle of the B_4C poisoned core strongly increases the Doppler constant (+60%) and reduces the sodium void reactivity worth (-20%). As a result the "poisoned" core option with $^{11}B_4C$ has similar plutonium burning characteristics and better safety parameters that the "dilution" option. Its main advantages result from the large diameter fuel pins which increase fuel lifetime using a standard fuel bundle. The use of $^{11}B_4C$ as filling material does not introduce additional technical problems.

It was concluded that the "poisoned reactor option" deserves further investigation

3.3. Advanced passive reactor shutdown systems

The meeting participants noted that in classic core disruptions (unprotected loss of flow - ULOF, and unprotected transient overpower - UTOP), the reactor can be shutdown only by inherent passive reactivity feedback mechanisms. Therefore, special measures to provide additional negative reactivity are needed.

Russia has developed a backup passive shutdown system based on hydraulically suspended absorber rods (HSAR) which drop into the core when the coolant flow rate in the core falls below 50% (fig. 3). Full-scale HSARs for the BN-600 and BN-800 reactors have been tested. This system is capable of shutting the reactor down in ULOF transient.

The German system for passive shutdown of the European Fast Reactor (EFR) uses enhanced thermal expansion of control rod drive lines (CRDL) to provide automatic scram (fig. 4). This system forces control rod into the core when pre-set coolant temperatures are exceeded. This device uses a hydraulic expansion module and two individual (coaxial arranged) drive lines. The sodium has a large thermal expansion coefficient than the container material. This system is capable of shutting the reactor down in ULOF and UTOP transients.

The demonstration fast breeder reactor (DFBR) in Japan includes the following passive backup shutdown systems:

- a gas expansion module (GEM), and
- a self-actuated shutdown systems (SASS) using a curie-point magnetic alloy.

In addition to GEM and SASS, a feasibility study of the enhanced thermal expansion of CRDL has been conducted.

The French presentation described and evaluated a third shutdown level for the EFR project which has been developed, implementing the following four new devices systems:

- a system which terminates the power to the absorber electromagnets after a loss of primary pump electric power supply,
- a device for CRDL passively enhanced thermal expansion,
- a device which overcomes control rod jamming by motorized insertion of absorber rods, and
- a mechanical stroke limitation device which passively terminates the withdrawal of the faulted control rod.

The inclusion of these systems in the EFR project leads to a significant improvement of the shutdown function.

The shutdown system of the Indian prototype fast breeder reactor (PFBR) incorporates diverse and redundant features and includes the following systems.

- a system which terminates the power supply to the absorber electromagnets after a loss of pump power,
- a gas expansion module,
- a cure point magnetic switch

It was concluded, that with the introduction of these passive backup shutdown systems, future advanced LMFRs will have a very high degree of safety.

3.4. Recommendations

The TCM participants recommended to the IAEA/IWGFR:

- to convene a consultants meeting in 1996 to assess the feasibility of using by Member States the REGAIN code developed in France,
- to establish a Data Bank on LMFR shutdown systems faults, and
- to convene a specialists meeting in 1998 to review and discuss the status of R&D on passive reactivity shutdown systems and reactor cores designed to have a large negative reactivity feedback mechanism.

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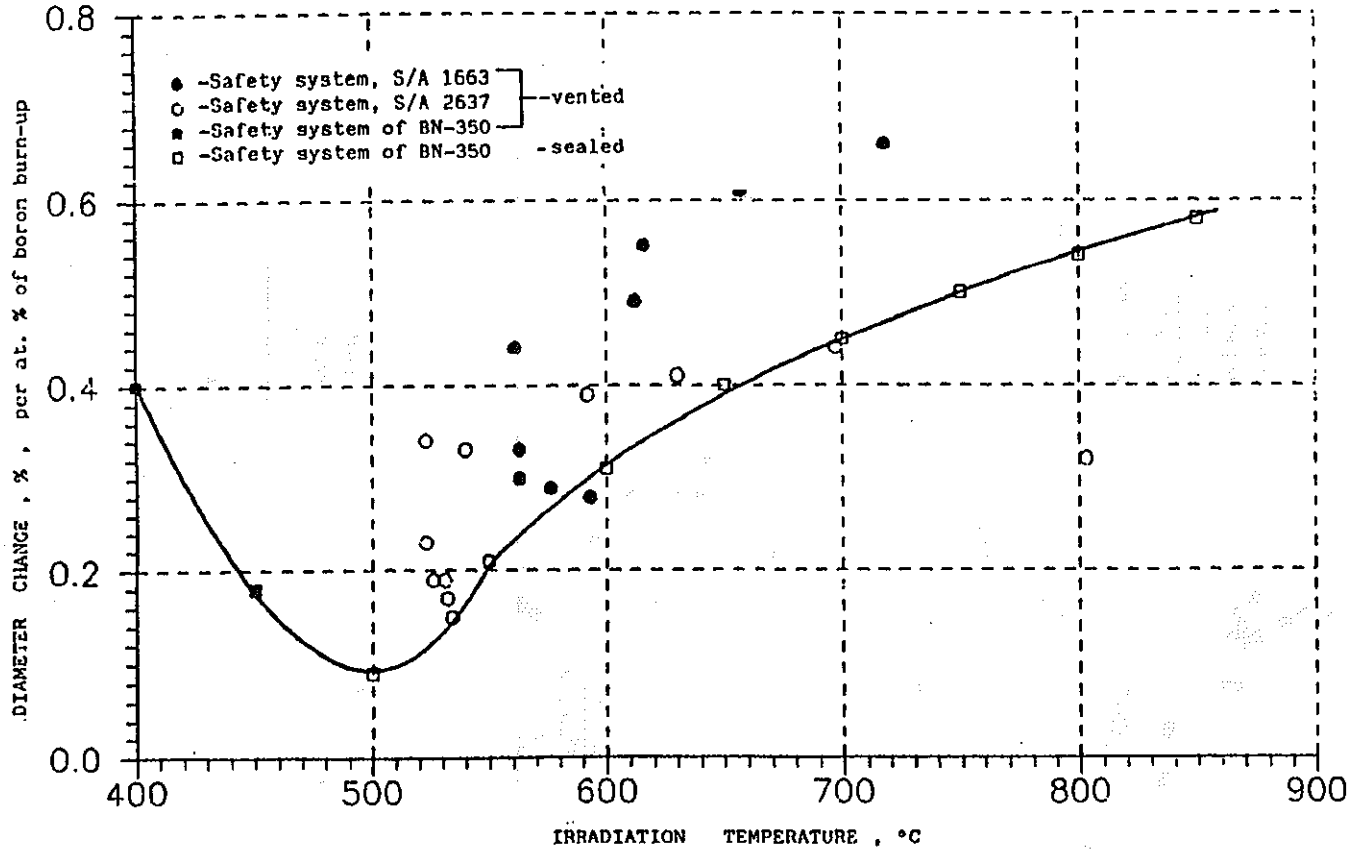
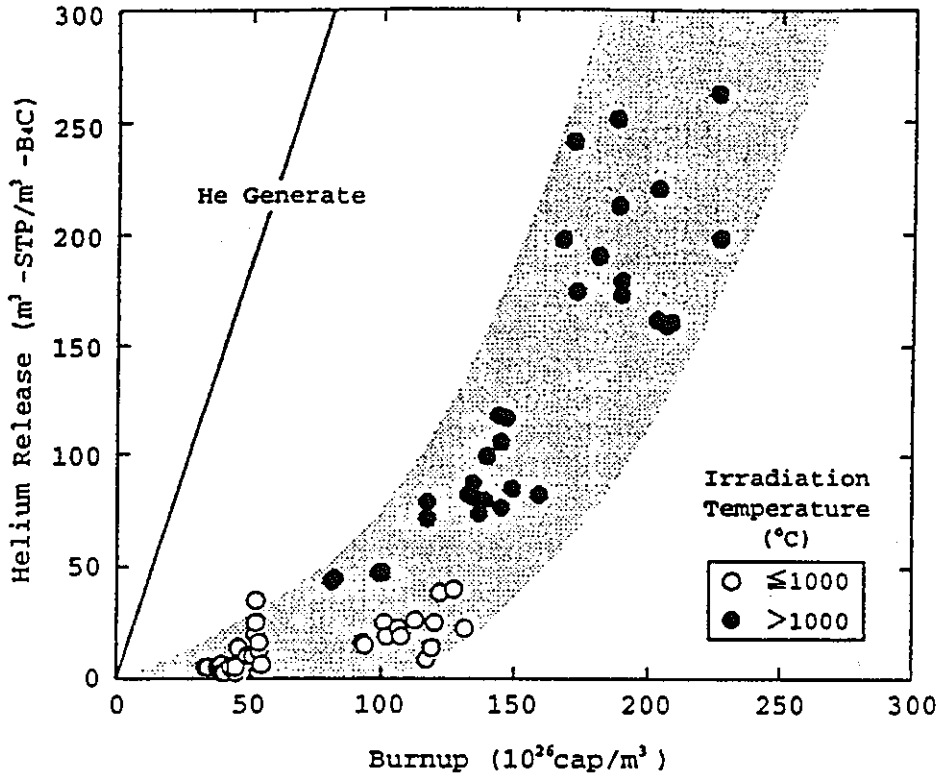
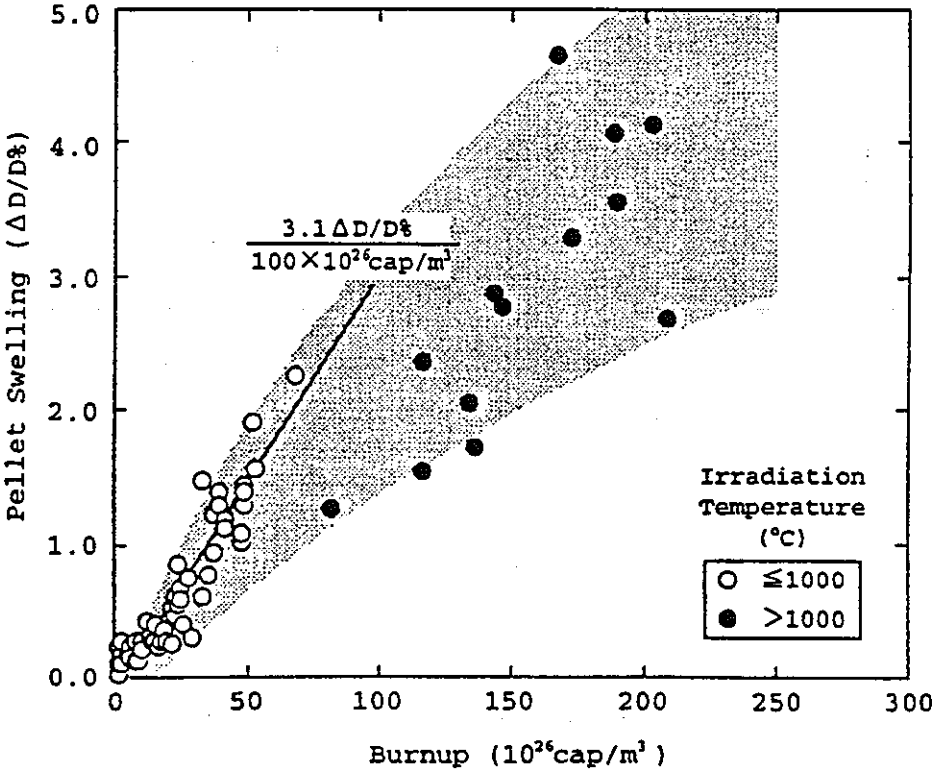


LINEAR SWELLING RATE OF BORON CARBIDE (60-90 at.%B-10)
AS A FUNCTION OF IRRADIATION TEMPERATURE.

Fig.1



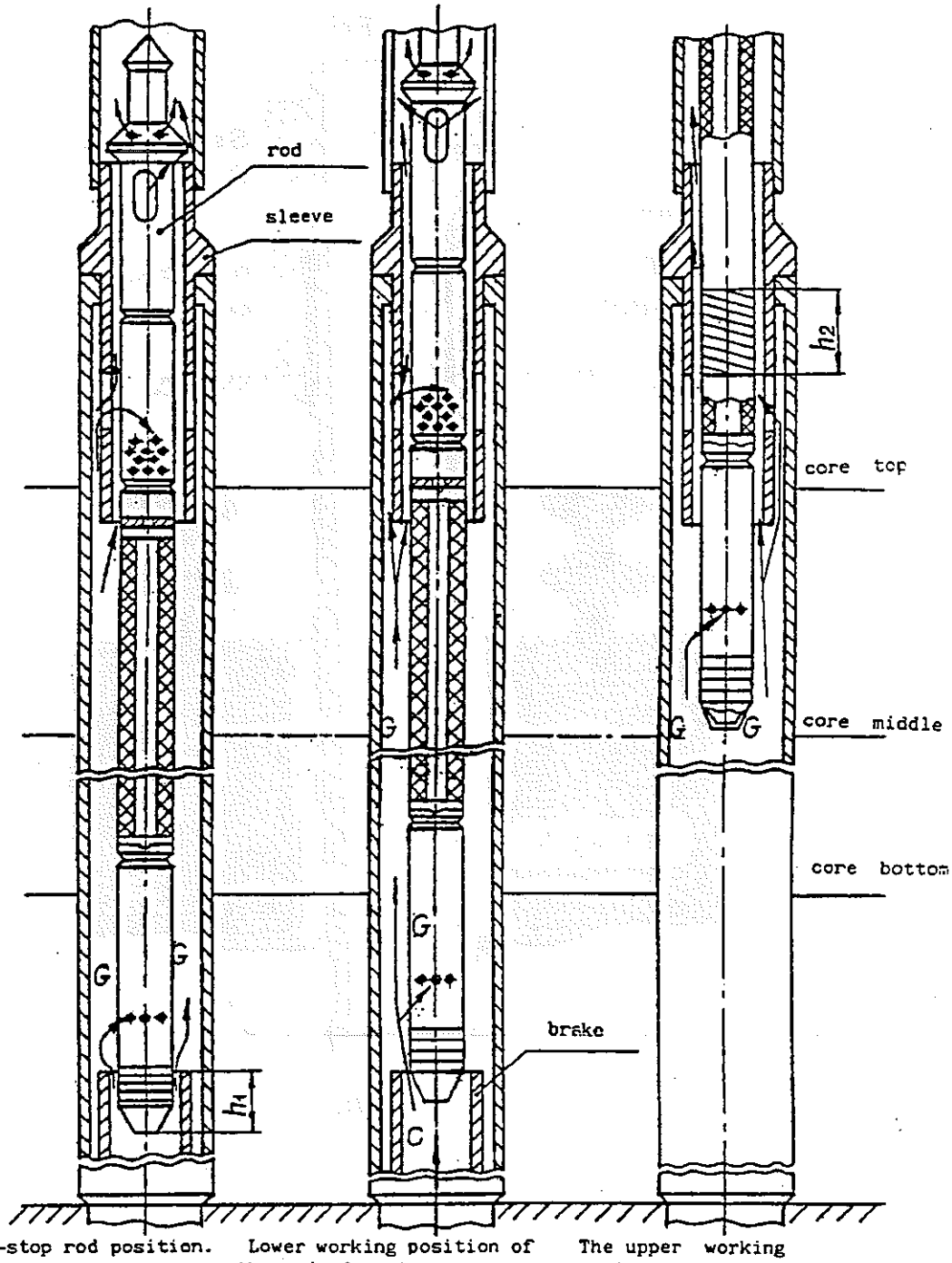
Burnup Dependence of Helium Release



Burnup Dependence of Pellet Swelling

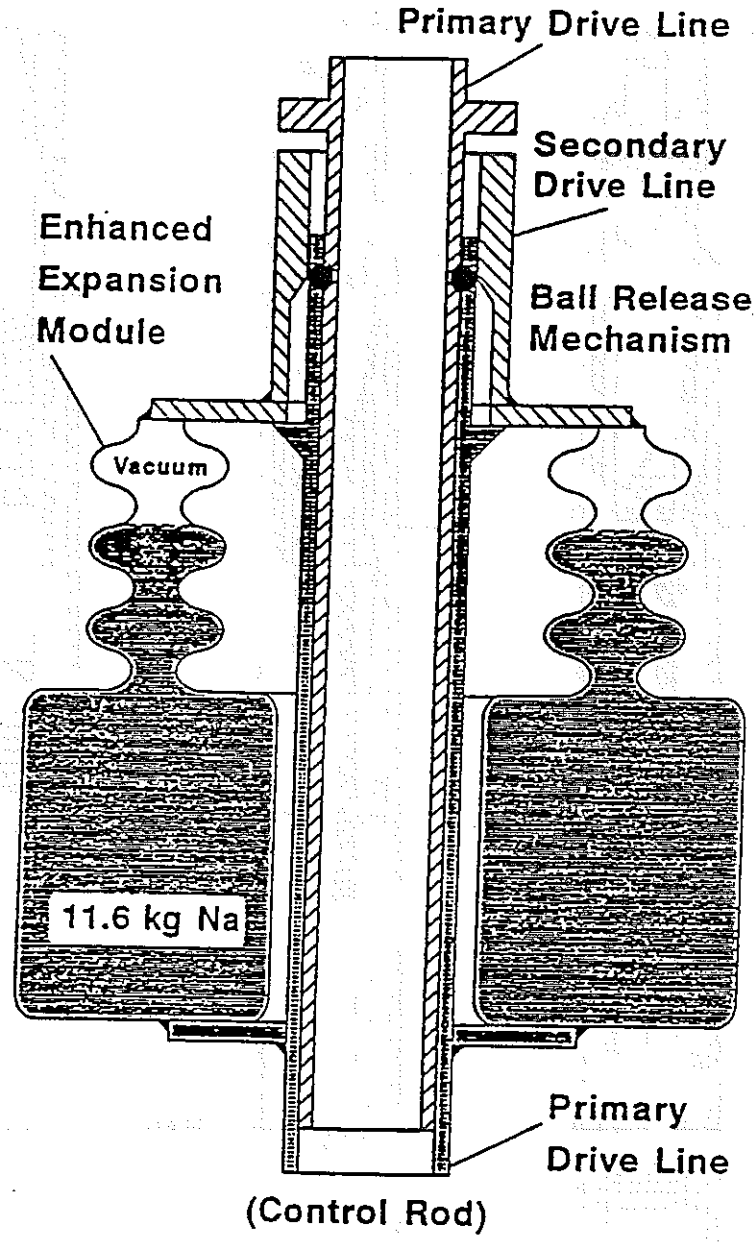
Fig. 2

PSS subassembly with hydraulically suspended rod for the BN-600 reactor.



On-stop rod position. Lower working position of the rod after its movement by the drive (or at its floating-up from the stop). The upper working position of the hydraulically suspended rod.

Fig. 3



Basic Elements of the ATHENa Device

Fig. 4

SUMMARY OF THE FINAL MEETING OF IAEA EXTENDED COORDINATED RESEARCH PROGRAMME (CRP) ON "ACOUSTIC SIGNAL PROCESSING FOR THE DETECTION OF SODIUM BOILING OR SODIUM/WATER REACTION IN LMFR", AT IPPE, OBNINSK, RUSSIA, FROM 11-14 JULY 1995.

1. Introduction

The final meeting of the Coordinated Research Programme, (CRP) on Acoustic Signal Processing for the detection of Sodium Boiling or Sodium/Water Reaction in LMFR was held in Obninsk, Russian Federation from 11 - 14 July 1995.

The meeting was attended by participants from Australia, France, India, Japan, The Netherlands, Russia and UK.

The purpose of the meeting was to review the results of the analysis of the benchmark data based on PFR End of Life Experiments on Acoustic leak detection provided by UK. In addition to reviewing the results of the 1995 Benchmark, the meeting prepared the final report of CRP and made recommendations for future activities.

The participants were welcomed to IPPE by Dr. Poplavskyi, Prof. Kirillov and Prof. Kozlov. They emphasized the importance of acoustic leak detection for protection of fast reactor steam generators.

Mr. Rinejski from the IAEA opened the meeting by reviewing the current status of fast reactor development and the activities of IWGFR.

The meeting was chaired by Mr. Currie from AEA -T, UK. The agenda proposed by the Scientific Secretary was approved. The agenda included the presentations of the results from the participants, followed by the presentation of the details of the benchmark data, which had been prepared by the UK participant. The agenda also included a general discussion on the results obtained and, a visit to IPPE facilities. The meeting closed after agreeing to the recommendations resulting from the discussions, final conclusions and recommendations of the CRP.

2. Objective of the meeting and 1995 benchmark data

The purpose of the meeting was to review the results of the analysis of the benchmark data based on PFR End of Life Experiments on Acoustic leak detection provided by UK. In addition to reviewing the results of the 1995 Benchmark the meeting prepared the final report of CRP and made recommendations for future activities.

The UK participant, Mr. Currie, presented the details of the 1995 benchmark data. In June 1994, a series of injections of argon, hydrogen and steam were performed in PFR Evaporator 3. Two injection locations were studied: one within the tube bundle region and one in the flowing sodium interspace between the tube bundle wrapper and the steam generator shell. Acoustic noise measurements were made at a number of transducer locations,

spatially distributed on the shell of Evaporator. These transducers were mounted on waveguides, welded to the shell. Data from two argon injections, two steam injections and one hydrogen injection in the tube bundle and from one argon, one steam and one hydrogen injection in the sodium interspace, recorded at four transducer locations, were selected for the IAEA 1995 Benchmark exercise (Column 3 in Table 1, Figure 1). The participants were not provided with any information which would enable them to identify which leak was which.

The plant condition during these injections was such that the acoustic background noise was lower than at full power operating conditions. It was agreed to include some separate full power background noise data for the Evaporator and for a Superheater. Accordingly, full power data recorded at two transducer locations in each unit have been included with the injection data.

The task set was to:

(a) compare and characterize the background noise from the evaporator and the superheater,
(b) characterize and identify the type of leak noise signals (steam or argon or hydrogen),
(c) estimate the onset and duration of the leak in the injection data files,
(d) evaluate the transmission characteristics of the leak noise signals from leak location to the transducers, and
(e) locate the leak.

In addition to presenting the details of the injections in the Benchmark test, the main characteristics of the data from these injections were compared with the other injections in the series, to enable the representations of the Benchmark data to be assessed and to present the trends observed during the experiments. The main trends observed were that the leak signal increased with leak rate initially, but quickly reached a plateau value and differences were observed between the characteristics of the noise recorded during argon and steam injections, probably due to the different gas bubble behavior.

During the discussion following this presentation, attention was focussed on the way in which the injections were initiated. The times presented in Table 1 are the times at which the injection valves were opened. These times are subject to an uncertainty of about ± 0.1s. In the case of argon and hydrogen injection, the gas would flow from the opening into the sodium very shortly afterwards. In the case of steam injections, there was an initial phase during which the injection line filled with water, displacing argon through the opening. This phase lasted for about 1.5 - 2.0 s for injection 3 and 8. In the case of injection 6, the data suggest that the injection line was been partially blocked and that water flow may not have been fully established until about 4,0 s after the line was opened. The uncertainties in injection start time are given in Table 1, including those resulting from the method of data preparation for the Benchmark.

3. Presentation of Results of the 1995 Benchmark data analysis

3.1. Australia

The Australian results were presented by Prof. Ledwidge. Data from the 1994 and 1995 benchmark tests was used to compare the performance of seven different signal processing strategies proposed for the detection of boiling or a sodium/water reaction in LMFBR. The general signal processing strategy relies on the signals from the normal background noise and the fault being additive, which gives rise to changes in the signal model in the time, frequency or probability domain. Two of the specific signal processing strategies are derived from an autoregressive model of the process, whilst the rest are implemented in the frequency domain using either global spectral distance measures or more particular spectral measures used in conjunction with wavelet analysis. The emphasis has been to make no assumptions about the nature of the signals from a fault and the background noise. The approach adopted by the Australian team had been to combine the leak signal from the clearest test case with full power background noise and use the combined signal to test the seven signal processing strategies. It was found that a combination of the use of wavelets with spectral distance measures proved to be most effective method considered.

During the discussion following the presentation the following points were raised:

1. The assumption that background noise and leak noise can be added linearly was questioned. But for some of the strategies assessed this assumption was not necessary.
2. Similarly, the assumption that the noise source was white with Gaussian amplitude distribution was questioned. It was felt that it is not necessary to assume a Gaussian distribution in order to apply the techniques.
3. It was suggested that the detection methods may be made more sensitive by selecting a subset of the AR (Auto Regressive) parameters which are more sensitive to abnormal situation.
4. Prof. Ledwidge stated the decision time would not exceed 20 000 samples following of the onset of the leak, and that the techniques were well enough proven for use as an alarm system in a real plant. However, a trip system may need to be based on a combination of several strategies to achieve an acceptable false trip rate.

3.2. France

The French presentation was made by Mr. Oriol. CEA had applied simple signal processing in order to determine the start time of the injections with most attention being directed to the problem of leak location. To determine the start time of the injections the frequency range of 200 - 1000 Hz was analyzed. The ALD (Acoustic Leak Detection) technique developed by EDF for SPX operates in this range.

The start time of the injections determined by this analysis are summarized in Table 1. Two techniques were used for leak location:

a) Pulse timing - in order to derive time delays from the cross correlations it was necessary to apply a 30 kHz high - pass filter. The results from application of this technique are shown in Fig. 2. The accuracy of this method was good.

b) Beam forming this technique assumes a direct transmission path through sodium from the leak to the detectors. In order to avoid aliasing problems the frequency range between 200 - 1000 Hz was chosen. Using this technique the axial location of the leak was incorrectly determined.

In the discussion which followed this presentation, the presence of the tube bundle wrapper and possibly significant transmission through the structure were identified as reasons for the poorer performance of the beam forming technique compared to 1994 benchmark.

3.3. India

The Indian presentation by Mr. Prabhakar described the results obtained from their analysis of PFR End of Life acoustic data.

For detection of the injection signal, features PSD (Power Spectral Density) - SUM and Determinant and Trace of the covariance matrix of the power spectral density were evaluated. Determinant and trace features were found to be sensitive and detected the injection signal in five out of the eight injection files; at all the four waveguides for injections 5 and 6 and at least at two waveguides in injections 1 and 7. Results for onset time are summarised in Table 1.

By analyzing the patterns of the injection spectra and Determinant and Trace outputs and by further considering the physical processes involved, an effort was made to predict the injection medium in five injection files.

The Indian presentation further assessed the capabilities of Determinant and Trace features to detect injection when the evaporator is operating at full power and concluded that the resulting S/N (Signal to Noise) ratios at waveguide - B (nearest to leak) for injections 5, 6 and 7 fall within the detection capability of these two features.

. Cross correlation analysis made by the Indian team did not yield satisfactory results for time delay estimates. The Indian presentation also discussed the similarities in the spectra of evaporator background noise at full power and injections 5, 6 and 7.

In the discussion which followed, it became clear that injections 1 and 8 have proven to be difficult for all of the participants. The Indian presentation was the most comprehensive in attempting to analyze all of the injections and addressing all of the tasks.

3.4. Japan

The Japanese presentation was made by Mr. Hayashi. The Japanese team applied a whitening filter based on univariate autoregressive model to estimate the start and durations of three injection (see Table 1). The three injections were chosen following a preliminary analysis of the full power noise in the Evaporator and the signals from 5 injections. The method based on UAR (Univariate Auto Regressive) background noise whitening filter can be used successfully for leak detection with sufficient signal to noise ratio. A potential technique for leak location has been proposed but not yet tested. During the discussion of the presentation the interesting result that the frequency component of 55 kHz gives a very clear

indication of all three injections analysed. Following the presentation of the UK participant, it was clear that this component was connected to the initiation of the injection.

3.5. Netherlands

The Dutch presentation was made by Prof. Hoogenboom. The analysis focused on the detection of the onset of the leak. Because rapid detection is an important aim, anomaly detection in the time domain was selected to avoid CPU intensive calculations. AR technique was applied to remove the correlated information from the signal. Statistical tests were used as an indication that anomaly had occurred. It was found that the success of this approach was dependent on the choice of algorithm used to determine the AR (Auto Regressive) coefficients. The injections start time derived from this approach are summarized in Table 1. Similar to other participants difficulty was found with injections 1, 2 and 8.

An attempt was made to locate the leak based on the relative increase in the variance of the residual noise at the different waveguides. The presence of spikes in preleak background noise caused false alarms.

In the discussion following this presentation:

a) the selection of the most appropriate model order was an interesting problem. There was no clear minimum in AIC (Akaike Information Criteria) nor FPE (Final Prediction Error). Values in the range of 20 to 60 had been used by various participants. Prof. Hoogenboom raised hypothesis that selection of a model order of about six could result in useful decrease in processing time without a significant reduction in sensitivity.

b) In a SGU at normal operating conditions spikes in the background noise can occur. Detection algorithms need to be able to reject these spikes. One suggestion for techniques based on AR models was to use more than one step prediction. This may suppress spikes, but sofar there is no experience of such techniques.

3.6. Russia

The Russian presentation was made by Mr. Volov. A similar technique (based on neural network), as for previous benchmark data analysis by IPPE was used. Adaptive filtering, at least with the same parameters as for previous benchmark data did not show any significant improvement of signal to noise ratio. Six injections have been identified with different reliability and the results are summarized in Table 1. One of the most powerful injections had been chosen as reference for network learning. Based on some complimentary data from PFR End of Life Experiments it was stated that injection 7 is argon. Injections 1 and 2 were the most difficult cases for the neural network which was trained on injection 6. Preliminary estimates show that being thaught on injection 6, it would be possible to recognize a leak twice the size of injection 6 in the acoustic sense at full power.

In the discussion following this presentation several issues were raised:

a) The network based on injection 6 correctly recognized the PFR full power data. By estimating what the signal to noise ratio would be if injection 6 had occurred at full

power, it was derived that network would correctly detect a leak, twice the size of injection at full power.

b) On the basis of IPPE analysis of other PFR End of Life data it was concluded that leak signals can be detected and the main problem will be the discrimination between leak noise and abnormal background noise.

c) A concern was raised about the need to train the network continuously in order to recognize all possible abnormal background noise. However, it was pointed out that the network trained on the low pump speed background noise prior to injection 6, successfully recognized full power background noise. This indicates that there is some robustness in the system towards new background noises.

Conclusions

1. The use of leak and background noise data recorded by the same instrumentation during the End of Life Experiments in PFR had removed the uncertainty in the previous benchmarks caused by the mixing of signal obtained from different rigs.
2. The acoustic signals from the injections were propagated to all sensors. The frequency components recorded at each waveguide were influenced by the medium of the injection fluid and the transmission path from the location of the injection.
3. Mechanical waveguides terminating in accelerometers are very suitable for monitoring acoustic noise in steam generators.
4. With a few exceptions there is a satisfactory correspondence in the times of onset of the injections found by different participants and different methods.
5. Some of the participants had correctly observed interesting features of some of the injections. They had observed that injection 1 and 8 were perhaps not steady and some had detected the noise due to the initial flow of argon during a water injection, followed by the noise of the water injection.
6. Determination of the fluid injected proved to be nearly impossible. There were no distinctive characteristics which allowed discrimination. The additional data presented by the UK participant were instructive in showing the similarities and differences between the results from the different injection fluids, derived from a larger number of injections.
7. Leak location proved to the difficult. Nevertheless, some success was achieved. The Tube Bundle Wrapper makes location of injection inside the Tube Bundle more difficult. Now that the participants know the location of the injection, they can repeat their analysis, concentrating on the leaks in the Sodium Interspace.
8. It appeared that there was a delay between the onset of the water injections and the creation of a detectable signal. This may explain why the injection 8 proved to be difficult to detect. The length of water injection period may have been too short.
9. The participants concluded that the experience by the analysis of PFR End of Life data has enriched the knowledge substantially. The participants further noted that the CRP has made significant contributions in evaluation different signal processing techniques and the results from the CRP could therefore provide invaluable data to the designers of Acoustic Leak Detection systems. In the long term, this would lead to enhancing the surveillance of steams generation and the economics of LMFRs.

Recommendations

The meeting discussed the results arising from the 1995 Benchmark and recognised that additional benefit could be derived from the analysis of the PFR data received by the participants. The following recommendations were proposed:

1. Participants who wish to update their contribution should submit any supplementary papers to the IAEA by 1 November 1995.
2. All participants should state which injection they would have detected if the injections had been made while the evaporator had been operating at full power and estimate:

   (I) the detection margins
   (II) the probability of a spurious trip
   (III) the time required to make a decision

3. All the participants agreed that although much progress was made, further work, based on the PFR data already in their possession would be valuable. A further meeting to review such additional analyses would be beneficial, but it was noted that this may be difficult to organise under the auspices of the IAEA. In spite of this, the participants requested the Scientific Secretary to convene a consultancy on this subject in the last quarter of 1996, if the results of their new analysis warrant it.
4. Participants should analyse the injections they had not considered for this RCM. They should perform some additional analyses on the tasks which had proved difficult, such as identification of the fluid injected and the location of the injection. They should also address the topics listed in Recommendation 2 above. Participants should address suggestions made at the RCM relating to the investigation of the effect of some variations in the various techniques. Particular attention should be paid to the question of the suitability of the methods for on-line detection in an operating steam generator with varying background noise.
5. The participants should send a summary of their papers to Mr. R. Currie in July 1996, in order for the value of a consultancy to be determined.

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MAIN RESULTS OF THE CRP AND RECOMMENDATIONS

Participants of the Co-ordinated Research Programme (CRP) on "Acoustic Signal Processing for the Detection of Boiling or Sodium - Water Reaction in LMFBR", included specialists from the following countries: Australia, France, Germany, India, Japan, Russia, the Netherlands and Untied Kingdom.

Research Coordination Meetings were held at IAEA, Vienna (Austria) in 1990, at Chester (UK) in 1991, at IAEA, Vienna in 1992 and 1993, at Kalpakkam (India) in 1994 and at Obninsk (Russia) in 1995.

Boiling noise data analysed during 1990 were obtained from Germany. The leak noise data were obtained from Russia, UK and Germany and the background noise data were obtained from UK (PFR) and France (SPX) and these data were synthesized to create test files with varying signal to noise ratio (S/N). These files were analysed during 1991 - 1994. In 1995, data were provided from PFR End of Life acoustic leak detection experiments.

The main results of the analysis can be summarized as follows:

1. A number of signal processing techniques were evolved/developed

   - Main square prediction error, distance in parameter space, and likelihood ratio tests based on autoregressive modelling approaches and wavelet functions of Australia.
   - Bandpass filtering and time domain beamforming technique of France.
   - Determinant and Trace of Covariance matrix constructed from power spectral density (PSD) ratio and adaptive learning network methodologies of India.
   - Twice squared processing of signals, background signal whitening filter based on univariate autoregressive modelling of Japan.
   - Adaptive filtering and neutral network methodology of Russia.
   - Autoregressive modelling approach by the Netherlands.
   - Pattern recognition, pulse counting and narrow band filtering of UK.

2. Acoustic boiling noise detection

   - Most participants who analysed the data were able to detect reliably boiling incidents when signal to noise ratio was as low as - 12 dB. This lies just within the margin of detection expected to be required for a reactor trip system based on boiling noise detection.
   - The detected boiling events were always detected within one second of the event occurring.
   - All the participants had difficulty in detection when the S/N ratio was less than - 12 dB. However, visual inspection of the output from the analysis techniques showed perceptible presence of boiling signals in the data.
   - All successful methods involved high frequency filtering of the signals.

3. Acoustic leak detection of synthesised data

- leak signals were distinctly detected with good reliability of S/N down to 16 dB. This lies very close to the detection margin expected to be required for a fast reactor steam generator protection system.
- Analysis methods also detected leak signals down to - 24 dB S/N but with less reliability. Discrimination and reliability get enhanced even at such low S/N ratios by filtering techniques if leak signals contain specific frequency bands.
- The response time of detection is always less than 1 s.
- Cluster analysis and n_0/N_0 technique are two methods available for improving the reliability of the detection system.
- Leak location was identified through time domain beamforming technique using multichannel analysis.

4. Analysis of PFR Data

- With a few exceptions, a satisfactory correspondence has been found in the times of onset of the injections determined by different participants using different methods. Detection capability in real field conditions has been established.
- Mechanical waveguides terminating in accelerometers are found very suitable for leak detection in steam generators.
- Leak location was difficult due to the complex internal geometry. Nevertheless some success was achieved by the French analysis.
- Acoustic signals form injection were transmitted to all the waveguides. The frequency content at each waveguide was influenced by the injection fluid and transmission path.

5. General

- The work carried out within the framework of this CRP has a general scientific value, since the results obtained can be employed in other types of reactors. Signal processing techniques developed in this CRP for anomaly (boiling or leak) detection may be applied for detecting other types of anomalies in thermomechanical equipment (like cavitation, rattling etc.) in fast and thermal reactors. Hence there is a merit in considering the setting up of IAEA research programmes in acoustic anomaly detection applicable to diverse types of reactors.

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Table 1

Injection	Fluid	Location	Leak rate (g/s)	Start (s)	Australia * (s)	France (s)	India (s)	Japan (s)	Netherlands (s)	Russia (s)
1	H_2	Bundle	0.05	1.95 - 2.01	N/A	8.49	1.56 - 1.79	1.9	-	-
2	Ar	Bundle	0.27	3.14 - 3.30	N/A	3.1	-	N/A	-	-
3	H_2O	Bundle	3.4	1.80 - 3.84	N/A	7.37	1.62	N/A	7.37	7.58
4	Ar	Interspace	0.5	4.46 - 4.65	N/A	5.13	0.95 - 2.25	N/A	4.83	5.43
5	H_2	Interspace	1.3	2.72 - 2.80	2.6 ++	2.65	2.39 - 2.67	N/A	2.61	2.77
6	H_2O	Bundle	7.65	2.40 - 6.40-	N/A	5.41	4.38 - 6.71	2.4 - 5.5	5.3	5.53
7	Ar	Bundle	2.4	3.22 - 3.31	N/A	5.58	3.41 - 3.9	N/A	3.57	4.4
8	H_2O	Interspace	2.2	3.65 - 5.65	N/A	-	-	3.65	-	4.51

Notes: The spread of detection times are based on results using different method and using data from different waveguides;
N/A - Not Analysed;
* Only used injection 5 to test 7 different signal processing strategies;
++ Typical value.
Onset times determined visually except by the Russian team who used an on-line technique.

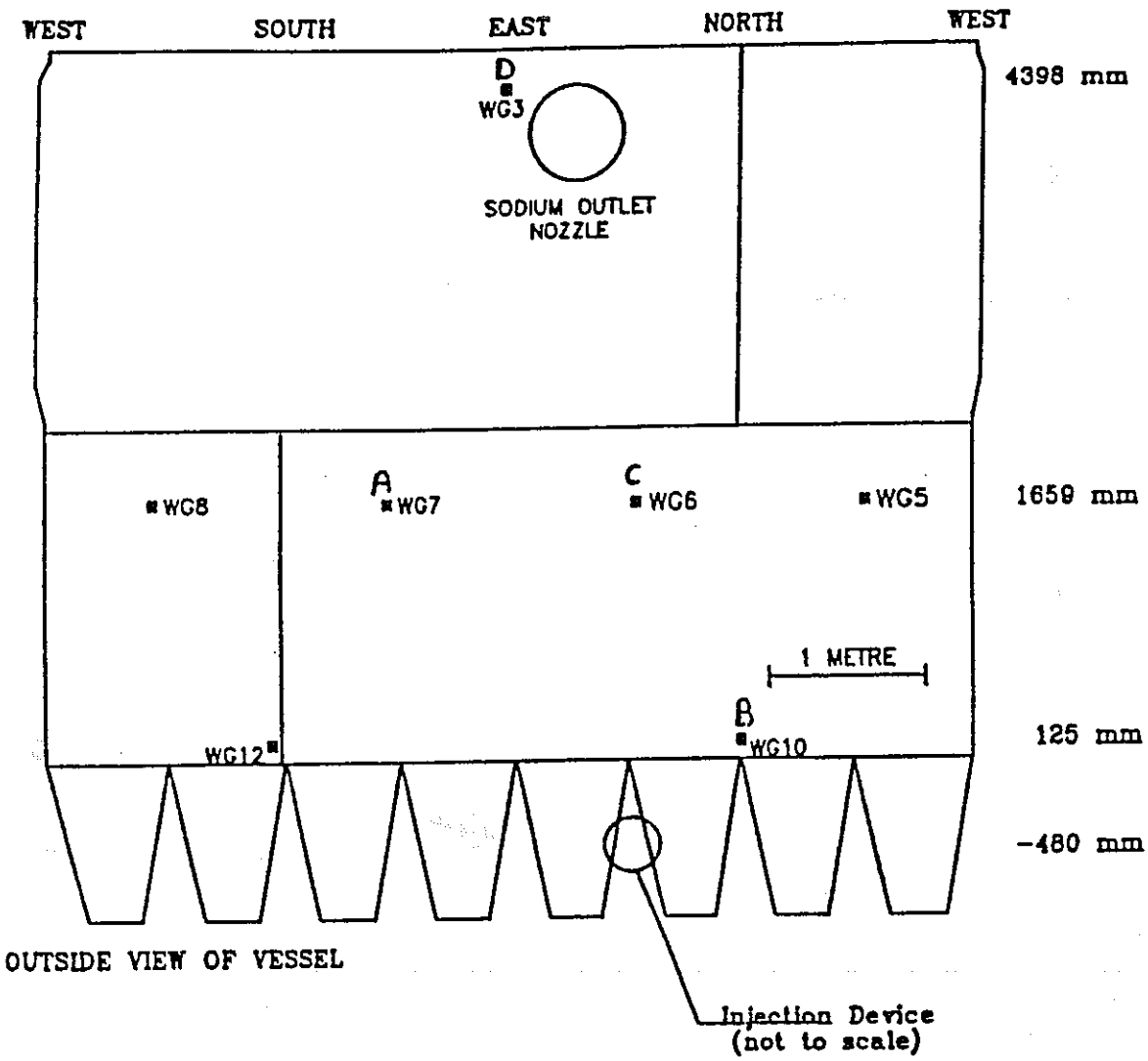

Figure 1 Schematic of Evaporator 3 showing the positions of the waveguides and the injection device

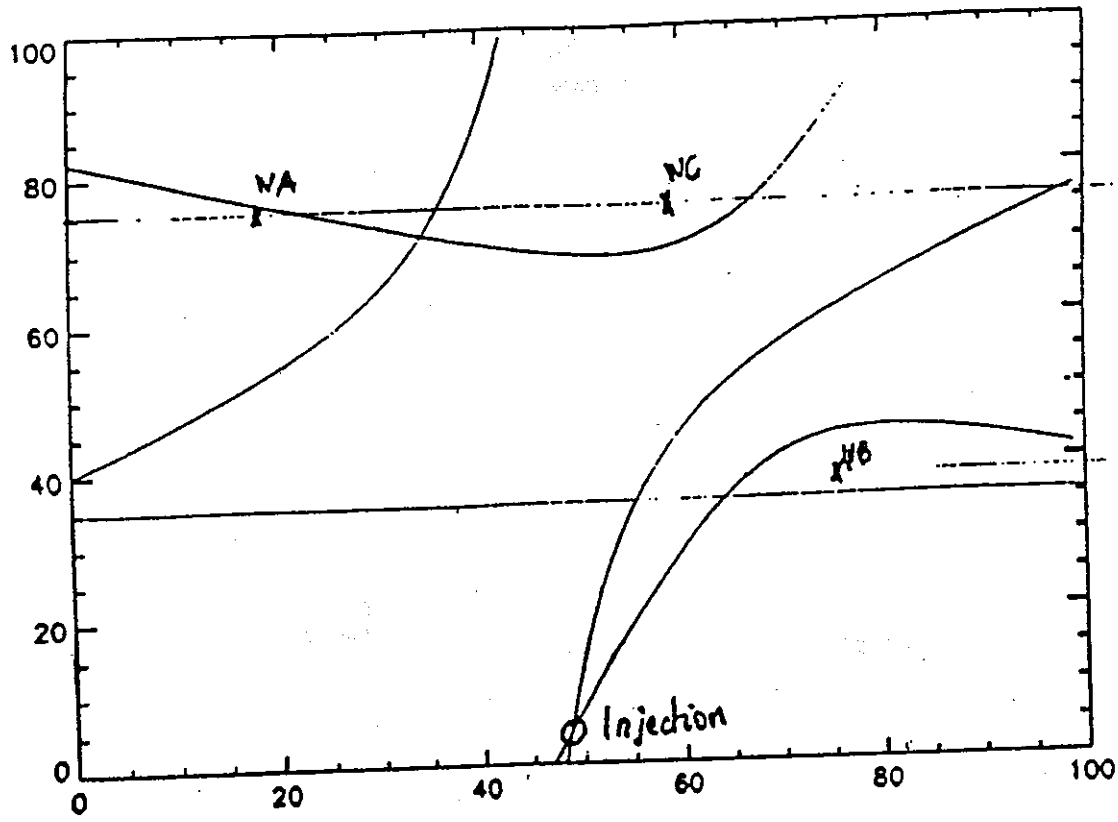

Figure 2 : Pulse timing location of the injection

Summary Report of the TCM on "Conceptual Designs of Advanced Fast Power Reactors"

1. INTRODUCTION

The IAEA Technical Committee Meeting (TCM) on "Conceptual Designs of Advanced Fast Power Reactors", was hosted by the Indira Gandhi Centre for Atomic Research, India, from 3-6 October 1995.

The countries that participated in the TCM were Brazil, China, India, Japan the Republic of Korea, and the Russian Federation. About thirty specialists participated in the TCM and were welcomed by Dr. P. Rodriguez, Director, Indira Gandhi Centre for Atomic Research (IGCAR). The meeting was opened by the Scientific Secretary, A. Rinejski of the International Atomic Energy Agency, and was chaired by Mr. S. B. Bhoje, Director, Reactor Group IGCAR, and Chairman of the Local Organizing Committee.

Dr. R. Chidambaram, Chairman of the Atomic Energy Commission and Secretary of the Department of Atomic Energy of India was unable to attend the opening personally, however, he sent his greetings via fax (Attachment I).

2. TECHNICAL BACKGROUND

The liquid metal reactor technology has demonstrated a number of effective designs examples, project realization and experience in construction and operation. Recent milestones of liquid metal fast reactor (LMFR) development are the restart of the 250 MWe Phenix and 1200 MWe Superphenix in France, the 15 years of commercial operation of the 600 MWe BN-600 in Russia, and the start-up tests of the 280 MWe reactor Monju in Japan. Of course, some incidents and/or failures have been experienced in the operation of the first prototype and demonstration reactors, but it should be noted that the mechanical failures in LMFRs were more a consequence of faulty detailed design decisions and/or fabrication errors than faults in the basic concept. The operating experience indicated that some LMFRs' design aspects should be improved.

The construction of BN-600, SPX1 and Monju have indicated that LMFRs are costly and take more time for construction than thermal reactors. Therefore, efforts are being made to reduce the capital costs and construction time.

The safety record of LMFR is good and many encouraging results have been obtained on controllability, shutdown systems, decay heat removal and other major systems and components. Certain improvements in safety are being considered, such as prevention, detection and mitigation of leaks and fires of sodium, passive shutdown and decay heat removal systems, core catcher and improved containment.

The development of simple, reliable, efficient and flexible systems and components is a primary objective in the design of advanced fast power reactors. Systems and components for developmental designs (advanced reactor designs which range from moderate modification of existing designs to entirely new design concepts) in general require much extensive testing and demonstration to verify component and system performance. Key issues are scaling effects for simulation plant configuration, design life and interactions among different systems.

Fast reactor programmes have been initiated in the People's Republic of China, the Republic of Korea and Brazil. International exchange of information and experience is becoming of increasing importance to these countries.

In view of these circumstances in May 1994 an IAEA International Working Group on Fast reactors recommended to convene a Technical Committee Meeting on Conceptual Designs of Advanced Fast Power Reactors.

The meeting participants reviewed and discussed the following general topics:

- Lessons learned from experience with demonstration and commercial size plants, applicable to design of advanced liquid metal fast reactors
- Developments in advanced LMFRs and their general design features
- Status of R&D programmes, findings and conclusions
- Anticipated performance of advanced LMFRs
- Future needs and trends in the subject area

A total of 18 papers were presented, and a technical tour of the Indira Gandhi Centre for Atomic Research (IGCAR) facilities, including experimental liquid metal fast breeder test reactor FTBR took place during the TCM.

3. SUMMARY OF TECHNICAL SESSIONS

3.1. Session 1: Chairman - Mr. R. Shankar Singh, India

Experience of LMFRs designing, construction and operation (N. G. Kuzavkov, Russia)

The developments in designs of BN-350[2] and BN-600 were discussed in detail, with emphasis on critical technical parameters that have been considered in the choices of individual concepts for these reactors. The BN-350 LMFR is the first nuclear reactor in the world producing heat for seawater desalination (80.10^3 tones per day) and electricity production (130 MWe), was commissioned in 1973. The BN-350 demonstrated reliable performance of reactor equipment and the plant achieved an average load factor of 85% at established power levels. The main factor limiting the reactor power level has been the failures in steam generators (SGs). The problem of deformation in core subassemblies, due to neutron irradiation experienced in the initial years, has been overcomed by changing to new structural materials. The 12 control rod drive mechanisms (CRDMs) (2 control rods, 3 shutdown rods and 7 reactivity compensation rods) have functioned normally. During the whole period of the reactor's operation, 56 planned refuelling cycles have been completed. Sodium aerosols deposition in shield plugs increased resistance to rotation. The intermediate heat exchanger (IHX) has operated for more than 160,000 h without any problems. The SG consist of 2 superheaters with U-shaped tubes and 2 evaporators with field tubes, inside which water flows under natural convection. The initial years operation of SG were unreliable on account of frequent tube-weld failures. The radiological release since commissioning has remained within acceptable levels. Detailed life extension studies for the

[2] Due to changes in the former USSR, it belongs to the Republic of Kazakhstan.

reactor are in progress and include ensuring reactor safety and structural integrity during seismic events.

15 years operating experience of BN-600 (V. V. Vylomov, Russia)

The paper discusses the operating experience of the BN-600 power plant. The BN-600 (1470 MWt/600 MWe, pool type) which was connected to grid on April 8, 1980 has successfully completed 15 years of operation. The commercial operation began in 1982. The initial U02 fuelled core consisted of two U-235 enrichment zones, i.e., 21% and 33%. In order to improve the fuel performance, the core was modified in 1987. The modifications included provision of three uranium enrichment zones with 17-21-26% enrichment, increasing the peak burnup from 7.3% to 8.3% and decreasing the linear heat rating from 51 to 47 KW/m. The decrease in linear heat rating increased the fuel column height from 75 to 100 cm. In 1993, with the use of advanced structural materials, the burnup was increased to the design value of 10% heavy atom (ha). Efforts are under way to further increase the burnup to 11% ha. The life of control rod guide tubes was increased (from 200 to 640 efpd) due to use of advanced structural material and back fitting of test results on materials. The operation of primary sodium pump (PSP) were successful except during the initial periods in which the reactor trips were caused by failures of gear couplings and shafts. The origin of these failures was traced to the shaft torsional vibrations. The experience with IHXs has been trouble free. The reactor has 3 secondary loops each consisting of a SG, a secondary sodium pump and a turbogenerator (TG). The SG is modular in construction with an evaporator, reheater (sodium reheat) and a superheater. The isolation valves in SG enable are isolated of faulty module and continue the reactor operation. There have been 12 instances of water sodium leaks during 15 years of the BN-600 reactor operation and about 30 small sodium leaks. The secondary sodium pumps have operated trouble free for about 105,000 h. As regards to the operation of TG, of late, the functioning of drain lines and valves are considered critical for the entire power plants' performance. The power plant performance has matched the performance of NPPs worldwide. The average plant availability is 76.6% with planned outages constituting 21.7% and unplanned outages constituting 1.7%. The 15-year operation of BN-600 has demonstrated reliable and safe operation of pool type sodium-cooled fast reactors and successful commercial exploitation of LMFR. The experience gained with operation LMFR in the Russian Federation has proved their high level of reliability and has revealed no fundamental problems associated with reactor physics and operation of various equipment. This experience will be useful in upgrading the design of next generation LMFR. The large leaks of secondary sodium happened on May 6, 1994 in a drain pipe of an intermediate heat exchanger (IHX) during repair work when the reactor was shutdown. The sodium leak was accompanied its burning and some equipment damage in the adjacent area. Approximately 1.2-1.3 m^3 of sodium were lost. The complex of means used in BN-600 to detect sodium leaks, to localize them and to fight with sodium fires allowed to avoid serious fire incidents. The mass of the burned sodium was of the order of several tens of kilograms.

3.2. Session 2: Chairman - Mr V. M. Poplavski, the Russian Federation

CEFR-25 Conceptual design and its recent progress, (Ma Dayuan, China)

The paper discusses the need for fast reactor development in China and gives the technical profile of the CEFR-25. CEFR-25 has a capacity of 65 MWt/25 MWe and may

use U02 fuel for first core and subsequent core will have MOX fuel. It is a sodium cooled pool type experimental fast reactor. There are 2 primary sodium pumps and 4 IHX in the main vessel, 2 secondary loops, single steam/water circuit and one TG set. The decay heat removal (DHR) is provided by two independent (0.5 MWt each) passive residual heat removal systems. The main vessel is top supported and is provided with a guard vessel. Two rotatable plugs with one straight pull machine are used for fuel handling. The primary sodium purification system is located outside the vessel. The paper also lists other major parameters that have been considered in the design of the reactor. The present planned burnup is 50,000 MWd/t with a future targeted burnup of 100,000 MWd/t. The concern has been to limit radioactivity releases during operational states (normal operation and anticipated operational occurrences) and accident states to within acceptable limits as the site selected is close to Beijing city (40 km). This project is the first step towards development LMFR in China and developments in LMFR technology is viewed from the angle of meeting the energy requirements in the next century.

An overview of REARA reference design, (Arthur Menezes, Brazil)

The paper discusses the need for development of fast reactor in Brazil with primary motive of gaining design and operating experience in LMFR systems. The reference design for 60 MWt/20 MWe REARA (REActor RApidos) is as follows. It is a pool type, U-10%, Zr metallic alloy fuelled reactor with active fuel height of 62cm and having 30 cm Ni reflectors at top and bottom. There are 6 B_4C control assemblies (4 primary + 2 secondary) for controlling/shutting down of the reactor. The neutronic calculations have been done using EXPANDA and CITATION codes and the corresponding burnup figures are given. The main parameters for primary sodium pump and IHX have been selected based on EBR-II and PRISM designs and process design of IHX has been done with computer code TCIPRO. The purpose of establishing the reference design is to put forth the case of LMFR to the "decision makers" and convince them of the potentials of LMFR technology.

The development and demonstration fast breeder reactor (DFBR) (Takashi Nakamura, Japan)

The DFBR is being developed as a next step of MONJU towards commercialization of LMFR around 2030. It is a 1600 MWt/660 MWe, PuO2-U02 fuelled, sodium cooled, top entry loop type reactor. The primary cooling system consists of a reactor vessel, three IHX vessels, three pump vessels and inverted U-shaped piping to connect these vessels. The argon cover gas system in these vessels is interconnected and maintained at 0.9 Kg/sq.cm. The diameter and height of the reactor vessel are 10.4 m and 16.0 m respectively and the thickness is 50 mm. The roof structure supports the vessel and is of box type construction (13.0 m dia, 6.2 m height and 60 mm thick plates). Both 316 FR and SUS 304 are used in construction of sodium cooling systems. The primary sodium flows on the tube side of IHX and the primary pump is of single stage single suction type. The design of SG has been based on "MONJU" design and is of integrated once-through type with helical coil arrangement for tubes. 9 Cr-1 Mo is used in construction of SG. The reactor has two types of shutdown systems to arrive at a level of safety equal to that of the LWR. The reactor containment building (RCB) is designed to withstand a temperature of 150°C and a pressure of 0.5 kg/sq.cm under condition of sodium ejection into RCB after a core disruptive accident (CDA) during which 100 Kg of Na is assumed to burn. The emergency power supply is provided by 3 independent diesel generator sets and the reactor protection uses a "2 out of

4 logic". The design of RCB incorporates suitable isolation in order to take care of seismic events. Rectangular and cylindrical reactor buildings were studied and the former is selected because of compact layout. In all the studies of the conceptual design phase of the DFBR, the aim has been to make the design harmonious with both safety and economic viability and to provide preparation for the basic design.

3.3. Session 3: Chairman - Mr. T. Nakamura, Japan

Prototype fast breeder reactor (PFBR) main options, (S.B. Bhoje, India)

The importance of fast breeders as the only alternative to coal in meeting the energy demands of India was highlighted. The significance of capital cost and construction period on economy was elaborated. Sodium coolant, pool type concept, oxide fuel, 20% cold work D9 for fuel clad and hexcan, SS 316LN for main hot leg components and sodium piping and modified 9Cr-1Mo steel (T91) for steam generator tubes have been selected for PFBR. Steam temperature of 763 K at 16.6 MPa and a single TG of 500 MWe gross output have been chosen.

Conceptual design of PFBR core, (S. M. Lee, India)

The mixed oxide fuel care of PFBR is conventional, homogenous core with two enrichment zones and radial and axial blankets of depleted U02. Pin diameter has been chosen on lower side at 6.6 mm for reducing the core plutonium inventory below 2 t. Considerations for the choice of number of pins per subassembly, integrated vs separate axial blankets, and other pin and subassembly parameters were discussed. No special schemes have been explored for reducing the maximum positive sodium voiding coefficient as the core size is moderate and the coefficient is about 3.5 $. Twelve control rods have been chosen in a layout optimised for maximum antishadow effect. The optimisation of layout of radial and axial shielding was also discussed.

Conceptual design of reactor assembly of PFBR, (A. Selvaraj, India)

The conceptual design of PFBR selected in 1995 was revised with the aim of simplification of design, compactness of assembly and ease in construction. The reduction in size has been possible by incorporating concentric core arrangement, adoption of elastomer seal for rotating plug, fuel handling with one transfer arm mechanism, incorporation of mechanical sealing arrangement for IHX at the penetration in inner vessel redan and reduction in number of components. Warm roof slab concept has been chosen to avoid sodium vapour deposition in the gaps between rotatable plugs. A single grid plate of fully bolted construction has been adopted to simplify manufacture. The main vessel cooling circuit consists of cooling pipes drawing coolant from leakage from core subassemblies and passing the coolant through thermal baffles on the main vessel. A core catcher is provided below the grid plate as defence in-depth philosophy. The sequence of erection of reactor assembly components was also explained.

Conceptual design of heat transport systems and components of PFBR-NSSS (S.C. Chetal, India)

Basis of selection of two primary pumps, four IHX, two secondary loops, two

secondary pumps and eight steam generators for PFBR was discussed. The details of design of above components were also presented. A single stage impeller design was chosen for primary pump. The non-return valve chosen for earlier design was eliminated to permit increased submergence under free sodium level. For IHX, straight tube design with primary sodium on shell side has been selected. Steam reheat cycle has been chosen for PFBR to effect cost reduction for SG and associated circuits, reduce construction time and provide ease in design and operation. Straight tube once through steam generator with expansion bend on each tube welded to tubesheet with raised spigot has been selected.

3.4. Session 4: Chairman - Mr. D. Hahn, Republic of Korea

Conceptual design of core component handling system in PFBR, (Sodhi, India)

In vessel handling in PFBR comprises of two rotatable plugs and a fixed transfer arm for handling subassemblies for in vessel transfer between core, internal, storage and in vessel transfer position (IVTP). From IVTP, subassemblies are transferred to ex-vessel transfer position (EVTP) by inclined fuel transfer mechanism. For spent fuel cooling outside the reactor, water pool storage has been chosen. The economic considerations in the choice of the design were explained.

Conceptual design of safety instrumentation for PFBR (G. Muralikrishan, India)

After explaining the design criteria for safety instrumentation in PFBR, the parameters to be measured were listed. The flux measurement for monitoring core power is proposed to be done in a location below the main vessel. However, for first startup, in-core fission chambers are provided. Provision has been made for monitoring outlet temperatures of all fuel subassemblies. Global monitors for delayed neutron detection in sodium and fission gas activity in cover gas have been provided for detection of failed fuel. For localisation of failed fuel sodium from individual subassemblies are monitored. Primary sodium flow is measured by permanent magnet flowmeters provided in a bypass line. Sodium leak will be detected by sodium aerosol detector and spark plug detectors. Water leak in steam generator is monitored by mass spectrometer based hydrogen leak detection systems. Protection logics for scram of the two shutdown systems are based on different principles for diversity. To achieve required reliability it is proposed to couple the two shutdown systems optically.

Safety considerations in the design of PFBR, (G.Vaidyanathan, India)

The general safety approach in PFBR design is governed by principles of redundancy, diversity, independence, fail safe design, sound design and operating experience of various LMFBRS. The temperature and power coefficients of reactivity are negative. Subassembly blockage is prevented by multiple opening entries and an adaptor at the exit. Two independent fast acting diverse shutdown systems have been chosen. Curie point magnetic switch triggering scram on secondary shutdown systems is proposed to be provided to protect against abnormal coolant temperature rise. Four decay heat removal circuits each comprising a dip heat exchanger in the hot pool of the reactor have been provided. The circuit is passive ensuring removal of decay heat from the reactor by natural convection. In addition, battery power to drive the primary pumps at 20% of the speed for one hour is also provided. Measures to control sodium fire were also presented. Sodium void reactivity is about 3.5$. Studies made for loss of flow without scram (LOFWS) for flow coast down times of 2s and

10 s showed that reduction in sodium void coefficient has the effect of delaying fuel melting by 10 to 40 minutes. Guillotine rupture of pump to diagrid pipe is considered very remote as leak before break argument seems to be applicable to this component. However, the guillotine rupture is proposed to be monitored by primary coolant flow measurement, fast response thermocouples and eddy current flow meters at selected subassembly locations and reactivity. A core catcher below the grid plate for post accident cooling of molten fuel debris from 7 subassemblies has been provided to protect against total instantaneous blockage of a single subassembly. The main vessel has been designed for an energy release of 200 MJ from core disruptive accident (CDA). The containment building is designed for overpressure of 25 KPa arising from burning of 500 kg of sodium ejected from reactor vessel during CDA.

3.5. Session 5: Chairman - Mr. D. Ma, China

Design of advanced reactor DN-600M, (N.G. Kuzavkov, Russia)

The design concept being adopted for BN-600M was presented. While the long-term objective of BN-600M is to provide nuclear power and fuel, the short-term objective is to burn the weapon grade plutonium, and minor actinides. The main concepts adopted for BN-600M are from BN-600 and BN-800. However, there are modifications towards enhancing the safety and reducing the cost. The shape and support of the main vessel are similar to that of BN-600. The main vessel is cooled by sodium leakage from diagrid. Additional safety features such as negative sodium void reactivity, hydraulically suspended absorber rods, provision of core catcher etc. are included. In order to avoid excessive heat generation in invessel storage subassembly boron carbide shielding is provided and storage subassemblies are cooled by natural convection. Four decay heat removal heat exchangers each of 15 MW capacity are provided while two are sufficient to limit the sodium temperatures below the design basis value. Towards reducing the cost it has only 2 primary pumps and IHXS. The plenum for feeding the primary pumps is common. However, the number of secondary loops has not yet been decided., which may be either 2 or 4 depending upon the type of SG selected. The specific weight of BN-600M will be reduced by 1.25 (2 secondary loops with modular SG) or 1.58 (vessel type SG) times that of BN-600. Fuel burn-up will be increased by 12.7% ha (I phase) and by 15.8% ha (II phase). Future works will be devoted to the SG, secondary pipe work, development of bellows, testing DHRs in sodium and water etc.

Intelligent type sodium instrumentation for LMFR, (D. Chen, China)

Details of construction and the performance of newly developed intelligent type sodium instrumentation which includes pipeline, immersed flowmeters, pressure measurement device and level gauge were discussed. The graduation characteristic equations for the corresponding transducer, using the temperature of medium as a parameter were presented. The need for the online compensation of the temperature effect of graduation characteristic was emphasised. These instrumentations which were calibrated in sodium experiments, possess good linearity. These instruments possess (1) self inspection, (2) electric shut-off protection, (3) setting of full scale, (4) setting of alarm limits etc. and satisfy the needs of LMFR applications.

Safety features and core performance of KALIMER
(D. Hahn, Republic of Korea)

The main options selected for the KALIMER (Korea Advanced LIquid MEtal Reactor) were discussed. It is a modular pool type sodium cooled LMFR and its construction is expected to be over by 2011. The electric power per module is 333 MWe. It is designed to use metal fuel, initially U-Zr and later Pu-U-Zr. It uses electromagnetic pumps for the primary as well as secondary circuits. The number of primary pumps in the primary circuit is 4 while that in the secondary circuit is 2. It adopts the integrated concept which combines SG, intermediate sodium pumps and sodium expansion tank in a single compact component. The other unique feature includes use of integrated SG elimination of rotating plugs and simplification of invessel transfer machine. Passive safety features of KALIMER design include, the reactor vessel auxiliary cooling system (RVACS) which assures safety grade decay heat removal and the self- actuated shut-down system (based on Curie paint magnet) and gas expansion module. It is designed to provide a strong inherent negative reactivity feedback with rising temperature. This along with RVACS, makes it capable of safely withstanding severe under-cooling and overpower transient events without scram.

3.6. Session 6: Chairman - Mr. D. Ma, China

Elevation configuration of KALIMER intermediate system
(Y. Sim, Republic of Korea)

The importance of nuclear power in Korea and the plans initiated by KAERI in developing the first FBR in Korea, namely KALIMER were explained. The relationship between elevation and natural circulation was analysed and a brief description of RVACS and SGACS were presented along with their expected ranges of heat removal capability. Based on which skeleton of decay heat removal strategy for KALIMER has been set up. Mention was also made of development of a fast running computer code LSYS in Korea in assessing the above mentioned decay heat removal capability.

Study and choice of main characteristics of fast reactor effective minor actinide burner, (V.M. Poplavsky, Russia)

The present strategy in Russia to solve power and ecological problem is to use FBRs to burn plutonium and minor actinides. It is feasible to develop a core with MOX fuel for effective plutonium utilisation on the basis of using of fuel pin structure with increased enrichment of plutonium and also using absorber shield, which also provided higher safety sodium void reactivity effect (Sodium void reactivity effect < 0). Advanced core of BN-800 type reactor for plutonium utilisation also allows spent fuel removed from VVER type reactors to be recycled and used leading to reduction in radiotoxicity. Considered scenarios in Russia in plutonium utilisation also ensure non-proliferation.

Advanced design of fast breeder reactor in PNC, (R. Mukaibo, Japan)

The background of nuclear power development through LMFBRs in Japan and the evolution of an advanced concept of reactor design for LMFBRs were presented. Diverse needs including reduction of impact to the global environment and enforcing greater thrust on non-proliferation aspects were introduced. Nitrides are considered as promising candidate

as fuel. Details of the core, the new fuel concept and related aspects of interface with plant design were described. It was mentioned that the design of an advanced recycle fast test reactor has just started which will facilitate to evaluate all the features considered in the advanced concept.

CONCLUSIONS

1. The meeting participants reviewed and discussed, in detail, the following general topics:

- operation experience of demonstration and commercial size plants;
- developments in advanced LMFRs, its general design features;
- status of R&D programme, findings and conclusions;
- predicted performance of advanced LMFRs;
- future needs and trends in the subject area

2. The liquid metal reactor technology has demonstrated a number of positive examples of effective designs, project realization and experience in construction and commercial operation. Research on LMFR during the last decades has significantly improved specialists understanding of LMFR designs, technology and safety. But there is a number of safety concerns which influence fast reactor licensing and safety analysis. In this particular context there still is a sodium leaks and fires. According to the experience gained with long-operating LMFR, sodium leakage took place in sodium containing systems and components. The BN-600 reactor had several small leaks and one of ~ $1m^3$ leaks in its secondary circuits during 15 years of operation. The potential risks associated with large sodium leaks and fires include: loss of function in the system in which the leak occurs, sodium-concrete reactions, sodium fire and associated phenomena. Prevention, detection and mitigation of sodium leaks, improved resistance of nuclear systems to fires and development of reactor equipment and systems designs for elimination of leaks is still an important direction of LMFR technology research.

3. Two principal design concepts have been discussed at the TCM: the loop and the pool type. The pool concept was chosen for all small, medium and large size advanced fast reactor: REARA (Brazil), CEFR (China), PFBR-500 (India), KALIMER (Korea, Republic of), BN-600 M (Russia) and EFR-1500 (European Fast Reactor) except DFBR-660 (Japan), which use the top-entry loop-type reactor.

The pool reactor placed in a guard vessel has been shown to have very attractive safety characteristics, resulting to a large extent from a liquid metal cooled reactor being a low pressure radioactive system with large thermal inertia. This type of design practically excluded unfavourable consequences of failures in the external radioactive systems and loss of reactor coolant.

The top-entry loop-type design was selected for the Japanese DFBR because of following considerations:

a) Major primary components such as the intermediate heat exchangers (IHX) and the pumps are outside of the reactor vessel, and this facilitates maintenance and repair.

b) The system has flexibility to introduce such innovative technologies as the electro-magnetic pump integrated component, which is needed for commercialization of the FBR.

c) Experience gained at the prototype "Monju" must be fully utilized. Considering that the top entry system is quite a new concept, the conceptual design study, the evaluation study of commercialization prospects and the water hydraulic tests using models of thermal-hydraulic properties peculiar to the top-entry system were conducted.

3. Although differences in conceptual design approaches were in evidence through the TCM, a number of common topics could be identified among the conceptual design approaches presented. These included improvements with regard to safety, design simplifications and reduction in cost. Safety improvements included among other the consideration of core catcher design (excluding of the recriticality, cooling capabilities), passive backup reactivity shutdown and decay heat removal systems, a strong inherent negative reactivity feedback with rising temperature. Economic competitiveness with alternatives should be improved due to the optimization of the number cooling loops and equipment and achievements of reducing its weights as well as the building volumes, findings of structural materials, fuel technology and core design to achieve high burnup, and limiting the number of safety graded systems.

** ** ** ** ** **

परमाणु ऊर्जा विभाग
DEPARTMENT OF ATOMIC ENERGY

ATTACHMENT I

FAX

FROM R. CHIDAMBARAM CHAIRMAN, ATOMIC ENERGY COMMISSION & SECRETARY, DEPARTMENT OF ATOMIC ENERGY, BOMBAY

FOR DR.A. RINEISKY SCIENTIFIC SECRETARY IAEA-IWGFR
CAMP: KALPAKKAM

I WOULD LIKE TO CONGRATULATE IAEA-IWGFR ON ARRANGING THE TECHNICAL COMMITTEE MEETING ON "CONCEPTUAL DESIGNS OF ADVANCED FAST POWER REACTORS (KALPAKKAM, OCTOBER 3 - 6 1995) WHICH IS OF GREAT RELEVANCE AT THIS STAGE OF FAST BREEDER REACTOR DEVELOPMENT STOP CONSIDERABLE WORK IS BEING DONE IN VARIOUS COUNTRIES FOR IMPROVING THE CONCEPTUAL DESIGNS OF FAST BREEDER REACTORS AND MAKE THEM ECONOMICALLY COMPETITIVE STOP I SEND MY BEST WISHES TO THE PARTICIPANTS AND WISH ALL SUCCESS TO THE TCM AND ALSO TO THE FBR PROGRAMMES IN ALL THE COUNTRIES STOP

KIND REGARDS

ANUSHAKTI BHAVAN, CSM MARG
Bombay 400 039
October 2 1995

R. Chidambaram
2.10.95

(R. Chidambaram)
Chairman, AEC/Secretary, DAE

PCC:

Dr. A. Rineisky
Scientific Secretary
IAEA-IWGFR Camp: Kalpakkam.

copy to : Shri S.B. Bhoje, Director, Reactor Group, IGCAR & Chairman, Local Organising Committee

SUMMARY REPORT OF THE SECOND CONSULTANCY ON "IAEA/CEC COMPARATIVE CALCULATIONS FOR ULOF (UNPROTECTED LOSS OF FLOW) IN THE BN-800 REACTOR", 26 - 28 JULY 1995, CEC HEADQUARTERS, BRUSSELS, BELGIUM

1. Organization of the Consultancy

At the Annual Meeting of IWGFR in May 1994, Russia proposed to begin an evaluation of the BN-800 reactor transient characteristics. After discussions with contributions of all member countries, this proposal was unanimously supported. Mr. Balz of CEC indicated that Commission of the European Communities (CEC) would support this benchmark. A joint IAEA/CEC benchmark exercise for a severe accident in a BN-800 with near zero void core has been endorsed by all participants. It was the second Consultancy. The first was held from 5 to 6 December 1994 at the IAEA Headquarters in Vienna.

The purpose of this Consultancy Meeting was to discuss the progress of the IAEA/CEC sponsored comparative calculations and to clarify and agree on aspects of the proposed exercise which were left open during the last Consultancy Meeting held in Vienna on 5-6 December 1994 [1]. The specific objectives of this consultancy were to:

- discuss the status and scope of contributions from participating organisations,
- review the case setup data provided by Russia and define additional requirements,
- review and discuss first results provided by the participating organisations, and
- define a detailed work programme up to the next consultancy meeting.

The Scientific Secretary of the IWGFR opened the consultancy and proposed Mr. Struwe as Chairman. The participants endorsed this proposal and approved the agenda which has been jointly prepared by Mr. Rinejski and Mr. van Goethem. The following persons attended the meeting:

I. Chvetsov, IPPE Obninsk, Russia
J.F. Frizonnet, CEA, France
G. Glinatsis, ENEA, Italy
G. van Goethem, CEC Brussels, Belgium
G. Keinhorst, CEC Brussels, Belgium
D.G. Martin, AEA Technology, United Kingdom
H. Niwa, PNC, Japan
A. Rinejski, IAEA Vienna, Austria
Om Pal Singh, IGCAR, India
D. Struwe, FZK, Germany

2. Scope and Status of the BN-800 Exercise

All participants re-affirmed their agreement with and commitment to the previously agreed scope of the BN-800 ULOF exercise [1,2]. However, the following modifications to the consultancy report of the last meeting were agreed upon: Page 4, second to last paragraph reads as follows: Dr. Niwa also reported that Hitachi have expressed an interest in participating in the transient calculations. Page 5, third paragraph reads as follows: To avoid

the artificial coherency experienced in the previous BN-800 calculational exercise [3], it was agreed that each of the channel zones would be modelled by 3 channels representing the different burnup levels associated with the 3-batch core loading scheme.

The actual status of the work to the exercise was reported by the participants as follows:

Russia (IPPE): A revised draft of the input data for the comparative calculations was prepared and distributed by IAEA to the participants at end of June 1995. In response to remarks provided by the French participant, an amendment to the distributed information was prepared [4,5]. In addition, first preliminary calculations were performed for the steady state fuels pin characterization of the peak power channel.

France (CEA): Due to the late availability of the input data, it was not yet possible to perform first preliminary calculations with GERMINAL. However, the available data base was thoroughly reviewed and an evaluation of the data consistency was executed. Problems encountered were documented and communicated to IPPE for comments. The respective answers need to be discussed at the meeting.

UK (AEA-Technology): First preliminary calculations with TRAFIC were performed for the fuels pin characterisation, making reasonable assumptions on missing data [6]. These assumptions need to be discussed and agreed upon between participants of the exercise.

Italy (ENEA): In this exercise, ENEA will concentrate their effort on two aspects: Complementary evaluation of the neutron physics input data with their own tools, and transient calculations of the thermohydraulic behaviour of the core and the primary circuits in single phase conditions with the NATURA code system. First preliminary results with regard to the neutron physics input data specifications have been performed to get a first impression on the problems involved [7].

Germany (FZK): FZK intends to apply the SAS4A-code system to this exercise. Recently developed new models have been introduced into the code system, which is developed jointly by FZK, IPSN and PNC. As consequence of these new models, some modifications of the already available models became necessary which could not be validated up-to-now in a satisfactory manner on experimental information. Therefore, it was not yet possible to perform calculations for the BN-800 comparative exercise. Current activities concentrated on a thorough review of the input data provided and on consistency checks. On this basis, it seems necessary to modify and reiterate part of the input data after discussion and approval at the meeting.

India (IGCAR): First preliminary calculations of the thermohydraulic core state that the specified EOEC-condition have been performed [8]. Results indicated some peculiar features of the steady state conditions which are interpreted as hints for some inconsistencies in the provided input data. These need to be resolved prior to entering into transient calculations. Aside that situation, IGCAR underlined their interest in contributing to all stages of the agreed upon comparative calculations with their own tools:

Japan (PNC): PNC will contribute most actively to the exercise from stage 3 onward. Reviewing the case set-up specification, it appears as if special problems could occur when entering into the transition phase analysis. These need careful evaluation because they could overcompensate advantages of the considered core design in its behaviour during the initiation phase of a ULOF-transient. In addition to these questions, a representative of Hitachi will review the neutron physics input data. However, due to the late arrival of the input data specifications, no activities could be initiated prior to this consultancy meeting.

3. Status and Review of the Case Setup

Mr. Chvetsov presented the updated input data specification for the BN-800 comparative calculations [4]. It became clear to all participants that the proposed core design differs from the BN-800 project design version considerably. The core power of the IPPE proposal is nearly 40% less than the BN-800 project design value. This difference is mainly due to three design features chosen differently in the project and by IPPE: The first one is the number of fissile subassemblies which is lower by 18% than in project design. The other is a slightly smaller core height, which is about 5% smaller in the IPPE proposal. The third change in the design characteristics of the IPPE proposal is a peak linear rating which is about 10% smaller in the IPPE case set-up than in the project design reference. However, Mr. Chvetsov pointed out that these differences did not alter the generic safety features of the reactor and plant design. He illustrated that by discussing the relative differences between the maximum positive sodium void worth value, the sodium void worth value of the core region and the total overall negative void worth value. These data were kept similar in the IPPE core design variant, when compared to the project design values. He admitted that the core design proposed for the comparative exercise was established especially for this purpose, as already mentioned in the proposal made to the IWGFR.

Participants of the meeting expressed their concern about this procedure, chosen by IPPE. They would have preferred to concentrate their analysis efforts on a core design, being much closer to the real design of BN-800. They made it clear that results of the envisaged exercise cannot be used directly for the BN-800 project design variant. Evaluation of consequences of a ULOF-accident in the real plant design would need a second round of calculations to be performed with actualized input data. Representatives of the IAEA and the CEC shared the concerns expressed by the participants to the meeting.

Mr. Chvetsov made it clear that it would have been very difficult to provide input data for the real BN-800 reactor core design for the comparative exercise, due to IPPE internal reasons. It was due to these difficulties, that IPPE undertook the tedious effort of preparing a new input data set for a slightly modified core design. However, IPPE felt that the chosen design variant is sufficiently close to the real design data so that results of the comparative exercise are of high value for the IPPE. This value lies mainly in the validation of the calculational procedures used by IPPE in comparison to results prepared by the alternative tools of the participants to the exercise. The other participants to the meeting declared that this reduced objective of the exercise does not fully represent their level of interest in the exercise. They would have preferred a case set-up being closer to the real case so that results of the study can be directly used for evaluation of the success of the proposed design measures to overcome consequences of a ULOF-initiator without entering directly into core destruction. However, they agreed on the conclusion that there remains still a

considerable interest in evaluating the transient behaviour of the currently provided core design.

As it appears that the currently prepared input data set represents a paper reactor, it was concluded that this will have some non negligible impact on the level of detail in which this design proposal will be investigated. At the time being it is too late to request for a modification of the input data set. The then-expected delay in the performance of the study was considered too serious in comparison to the gain in quality of the results. Keeping the scheduled planning was felt to be more important. Consequently, an agreement was reached between all participants to modify the titles of all reports of ongoing activities. The new title is as follows:

> EC/IAEA Benchmark/Comparative calculations for a severe accident in the IPPE core design variant with a near-zero void coefficient of the BN-800 type reactor (unprotected loss of flow, ULOF)

Dr. Chvetsov offered to the participants of the meeting that IPPE prepares a report in which differences between the IPPE core design variant and the project core design basis are clearly identified and discussed. Participants of the meeting appreciated the offer and recommended that a representative of the project should join this effort to guarantee that the latest developments on the project side and their view on the evaluation of differences are included in this report.

In addition to the above discussed differences in design features, further changes have become necessary in the IPPE proposals. These concern adjustments of the hydraulic characteristics of the modified core design to the primary circuits design characteristics. Throttling characteristics of individual subassemblies needed to be changed. The singular pressure drop in the subassembly inlet had to be considerably increased to match the inlet plenum pressure as provided by the primary pumps. As a consequence of this artificial adjustments, the pressure drop in the subassembly inlet increases to values of 3 to 4.3 bars. This feature of the IPPE core design variant leads to a situation where singular pressure drop in the inlet is as large as the frictional pressure drops along fuel pin height. Such a redistribution of the total pressure drop would have considerable consequences on the two-phase flow development. Agreement was reached that IPPE has to provide modified input data for the pressure drop redistribution along subsassembly height on the basis of realistic characterization of the pressure drop distribution in the real design. It is expected that this results in a more realistic pressure drop distribution in the subassembly inlet to the frictional pressure drop along fuel pin height.

In addition to these more generic discussions on the characteristic of the case set-up, all input data were discussed in detail. To acquire insight into the currently established situation, it was of the utmost importance that results of preliminary calculations have been already available, namely those from our colleagues from the UK, Italy and India [6,7,8]. Results of these discussions are reported hereafter in short, since they were achieved only after a lengthy and rather complex evaluation of the situation. Comments refer to the items of the input data documentation distributed in June 1995 by IAEA [4,5].

Item 1. Arrangement of the core, blanket and spent subassembly storage

1.8 a) The outer pellet radius of radial blanket subassemblies still needs to be reasonably defined.

1.8 b) The outer pellet radius of axial blanket zones of fissile subassemblies is to be taken as 0.0028 m.

1.8 c) The geometrical details of the chamfered edges of fissile pellets have to be provided.

1.15 It was agreed by all partners that the thermal heat transfer coefficient between fuel and clad will be calculated by the partners using their individual modelling capabilities.

In addition to these data, the following data still need to be specified:

- fuels as fabricated grain size
- more precise documentation of the upper fuels pin tail ends geometrical design features (specification of the pellet stack hold down device and its material composition). This information is necessary for an improved determination of the actual length of the upper fission gas plenum length and more importantly for evaluation of the smeared out materials composition of zone 17
- complete information of design details of the radial blanket subassemblies. The most important items here are: residence time of radial blanket subassemblies as used to calculate the respective materials composition in the neutronics calculation, information on chamfering of blanket pellets, wire spacers dimensions of blanket pins, fuels specifications of blanket pellets, upper plenum length of blanket pins and dpa values of the clad material for blanket pins in accordance with the above mentioned residence time.

Item 2. Thermohydraulic Characteristics

Participants of the meeting took note of modifications to the original input data specification as documented in the telefax to Messrs. Struwe, Frizonnet and Rinejski from Mr. Kuznetsov dated 21st of July 1995 [5]. However, the specification of inlet throttling was not accepted. A request of a reasonable modification was formulated unanimously, which corrects the currently provided data in accordance with the realistically established design solutions of the project-related subasssembly design. Participants of the meeting were aware of the fact that this request might lead also to modified recommendations on frictional pressure losses along fuel pin height and/or modifications of the inlet plenum pressure. The respective questions need an urgent solution, hopefully in accord with specialists from the BN-800 design team.

Agreement was reached that the channel dependent SA-power, as specified in Table 2.4, will be taken as reference for determining the normalised power distribution in radial direction. (Channels 1 to 12). This normalised distribution defines the one used also for determining channel dependent contributions of the power production in the lower axial blanket region to the different coolant channel powers.

Review of the provided information led the participants to conclude that information on compensating as well as on passive safety rods is missing. The most important part

concerns the pellet stack length of the B4C pellets of the compensating rods and the passive safety rods and their position relative to the given total length. Furthermore, information on the atomic densities of the safety and compensating rods is missing as well as the materials' composition of the zones with the passive safety rods in their rod-in position. This information is necessary to perform the complementary neutron physics calculations in a reasonably accurate manner.

Item 3. Power and Irradiation Data

Input data provided for this item were discussed in great detail, to develop a strategy for the construction, from the provided information, a reasonable input of a three batch loading scheme, being representative of an end-of-equilibrium-cycle core configuration. (EOEC-condition). At the end, the following strategy was unanimously agreed upon by all participating partners: Distribution functions of the core fuel rating as provided in Table 3.2 are taken as reference for normalisation of axial distributions of linear ratings, assumed to be constant for all considered burn-up stages, i.e., for the 140 d, the 280 d and the 420 d irradiation time periods. Linear ratings of fuel pins of the LEZ, MEZ and HEZ reactor zones, as specified in Table 3.4, are taken as representative for the highest rated subassembly in each of the three enrichment zones. The specified values at the beginning and end of a run are taken as the ones after 140 d and 280 d preirradiation time period respectively. It was agreed to assume that the variation of the linear rating is linear in time and can be extrapolated to the beginning of the irradiation as well as to the end of the irradiation. In addition, it was agreed to assume that the linearisation holds for all subassemblies of one enrichment zone in the same manner as specified for the highest rated subassembly in one enrichment zone. Applying this strategy one can construct the power distribution for an EOEC-configuration from the available information for the involved 30 representative coolant channels. Due to the relatively high power production in the radial blanket subassembly, it becomes necessary to simulate its behaviour as well. Participants of the meeting were aware of the fact that this procedure will lead to a thermal power output of the plant which varies over time and deviates slightly from the as specified integral data.

The irradiation dose accumulated versus burn-up will be chosen in such a way that the values, as given in Table 3.7, are met in the peak power position after 280 d irradiation in the highest rated subassembly of one enrichment zone. The relation between clad damage and fuel rating is kept constant during the irradiation time period.

All the above mentioned agreements serve to meet the original agreement of simulating an EOEC-configuration during transients. So far, participants do not accept the altered specification in the input data provided by IPPE under item 3.4.2 to restrict the analysis to the one of core configuration, with an average operation time period of 280 days irradiation time period only.

Item 4. Materials

It was unanimously agreed, that each participant in the exercise uses his own data base of materials property libraries and modelling approaches to simulate the materials behaviour during irradiation and during transients. However, the respective libraries should be chosen in such a way that they represent as closely as possible the properties specified in

chapter 4. of the input data specification. This modifies part of the given specifications as far as the use of correlations for determination of the fuel conductivity is concerned, as well as it modifies the calculational routes to determine fuel and clad swelling. Agreement was reached to take 1400°C as solidus temperature of the clad material and 1410°C as the respective liquids temperature. The chemical composition of the clad material is close to the Ti-stabilised 316 CW stainless steel applied in another fast reactor project. A need was identified of getting access to high temperature clad creep data for this material. However, participants were aware of the problems involved and are prepared to tackle that item later, in more depth. The fuel melting temperatures will be taken dependent on the enrichment.

Item 5. Neutron Physics Data

Review of the neutron physics input data resulted in a rather controversial discussion of the calculational route chosen by IPPE for preparation of the respective information. Doubts were raised by the Italian representative whether it is really acceptable to transform the complex core arrangement of BN-800 into a simple r-z-geometry with concentric rings representing control and safety rod positions as given in Fig. 8 of the Input Data Specifications [4]. In addition, concerns were raised that the fissile core region is simulated only by one material composition in axial direction, for each enrichment fuel radial zone. The choices of materials compositions of zones representing control and safety rod positions were identified to be questionable as well, especially with regard to the axial resolution of the actually simulated rod-in or rod-out position. From the information provided, it is felt that details of the impact of different burn-up levels of the individual subassemblies are not considered in the data base.

Part of the above mentioned questions will be clarified when results of complementary calculations become available. However, to do so, it is necessary that IPPE provides additional information. This concerns the following items:

- pellet stack length of B4C for compensating and safety rods and their positioning within the total length,
- materials composition of the reactor zones with passive safety rods in the rod-in position,
- atomic densities of safety and compensating rods.

In addition, it was felt necessary to provide a better resolution of the axial power profile in the lower axial blanket.

Item 6. Kinetics parameters

On basis of the above mentioned items of discussion, a need was identified to review major parts of the kinetics parameters provided by IPPE in their Input Data Specifications.

The following requests were agreed upon:

- provide improved axial dependencies of the Doppler coefficients (Table 5.6, the current structure of the three zone-wise constant coefficient in axial direction is felt too coarse)

- provide fuel and steel reactivity worth tables with a reasonably fine resolution in axial direction (Tables 5.3 and 5.4)
- reevaluate the reactivity feedback due to axial fuel pin expansion as given in Table 5.8 (the provided information seems erroneous from their values as well as their unit).

A more basic discussion developed during a review of the recommendation of IPPE for determining reactivity feedback contributions due to radial core expansion. Due to the IPPE recommendation, this reactivity feedback is coupled to the temperature increase at the load pad level of the individual radial subassembly rows and to the temperature increase at the diagrid of the individual radial subassembly rows respectively. When multiplied with the linear expansion coefficient of the ferritic steel liner and the diagrid linear expansion coefficient respectively, the temperature increase is converted into a radial expansion which then results in a considerable negative reactivity feedback. The time delay constants of the heat-up of the load pad level relative to the coolant temperature heat-up are estimated to amount to 1.5 s and the time delay constant of the diagrid to 70 s.

The recommended procedure to determine reactivity feedbacks from the load pad expansion is based on the assumption that there is sufficient free volume available at the outer periphery of the subassembly arrangement. However, Fig. 4 of the Input Specification indicates the core arrangement close to the core barrel with the restraint ring. From the figure it cannot be deduced that there is free space available for an unrestrained thermal radial expansion at the load pad level. So far, participants got the impression that the design solution of the BN-800 reactor represents a passively restrained core clamping system which in principal is very close to the design solution of SNR-300. This implies that radial core expansion at the load pad level is determined by the heat-up of the core restraint ring and not by the heat-up of individual subassemblies at the load pad level. The delay time constant of this heat-up versus coolant heat-up in individual subassemblies would be by an order of magnitude or larger than the quoted 1.5 s, mentioned in the Input Data Specification. In addition, participants of the meeting raised their concerns about acceptance of the recommended calculational route for determination of the reactivity feedback due to the load pad expansion on the background of their knowledge. It was noted even in free standing core design variants, as established in PHENIX and SUPEPPHENIX, the efficiency of the quoted reactivity feedback effect could not be qualified during operation. Theoretical analyses has shown that friction on load pad levels prevents the assumed free radial expansion. In PHENIX and SUPERPHENIX reactivity feedback effects, due to an early activation of the radial expansion at the load pad level are therefore, not considered.

From the information made available about the BN-800 core clamping system nothing can be identified which supports the view that the situation there is so much different than in other core designs. Participants therefore requested that IPPE provides arguments and/or experimental qualification for the quoted efficiency of the load pad radial expansion feedback. If this controversial issue cannot be resolved it is to be expected that participants of the exercise do not simulate the effect as recommended in the Input Data Specifications. This issue should be discussed in more detail at the next Consultancy Meeting.

Due to the short time period between distribution of the Input Data and the Consultancy Meeting, it was not possible for the participants to review all recommended

input data specification for calculation of the control rod drive line feedback and for the primary loop simulation. This data will be discussed in more depth at one of the forthcoming Consultancy Meetings. However, the Italian representative from ENEA presented the main features of the NATURA-code system, which is intended to be used for performing complementary analyses of the primary loops transient behaviour during the considered accident scenarios.

In summary, it can be stated that participants appreciated the big effort of IPPE to prepare the Input Data for the comparative exercise. On basis of first preliminary calculations and on basis of a thorough review of the information made available, some inconsistencies, as well as areas of missing information were identified. Discussion of these items led to requests for additional information, as well as revision of the already provided information. The respective items are listed in Appendix I. To enter quickly into the next steps of the exercise, it was felt necessary that information with regard to items 1. up to 11. should be provided up to 15th of September 1995 and with regard to items 12. up to 16. up to 30th of October 1995. If these target dates are not met, progress of the comparative exercise will be significantly delayed.

4. Alternative Initiators

The IPPE-proposal of the comparative exercise to the IWGFR included the request to analyse consequences of initiators of core disruption, other than ULOF-transients, also within the scope of this exercise. At the first Consultancy Meeting, participants agreed to make proposals for other initiators at this meeting on basis of the procedures chosen in their national projects. Reports by the participants can be summarized as follows:

Japan: Within MONJU licensing the following initiators were considered in the approach to evaluate consequences of BDBA: ULOF, UTOP (3 c/s) local faults, Loss-of-piping integrity and ULOHS. This approach agreed upon around 1980 is now open for modifications in view of the advanced knowledge about the reliability of active and passive preventive safety features and the involved time scales from the initiating event up to partial core destruction. For the DFBR-plant design interest concentrates dominantly on the ULOF- and UTOP (3c/s)-initiator. More recent safety evaluations for the JOYO-reactor however included steeper reactivity ramp rates in the analysis (1 to 5 $/s) to evaluate safety margins of the plant design. On this background it seems reasonable to complement the reference ULOF-calculations of the BN-800 reactor design by calculations evaluating safety margins to demonstrate that it is possible to avoid so called "cliff-edge-effects." For this purpose it seems reasonable to consider different coolant flow halving times (0 up to 4 s) and UTOP-initiators with reactivity ramp rates of different steepness (a few c/s up to a few $/s).

Russia: The Russian delegate did not see the necessity of an immediate decision on this question. It should be postponed up to a time when first results of the agreed upon comparison exercise become available. In the past, IPPE covered in their analyses the following initiators of core disruption: ULOF, UTOPs starting from different initial states, subassembly blockages, an uncontrolled

closure of reverse valves in a damaged loop with the available operational safety system.

India: For the FBTR, the following initiators have been considered for core disruptive accident analyses: ULOF, UTOP and LOCA-scenarios. For the PFBR, these kind of analyses have been complemented by analyses of ULOHS-scenarios and consequence analyses of subassembly blockages.

France: For core disruptive accident analyses, the following initiators are considered: ULOF, SA-inlet blockage, ULOHS and UTOPs. For UTOPs the unlimited control rod withdrawal accident is of special importance, but alternatively consequences of a very rapid TOP with a limited reactivity insertion were analysed, simulating consequences of a rapid gas bubble passage through the core. For this initiator, it is assumed that gas accumulates somewhere within the grid plate region, and this gas is then nearly instantaneously released and swept through the core region. However, in case of SPX-1 this accident does not lead to fuel pin failures, due to the limited integrated energy release.

After discussion of the actual situation in the different member states, participants of the meeting agreed with the Russian position that there is no urgent need for an immediate decision on the subject. This item should be discussed again when first results become available from the agreed-upon reference case calculations.

5. Future Planning

Participants agreed to concentrate their efforts in the near future on providing calculations of the steady state core characterization at the EOEC-core state, as agreed-upon during the meeting. In addition, results of the complementary calculations on the neutron physics data are to be provided before the next consultancy meeting. To facilitate comparison of results, agreement has been reached on an output data format for the BN-800 steady state calculations. The respective specifications are given in Appendix III of this consultancy report. A format specification for comparison of the neutron physics data will be provided by G. Glinatsis from ENEA Italy.

To improve information flow between participants of the exercise and IPPE, it was decided that any requests to IPPE and any information from IPPE to the partners should be sent to the secretary of the IWGFR, i.e. Mr. Rinejski, who then informs all other participants of the exercise. The actions of the partners and of IPPE agreed at this meeting are listed in Appendix I and Appendix II.

6. Date and place of the next consultancy meeting

The next consultancy meeting is foreseen to be held at Vienna from 11-13 December, 1995. This meeting will take place only if actions 1 to 11 of Appendix I are fulfilled, by 15th of September 1995 and actions 12 to 16 by 30th of October 1995. If considerable delays would appear, the meeting will be postponed dependent on the necessary time to provide calculations on the steady state calculations and the neutron physics data. In addition to these items, it is intended to compare safety related design parameters of the IPPE core

design variant and the BN-800 project design and to evaluate arguments and experimental qualification of the efficiency of the load pad radial expansion feedback. Participants of the meeting are invited to contribute to the latter item with own contributions on the background of results available from their investigations of the problem related to realized projects.

REFERENCES

[1] Report of the consultancy on "IAEA/CEC Comparative Calculations for ULOF in the BN-800 Reactor" Vienna International Center (5h to 6th december 1994)

[2] IAEA, Scope of an International Benchmark Exercise Supported by IAEA/CEC on Calculations for ULOF in BN-800 (August 1994)

[3] D. Struwe et. al.: "Synthesis of Transient Calculations for a BN-800 Type Benchmark Exercise of an Unprotected Loss-of-Flow Accident" Int. Top. Meeting on Sodium Cooled Fast Reactor Safety, Obninsk, Russia, (October 3-7, 1994)

[4] IPPE, INPUT DATA, Obninsk (June 1995)

[5] Telefax by IPPE, Correction of INPUT DATA (July 1995)

[6] D.G. Martin; "Input data assumed in TRAFIC2.6j calculations and first preliminary results" (July 1995)

[7] G. Glinatsis; "BN-800 Neutronic Preliminary Evaluation" (July 1995)

[8] Om Pal Singh, R. Harish; "IAEA/CEC Comparative Calculation for Severe Accident (Unprotected Loss-of-Flow, ULOF) in BN-800 Reactor" (July 1995)

APPENDIX I: LIST OF ACTIONS TO IPPE

1. Chamfering dimensions of fissile pellets

2. As fabricated fuels grain size

3. Upper fuel pin tail end
   - upper fission gas plenum length
   - smeared out materials composition of zone 17

4. Outer pellet radius of the axial breeder zone

5. Radial blanket subassemblies
   - residence time of blanket SA
   - outer pellet radius of blanket pellets
   - chamfering of blanket pellets
   - wire spacers for blanket fuel
   - fuels specification for blanket material
   - upper plenum length
   - dpa in the blanket fuel element

6. Corrective input of inlet throttling to match the project design characteristics --> this in turn might lead to a modified inlet plenum pressure or/and to a modification of the frictional pressure losses

7. High temperature clad creep data

8. Pellet stack length of B4C compensating rods/B4C control rods and its position within total length

9. Material composition of the zones with passive safety rods in the rod-in position

10. Atomic density of safety and compensating rods

11. Axial power distribution in the lower axial blanket

12. Improved axial shape functions of the Doppler coefficients

13. Fuel and steel reactivity worth tables still to be provided but in a reasonable shape

14. Reactivity feedback due to axial fuel pin expansion needs to be reevaluated.

15. Arguments and experimental qualification of the efficiency of the load pad radial expansion feedback has to be provided. Otherwise this feedback effect will not be simulated as recommended in the Input Data specification

16. Comparison of safety related design parameters of the IPPE core design variant and the BN-800 project design is to be provided. A representative of the designers project team should participate in this exercise.

APPENDIX II: LIST OF ACTIONS TO PARTICIPANTS

1. Provide results of calculations of the steady state core characterization at the EOEC-core state (see output data specification as given in Appendix III).

2. Provide results of complementary calculations of neutron physics data.

3. Provide information on investigations about the efficiency of load pad radial expansion reactivity feedback.

Appendix III: Agreed Output Data Format for the BN800 Steady State Calculations

Output is required from channels 1, 6 and 10 at the end of each of the three irradiation cycles.

Table 1: Information relating to the whole pin at the end of each cycle (at power), i.e. 9 numbers for each item:
Fission gas generated (m^3 at $STP.kg^{-1}$)
Fraction of fission gas released
Gas pressure in the pin (MPa)

Table 2: Information relating to the peak power node at the end of cycles 1, 2 and 3, i.e. 9 numbers for each item.
Centre hole radius (mm)
fuel-clad heat transfer coefficient ($kW.m^{-2}.K^{-1}$)
inner surface fuel temperature (K)
outer surface fuel temperature (K)
clad mid wall (or average) temperature (K)
bulk coolant temperature (K)

List of Figures

These figures are intended to show the axial variation of a number of variables. Each figure should contain 9 lines, relating to the three channels, at the end of each of the three cycles, at power. However, if the 9 lines prove difficult to distinguish from each other, it is recommended that they be separated into 3 graphs. For example, fig.1 would then become figs.1(a), 1(b) and 1(c). Fig.1(a) would show the fuel inner radius of channel 1 (LEZ) at the end of the 1st, 2nd and 3rd cycles; likewise figs.1(b) and 1(c) would show the channel 6(MEZ) and channel 10(HEZ) results respectively.

Fig.1: Fuel inner radius (mm)
Fig.2: Fuel outer radius (mm)
Fig.3: Fractional gas release
Fig.4: Radial fuel-clad gap (μm)
Fig.5: Clad diametral expansion (%)
Fig.6: Fuel-clad heat transfer coefficient ($kW.m^{-2}.K^{-1}$)
Fig.7: Fuel inner surface temperature (K)
Fig.8: Fuel outer surface temperature (K)

This is the information required for the comparative exercise. However, any participant wishing to draw attention to any other interesting features of their output data is at liberty to include this additional information in their contribution.

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SUMMARY REPORT OF THE CONSULTANCY ON "INTERCOMPARISON OF ANALYSIS METHODS FOR SEISMICALLY ISOLATED NUCLEAR STRUCUTRES

1. INTRODUCTION AND SCOPE

At the IAEA Specialists' Meeting on "Seismic Isolated Technology", held in San Jose (California, USA) in 1992, participants proposed to begin a study of the behaviour of isolators and isolated structures, to evaluate the benefits of seismic isolation for nuclear plants and to compare the analysis methods used in the various participating countries.

The International Working Group on Fast Reactors endorsed the proposal, then the Coordinated Research Programme on "Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures" was approved by the Agency's Committee for Contractual Scientific Services. Thus, a Consultancy was convened with the following purposes:

- to verify the agreement of the countries which are the most active in seismic isolation development, to provide required experimental data and to analyze the data provided by others;
- to discuss the experimental tests and select a test data base for the benchmark exercises;
- to discuss the features of numerical models and codes used, in order to clarify the details of the data for calculations.

The Consultancy Meeting was held at the IAEA Headquarters in Vienna, 14-16 November, 1995. It was attended by representatives of Italy, Japan, and the Russian Federation. The USA expressed their interest in participating in subsequent activities, sending relevant materials (see Attachment I), and making experimental data available (Attachment II).

The agenda of the meeting, proposed by the Scientific Secretary, was approved without any modification. The meeting was opened by the Scientific Secretary, A. Rinejski of the International Atomic Energy Agency and was chaired by A. Martelli of ENEA, Italy. Mr. Martelli introduced the technical discussion on proposed activities.

2. PRESENTATION OF EXPERIMENTAL DATA

First set of presentations concerned the experimental data on isolators and isolated structures of possible interest for the benchmark study. These were given by V. Belayev for the Russian Federation, K. Hirata for Japan and A. Martelli for Italy and the USA (the latter on behalf of J.M. Kelly of EERC - UBC). Summaries of these presentations are reported below.

2.1. Russian experimental results on isolators and isolated structures and numerical methods

Basic technical approach to low-frequency seismic isolation of the building is based on application of pneumatic shock absorbers of high load-carrying capacity up to 10 MN

(vertical isolation) in combination with structures supports, ensuring the required motion transformation in the horizontal plane. Dynamic force characteristics of the seismic isolation system is substantially nonlinear. In vicinity of the static position of balance in the characteristics there is a part of the curves with increased stiffness that creates a zone which is nonsusceptible to the dynamic loads, the amplitudes of which do not surpass the preset level. In order to increase the energy dissipation the system is supplied with additional viscous dampers. The selection of the dynamic characteristics of elastic and damping elements of the seismic isolation system was carried out to satisfy a series of design requirements, determined by the operation conditions of the safety systems and the interface systems which connect the reactor building and adjacent buildings. Procedure of seismic isolation selection and design is step-by-step as a rule. It incorporates steps of preliminary selection of facility parameters, design assessment of dynamic efficiency of system, selection of structural layout and refined calculation. Thus, taking into consideration the results obtained in various steps, the necessary corrections and verification of the system parameters is carried out: stiffness and inertial characteristics of isolated building, characteristics of isolation devices, conditions of installation and fixing of internal equipment.

As a criteria of dynamic seismic isolation efficiency the condition of relation between the required and actual seismic shock stability is used. Criteria of required seismic shock stability are determined when designing the seismic isolation system.

The present practice in Russia shows the feasibility of simulation of the real non-stationary effect of seismic waves on large structures through underground explosions. The Russian computer codes are based on original methodology and results of model analysis and full-scale seismic testing on 3D seismic excitation platform (load-carrying capacity up to 500 t). For simulation of loads and displacements of elements of seismically isolated buildings the technology of buried underground explosions was used. Seismic explosive technology permits to expand substantially (in comparison with seismic platforms which have mechanical drive) the possibilities to reproduce the effects of severe earthquakes, especially for the high floors of buildings. The efficiency of the technology in question, including the economical efficiency, has been confirmed during the testings of VVER-640 reactor building.

2.2. Japanese Experimental Data on Isolators and Isolated Structure

In Japan, extensive experiments on both isolators and isolated structures have been carried out so far. In the presentation, shaking table tests conducted by CRIEPI, and earthquake monitoring of an existing building are introduced. CRIEPI completed two kinds of shaking table tests on model-isolated structure. One is a test using frame model; the goal of the experiment is to grasp the vibrational characteristics of the isolated structure under design earthquake motion and to obtain data to compare with analysis. Another is a test using rigid mass model, aiming at grasping the vibration behaviour of the isolated structure in its ultimate state, and earthquake motions of up to 6 times as large as the design are inputted. In both experiments three types of isolators are used, i.e., natural rubber bearing (NRB) with steel damper, lead rubber bearing (LRB) and high damping rubber bearing (HDR). Loading tests on these isolators in both horizontal and vertical directions were conducted. As for existing base isolated structure, snap back test was conducted and earthquake observation is going on.

As for data availability, data on isolator tests and shaking table tests conducted in CRIEPI may be available, and data on existing building introduced in the presentation will be possibly available.

2.3. U.S. experimental results on isolators and isolated structures

Since Prof. J.M. Kelly of EERC-UBC, USA, was unable to attend the meeting, A. Martelli made some short remarks on the tests performed in the USA on isolators and isolated structures which may be of interest for the CRP. In particular, he mentioned that a set of scaled isolators were manufactured in Italy by ALGA S.P.A in the framework of the U.S. ALMR project. ALGA developed a rather hard rubber compound (G=1.4 MPa) as required by the U.S. team. These isolators were successfully tested at EERC-UBS, together with similar isolators which had been manufactured in Japan. It has been planned to perform shake table tests on a PRISM mock-up supported by such bearings (part of which should be in Italy). A. Martelli expressed the opinion that the results of tests on the aforesaid bearings could be very useful for the CRP activities, together with those of tests foreseen on the PRISM mock-up, if available. In fact, the ALMR isolators have a much larger G value with respect to the HDRBs used in Italy and Japan (see below). With regard to the isolated structures, it was noted by A. Martelli that data related to further mock-ups with steel-frame superstructure are available at EERC - UBC, which could be used for the CRP, should not the results of tests on the previously mentioned mock-up be available on time for the CRP activities.

2.4. Italian experimental results on isolators and isolated structures

A. Martelli (ENEA) mentioned that wide-ranging experimental campaigns have been performed in Italy since 1990 on High Damping Rubber Bearings (HDRBs) and structures provided with such bearings. The tests which have been already completed concern the HDRBs which have been installed in isolated buildings in Italy, mainly those of the Administration Center of the National Telephone Company (TELECOM Italia) at Ancona and the twin conventionally founded and isolated houses at Squillace (Lalabria). They included experiments on the full scale isolators, isolators scaled to different sizes, rigid masses supported by both full scale bearings (1,000 ton mass, subjected to pull-back tests and scaled bearings, 40 ton mass, also subjected to shake table tests with one-, two- and three-directional simultaneous excitations) and finally, the actual isolated buildings. The latter included both forced and free vibration tests, which were performed to a maximum value of the initial displacement of 110 mm for the TELECOM-Italia building. The tests being in progress at present which were mentioned by A. Martelli are as follows:

1) A project of the National Utility (ENEL, S.p.A) for the seismic protection of electric equipment using HDRBs and other isolators;
2) A project funded by the European Commission concerning the optimization of HDRBs;
3) A project being performed by JRC Ispra in cooperation with Italian partners, which has already concerned pseudodynamic tests on large scale structures supported by HDRBs;
4) The cooperation of ENEA with U.S. partners for the ALMR project.

Finally, new activities concerning seismic isolation which are beginning in Italy in the framework of EC funded projects were mentioned by A. Martelli, namely:

- the optimization of energy dissipation devices and hydraulic couplers and evaluation of the related benefits on the design of structures;
- the development of innovative techniques to improve stability, including seismic protection, of cultural heritage;
- the extension of design guidelines previously developed by ENEA for isolated nuclear plants to three dimensional pneumatic systems developed by the Russian Federation and "Rolling- Ball Rubber - Layer" systems developed by MRPRA in the UK.

3. DISCUSSION AND DEFINITION OF DATA FOR THE CRP

At the conclusion of these presentations, a discussion was held among the participants, in order to select the test data on isolators and isolated structures to be used for the benchmark exercise. It was decided that the data on isolators shall be provided by each participating country to A. Rinejski, no later than December 15, 1995, while those on isolated structures may be made available later (in 1 to 1.5 years). The reason is that the first phase (first and second year) of the benchmark study will be devoted to the isolators only. For the same reason, detailed agreements were made only as regards the isolators, while only the types were proposed for the isolated structures.

As regards the isolators, the data to be provided shall include:

Italy: high damping rubber bearing with soft compound (G=0.4MPa)
Japan: rubber bearings (possibly lead plug rubber bearings);
Russian Federation: pneumatic devices and viscous dampers;
USA: High damping rubber bearings, possibly with hard compound (G= 1.4 MPa)

K. Hirata will ask for the necessary authorizations concerning the distribution of Japanese data related to the lead plug bearings. He will confirm the availability of such data to A. Rinejski by the end of November 1995.

J.M. Kelly will be asked by A. Martelli to confirm the availability of the aforesaid data by the same date.

It is noted that the isolators to be considered in this phase of the study shall possibly be those also used in the isolated structures to be analysed in the second phase (3rd and 4th year).

It was stressed that the data to be provided for the isolators shall concern:

- all the geometric, operating, and material data;
- experimental stabilized cycles at various shear strain levels (from small to very large shear including the design value) under constant vertical compression load.
- experimental loading curves for the evaluation of vertical stiffness at zero lateral displacement, for different vertical loads including the design value;
- for rubber bearings, all the data necessary for the definition of the hyperplastic model in the detailed bearing model.

The experimental data shall be provided on diskette in ASCII format (IBM compatible). Detailed drawings shall be provided for the different isolation devices.

As regards the isolated structures, it has been preliminarly decided that the following contributions will or may be provided:

Japan: Rigid mass mock-ups supported by the previously tested lead plug rubber bearings.
Italy: Flexible superstructure mock-up supported by the previously tested high damping rubber bearings with soft compound (to be tested at ISMES very soon).
USA: Flexible mock-up possibly supported by the previously mentioned high damping rubber bearings with hard compound (if it will be tested on time - otherwise a different mock-up).
K. Hirata will confirm the availability of aforesaid Japanese test data, similar to those concerning the isolators.
A. Martelli will ask for the ENEL and EC approvals for making the Italian mock-up test results available.
J.M. Kelly will be asked by A. Martelli to express his opinion on the feasibility in due time, of structure mock-up tests using the ALMR isolators.

In addition, the participants of the Consultancy Meeting decided to ask other partners in the benchmark exercise to make some of their experimental data available, in particular:

South Korea should be asked to provide data concerning the test model used for isolated spent fuel storage pools;

JRC Ispra should be asked to provide data concerning pseudodynamic tests on large scale isolated structures supported by high damping rubber bearings.

4. DESCRIPTION OF NUMERICAL METHODS

The second set of presentations at the consultancy concerned the numerical methods and computer codes used in the participating countries for the anlysis of isolators and isolated structures. These were given again by the previously mentioned representatives of the Russian Federatin, Japan and Italy.

The presentations of Italy and Japan are summarized below. That of the Russian Federation has already been summarized above.

Numerical methods and computer codes used in Italy for the analysis of isolators and isolated structures

A. Martelli pointed out that adequate calculations of rubber bearings can be performed only if the parameters necessary for the definition of hyperplastic model of the rubber are made available, in addition to a detailed description of the geometric characteristics. He distributed a document issued by ENEA which details these items. A. Martelli described the numerous finite - element calculations which were performed in Italy to analyse the behaviour of various HDRBs, especially those manufactured and tested in the framework of the EC-funded project for the optimization of HDRBs. He showed that the agreement

between measured data and numerical results is excellent for all considered combinations of parameters (G, shape factor, attachment system, etc.). He also presented the results of the study which was carried out at ENEA to "further optimize" such bearings (so as to increase the shear strain value leading to failure) and to design "squatter" bearings (so as to study the effects of the secondary shape factor). The results of the tests which will be performed very soon will allow for checking if the forecast of ENEA was adequate. Anyway, the models developed already look very satisfactory, thus similar models will also be used by ENEA in the CRP activities, provided that all the necessary data are made available by the other partners.

As regards the simplified bearing models, A. Martelli clarified that these are necessary to evaluate the response of isolated structures. Up to now, ENEA has made use of spring & damper models, implemented on the special purpose programme ISOLAE, which also includes oscillators describing the superstructure. The need for developing ISOLAE was that computer codes like ABAQUS, which are usually adopted for structural analysis, do not include the possibility of assuming hysteretic damping (i.e. damping varying with displacement). On the contrary ISOLAE accounts for variations of both horizontal stiffness and damping with displacement. Thus, the procedure using ISOLAE consists of the following steps:

- definition of oscillators describing the superstructure;
- definition of isolators spring/damper models;
- calculation with ISOLAE of the seismic motion at the base of the superstructure;
- calculation of the superstructure with ABAQUS;
- iteration of the procedure, if necessary (by defining a more adequate set of oscillators describing the superstructure).

However, a new procedure is now being developed, which consists of implementing a variable spring and an elastic-perfectly plastic beam, in ABAQUS (which is possible): this should permit to simulate hysteresis loops correctly. Should this new methods be successfully checked, it will be used in the CRP calculations of ENEA. Otherwise, the previous methods (i.e. that adopting ISOLAE) will be utilized.

Numerical Methods and Computer Codes used in Japan

In Japan, computer codes used for seismic response analysis are developed in various institutes or companies, and so called "general purpose structural analysis programmes" are generally not used. In the modelling of isolator, several models are used. B_i-linear model is most simple and widely used, (for design purpose it is enough); also Ramberg-Osgood model can be used, as both models are incorporated as typical non-linear elements into non-linear response analysis code. For more detailed analysis a non-linear spring, in parallel with a bilinear sping can be used for modelling the isolators. As for the vertical direction, a linear spring is used for design purposes as far as the tensile fore is not applied to the isolator. In the case where the effect such as dependency of vertical stiffness on horizontal displacement has to be considered and the models considering this effect has to be incorporated. For the purpose of predicting the ultimate behaviour of the isolated structure or the simulation of shake table test, several models taking into account the hardening and other phenomena occurring in the rubber bearing are proposed. If horizontal two directional earthquake input is considered, so called "multiple shear spring model" is used.

In the response analyis for design purposes, isolators are usually grouped into three spring elements, i.e. horizontal, vertical and rotational (rocking). However, considering some dependency of mechanical characteristics, such as the effect of vertical loading on horizontal stiffness, etc., the grouping is not recommended. Superstructure of the isolated building is modelled in the same way as for the fixed-base structure; for base isolated nuclear reactor buildings non-linear resorting force characteristics are used as for fixed-base one. Soil structure interaction (SSI) is taken into account using horizontal, vertical and rocking soil springs, although the effect of SSI is small for the isolated structure, especially on hard soil. Since the base isolated structure behaves non-linearly, the equation of motion is solved in general by a direct integration method; however, in some cases equivalent linearization technique is used.

5. CONCLUSIONS

It was confirmed that the calculations will be performed by the countries participating in the Consultancy Meeting, for the isolators and structures described above. J.M. Kelly will inform A. Rinejski if the U.S. will only provide experimental data, or will also participate in the calculations. This question will be addressed by A. Martelli to J.M. Kelly. Other countries and international organizations which have already informed the IAEA that they would like to participate in the benchmark exercise (e.g. South Korea, India, JRC Ispra) will be asked to base their calculations on the aforesaid data.

With regard to the planning of the activities the proposed schedule for the first two years is the following:

- December 15, 1995: receipt of experimental data on isolators by IAEA
- January 20, 1996: distribution of the above data to the organizations involved in the benchmark (CRP).
- February-May, 1996: analysis of the above data, first calculations and first discussion among the participant in the CRP (first RCM at St. Petersburg, Russia).
- June 1996 - December 1997: Completion of activities concerning isolation; definition of isolated structures to be analysed; distribution of the related data.

The first Research Coordination Meeting (RCM) will be held at St. Petersburg (Russia) from 27 - 31 May 1996. The RCM will discuss in detail the experimental data on isolators, by clarifying all possible doubts, and perform a first comparison of the results of calculations, possibly already carried out, based on such data. The organizations which have provided the data will make detailed presentations on the experiments which led to such data and on the calculations were based on them.

The participants of the Consultancy Meeting agreed that the CRP will be very beneficial for enhancing knowledge on seismic isolation technology and improving international cooperation on such topic, which has a great potential in the seismic safety of advanced reactors.

The Scientific Secretary of the IWGFR distributed the form "Proposal for Research Agreement" to the participants of the Consultancy Meeting. This form will be also sent by IAEA to the other countries/organizations which expressed interest in participating in the CRP. This document must be filled out and sent to the Agency by the end of December 1995.

The participants of the Consultancy Meeting thank the Agency for the excellent organization of the meeting itself, which made it possible for all the expected results to be achieved.

During the closing of the meeting, the Scientific Secretary expressed the Agency's thanks to A. Martelli, Chairman of the Meeting, for his very effective guidance of the meeting and for the constructive discussion.

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ATTACHMENT I

UNITED STATES MISSION TO THE
UNITED NATIONS SYSTEM ORGANIZATIONS IN VIENNA

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622-I3-CT-2738

October 30, 1995

Mr. B. A. Semenov
Deputy Director General
Department of Nuclear Energy and Safety
Room A2680
International Atomic Energy Agency
Vienna International Center

95/11/02 3
cc: Semenov
Kupitani
Rinejski
622
X

Dear Mr. Semenov:

This is in response to your letter dated 12 October, 1995, requiring a Consultants' advice on "Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures" 14-16 November, 1995, in Vienna. The U.S. Government would like to inform the agency that Mr. James M. Kelly is not able to participate at subject meeting and that there is no other U.S. participant available. However, Mr. Kelly will provide to the IAEA Scientific Secretary, Mr. A. Rinjeski, Division of Nuclear Power, information on The Wide Range of Isolator Test Programs for Application to Seismically-Isolated Nuclear Structures that have been conducted at the Earthquake Engineering Research Center.

For our records would you please provide the U.S. Mission with a list of attendees to this meeting at your earliest convenience. If we can be of further assistance in this matter, please do not hesitate to contact us.

Sincerely,

Jim Richardson
Nuclear Safety Attache

Summary of Experimental Research on Elastomeric Seismic Isolation Bearings at the University of California at Berkeley

James M. Kelly

Peter W. Clark

Earthquake Engineering Research Center

University of California at Berkeley

Background

- Research was initiated at EERC in the late 1970's and continues today
- A number of earthquake simulator tests and component tests of full-size and reduced-scale bearings have been performed — all of this data is still available at EERC
- Initial shake table test programs focussed on natural rubber bearings with various kinds of plugs (inserted in the center of the bearing) to enhance the damping capabilities of the system
- Later tests examined response of isolated bridge systems, tall frames subject to uplift, combined rubber/sliding systems, very soft isolation systems for soft soil sites, and ultimate behavior of isolated structures
- Component tests have investigated the response of isolators with widely-varying properties, including low shape factor/high shape factor, dowelled/bolted/recessed end connections, soft compounds, stiff compounds, neoprene, etc.
- More recent studies have looked at ultimate capacities of bearings and isolated structures

Bearing Test Programs

- Full-size prototypes for the Foothill Communities Law and Justice Center (FCLJC), the first base isolated building in the U.S., were tested in 1983. These bearings were recently re-tested to determine the variation in properties with age

- Bearings for a bridge test model have been tested extensively to quantify the buckling behavior of isolators

- Both high shape factor and low shape factor isolators have been tested as part of nuclear programs — of particular interest was their behavior under various axial loads and their performance with different end connection details

- A number of Japanese bearings have been thoroughly tested, with an emphasis on their behavior under extreme loadings — the results indicate that properly designed and manufactured isolators possess substantial overstrength and will not be the weak link of an isolated structure

Bearing Test Programs (continued)

- Full-size isolators for several building projects have been evaluated, including bearings for three low-cost housing projects in developing nations

- Bearings having soft compounds appropriate for regions of low seismicity or soft-soil sites have been characterized

- High-modulus elastomers for U.S. nuclear applications were most recently tested — a very thorough test program was performed to quantify the variations in their mechanical properties due to axial loading, loading frequency, strain cycling, etc.

- Interaction with elastomer compounders and bearing manufacturers has been developed over the years to tailor isolation components to particular applications

- All of this test data has been documented and archived and is available for future study

Earthquake Simulator Test Programs

- First tests examined the response of a five-story steel frame on elastomeric systems with various types of damping elements — plugs of different materials, external yielding steel dampers, etc.

- Later tests investigated the behavior of a nine-story steel frame subjected to uplift — special devices designed to prevent localized uplift and limit shear displacements were also studied

- Hybrid isolation systems incorporating both elastomeric bearings and sliding bearings were tested — such systems have seen applications in several buildings in the United States

- Tests of the five-story steel frame on low-modulus isolators indicated that isolation is a viable earthquake-protection scheme at soft-soil sites

- A bridge deck model has also been tested with particular emphasis on the instabilities in bearings with dowelled end connections

- The most recent tests were performed on a three-story reinforced concrete frame which exhibited substantial nonlinear behavior under extreme inputs — the purpose of these tests was to investigate the ultimate behavior of isolated structures and to demonstrate that properly designed isolators have capacities far in excess of those assumed in their design

Summary Report of the Third Consultancy on "IAEA/EC Comparative Calculation for ULOF (unprotected loss of flow) in the BN-800 Reactor", 11-13 December 1995, IAEA Headquarters, Vienna, Austria

1. PURPOSE AND ORGANIZATION OF THE CONSULTANCY

The purpose of this consultancy meeting was to discuss the progress of the IAEA/EC-sponsored comparative calculations and to clarify and agree on aspects of the proposed exercise, left open during the last consultancy meeting held in Brussels, 26-28 July 1995. The specific objectives of this consultancy were to:

- discuss and agree upon the neutronics and steady state fuel pin input data,
- review and discuss first results provided by the participating organizations, and define tasks for the next consultancy meeting.

The consultancy was opened by Poong-Eil Juhn, Director of the Division of Nuclear Power. He welcomed the participants on behalf of the IAEA and gave a briefing on the status of nuclear energy and fast reactor development. Mr. Juhn noted the magnitude of the benchmark problem to establish a basis, so as to evaluate the pros and cons of a BN-800 type reactor under, hypothetically severe conditions.

Mr. G. Van Goethem welcomed the participants on behalf of the European Commission (EC). He pointed out that EC supports this benchmark exercise, and thanked the participants for contributions to the comparative calculations for a severe accident in BN-800 reactor. He wished them a very productive time.

The Scientific Secretary of the IWGFR formulated the purpose of the consultancy, proposed the Chairman (Dr. D. Struwe, Germany) and the programme of the consultancy meeting (Attachment 1). A total of eleven presentations were made by the participants (Attachment 2).

2. ANALYSIS PERFORMED BY ORGANIZATIONS

2.1. Steady state fuel characterisation /1-5/

Five countries - France, Germany, India, Russia and United Kingdom - participated in the pre-transient calculations of the pin characteristics. The names of codes employed were SAS4A, GERMINAL, TRAFIC, KONDOR and "unspecified", respectively. The only results for a radial breeders pin came from the UK and Germany; both of these sets of calculations also considered heating of the sodium flowing past the fuel pins from the lower axial breeder.

GERMINAL 1-2 and SAS4A codes include JOG models in their calculations. In addition, the effect of fuel cracking with relocation, such as might be expected during a transient, on the thermal behaviour and, consequently on the gas release, was investigated in the German contribution. In the case of the Russian and Indian codes, calculation of the fission gas release was not consistent with thermal results, due to the fact that thermal evolution calculation and gas release modelling are not correlated.

Presenters of the Indian and Russian results were asked to give some supplementary information, namely values of the linear rating, burn-up and neutron dose during the course of the irradiation at the peak-rated position.

Before moving on to the next stage of the exercise, participants were requested to repeat their calculations using modified input data. These were:

- Russian clad swelling rule,
- Modified linear rating dependence during the three cycles, as in tables 3.2.a to c of the IPPE input data /5/. Linear ratings are to be at power (c) for the radial breeder, use a total residence time of 490 days and assume that the total power generated in all the breeder pins after that time is 51.47 MW. The power is assumed to increase linearly with time from zero, and
- Linear ratings must be re-normalized, in order to get a total reactor power of 1500 MW at the end of a cycle.

2.2. Neutron physics analysis /4-8/

Within this topic, different contributions were given from: IGCAR (India), HITACHI (Japan), and ENEA (Italy) concerning:

- power distribution;
- Doppler coefficients;
- reactivity worth and reactivity feedback coefficients;
- sodium void reactivity;

carried out by different calculation approach, methodology and basic-data, for a comparison with the updated IPPE-input data for the BN-800 type reactor.

Comparative analysis of the results of the power distribution show:

- a very good agreement between HITACHI and ENEA results in fuel zones (using the diffusion approach) and structural zones (transport approach);
- a discrepancy between HITACHI - ENEA results and IPPE results, particularly in B_4C - zones;
- a discrepancy between ENEA and IPPE results regarding the power distribution (per channels).

Because this last item will be repeated for all the other parameters participants proposed to define, from IPPE, the exact composition of channels in the hexagonal-core scheme.

Concerning the Doppler coefficient the HITACHI and ENEA values are to be compared with the small IPPE value and this requires clarification as well as its distribution per channel. The Doppler coefficient values in the temperature range ($T_{f,o}$; T_f) as well as the applicable temperature range should be specified.

For the fuel density reactivity worth the difference between ENEA and IPPE results

is 9.4%, and may be justified by the different core geometry model: 3D instead of 2D but this is not sufficient to explain the large differences for the stainless steel and sodium density reactivity worth.

Another question relating to the densities reactivity worth is the discrepancy between IGCAR and IPPE over the net reactivity change in power raising of the reactor.

More complicate situation is the Na-void reactivity value:

- IPPE has "proposed" the value: - 3.82$
  carried out -most probably - in diffusion theory: 2D code to compare with:
- - 1.13 $ from HITACHI diffusion theory 2D code and
- - 1.02 $ from ENEA diffusion theory 3D code.

In transport solution, the HITACHI and ENEA results are not directly comparable because of the difference in methodology and voided zones.

It was noted that the 3D homogenous solution proposed by HITACHI must be corrected by the fission products (FPs) contribution which is not less than + $0.8, as well as the 3D heterogeneous solution. Also the HITACHI conclusion that - $0.9 is expected at the end of cycle (EOC) condition for whole core Na-void reactivity must be corrected for:

- systematic errors $\geq$ 1 $
- real axial burnup distribution, and
- control rods partially inserted in the core at EOC.

After a discussion the participants agreed that: (a) several points need to be discussed with the IPPE in order to understand and explain the different performances; and (b) supplementary work is needed for a correct calculation of Na-void reactivity value.

2.3. Pre-boiling phase calculations /8-10/

The HITACHI representative presented the preliminary results of the pre-boiling phase ULOF (unprotected loss of flow) transient evaluation, in which the ULOF conditions and end of equilibrium cycle (EOEC) core data were as given by the IPPE input data. The analysis does not include the axial heat conduction in the B_4C reflector region. The results showed that the core could be sluggish in response to ULOF transient, mainly because of the negative core radial expansion reactivity if the load pad free expansion was assumed in the calculation model.

The representative of the Power Reactor and Nuclear Fuel Development Corporation (PNC, Japan) presented improvement and preliminary application of plenum gas blowout model in SAS4A. The model validation has been performed through CABRI experimental analysis. The discussion was about the modeling of the orifice model which might be used in the BN-800 cases. The presenter noted that this is a normal model of gas flow through an orifice, which could be found in literature, but the cross sectional area of the rip which is an important parameter, depends on the rupture mode, that is, creep failure or melt-through.

2.4. Qualification of the efficiency of the load pad radial expansion feedback /5,9,11/

After a discussion, majority of the meeting's participants agreed that:

- the state of the art concerning the evaluation of radial expansion reactivity feedback is not yet sufficient to take into account in the comparative calculation,
- it is best to omit this effect if experimental data demonstrating its validity could not provided.

3. TASKS FOR THE NEXT MEETING

The meeting's participants agreed on the following tasks for the next meeting:

- updating of fuel pin characterization to EOEC conditions;
- preparation of an alternative neutron physics input data set based on transport theory;
- transient analysis either to first fuel pin failure or to 50s into the transient boiling time;
- preparation of proposals of how to come to a commonly accepted final report

4. DATE AND VENUE OF THE FOURTH CONSULTANCY MEETING

It was agreed that the fourth consultancy meeting should be held at the EC Headquarters in Brussels, Belgium, from 25 to 27 June 1996.

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Consultancy on the IAEA/CEC Comparative Calculations for a Severe Accident in the IPPE Core Design Variant with a Near-zero Void Coefficient of the BN-800 Type Reactor, (unprotected loss-of-flow, ULOF)

IAEA Headquarters, Vienna, 11-13 December 1995
Agenda

Chairman: Dr. D. Struwe (Germany)

Monday, 11 December 1995

1. Opening (IAEA, EC)
2. Approval of the agenda (all)
3. Approval of the report of the 2nd Consultancy in Brussels, 26-28 July 1995 (all)
4. Review of actions (all)
5. Comparison of safety related design parameters of the IPPE core design variant and the BN-800 project design
6. Status of the case set-up. Quick review of the information provided by IPPE (all)
7. Neutron physics input data
   - 7.1. IPPE reference calculation (new results)
   - 7.2. Complementary results: Italy (ENEA), Japan (Hitachi), India (IGCAR)
   - 7.3. Discussion and conclusion

At 12.30 - Lunch at the VIC Restaurant - IAEA Hospitality

Tuesday, 12 December 1995

8. Steady state fuel pin characterization
   - 8.1. Presentation of results
     - 8.1.1. Russia (IPPE)
     - 8.1.2. India (IGCAR)
     - 8.1.3. UK (UKAEA)
     - 8.1.4. France (IPSN)
     - 8.1.5. Germany (FZK)
   - 8.2. Discussion and conclusion

9 Pre-boiling phase ULOF transient evaluation Japan (Hitachi)

Wednesday, 13 December 1995

10. Qualification of the efficiency of the load pad radial expansion feedback
    - 10.1 IPPE evaluation results
    - 10.2. Complementary information from other participants
    - 10.3. Discussion and Conclusion
11. Work programme up to the next meeting
12. Date and place of the next meeting
13. Drafting the minutes of the Consultancy (all)

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ATTACHMENT 2

Written materials presented at the Consultancy Meeting on IAEA/EC Comparative Calculation for a Severe Accident in the core of the BN-800 Reactor (unprotected loss-of-flow, ULOF), 11-13 December 1995, at the IAEA Headquarters.

1. Steady-state calculations of a BN-800 type reactor with GERMINAL 1;2.
   F. Lemoine, (France)

2. First preliminary results of the BN-800 core characterisation at EOEC conditions.
   D. Struwe (Germany)

3. Some TRAFIC calculations of the behaviour of fuel and breeder pins during the normal operation of a BN-800 reactor.
   D.G. Martin (UK)

4. Steady state calculations and fuel pin characterization for end of equilibrium core in BN-800 reactor with near-zero sodium void coefficient of reactivity.
   R. Harish, (India)

5. Input data for EC/IAEA comparative calculations for a severe accident in the IPPE core design variant of the BN-800 type reactor.
   I. Kuznetsov, L. Zabudko (Russia)

6. Neutronic evaluation of the BN-800 type reactor.
   K. Kawashima (HITACHI, Japan)

7. BN-800 Neutronic analysis.
   G. Glinatsis (Italy)

8. Improvement and preliminary application of plenum gas blowout model in SAS4A.
   H. Niwa, (PNC, Japan).

9. On the radial core expansion effect.
   H. Niwa (PNC, Japan).

10. Pre-boiling phase ULOF transient evaluation.
    K. Kawashima (HITACHI, Japan).

11. Load pad reactivity effect in Phenix and Superphenix reactors.
    J.M. Frizonnet (France).

** ** ** ** ** **

Report of the Consultancy on "Harmonization and validation of fast reactor thermomechanical and thermohydraulic codes using experimental data", VIC, Vienna, 1-2 Feb. 1996

The purpose of the consultancy was to prepare experimental input data for the Agency's planned Co-ordinated Research Programme (CRP) on "Harmonization and validation of fast reactor thermomechanical and thermohydraulic codes using experimental data". A benchmark exercise on "Tee junction of LMFR secondary circuit" was proposed by France in the scope of the said CRP.

The physical phenomenon chosen here deals with the mixture of two flows of different temperatures. In a LMFR, several areas of the reactor are submitted to this problem. They are often difficult to design, because of the complexity of the phenomena involved. This is one of the major problems of the LMFRs. This problem has been encountered in the Phenix reactor on the secondary loop, where defects in a tee junction zone were detected during a campaign of inspections. The tee junction component in the Phenix reactor has been replaced by a new one, with a number of thermocouples to investigate the temperature fluctuation in the mixing area.

At the Annual Meeting of the IWGFR in May 1995 it was agreed that the Scientific Secretary of the IWGFR be in contact with French experts to prepare the exercise aimed at testing the ability of analytical methods.

The proposed work has been done at the consultancy, and the results are given below.

Mr. M. Sperandio, Head of the Department of Novatome-Framatome, Lyon, France, informed the Agency about the re-start of the French LMFRs Phenix and Superphenix in 1994. The reactors' power was increased in successive steps.

1. Presentation of the benchmark

The considered technical problem deals with the mixing of two flows of different temperatures in the secondary circuit of the French LMFR PHENIX, which is a 250 MWe demonstration plant, with three secondary loops, modular steam generators and integrated primary circuit. It has been operating since 1974. The main features of the circuit in the tee junction area are shown in figures (Appendix III). During nominal operation, sodium flows into the main pipe of the secondary circuit at 340°C. A small pipe, connected with a tee junction to the main pipe, discharges sodium into the main pipe at 430°C. Two convergent flows of different temperatures (ΔT = 90°C) are therefore mixed in the tee junction area.

2. Scope and objective of the benchmark

The work of each participant of the benchmark is to evaluate the thermohydraulic behaviour of the phenomenon described previously, and to foresee the thermomechanical consequences on the main pipe of the secondary circuit of PHENIX. The final objective of the benchmark is to quantify the major differences in the procedures used by the participants and to confront the results with the actual phenomena observed on site.

3. Technical data of the benchmark

Geometrical characteristics of the component: The main pipe in the tee junction area, consists of a horizontal straight part, an elbow, and a vertical straight part where the tee junction is connected. In the straight parts, the main pipe has the following characteristics: inner diameter ϕ_i = 494 mm, thickness t = 7 mm. The elbow is characterized by the following data: angle a = 90°, radius R = 762 mm, thickness t = 9.53 mm. A circumferential weld is located 160 mm downstream from the tee junction, in the vertical straight part of the main pipe. The small pipe connected to the main pipe has the following characteristics at the tee junction: inner diameter ϕ_i = 68 mm, thickness t = 8.5mm. The detailed geometry is represented on figures (Appendix III).

Operating conditions: The secondary circuit of PHENIX has been operating during 90 000 hours at nominal power. The corresponding operating conditions are the following: In the main pipe : sodium at T_{1Na} = 340°C, flow rate Q_1 = 800 kg/s, internal pressure P_1 = 2.2 bar, and in the small pipe:sodium at T_{2Na} = 430°C, flow rate Q_2 = 7 kg/s, internal pressure P_2 = 2.9 bar. The direction of the flows is mentioned on figures.

Forces and moments in the pipes: Internal pressures inside pipes, dead weight of pipes, weight of sodium and thermal expansion of pipes, induce forces and moments. The corresponding membrane stress in the main pipe, in the tee junction area, is: -10 MPa.

Fluid: The fluid flowing inside pipes is sodium. The main characteristics (density, thermal conductivity, specific heat coefficient, thermal expansion coefficient and kinematic viscosity) are provided in Appendix I.

Base material: Both pipes are made of AISI 304 stainless steel, grade: Z5 CN 18.10. This steel corresponds to the 2S-type of RCC-MR code (ref.1).

Weld material: The circumferential weld has been made by a plasma welding in two passes, the first pass without weld metal, the second one using a wire grade 16-8-2 (single-V groove).The specifications of this wire are consistent with section RS 9523.2 of RCC-MR code (ref.1).

Material characteristics: Base material : The main characteristics (density, thermal conductivity, specific heat coefficient, Young's modulus, mean thermal expansion, yield stress, mean stress-strain curve, cyclic curve, best fit fatigue curve) of the base material (AISI 304) are provided in appendix II. They are taken from the 2S-type and 1S-type data of RCC-MR code. Weld material: the characteristics of the weld joint are assumed to be those of the base material, except for the fatigue curve which is provided in appendix II.

4. The main stages of the work:

1) Thermohydraulic and thermal analysis,
2) Thermomechanical and fracture mechanics analysis, and
3) Conclusion of the whole analyses, a final report preparation

The exercise should be concluded within 3 years, when, a final report will be published.

1) Thermohydraulic and thermal analyses: After the completion of thermohydraulic and thermal calculations, each participant to the benchmark shall: (a) characterize the mixing area, which means locating, defining the dimensions of the mixing area, defining the parts of the pipe subjected to the mixing phenomenon; (b) provide isothermal lines (mean temperatures) on the parts of the pipe as defined previously; (c) provide the temperature signals (temperature variations as a function of time) on inner and outer skins of the pipe at points judged significant, as well as at the points where the thermocouples represented on figures are located and (d) provide the results of a spectral analysis of the metal temperatures at the points defined above.

2) Thermomechanical and fracture mechanics results: After the completion of the thermomechanical and fracture mechanics calculations, each participant of the benchmark shall: (e) define the parts of the component subjected to thermomechanical damages, in the base material as well as in the weld joint; (f) quantify the damages on the component, which means: (f.1) defining whether cracks may be initiated or not, (f.2) defining the time needed to initiate cracks (if any), (f.3) characterizing the crack propagation as a function of the number of cycles applied (if any), (f.4) defining the depth of the cracks (if any) after 90,000 hours of operation.

3) Final report: Each participant shall produce a final report following the contents indicated below:

1. Objective
2. Thermohydraulic and thermal analyses
    - 2.1. Computer code description
    - 2.2. Model description
        - 2.2.1. Calculation domain presentation
        - 2.2.2. Boundary conditions
        - 2.2.3. Fluid/wall heat transfer
        - 2.2.4. Numerical scheme
    - 2.3. Results
        - 2.3.1. Average flow pattern (including results as mentioned in a/)
        - 2.3.2. Fluctuation characteristics in sodium
        - 2.3.3. Transmission of the fluctuations through the wall (including results as mentioned in b/,c/,d/)
    - 2.4. Discussion about damage risks of the component
3. Thermomechanical and fracture mechanics analysis
    - 3.1. Thermomechanical analysis
        - 3.1.1. Computer Code description
        - 3.1.2. Model description
            - 3.1.2.1. Calculation domain presentation
            - 3.1.2.2. Boundary conditions
        - 3.1.3. Stress fields
        - 3.1.4. Damage (including results as mentioned in e/,f.1/,f.2/)
    - 3.2. Fracture mechanics results

3.2.1. Method description
3.2.2. Hypotheses (including initial crack size)
3.2.3. Propagation analysis results (including results as mentioned in f.3/,f.4/)
3.3. Discussion about damages of the component
4. Conclusion of the whole analysis

References

Reference 1 : RCC-MR code - May 1993 Edition

APPENDIX I

Characteristics of liquid sodium

The characteristics of liquid sodium are given as a function of temperature in the table below:

Temperature (°C)	(K)	density (kg/m³)	kinematic viscosity (10^{-06} m²/s)	thermal expansion (10^{-04} °C⁻¹)	thermal conductivity (W/m.°C)	specific heat coefficient (J/kg.°C)
340	613	870.5	0.3614	2.731	74.56	1293.
350	623	868.1	0.3548	2.741	74.06	1290.
360	633	865.7	0.3485	2.750	73.56	1288.
370	643	863.3	0.3425	2.760	73.06	1285.
380	653	860.9	0.3367	2.770	72.57	1283.
390	663	858.6	0.3312	2.779	72.08	1281.
400	673	856.2	0.3260	2.788	71.59	1278.
410	683	853.8	0.3210	2.798	71.11	1276.
420	693	851.4	0.3161	2.808	70.62	1274.
430	703	849.0	0.3115	2.817	70.14	1273.

APPENDIX II

Characteristics of steel: Steel density ϱ, specific heat coefficient Cp, thermal conductivity λ, mean thermal expansion αm, young's modulus E, minimum yield stress at 0.2%, and mean stress-strain curves, cyclic curves, best fit fatigue curves (strain controlled) are given as a function of temperature in the tables and figures below:

Temperature (°C)	300	350	400	450
ϱ (kg/m³)	7846	7824	7803	7781
Cp (J/kg.K)	529	539	550	560
λ(W/m.K)	17.95	18.67	19.39	20.10
α_m (10^{-6}/K)	17.5	17.7	17.9	18.1
E (10^{-3}.MPa)	170	166	161	157
$R_{0.002min}$(N/mm²)	132	126	121	118

Cyclic curves

The cyclic curves are given by the following formula:

$$\overline{\Delta\varepsilon_t} = 100.\frac{2(1+\nu)}{3E}\overline{\Delta\sigma} + (\frac{\overline{\Delta\sigma}}{K})^{1/m}$$

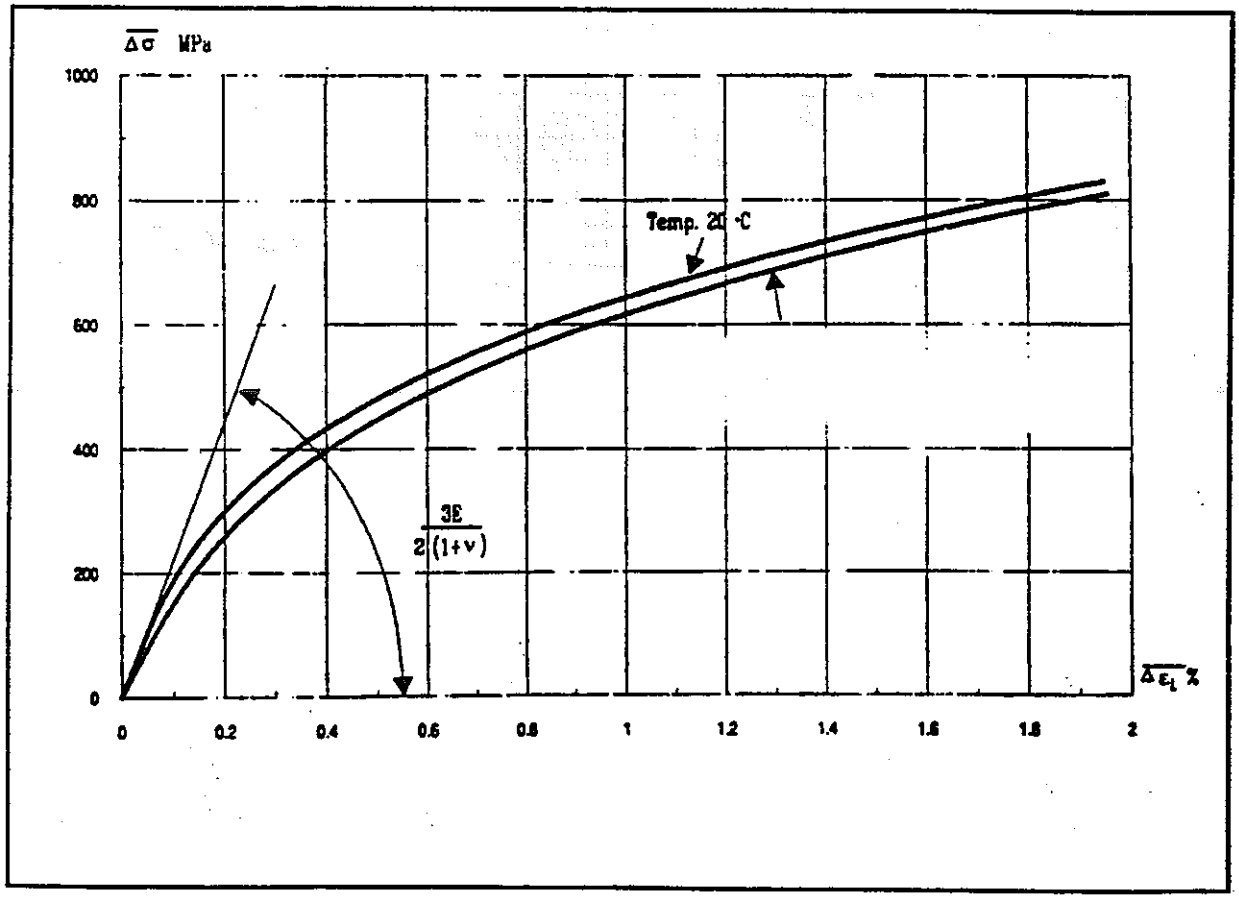
$\overline{\Delta\sigma}$: *equivalent stress range (MPa)*

$\overline{\Delta\varepsilon}$: *total strain range (%)*

E : Young modulus

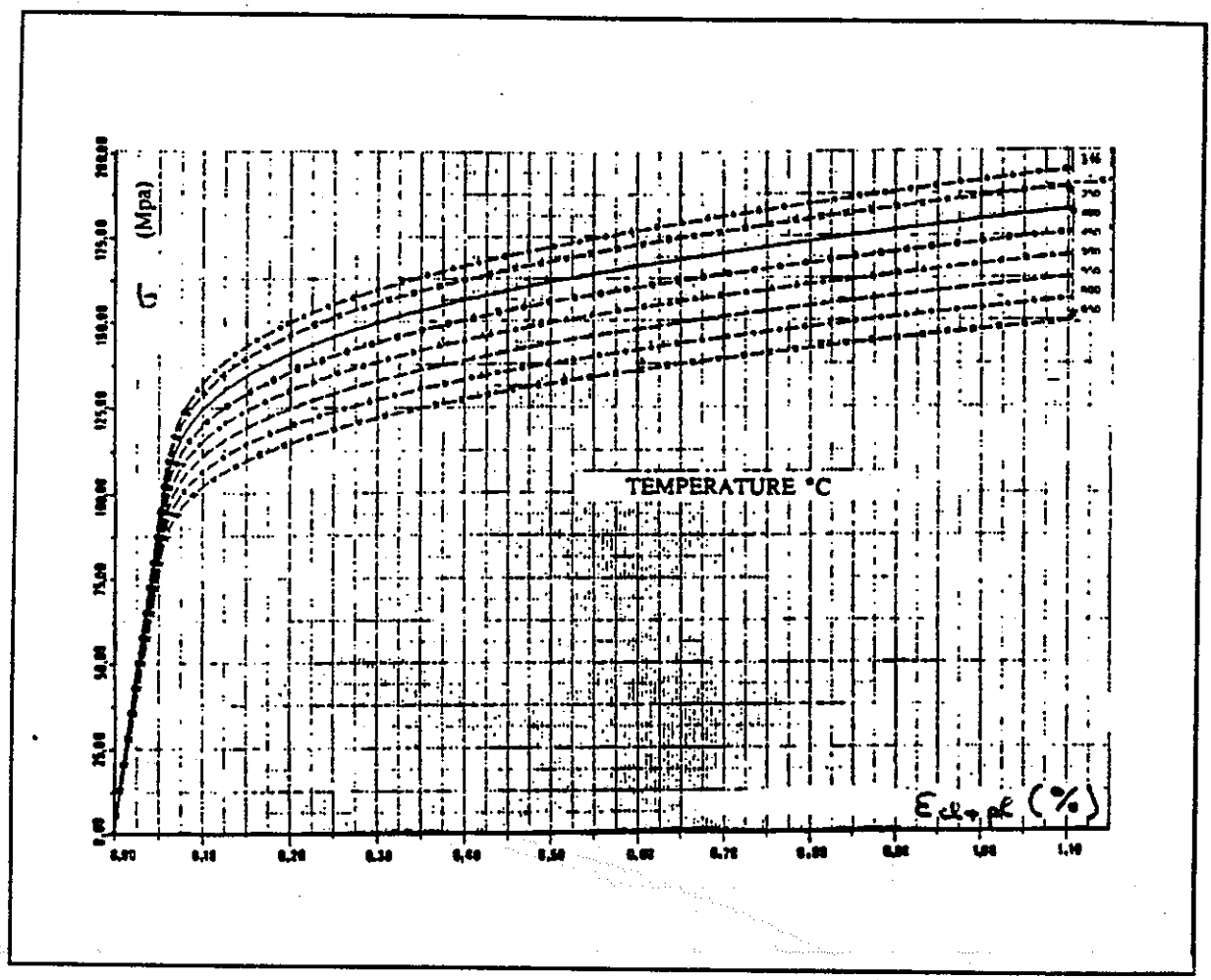
K,m : coefficients : K = 718, m = 0.319

They are represented in the figure below:



Mean stress-strain curves

The mean stress-strain curves are given as a function of temperature in the figure below:



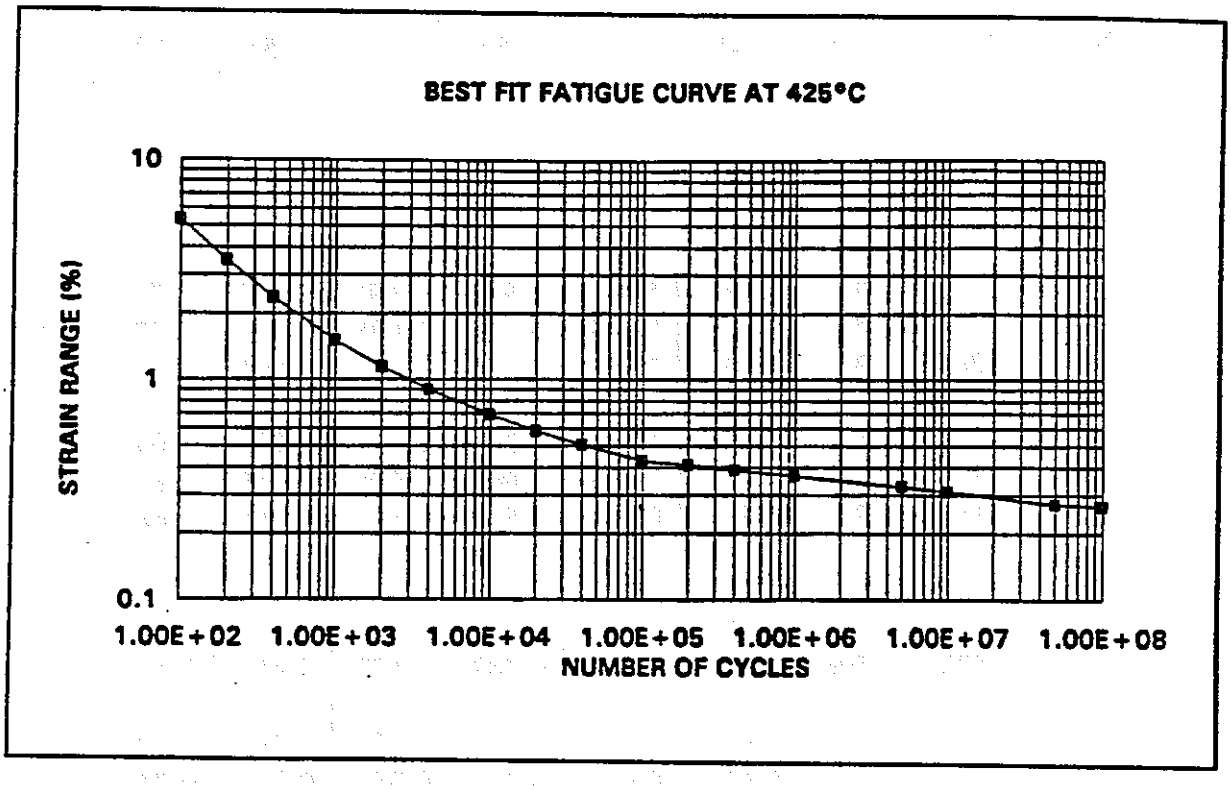
Best fit fatigue curves (strain controlled)

The best fit fatigue curves (strain controlled) for base material, are given for different temperatures in the table below, and at 425°C on the figure below:

Number of Cycles N	Temperature (°C)								
	20	425	450	475	500	525	550	600	650
10									
2.10									
4.10									
10^2	7.377	5.334	5.265	5.206	4.951	4.695	4.440	3.782	3.183
2.10^2	4.797	3.507	3.459	3.419	3.234	3.038	2.853	2.374	1.964
4.10^2	3.265	2.361	2.332	2.303	2.167	2.041	1.905	1.584	1.322
10^3	2.084	1.505	1.496	1.476	1.399	1.312	1.235	1.042	0.892
2.10^3	1.535	1.142	1.132	1.123	1.065	1.008	0.949	0.817	0.709
4.10^3	1.165	0.899	0.894	0.890	0.848	0.807	0.765	0.667	0.588
10^4	0.832	0.690	0.690	0.689	0.661	0.633	0.605	0.535	0.479
2.10^4	0.655	0.585	0.585	0.585	0.564	0.541	0.519	0.465	0.419
4.10^4	0.520	0.506	0.506	0.506	0.489	0.473	0.456	0.411	0.373
10^5		0.429	0.429	0.429	0.416	0.404	0.391	0.356	0.327
2.10^5		0.411	0.411	0.411	0.399	0.388	0.375	0.341	
4.10^5		0.390	0.390	0.390	0.378	0.367	0.355	0.322	
10^6		0.367	0.367	0.367	0.355	0.345	0.334	0.304	
5.10^6		0.329	0.329	0.329	0.319	0.310	0.300	0.273	
10^7		0.312	0.312	0.312	0.302	0.293	0.284	0.258	
5.10^7		0.273	0.273	0.273	0.265	0.257	0.249	0.227	
10^8		0.266	0.266	0.266	0.258	0.250	0.242	0.220	

Best fit fatigue curve for weld joint

The best fit fatigue curve at 425°C (strain controlled) for weld joint is represented on the figure below and tabulated in the table below:



Number of cycles N	10^2	2.10^2	4.10^2	10^3	2.10^3	4.10^3	10^4	2.10^4
strain range (%)	4.2672	2.8056	1.8888	1.2040	0.9136	0.7192	0.5520	0.468

4.10^4	10^5	4.10^5	10^6	2.10^6	4.10^6	10^7	2.10^7	10^8
0.4048	0.3432	0.3288	0.3120	0.2936	0.2632	0.2496	0.2184	0.213

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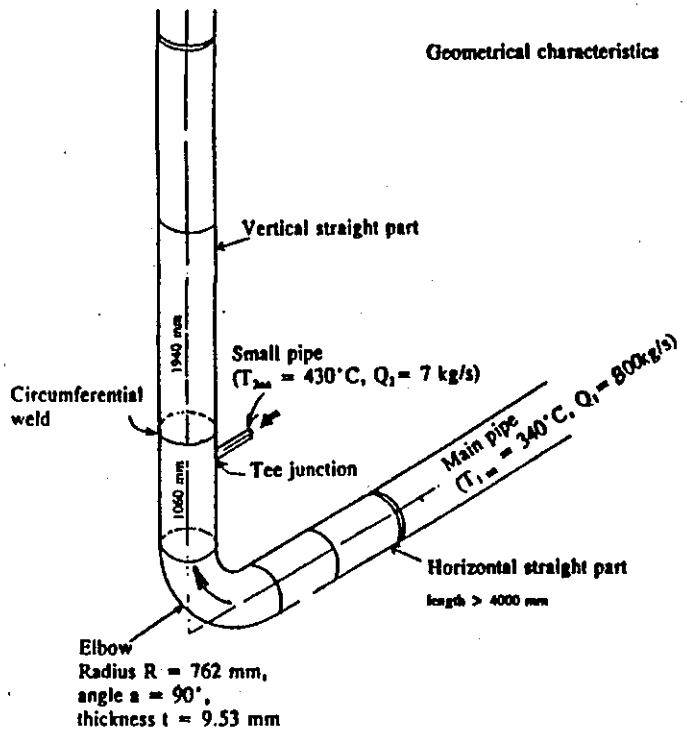

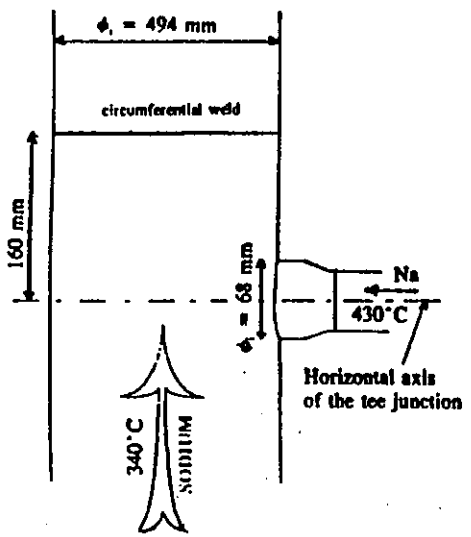

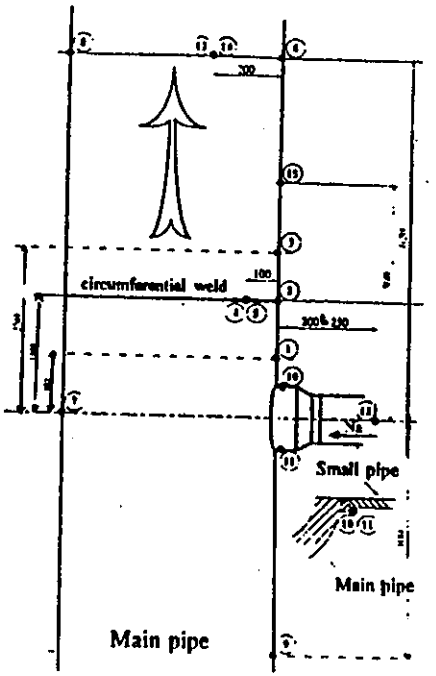

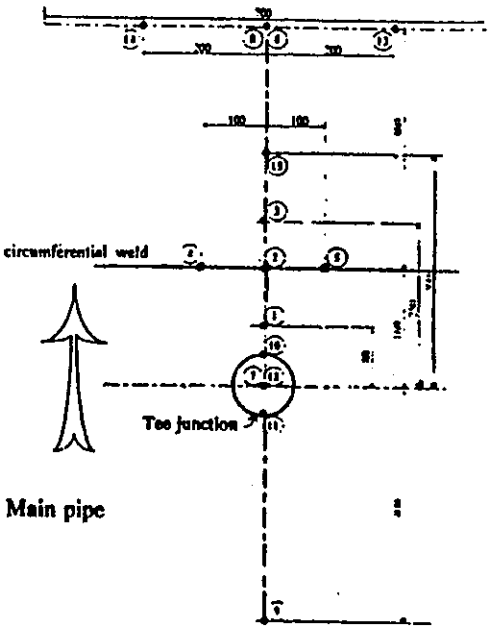

Location of thermocouples
(curvilinear abscissa)

INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors (IWGFR)

REVIEW OF THE IWGFR ACTIVITIES

FOR THE

PERIOD SINCE THE 28TH ANNUAL MEETING

APRIL 1996

REVIEW OF THE IWGFR ACTIVITIES

The following activities of the IAEA within the framework of the IWGFR have been carried out since the previous meeting of the Working Group.

1. Technical Committee Meetings (TCMs)

   a) Absorber Material, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles (Obninsk, Russia, 3-7 July 1995)

   b) Conceptual Designs of Advanced Fast Power Reactors, (Kalpakkam, India, 3-6 October 1995)

2. Research Co-ordination Meetings (RCMs)

   a) Intercomparison of LMFR Seismic Analysis Codes (Bologna, Italy, 30 May-2 June 1995)

   b) Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium/Water Reaction in LMFR, (Obninsk, Russia, 11-14 July 1995)

3. Consultancies

   a,b) IAEA/CEC Comparative Calculations for ULOF in the BN-800 Reactor (Brussels, Belgium, 26-28 July 1995, and Vienna, 11-13 December 1995)

   c) Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures (Vienna, 14-16 November 1995)

   d) Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes using Experimental Data (Vienna, 1-2 February 1996)

Taking into account that the complete reports of the above mentioned meetings have been sent to all IWGFR members in due course, only short information on the meetings are given.

Technical Committee Meeting on "Absorber Materials, Control Rods and Designs of Backup Reactivity Shutdown Systems For Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles", at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia, 3 - 7 July, 1995 (Countries participated: France, Germany, India, Japan, Kazakhstan, Russia and Georgia (observer), China sent two papers, which were included in the proceedings: 21 papers were presented, 35 specialists participated in the TCM)

Meeting participants reviewed and discussed the following topics:

- Status of development of designs of control rods for fast reactors and experience with their operation.
- Properties and behaviour of absorber materials for control rods; results of post irradiation examination of absorber materials, mechanisms affecting the properties and behaviour.
- Design of a backup reactivity shutdown system functioning on the following passive mechanisms:
  - curie point electromagnetic mechanism,
  - enhancement of thermal expansion of absorber rod drive lines,
  - hydraulically suspended control rod,
  - gas expansion modules in the core,
  - others.
- The possibility of optimizing the reactivity coefficients and the efficiency of Pu and minor actinides burning by using absorber/moderator materials in the core.

Conclusions

- The problems of reliable and effective control rods for FRs are being solved,
- The "poisoned reactor-burner option" (with $^{11}B_4C$) deserves further investigation,
- With the introduction of passive backup shutdown systems, future advanced LMFRs will have a very high degree of safety.

The TCM participants recommended to the IWGFR:

- To convene a consultants meeting in 1996 to assess the feasibility of using by Member States the REGAIN code developed in France,
- To establish a Data Bank on LMFR shutdown systems faults, and
- To convene a specialists meeting in 1998 to review and discuss the status of R&D on passive reactivity shutdown systems and reactor cores designed with strong negative reactivity feedback mechanism.

Technical Committee on "Conceptual Designs of Advanced Fast Power Reactors", Kalpakkam, India, 3-6 October 1995

(Countries participated in the TCM were: Brazil, China, India, Japan, Republic of Korea, and the Russian Federation. About thirty specialists participated in the TCM. A total of 18 papers were presented).

The meeting participants reviewed and discussed the following general topics:

- experience with demonstration and commercial size of LMFRs applicable to advanced fast power reactor designs,
- developments in advanced LMFRs, its design features,
- status of R&D programmes, findings and conclusions,
- predicted performance of advanced LMFRs.

a) Experience with LMFRs:

Major results and analysis of events, faults and operational experience with BN-350 and BN-600:

b) Advanced Projects:

Reactor block systems and components (overall design, core design, reactor assembly, main heat transport systems, fuel and component handling and storage systems, instrumentation and control, safety systems) of advanced reactors: BN-600M and BN-800M (Russia), DFBR-660 (JAPCO, Japan) and Advanced Design of Fast Breeder Reactor (PNC, Japan), PFBR-500 (India), CEFR-25 (China), KALIMER-300 (Republic of Korea) and REARA (Brazil)

Recommendation

The participants found the meeting very useful in knowing design trends in each of the participating countries and recommended that periodic (~ 3 years) meetings should be held to discuss in detail the progress made in the fast reactor designs.

IAEA/CEC Comparative calculation for a severe accident (ULOF) in BN-800 reactor with near zero void effect, Consultancies: EC, Brussels, 26-28 July 1995, IAEA, Vienna, 11-13 December 1995 (France, Germany, India, Italy, Japan, Russia, United Kingdom took part).

1. It was agreed that the main emphasis should continue to be placed on analysis of two ULOF cases:

   - Fully unprotected loss of flow (all safety rods are out of operation)

   - Partly unprotected loss of flow (all active safety rods are out of operation but all passive safety rods keep their ability to work normally)

2. Stages of benchmark:

   I. Case setup (input data specification)
   II. Steady-state fuel pin characterization
   III. Pre boiling transient
   IV. Boiling transient
   V. Post-failure transient

3. Scope of proposed contributions and main interest of participants

Organization	FZK, CEA, AEA	PNC	Hitachi	IGCAR	ENEA	IPPE
Interest	All stages	III, IV, V	I	All stages	I, II	I, II, III, IV
Codes	Germinal (CEA) TRAFIC (AEA) SAS4A (FZK) FRAX5D (AEA)	SAS4A [Simmer]	SAS3D		SAS3D	GRIF-A3D GRIF-SM DINROS

4. Consultancy meeting in Brussels, 25-27 June 1996.

   - Report by each participant about results and problems encountered in the execution of the calculations,
   - General conclusions about the first phase,
   - Organization of the second phase of the benchmark exercise to be started in September 1996.

COORDINATED RESEARCH PROGRAMME ON "INTERCOMPARISON OF LMFR SEISMIC ANALYSIS CODES"

The final RCM was held from 30 May - 2 June 1995 in Bologna, Italy. France, India (IGCAR, BARC), Italy, Japan (PNC, Hitachi, Mitsubishi and Toshiba) and Russia participated in the RCM.

- Verifications and improvements of the reactor core seismic analysis codes using the Core Mock-up Experiments.

- Three sets of experiments data for benchmark analysis were provided by Italy, France and Japan for PEC, Rapsodie and Monju Mock-Ups respectively.

- The codes used for the structural design of the core and reactor internals were validated in three stages by comparison of the analytical results with the Italian PEC reactor core experiments (I stage, 1st RCM), with the French and Japanese data (II stage, 2nd RCM) and by intercomparison of the codes and results (III Stage, Final RCM).

- The proceedings of the 1st and 2nd stages "Validation of the Seismic Analysis Codes using the Reactor Core Experiments" (Volumes 1 and 2) have been published and distributed. Volume 3: Comparison of Experimental Results with Computer Predictions for Reactor Cores to be published in 1996.

- The final RCM noted that the studies carried out within the framework of this CRP has a general Scientific value, since the results obtained can be employed in different types of reactors.

Coordinated Research Programme on "Acoustic Signal Processing for the Detection of Sodium Boiling or Na/H_20 Reaction in LMFR"

- The first stage (1986-1990) concerned acoustic monitoring of the reactor core by detecting sodium boiling which produce acoustic effects.

- The research programme has shown that there is a range of techniques available to detect boiling.

- The second (1990-1995) stage of the CRP covered the acoustic detection of leaks in Steam Generators Unit (SGU). The objective was to apply and develop/modify the techniques developed for boiling detection to detect acoustic leak noise in SGUs.

- Based on adaptive techniques methods are to be trained on leak data.

- Four sets of synthesized data (PFR, BN-600, SPX1 and ASB loop) including multichannel data were prepared and made available for analysis.

- The possibility to detect leaks between 0,1 and 1,0 g/s in a few seconds was demonstrated.

- In 1994 a number of end of life acoustic leak detection experiments have been performed on PFR steam generator by injecting argon, hydrogen and water/steam. Some of this data was made available to the CRP for analysis in 1995 the concluding year of CRP. This has provided an opportunity to the participants to test the ALD techniques on actual field data and removed the uncertainty in the previous benchmark caused by the mixing of signals obtained from different rigs.

- The work on this CRP would lead to the development of a reliable diagnostic system for enhancing the safety surveillance of SGU.

- The final RCM was held in Obninsk, Russia, from 11-14 July 1995. Countries particiated were: Australia, France, India, Japan, the Netherlands, Russia, UK. Germany sent the final report.

- The CRP has evaluated different signal processing techniques both on synthesized data and real field data and hence the results could provide invaluable information to the designers of ALD systems. In the long run, this would lead to enhancing the economics of LMFRs through better surveillance on steam generators.

- The work carried out within the framework of this CRP has a general scientific value. Signal processing techniques developed in this CRP for anomaly (boiling or leak) detection can be applied for detecting other types of anomalies in thermomechanical equipments (like cavitation, rattling etc.) in fast and thermal reactors.

NEW CRP ON "INTERCOMPARISON OF ANALYSIS METHODS FOR SEISMICALLY ISOLATED NUCLEAR STRUCTURES", (1996-1999)

Participating countries: **India, Italy, Japan, Republic of Korea, Russia, UK, USA and EC.**

- **The purpose of the CRP is to validate reliable numerical methods used for both detailed evaluation of dynamic behaviour isolation devices and isolated nuclear structures of power plants**

- **The Consultancy (Vienna, 14-16 November 1995, countries participated: Italy, Japan, and Russia) discussed a data base for benchmark exercises. It was agreed that the experimental data will be provided by Italy, Japan, Republic of Korea, Russia and USA.**

- **The first RCM will be held in St. Petersburg, Russia, from 27-31 May 1996, to discuss in detail the experimental data on seismic isolators and to perform a first comparison of the results of calculation based on such data and to approve a work plan for the CRP.**

The RCM will include a visit to the Seismic Center of the Scientific Research Institute for Capital Construction of the Ministry of Defence of the Russian Federation with a demonstration of dynamic testing of seismic protection systems.

NEW CRP ON "HARMONIZATION AND VALIDATION OF FAST REACTOR THERMOMECHANICAL AND THERMOHYDRAULIC CODES USING EXPERIMENTAL DATA (1996-1998)

- At the IWG meeting in May 1995 it was agreed that the Scientific Secretary be in contact with French experts to prepare the exercise. The proposed work has been done. A draft input data has been distributed in January 1996.

- A benchmark exercise on "Tee Junction of LMFR secondary circuit" was proposed by France in the scope of the said CRP.

- At the consultancy (1-2 Feb. 1996, Vienna, Mr. M. Sperandio, Head of the Department of Novatome-Framatome, Lyon, France) the technical data of the benchmark and comments made by Japan, Republic of Korea, China, Russia and UK have been discussed. Now, the CRP participants have the whole technical data for analyses.

- The physical phenomenon chosen here deals with the mixture of two flows of different temperatures. This problem has been encountered in the Phenix reactor on the secondary loop, where defects in a tee junction zone were detected during a campaign of inspections. The tee junction component in the Phenix reactor has been replaced by a new one, with a number of thermocouples to investigate the temperature fluctuation in the mixing area.

- The work of each participant of the benchmark is to evaluate the thermohydraulic behaviour of the mixture of two flows of different temperatures, and to foresee the thermomechanical consequences on the main pipe. The final objective of the benchmark is to quantify the major differences in the procedures used by the participants and to confront the results with the actual phenomena observed on site.

- We make all necessary formalities for participation in the CRP: France, India, Japan, Republic of Korea, Russia and may be other countries. It will be a joint IAEA/EC benchmark. First RCM plan to take place at the end of this year.

ARRANGEMENTS FOR MEETINGS IN 1996

a) TCM on "Creep Fatigue Damage Rules to be used in Fast Reactor Design", Manchester, UK from 11-13 June 1996.

An official agreement of UK to host the meeting has been received at the IAEA in January 1996.

France, India, Japan, Republic of Korea, Russian Federation and UK will participate in the TCM.

b) TCM on "Evaluation of Radioactive Materials Release and Transport in Fast Reactors", O-arai Engineering Center, 11-13 November 1996.

The meeting is under preparation.

An official agreement that "the Government of Japan is, in principle, willing to host this meeting" has been received at the IAEA in April 1996. In May, we will ask "Should the Government of Japan wish to host the meeting in question" (The second host Government Agreement letter).

We ask Dr. Yamashita to get an answer as soon as possible. After that we will send the official invitation letters to Member States. We ask the Members to speed up the nomination of the participants.

PUBLICATIONS

Series and No:	Title
IAEA-TECDOC-798	Validation of seismic analysis codes using reactor core experiments
IAEA-TECDOC-817	Influence of low dose irradiation on the design criteria of fixed internals in fast reactors
IAEA-TECDOC-819	Earthquakes, isolation, energy dissipation and control of vibrations of structures for nuclear and industrial facilities and building
IAEA-TECDOC-829	Verifications and improvements of reactor core seismic analysis codes using core mock-up experiments
IAEA-TECDOC-840	Unconventional options for plutonium disposition
IAEA-TECDOC-850	Fuel technology and performance of non-water cooled reactors
IAEA-TECDOC-866	Fast reactor data base
IAEA-TECDOC-876	Progress in liquid metal fast reactor technology
IWGFR/90	Correlations between material properties and thermohydraulics conditions in LMFRs

FAST REACTOR TECHNOLOGY: STATUS REPORT

1. Background and overview

2. Operation experience with prototype and demonstration LMFR

   2.1. Operation experience of BN-350
   2.2. Operation experience of Phenix
   2.3. Operation experience of PFR
   2.4. Operation experience of Super Phenix
   2.5. Operation experience of BN-600

3. Experience in construction and pre-operational testing of the prototype LMFR

   3.1. Construction and pre-operational testing of the LMFR SNR-300
   3.2. Construction and pre-operational testing of the LMFR "Monju"

4. LMFR physics and core design

5. LMFR safety

   5.1. New trends in safety principles and goals
   5.2. Emergency heat removal by natural and forced convection

6. Some aspects of LMFR's main components instrumentation and inspection

   6.1. Steam generator: design, instrumentation and inspection
   6.2. Reactor core and other components

6.3. Ultrasonic instrumentation
6.4. In-service inspection and repair

7. Core structural material and fuel technology for high burn-up

7.1. Introduction
7.2. Behaviour of materials for fuel pin cladding and subassembly duct (wrapper)
7.3. Irradiation performance of oxide fuel elements
7.4. Irradiation performance of metallic fuel elements
7.5. Irradiation performance of advanced fuel elements
7.6. Irradiation performance of absorber elements

8. LMFR engineering

8.1. Sodium pumps
8.2. Thermal-hydraulics of the primary circuit
8.3. Decommissioning of the experimental LMFR Rapsodie

9. LMFR plants design in progress

9.1. BN-1600M reactor plant
9.2. The european fast reactor (EFR)
9.3. BN-800 reactor plant
9.4. BN-600M advanced LMFR
9.5. DFBR-660 reactor plant
9.6. PFBR-500 reactor plant
9.7. ALMR technology development
9.8. BMN-170 modular LMFR
9.9. China experimental fast reactor (CEFR-25)

10. Summary and future trends

INTERNATIONAL WORKING GROUP ON FAST REACTORS (IWGFR)
Technical Committee Meeting - 29th Annual Meeting
of the Working Group on Fast Reactors,
Aktau, Kazakhstan, 14-17 May 1996

List of Proposed Topics for the IWGFR Technical Committee Meetings

No:	*Title*	*Country*	*Proposed for consideration at the IWGFR meeting in:*	*Remarks*
1.	Unusual occurrences in LMFR	India	1991	
2.	Operability with failed fuel pins	Japan	1985	
~~3.~~	~~Reliability data evaluation for fast reactors~~	~~Japan~~	~~1991~~	日本より取下げ
~~4.~~	~~The use of simplified methods to replace full inelastic analysis~~	~~UK~~	~~1991~~	イギリスより取下げ
5.	Core performance monitoring	~~USA~~ India	1992	アメリカ取下げ インド引受け
6.	Experimental fast reactors: Experience, main scientific and engineering results, new projects	China	1993	
7.	The problems of complex mass exchange in LMFR (primary and secondary circuit)	Russia	1994	
8.	Influence of high dose irradiation on fuel and structural materials behaviour	Russia	1995	Proposed to hold in Spring-Summer 1997
9.	Methods and codes for calculations of fuel and absorber pins and subassemblies thermohydraulic parameters in the real reactor conditions	Russia	1996	
10.	Sodium removal and disposal from LMFRs in normal operation and in the framework of decommissioning	France	1996	Proposed to hold in Autumn 1997, in Aix-en-Provence, France

JUSTIFICATIONS FORMULATED FOR PROPOSED MEETINGS

1. Unusual Occurrences in Fast Breeder Reactors, India

Design and operating of Fast Breeder Reactors (FBR) is still in evolution stage and only a small number of FBRs are in operation around the world. Specialists operating these FBRs have gained valuable experience from unusual occurrences that took place in the reactors. These unusual occurrences, lessons learnt and measures to prevent recurrences are not well reported in conferences. All the unusual occurrences and wherever reported are very brief and lack details. Hence there is a need for specialists' operating FBRs to share their knowledge on unusual occurrences. The specialist meeting caters the following requirements.

- It benefits the specialists from one country to learn from other country's experience and avoid recurrences of similar occurrences.
- It also helps to know measures to be adopted in the design and operation of future reactors to avoid similar occurrences.
- To review unusual occurrences in all the FBRs and generate reliability data for components.
- To improve reports on unusual occurrences.

2. Operability with Failed Fuel Pins, Japan

At the Specialists' Meeting on "Fuel failure detection and location in LMFBRs" held at the Kernforschungszentrum Karlsruhe in 1981 methods and experience for the detection and location of failed fuel assemblies were reviewed and discussed.

Since that time an additional experience of the reactor operation with failed fuels were obtained in several countries and some testing was carried out aiming at getting knowledge on the behaviour of failed part of fuel pin as well as of fission gas or fission products in the primary circuit of LMFBRs.

Simulation test for gas leak type failure is planned at JOYO, the Experimental Fast Reactor in Japan in early 1985.

It may be beneficial for each country, if problems related to the experience and testing of operation with failed fuel are discussed.

3. Reliability Data Evaluation for Fast Reactors, Japan

Objectives:

a. To review on reliability data analysis on operation and maintenance from fast reactors and various sodium facilities.
b. To compile reliability/maintainability data of fast reactor specific components (special emphasis on safety-important components).
c. To review abnormal events occurred at fast reactors and to provide useful insights for safety evaluation of fast reactors (Special emphasis on aging effect of components).

4. The Use of Simplified Methods to replace Full Inelastic Analysis, UK

It is not becoming clear that a full inelastic analysis, under the elastic-plastic creep cycling conditions which occur in a fast reactor, is too extensive for use as a design tool. In addition, the results are uncertain

because no satisfactory constitutive equations have yet been developed which describe the behavior of the materials under these conditions.

Conversely, the simple, elastic analysis, consistent with the capability of most design offices, is often too pessimistic. It often predicts that fast reactor components, which are perfectly acceptable, should be rejected.

Many of the pessimism associated with the elastic analysis route for design can be eliminated by simplified methods, such as those based on shakedown and modified BREE diagrams. Such simplified methods are readily applied in the design office and involve no more complexity than elastic analysis. At the same time, they are very little more conservative than the full inelastic analysis.

The growing importance of simplified methods has been recognized by some, but not all, of the countries involved in fast reactor design. The purpose of the meeting would be to exchange information about what is available, and use existing examples to demonstrate the efficiency of the procedure.

5. Core Performance Monitoring, USA

Core performance monitoring during burnup, including reactivity, fuel integrity, thermal margins, sodium purity, flow stability, structural integrity, etc.

Extensive data, not well known, is available from the US, UK, France and CIS, and detailed plans are certainly available from Japan and perhaps others. This data is a key input to design development and optimization of advanced cores. Reliability of data is essential to safety cases. This data is the only confirmation of much nuclear data, thermal hydraulics performance prediction. Data exchange on instrument and measurement techniques, methods validations, and safe performance confirmation at a level short of commercial and patentable design specification would be of great interest and mutual value; confirmatory information spanning a variety of conditions is of immeasurable more value than individual data in isolation.

6. Experimental Fast Reactors: Experience, main Scientific and Engineering Results, New Projects, China

It is obvious that the FBR development, especially for electricity, breeding purposes has been gotten a negative interferences in some developed countries due to they don't urgently need more electricity now and near term, and facing a strong anti-nuclear actions, and also they have enough money to buy or import oil for electricity production if they need. But it is not the case for developing countries and almost impossible to use oil when they would have developed in rather higher economic level because of limited petroleum resources. So naturally it is gradually more interesting in FBRs in developing countries.

Fast reactor technology is much complicated due to using sodium and operating in high temperature. It must be started from basic research and experimental fast reactor for any countries to master the engineering technology of FBRs. It is reasonable to make a good opportunity to exchange experiences on experimental FRs.

Objectives:

a) to review and exchange the national FBR development strategy and projects
b) to exchange operating experiences and main scientific and engineering results on experimental FRs
c) to review experiences on the enhancing experimental FBR's safety
d) to exchange experiences of multi-utilization of experimental FRs.

7. The Problems of Complex Mass Exchange in Fast Reactors Sodium Coolant Circuits (Primary and secondary) and its possible Influence on the Reactor Safety, Russia

In the frames of the meeting can be considered the following items:

- Impurities sources in Fast Reactors sodium coolant (including here accidental ones such as oil ingress, air penetration etc),
- Distribution of impurities between the different zones of reactor sodium circuits and protective gas plenum,
- Accumulation of impurities in the reactor sodium circuits stagnant zones and other preferential sites. Washing out of impurities from the sites of theirs localization and influence of above phenomena on the reactor thermo-hydraulics and other parameters,
- Behaviour of impurities particulate phase in sodium flows,
- Behaviour of the radioactive impurities correlation in their distribution on the reactor vessel and units surfaces.

8. Influence of High Dose Irradiation on Fuel and Structural Materials Behaviour, Russia

Up to now a lot of experience has been accumulated in various countries on high dose irradiation behaviour of fuel pins and subassemblies in FBRs. There are some critical issues topical both for the breeding type core and for the burning type core as well which need to be discussed:

- High dose behaviour of fuel cladding and subassembly wrapper materials (advanced austenitic, ferritic-martensitic, ODS-steels): swelling and irradiation creep variability, thermal creep anisotropy, mechanical behavior after irradiation, etc.
- Chemical interaction between fuel and cladding depending on different factors (particularly, the influence of Pu content).
- Different aspects of fuel-cladding mechanical interactions during transients and SAs shuffling.

Data exchange and discussions on above mentioned topics would be of great interest and mutual value. The meeting may be held in Obninsk, Russia.

Main topics:

1) High dose behaviour of fuel-cladding and subassembly wrapper-materials:
   - advanced austenitic, - ferritic - martensitic, ODS-steel
   - 1.1. Swelling and irradiation creep variability
   - 1.2. Thermal creep anisotropy
   - 1.3. Mechanical properties
2) Chemical interaction between fuel and cladding depending on different factors (particularly, the influence of Pu content)
3) Different aspects of fuel-cladding mechanical interactions during transients and SAs shuffling.

9. Methods and codes for calculations of fuel and absorber pins and assemblies thermohydraulic parameters in the real reactor conditions, Russia

The core is the most loaded component of the reactor reliable performance of which largely determines operation of the plant.

The temperature field of the fuel and control assemblies formed under the effect of many factors (non-uniform power generation, deformation due to swelling and thermal stresses, interchannel and inter-subassembly heat exchange, etc.) primarily affects the core performance.

Investigations on thermal hydraulics of the above core components carried out during last time have allowed an influence of many factors on temperature fields to be taken into account. It should be noted that these investigations have brought thermohydraulic substantiation of the reactor core performance to a higher qualitative level.

The exchange of information on the results gained, on analytical approaches and codes to support projects of experimental and power liquid metal fast reactors, as well as the experience of their use seems to be of interest and it would allow to define problems requiring further research and development.

The purpose of the meeting is to exchange information and to define the lines of investigations in the field of advanced LMFBR core thermal hydraulics.

Objectives

1. Status of thermal hydraulic investigations of the reactor core components (experimental and analytical results, trends of future investigations).
2. Results of experimental and analytical research on the influence of various factors upon temperature behaviour of pins, subassemblies and the reactor core as a whole.
3. Engineering and precise numerical calculation procedures. Codes and closing relations. Verification.
4. Features of core transient modeling.
5. Experience of using the codes for temperature behaviour calculations. The problems of the core parameters optimization from the viewpoint of thermohydraulics.

The meeting may be held in Obninsk, Russia.

10. Sodium removal and disposal from LMFRs in normal operation and in the framework of decommissioning, France

Objective:

There are several Fast Breeder Reactors (FBR) and sodium facilities in operation or under decommissioning, throughout the world. To deal with sodium removal or disposal, several special processes have to be used which are peculiar to this type of reactor.

In normal operation:

- Sodium quality versus dissolved oxygen and hydrogen is controlled using cold traps; when full, they have to be processed;
- Fuel assemblies have to be washed and sometimes major components such as IHX and pumps have to be cleaned and decontaminated for repair. Several processes have been developed with the aim to use there elements again after requalification.

For decommissioning:

- Sodium has to be purified, decontaminated and deactivated for reuse or easy disposal,
- Components have to be cleaned in devoted cleaning pits; the walls of vessels and loops, after sodium

draining, can also be cleaned in situ,
- A decontamination strategy for structures has to be set up,
- Finally, the sodium bulk is stored or destroyed.

Proven processes exist for these various needs, but some have only been demonstrated on structures of limited size and complexity. For these processes, the designer and the operator have to deal with two main aspects;

- Hydrogen and sodium risk assessment during operations,
- Management of sodium wastes (cold traps, cesium traps, NaK rough wastes,...) and effluents (caustic soda, gaseous wastes,...)

The aim of this meeting is to focus on these various issues and encourage technical exchanges between specialists involved in R&D or decommissioning operations.

Main topics:

The following topics will be considered during the meeting:.

a) Purification and decontamination of the sodium
b) Sodium draining and storage
c) Cleaning of components and walls
d) Decontamination of components and walls
e) Sodium disposal
f) Safety (hydrogen, sodium, ...)
g) Management of sodium wastes
h) Management of effluents.

Papers may deal with R&D studies, experimental feedback, design of processes (mock-up tests, modelling,...), but does not include detailed description of dismantling processes as well as topics of sodium waste storage and effluent storage.

Organization:

France proposes to organize such a meeting in Aix-en-Provence, in autumn 1997.

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IWGFRSPM.SAR
rt/24 April 96

Subject : IAEA Technical Committee Meeting on "Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors"

Purpose of the meeting :

To provide a forum to review and discuss the state-of-the-art of the experimental data and relevant computer codes for evaluation of radioactive materials release and sodium fires in Fast Reactors

Date and topics : 11 - 13 November, 1996

- Release from overheated fuel elements
- Transfer to liquid sodium (behavior with gas bubble)
- In-sodium behavior (solubility, deposition and distribution)
- Release from liquid sodium to gas phase
- Behavior in cover gas

14 November, 1996

- Sodium fires and aerosol behavior
- Sodium-concrete reaction and hydrogen burn

Place : O-arai Engineering Center
Power Reactor and Nuclear Fuel Development Corporation
4002 Narita, O-arai, Higashi-Ibaraki,
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INTERNATIONAL ATOMIC ENERGY AGENCY

Technical Committee on Fast Reactor Fuel Development and Performance
9-12 December 1996, Vienna

INFORMATION SHEET

1. Introduction

The Fast Reactor (FR) fuel development program, in the initial years, aimed at developing uranium or uranium-plutonium (MOX) based oxide fuels that would operate safely, breed efficiently and generate electricity economically. They were used as reference fuels in most of the first generation of demonstration, prototype and commercial FRs. Significant experience had been accumulated in France, Germany, India, Japan, Kazakhstan, Russia, UK and USA with burnups up to ~100 MWd/kgHM and radiation damage doses up to ~90-95 dpa. At that time mixed uranium plutonium monocarbide, mononitride and the metallic fuel U-Pu-Zr were recognized as advanced FR fuels because of their higher breeding ratio, shorter doubling time, better thermal conductivity and excellent chemical compatibility with sodium coolant, compared to the oxide fuel. In case of oxide fuels, all aspects of the fuel cycle have been demonstrated on a commercial scale, namely, fabrication, irradiation, reprocessing and refabrication. Experience on advanced FR fuels is more limited.

A lot of effort has also been directed in different countries towards developing suitable fuel cladding materials for FR fuel rods. Irradiation swelling of the cladding material could be a life limiting factor. The reference cladding material for FR fuel is D-9 stainless steel but ferritic stainless steel HT-9 and oxide dispersed stainless steel are emerging as advanced cladding material because they have very low void swelling.

Due to many reasons FR activities were diminished to practically zero level during the last years in Germany, UK and USA. Nevertheless, continuing effort in France, India, Japan and Russia has shown that further increase of burnup up to ~150 MWd/kgHM seems to be possible in the near future. Under these burnups FR fuel cycle will compete with LWR cycle in economics.

The recent trend in FR technology is to burn, but not to breed plutonium. Accordingly, a number of plutonium bearing inert matrix fuels are under development. These are for example cercers: PuN-ZrN, PuC-ZrC, PuO_2-ZrO_2, PuO_2-Al_2O_3, PuO_2-CeO_2, PuO_2-zircon, PuO_2-spinel, etc. and cermets: PuO_2-SS and PuO_2-Zr. Pu content in these fuels could reach 45-50 %. In coming years the thrust areas here would be to develop economically viable fabrication flowsheets for plutonium fuels in inert matrix, evaluation of their out-of-pile and in-pile thermophysical and thermochemical properties, irradiation testing and post irradiation examination.

The Agency's activities in the area of FR fuel development are co-ordinated by the Advisory Group on Advanced Fuel Technology and Performance and the International Working Group on Fast Reactors. The last meeting on this subject was conducted in 1987. To review the progress made since that time the Agency decided to hold this Technical Committee Meeting.

2. Purpose of the Meeting

The purpose of the meeting is:

- to exchange information on technology, design, operation experience and performance (including modelling) at high burnups of currently used or tested FR fuels (rods and their components) accumulated during the last 5-7 years;

- to discuss the issues related to further increase of lifetime fuel characteristics with emphasis on analysis of factors limiting burnup and remedies (with regard to the optimization of fuel rod technology, design and materials);

- to consider and discuss issues related to the development of advanced fuels for FR cores for high burnup operation and/or weapon grade Pu utilization.

3. **List of Topics**

- FR driver and advanced fuel [UO_2, $(U,Pu)O_2$, UC, (U,Pu)C, UN, (U,Pu)N,U-Pu-Zr, etc.] performance under high burnups (on the basis of analysis of operational experience and results of irradiation tests and post-irradiation examination) including impact of transients on life time and operational characteristics of fuel rods, assemblies and their components.

- R&D on FR fuel fabrication technology and rod design (including modelling): current status and R&D to increase fuel life time and reach target burnups.

- R&D activities on fuels with inert matrix to burn weapon grade Pu (cercers and cermets).

- R&D activities on high burnup advanced FR fuels [oxide, carbide, nitride, metallic, etc (e.g. with increased up to 45-50% Pu content)] to provide maximum Pu utilization, including issues concerning the reprocessing of these fuels.

4. **Deadline for Papers and Participation in the Meeting**

The meeting may be attended only upon designation. Participants should complete the attached Form "A" and send it, together with an abstract of approximately 500 words to the appropriate official authority (Ministry of Foreign Affairs or National Atomic Energy Authority) for subsequent transmission to the IAEA not later than 12 October 1996, with a copy to the Scientific Secretary, Mr. V. Onoufriev.

The papers with figures in a form suitable for duplication are to be submitted to the Agency during the meeting (a diskette with the paper in World Perfect format should also be provided).

Prospective participants whose designations have been received by the IAEA will be notified directly.

5. **Expenditures**

In accordance with the established rules, Governments or other national authorities are expected to bear the travel and other costs of designated participants at the Technical Committee.

Please note that compensation is not payable by the Agency for any damage to or loss of the participants' personal property or for illness, injury or death attributable to their relationship with the Agency under the present arrangements. However, while travelling under the authority and at the request of the Agency, the designated expert(s) will be covered under the Agency's insurance policy for inter alia permanent total disability or death up to an amount of US$ 100.000 and medical expenses up to an amount of US $10.000 in case of an accident related to such travel, subject to the terms of the insurance policy.

6. **Working Language**

The working language of the meeting will be English. All communications, abstracts, and papers must be sent to the Agency in English.

7. **Visas**

Designated participants who require a visa to enter Austria should submit the necessary application to the nearest diplomatic or consular representative as soon as possible.

8. Meeting Site

The meeting will be held at the Vienna International Centre, Wagramerstrasse 5, Vienna, Austria. The Centre is near the VIC-Kaisermühlen Metro Station. The meeting will start at 10.00 a.m. on Monday, 9 December 1996.

9. Channels of Communications

The Participation Form, together with two copies of an abstract should be sent to the competent official authority (Ministry of Foreign Affairs or National Atomic Energy Authority) for subsequent transmission to the International Atomic Energy Agency.

10. Secretariat

The Scientific Secretary for the organizational part of the meeting is **Mr. V. Onoufriev** of the Division of Nuclear Fuel Cycle and Waste Management and the Co-Scientific Secretary, Mr. A. Rineiskii of the Division of Nuclear Power.

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STATUS OF FAST REACTOR ACTIVITIES IN BRAZIL[1]

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Abstract

This text describes the present status of fast reactor activities in Brazil, emphasizing the strategies being used to preserve this reactor concept as a viable alternative for future electricity generation in the country. The program is mostly research-oriented and has the objective of establishing a consistent knowledge basis which can serve as a support for the transition to the activities more directly related to design, construction and operation of an experimental fast reactor. Due to the present economic difficulties, the program is still modest but it is gradually growing. A report which has been finalized in December, 1995 and submitted to the authorities indicates the existence of the grounds for enlarging and consolidating the program.

1- Introduction

Brazil is a large country, with a population of a little more than 150 million distributed on its 8,5 million square kilometers. The "per capita" electricity consumption is of the order of 1500 kWh, a value which is low when compared to most advanced countries. Around 95% of the electricity come from hydric resources, most of them located in the southeast region of the country. The estimated hydroelectric potential is 260 GW, from which around 60 GW is being used. A recent study made by the Brazilian Federal Energy Board -ELETROBRÁS-, which takes into consideration different scenarios for economic growth, the resources available today and also the possibility of exploiting new resources, indicates that in the second decade of the next century the hydroelectric potential will be exhausted and, from that time on, the demand will have to be supplied by a growing number of thermoelectric power plants, either conventional or nuclear.

Brazil has presently one reactor (PWR/Westinghouse, 626 MWe) in operation and a second one (PWR/Siemens, 1300 MWe) has recently received authorization for completion, which is planned for 1999. A third reactor (PWR/Siemens, 1300 MWe), originally planned to be assembled in the same site as the other two is still awaiting for a government decision to its construction, amidst discussions related to construction costs and the benefits of nuclear energy. Although the participation of nuclear energy in the electricity grid is still very small (less than 1%) and hydroelectricity is potentially abundant, our view of the future indicates that a fast reactor program should be consolidated in order to provide a consistent basis for discussing the utilization of fast reactors for future electricity generation in the country.

Only recently the Brazilian economy is giving signs of stabilization, after a long period of very high inflation rates. Also, both internal and external national debts resulted, among other things, in a reduction of the investments in nuclear power Research and Development (R&D) activities. Considering the future importance of fast reactors, all efforts have been made in order to keep this option alive, in spite of the tight budgets. As a result, the fast reactor program is small and the plans for its growth (which is to be gradual) envision to avoid an excessive demand on the already scarce financial resources, which could increase the risk of its discontinuation, as it happened with the past fast reactor activities in Brazil.

2- Historical Background

In 1969, the Instituto de Engenharia Nuclear-IEN (Nuclear Engineering Institute) initiated its activities related to fast reactors. As part of a contract with TECHNICATOME (France), a ther-

[1] Presented at the 29th Annual Meeting of the IWGFR (Aktau, Kazakhstan)

mal-fast reactor has been designed, but it was not constructed due to problems with fuel supply. In 1972, a small sodium loop (100 kW) was inaugurated for studying heat transfer and several aspects of sodium technology and is still in operation.

In 1981 a decision to intensify the activities in fast reactors resulted in a cooperation agreement with ENEA (Italy), mainly for training in several areas. Also a contract with ANSALDO/NIRA (Italy) was signed to design and construct three modern sodium loops and auxiliary systems. Due to insufficient funding the loops were never constructed. Discussions are presently taking place with the aim of assembling one of the loops, which is capable of serving as a bed for experiments related to sodium purification and transferring.

In 1985, a new technical cooperation agreement was signed, now with Argentina, for a joint work in the fast reactor area. Many discussions took place, but nothing has been done, mainly due to the fact that discussions were too focused on the design and construction of an experimental fast reactor, which neither Brazil nor Argentina needed at that time.

At the Instituto de Estudos Avancados-IEAv (Advanced Studies Institute), fast reactor activities were initiated in 1979, when a technical group has been created with the objective of doing research on the utilization of nuclear reactors for space applications, with the main focus on neutronics and thermal-hydraulics. That work gave the group a better understanding of the importance of fast reactors for electricity generation. It was then decided to re-direct the work towards investigating different fast reactor core and blanket configurations for efficient utilization of Thorium, due to its large resources in Brazil (estimated: one million tons). Several ideas came up and a good number of technical papers have been published. In 1988, fast reactor activities at IEAv were discontinued not only because of the lack of an well-defined long term program, but also due to the many difficulties in having the participation of other research institutions in Brazil, mostly involved with thermal reactor work.

The main reasons for all these setbacks can be also associated to the following: a) the initiatives were isolated, with no search for institutional cooperation; b) the country has been, for a long time, going through economic problems; c) the public opposition to nuclear energy was high and; d) there has been too much emphasis on the reactor itself, with no emphasis at all on the scientific and technological gains which would result from a fast reactor program with multi-disciplinary characteristics. These factors, and maybe some others, have never stimulated the "decision makers" to take seriously a long term and certainly costly R&D fast reactor program.

In 1992, after many discussions, we succeeded in convincing the authorities that a long term fast reactor program should be maintained, even with modest proportions, if the country wished to keep this reactor concept as an alternative for a future use. A long term R&D project was then established with the objective of having, within 25-30 years, an experimental reactor in operation, in which relevant experiments could be performed. For this program, the cooperation among institutions would be a key factor.

Between 1992 and 1995 efforts have been directed to what we called Feasibility Phase, which had, among others, the objective of submitting to the Brazilian authorities a detailed report composed of six parts: 1) Global View; 2) Reference Design (Primary Circuit) for an Experimental Fast Reactor; 3) Survey of Brazilian Research Institutions; 4) Survey of Brazilian Industrial Park; 5) Survey of Brazilian Universities and; 6) Preliminary Planning for Next Phases. This report, which has been finalized in December, 1995 and is presently being analyzed by the authorities, shows the potential for enlarging the present group of activities and also for including others which have been left aside, mainly due to the shortage of people.

3- Managerial Strategies

As a general rule, countries with fast reactor programs justify them based on the fact that this class of reactors, among other things, utilize more efficiently the uranium resources. For those countries this represents a good argument, because it is supported by the considerable participation of thermal reactors (which lead to a rather inefficient uranium utilization) in their electricity grids.

The arguments related to the generation of less waste (in fast vs. thermal reactors) follow the same line of reasoning. Although these arguments may be consistent for some countries, for Brazil they sound a bit anachronic, due to the present very low participation (less than 1%) of nuclear energy in the electricity grid, with an insignificant demand on our uranium resources, estimated in 200,000 tons.

For some years we have tried to consolidate a fast reactor program using the arguments of Japan, United States, France and others: the main focus was on the reactor and the efforts were almost uniquely devoted to the development of conceptual designs, some of them very interesting but incapable of strongly motivating the Brazilian "decision-makers" and also a large part of the nuclear community. The reason for this is very simple to understand and can be attributed to the generalized perception that the only "product" of the program would be the reactor, whose necessity is, in principle, distant in the future. The belief that "decision-makers" are more efficiently convinced by facts rather than by arguments which certainly will be irrefutable in the future but sound vague and difficult to be evaluated today, have made us switch, in 1992, to a strategy in which the efforts towards the materialization of the reactor would be temporarily de-emphasized in favor of research-oriented activities in areas such as safety analysis, materials, fuel recycling and others, in order to show that a fast reactor program can produce, along its way, important scientific and technological gains for the country. This strategy should be associated to the generation of the maximum number of important and easy to be evaluated results which could be useful not only for the fast reactor program itself but also to other sectors.

The "environment" which prevailed at the beginning of the Feasibility Phase was characterized by tight budgets and also by a strong disbelief from a large portion of the Brazilian nuclear community with respect to big projects. In order to minimize the chances of a new setback, we also decided to establish the following group of strategies:

- **For motivating the internal technical team:** the project is to be conducted in phases of duration no longer than 3-4 years, with the objectives for each phase being defined in such way as to create a feeling that they can be reached, even with the present economic difficulties.

- **For motivating the participation of other research institutions:** the approximation with the institutions must emphasize the several technical challenges in areas such as reactor safety, fuel, materials and so on, and to avoid (at least until the program is entirely consolidated) discussing merits of fast reactors with a community biased towards thermal reactors.

4- Present Status

Research activities are being underway in the following areas:

1. U-Zr alloys are being studied in cooperation with the Instituto de Pesquisas Energéticas e Nucleares-IPEN (Nuclear Research Institute), with the objective of learning how to fabricate and characterize them, in laboratorial scale;
2. Also with the same Institute, metal fuel recycling is being investigated using electrorefining techniques. We expect very soon to have a first sample of actinide electrodeposited;
3. A laboratorial scale direct current electromagnetic pump has been developed and it is being tested at IEAv, using mercury as working fluid. Work is continuing towards its optimization and also towards the development of a small scale alternating current pump;
4. Activities related to HT-9 ferritic steel have been initiated in cooperation with IPEN and with the Instituto de Pesquisas Tecnológicas-IPT (Technological Research Institute). A first sample is presently being characterized;
5. A small thermal-hydraulic nuclear laboratory was set within a collaboration between IEAv and the Instituto Tecnológico de Aeronáutica-ITA (Aeronautic Technological Institute), in order to perform experiments related to natural convection phenomena. ITA is an engineering school, which implies that beyond the technological research aspects, also the educational ones are in-

volved. An experiment for validation of the thermal-hydraulic code NACISS is planned to be initiated in the first semester of 1996;

6. A reference design for the primary circuit of a 60 MWt experimental fast reactor is completed. Some parameters and characteristics have been selected mostly based on the Integral Fast Reactor (IFR) concept. Fuel pin dimensions and other data were used for calculations which lead to a general core configuration. Some calculations were also independently performed at IEN, and the differences in methodologies are being evaluated.

Together with the above activities there are some others related to shielding analysis, actinide burning calculations, electromagnetic pumps and the development of a software for simulating the fast reactor circuits.

5- Final Remarks

The consolidation of the present fast reactor activities in Brazil must contemplate both the achievement of important technical results and also a step-by-step dissemination of a "fast reactor culture", with the objective of increasing the number of people who can evaluate the importance of fast reactors on a more rational basis. In order to establish a fast reactor "culture" we have been, among other things, stimulating the participation in as many technical meetings as possible, even in the ones which are not entirely devoted to nuclear energy. From 1992 to 1995, about 25 articles have been either published or presented in those meetings, with good receptivity.

So far, we have been successful in getting the cooperation of excellent technical groups, from different research institutions, by proposing their involvement in activities which may represent a challenge to their technical expertise. We are aware that there is still a lot to be done and also a lot to be learned, mainly from the countries with more advanced fast reactor programs. In this respect, the recently signed technical cooperation agreement between Brazil and India may represent an important starting point.

It is our feeling that continuation of fast reactor activities in Brazil depends upon the strengthening of the cooperative work already initiated, and also on the generation of observable results which can easily be evaluated by our "decision makers". All efforts have also been made to convince them that an operating reactor is not the only relevant outcome of a fast reactor program. This is the main reason why we have decided to consolidate it on the basis of research-oriented activities rather than on activities strongly directed towards reactor design.

Following the Feasibility Phase, which ended in December, 1995 we intend to increase the level of details of the Reference Design. Considering that an operating experimental reactor is still distant in the future, for all purposes the Reference Design is our "experimental installation" and is to be used as the basis for calculations related to safety analysis, neutronics, thermal-hydraulics and so on. Experimental and theoretical research activities are also to be continued in a few areas, in order to produce results that can used both for the long term objectives and also as assets for larger medium term funding.

Finally, it must be emphasized that even modest or small programs from developing countries may give an important contribution to the fast reactor field, mainly through intellectual work and also through the execution of a few experiments. Our long-term objective is to establish a basic know-how which can be useful for an eventual transition to fast power reactors in Brazil. The interaction with the more advanced programs can serve as a guide, allowing us to reach a degree of maturity which will result in benefits both to Brazil and also to the international fast reactor community.

The Status of Fast Reactor Technology Development in China

For the Twenty-ninth Annual Meeting
of the International Working Group
on Fast Reactors, IAEA

by XU MI
China Institute of Atomic Energy

May 14-17, 1996, Aktau, Kazakhstan

The Status of Fast Reactor Technology Development in China

by XU MI
China Institute of Atomic Energy

Abstract

Since last annual meeting of the IWGFR, Chinese nuclear power plants, especially Qing Shan 300 MWe, have maintained rather good operation records. The Chinese Experimental Fast Reactor (CEFR) with the power 65 MWt, 25MWe is still in the preliminary design stage. The results based on its conceptual design are given in this paper. Few R&D results concerning materials, instrumentation, neutronics study and sodium loops, have been obtained last year due to lack of budgets. As an important progress for the CEFR project, the State Planning Commission has approved this project in the end of the year 1995. It means that after the approval of its feasibility study reports, the CEFR will be becoming a National Annual Project.

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1. Introduction

As a developing country China has a rather fast development in her national economy. During last five years the annual average increasing 12% of the gross national products (GNP) was reached. Meantime China has become one of the countries with a very fast growth of the electricity production. From the years 1991 to 1995 the total electricity power capacity increased 75 GWe, and 210 GWe total capacity was reached in the year 1995[1]. But the electricity production is still far from meeting the demands for development of the national economy and for an improvement in the standard of living.

The nuclear power has started to be developed in China, which will play a role of the supplementary energy supply. Qing Shan 300 MWe PWR nuclear power plant had an annual load factor of as high as 68% for the first year of its Commercial operation in the year 1994 and increased further to 84% in the year 1995. The Guangdong Daya Bay NPP (2×900MWe PWR) is owned by a Joint Venture company setting up by Guangdong and Hong Kong. The commercial operation of the first unit of Daya Bay plant began on Februay 1, 1994 and that of the second unit on May 6, 1994. The load factors of the unit 1 and unit 2 were 45.22% and 77.92% in 1995 respectively. The status of planned nuclear power plants is presented in Table 1. The other additional nuclear plants are being reviewed for some provinces and cities, especially those in the south -east coastal areas such as Shangdong, Jiangsu, Fujing etc,.

Table 1 Chinese NPP Planned

Name	Power (MWe)	Reactor	First concrete	Connected to grid	Commercial operation
Qing Shan phase 2	No. 1 600	PWR	1996-06-01	2002-06	
	No. 2 600	PWR	1997-04-01		
Guangdong Ling Ao phase 1	No. 1 984	PWR	1997-05-15	2002-04-15	2002-07-15
	No. 2 984	PWR	1998-01-15	2002-12-15	2003-05-15
Qing Shan Phase 3	No. 1 700	HWR			
	No. 2 700	HWR			
Liaoning	No. 1 1000	PWR			
	No. 2 1000	PWR			

Concerning the fast reactor development in China, as its first step, the Chinese Experimental Fast Reactor (CEFR) with the power 65MWt matched with a 25 MWe turbine-generator is still in the preliminary design stage. The feasibility study reports have been prepared and submitted to the authorities.

2. Design of the CEFR[2]

The conceptual design of the CEFR was started in 1990 and completed in the end of 1993 including the confirmation and optimization of some important design selections and characteristics. Having spent almost whole 1994 for its preparation, the CEFR preliminary design has been started in the early of 1995. Based on the conceptual design, some important design results are as following.

2.1 Reactor Core

As shown in Fig 1, the CEFR reactor core is composed of 82 fuel subassmblies with $(Pu, U)O_2$ fuel, 8 control and safety subassemblies using B_4C enriched with B-10, 158 blanket subassemblies, 378 stainless steel reflector subassemblies and 54 primary storage positions for spent fuel subassemblies. The centre to centre distance of neighbouring subassemblies is 61.5 mm.

In each fuel subassembly there are 61 fuel pins of 6 mm diameter in triangular arrangement, using wire wrapped positionning to keep 7 mm lattice pitch of fuel pin. Each fuel subassembly composed of a handling head matched with fuel handling devices, hexagonal tube in which the fuel pins are included and a cylindrical foot playing the roles of positionning and introducing of sodium coolant. The structure drawing of the fuel subassembly is given in Fig. 2

The recator is equiped with two independent shut down systems, the first of which is composed of 4 compensation and 2 regulation subassemblies with a rapid drop down time less than 1.5 seconds, the second of which includes 2 safety subassemblies with a rapid drop down time less than 0.7 seconds.

2.2 Reactor Block

The CEFR reactor block as shown in Fig 3 is composed of a reactor top deck in which there are double rotating plugs and some components belong to the primary circuit, main and guard vessels and internal structures on which the reactor core and its grid plenum are supported. A small gap

between the main vessel and guard vessel is selected to ensure that the reactor core will be still immersed in the sodium and the natural convection of primary sodium will be still kept when a leakage on the main vessel and the lowest primary sodium level kept by the guard vessel have happened. For this concept, as a penalty, there is no any possibility to realize inspections with an inspector movable on the surface of the main vessel. The double rotating plugs are traditional. A straight moving fuel handling machine is eccentrically mounted on the small plug. With the double rotating plugs and the fuel handling machine, any movement of subassembly could be realized between any core subassembly positions or between core subassembly position and transfer pot which is connected with a transfer elevator by which the core subassemblies could one by one enter into and go out the reactor vessel.

2.3 Heat Transfer Systems

The sodium pool type has been selected for the CEFR primary circuit arrangement. The primary circuit includes two loops, each one has one primary pump and two intermediate heat exchangers. In the normal operation, the 360℃ sodium from cold pool is pumped into the grid plenum through high pressure tube which has a double wall structure. The sodium temperature will be 530℃ when it leaves from the reactor core and then mixed with sodium of hot pool, it enters into the intermediate heat exchangers with the temperature about 514℃, and leaves from intermediate heat exchangers to the cold pool.

The secondary circuit is also composed of two loops, each one includes one secondary pump, one steam evaporator, one steam-water seperator, one super-heater and valves. The steam generators will produce 93.5 t/h dry steam with the temperature 480℃ and the pressure 10 MPa for a turbine-generator. The heat in the condenser will dissipate to the air by a cooling tower.

Two independent passive residual heat removal systems are equiped in the reactor, each one includes one independent Na-Na heat exchanger immersed in the primary sodium and two sodium-air cooler with a high stack.

2.4 Main Design Parameters

Parameter	Unit	Value
Thermal Power	MW	65.5
Electric Power, net	MW	20

Reactor Core		
Height	cm	50
Diameter Equivalent	cm	58.5
Fuel		$(Pu, U)O_2$
Pu, total	kg	121.6
Pu-239	kg	93.2
U-239 (enrichment)	kg	97.6 (30%)
Linear Power max.	W/cm	430
Neutron Flux	$n/cm^2 \cdot s$	2.97×10^{15}
Burn-up, target max.	MWd/t	100000
Burn-up, first load max.	MWd/t	50000
Inlet Temp of the Core	℃	360
Outlet Temp of the Core	℃	530
Diameter of Main Vessel	m	8.0
Primary Circuit		
Number of Loops		2
Quantity of Sodium	t	300
Flow Rate, total	t/h	1400
Number of IHX per Loop		2
Secondary Circuit		
Number of Loops		2
Quantity of Sodium	t	48.2
Flow Rate	t/h	986.4
Tertiary Circuit		
Steam Temperature	℃	480
Steam Pressure	MPa	10
Flow Rate	t/h	93.5
Plant Life	a	30

2.5 DBA and BDBA to be Analysed

Following design basis accidents (DBA) and beyond design basis accidents (BDBA) are planncd to be analysed in the preliminary design stage.

For DBA:

(1) leakage on a tube of auxiliary systems of the primary circuit, out of main vessel;

(2) leakage on the cover gas system of primary circuit;

(3) blockage of flow channel in the fuel subassembly;

(4) the hottest fuel subassembly moved from sodium to transfer cell with Ar gas environment as a handling mistake;

(5) leakage on the decay tank of radioactive Ar gas.

For BDBA:

(1) loss of both grid power and emergency power supply, and in the same time loss of the effectiveness of all control and shutdown systems;

(2) guillotine rupture of a tube of auxiliary systems of primary circuit, out of main vessel;

(3) loss of both grid power and emergency power supply, and in the same time loss of effectiveness of the residual heat removal systems;

(4) unanticipated withdrawal of one regulation subassembly, and in the same time loss of effectiveness of all control and shut-down systems;

(5) unanticipated close down of two check valves at outlet of main pumps on two primary loops;

(6) leakage in the same time, of both main and guard vessels

3. Research and Development

3.1 Materials

Some researches have been aimed at materials for steam generators, such as 2.25Cr-1 Mo, 9Cr-1 Mo, Incoloy -800 and 2.25Cr-1 Mo-Nb which are all domestic products. Tests are conducted in the sodium convection corrosion loop with high temperature. The 321 stainless steel has been used as structure material for this test loop. The test parameters are as following:

test temperature	550℃
diffusion cold trap temperature	120℃
sodium flow rate	6 cm/s
oxygen in sodium	~15PPm
carbon in sodium	~3PPm

After exposure for 1500 h to flowing sodium at 550 ℃, weight measurements, microstructural examinations and microhardness measurements were carried out. Electron microprobe traces were made on the specimens to establish the chromium, molybdenum and manganese profiles. Decarburization was found in the low chromium 2.25 Cr-1 Mo steel and carburization was found in the high chromium 9 Cr-1 Mo The Nb stabilized 2. 25 Cr- 1 Mo- Nb steel was more resistant to decarburization than 2.25 Cr-1 Mo steel. The Nb/C ratio had a significant effect on the carbon stabilization. The extent of carbon transfer of the Incoloy-800 was lower than that of the 9 Cr-1 Mo. These phenomen were illustrated by the curves of the surface microhardness for different materials (Fig 4). In addition, the changes in the carbides distribution were evident for Cr -Mo steels because of carbon tranfer. [3]

3.2 Instrumentation

In the operation of the fast reactor steam generators, for safety reasons, it's important to diagnose immediately the leak of water or steam

if happened. Among various diagnosis techniques developed recently, the detection of an increase of hydrogen contents in sodium and in cover gas, if a sodium-water reaction has happened, is a unique on-line diagnosis techniquc with very high sensibility. A micro-hydrogen detection system has been developed by The Liquid Metal Technology Laboratory of Tsinghua University, for which a sodium loop and a calibration system have been established at same time. The test system sketch is shown in Fig 5. The experiments in the sodium test loop includes static state calibration and dynamic state calibration by changing cold trap's temperature and injecting hydrogen. The results present that the hydrogen detection system in sodium has a superior sensibility over 5×10^{-9} which is good enough for the requirement of the FBR steam generator leak detection and its performance data and operation experience are provided for designing and operating the hydrogen detection system used for the FBR steam generators. [4], [5]

3.3 Neutronics

The Zirconium Hydride Solid Zero Power Facility has been established using hydrogen containing solid materials as moderator, plexiglass as reflector and U_3O_8 with enrichment of 20% as fuel. Since its initial criticality the physical characteristics and safety features were measured. The reactor has sound reactivity stability and high sensibility, better than other water lattice facilities. It was successfully used for the personel training and for proving and testing the reactor control instruments and experiment devices. [6]

3.4 Sodium test loops

Two sodium test loops ESPESSO and CEDI, 110m³/h and 320m³/h respectively, ceded from the ENEA, Italy, are under reconstruction. It is planned to use ESPESSO for the endurance test and thermo-shock test of the CEFR's core subassembly' models. CEDI will be used for proving of some sodiun components if neccessary.

4. CEFR Engineering Preparation

The Feasibility Study Reports of the CEFR have been prepared, including General Report, CEFR Design Report, Site Selection, Safety Evaluation, Environment Impact Evaluation and Capital Cost Estimation, in which the reports concerning design, safety and environment impact have been approved by the China National Nuclear Corporation (CNNC), the Safety and Protection Bureau of CNNC and the China National Environment Protection Administration (CNEPA) respectively from the year 1994 to January, 1996.

The detailed exploration of the CEFR's site has been completed in the year 1995. It is pointed that the basic intensity of earthquake in this region is 7 degree in 12 degrees devided. The design basis vibratory ground motion SL-1 is 107 gal and limited safety vibratory ground motion SL -2 is 214 gal. The floor response spectrum and time range curves have been obtained.

In the end of 1995 the CEFR project has been approved by the State Planning Commission (SPC). It means that if the Feasibility Study Reports are all passed by the Authorities, the CEFR project will be arranged by the SPC as an annual capital construction project. So, now the approval of the Feasibility Stacdy Reports of the CEFR is becoming the most important issue.

1	Fuel Subassembly	82
·	Blanket Subassembly	158
2	Reflector Subassembly	378
	Spent Fuel Storage	54
	Safety Rod	2
	Compensation Rod	4
	Regulation Rod	2
S	Neutron Source	2
E	Irradiation Facility	1
3	Irradiation Loop	1

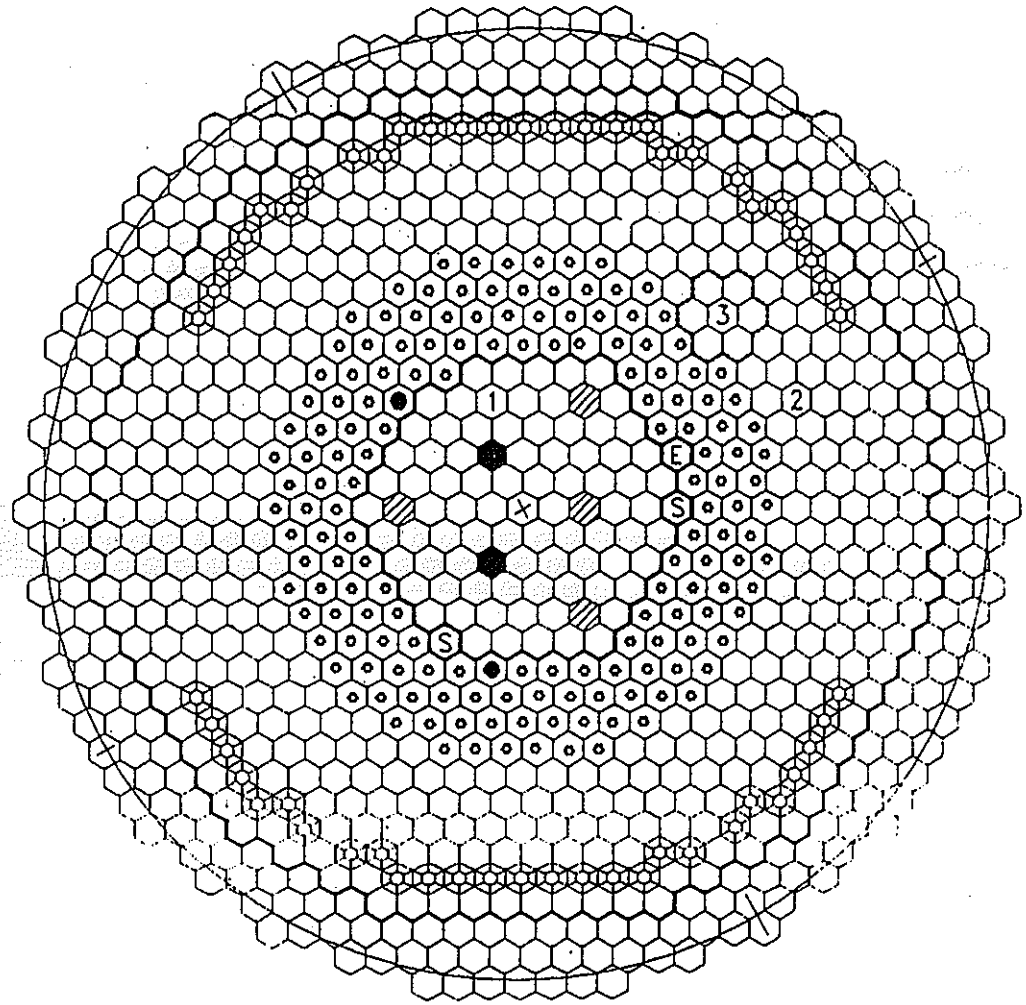
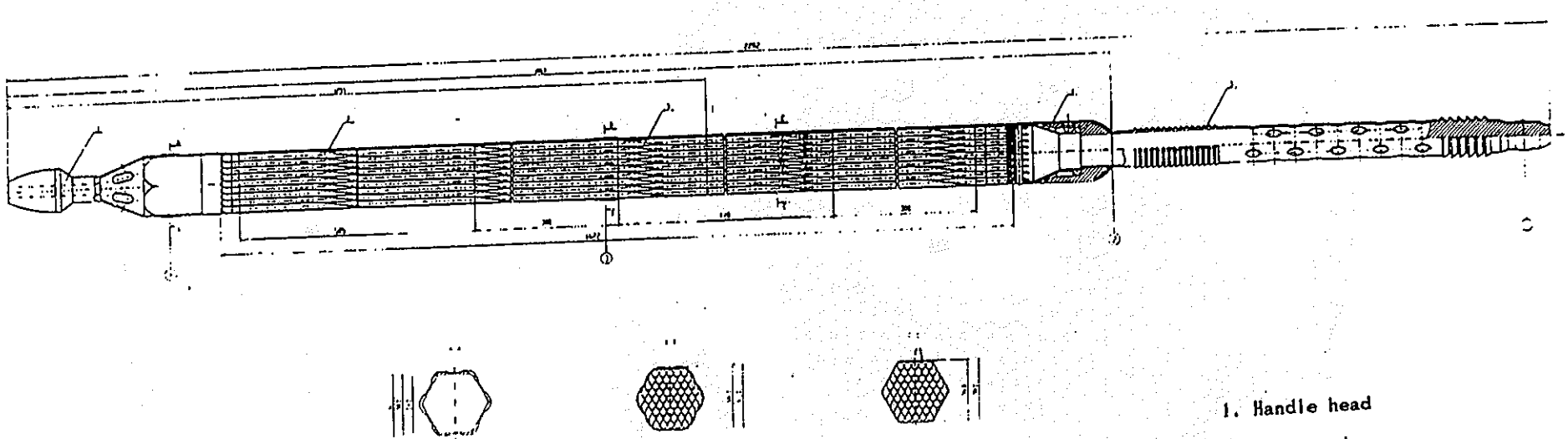


Fig 1 CEFR Core



1. Handle head
2. Hexagonal
3. Fuel Bundle
4. Grid
5. Foot

Fig 2 Fuel Subassembly

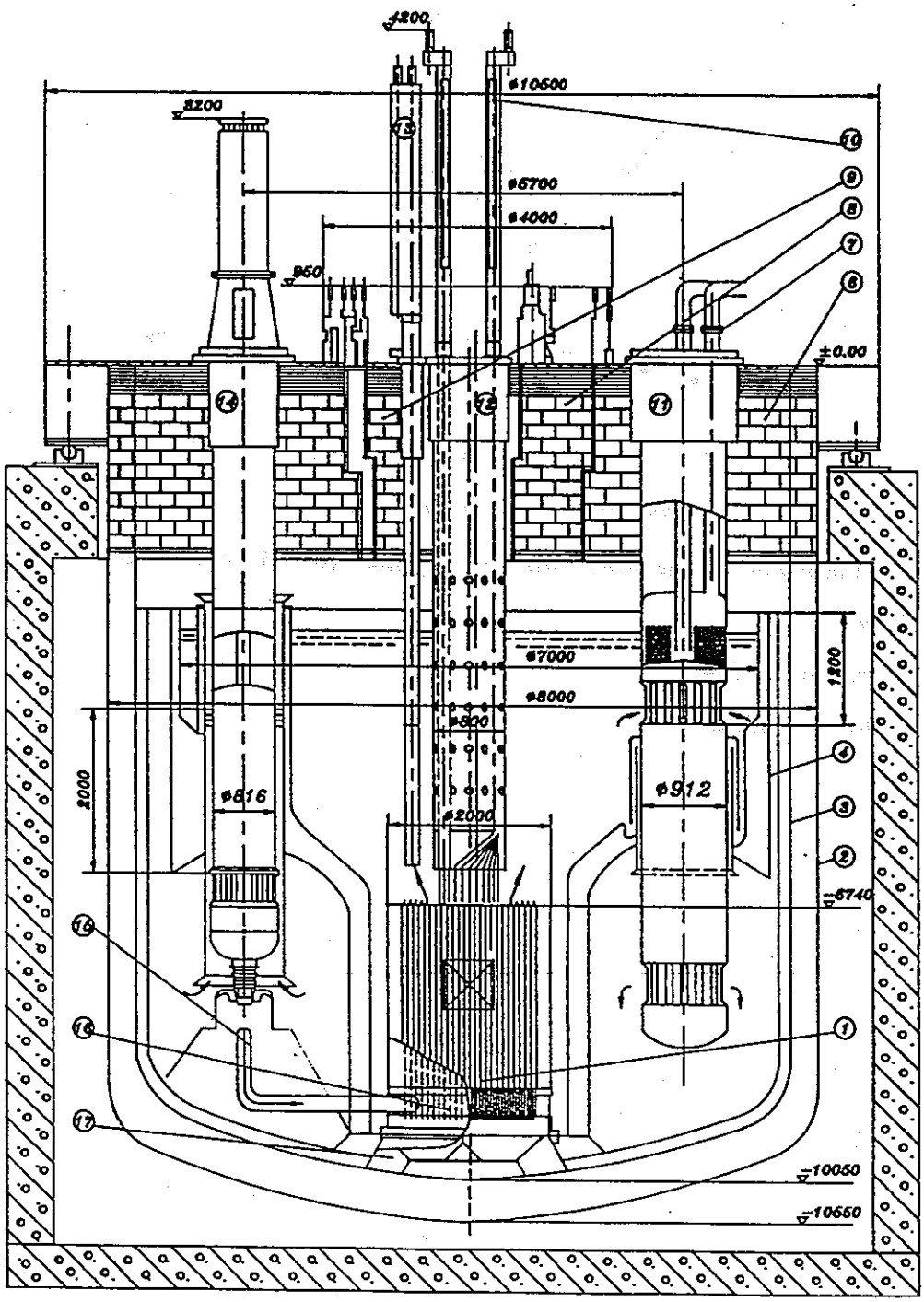


Fig. 3 *Reactor block*

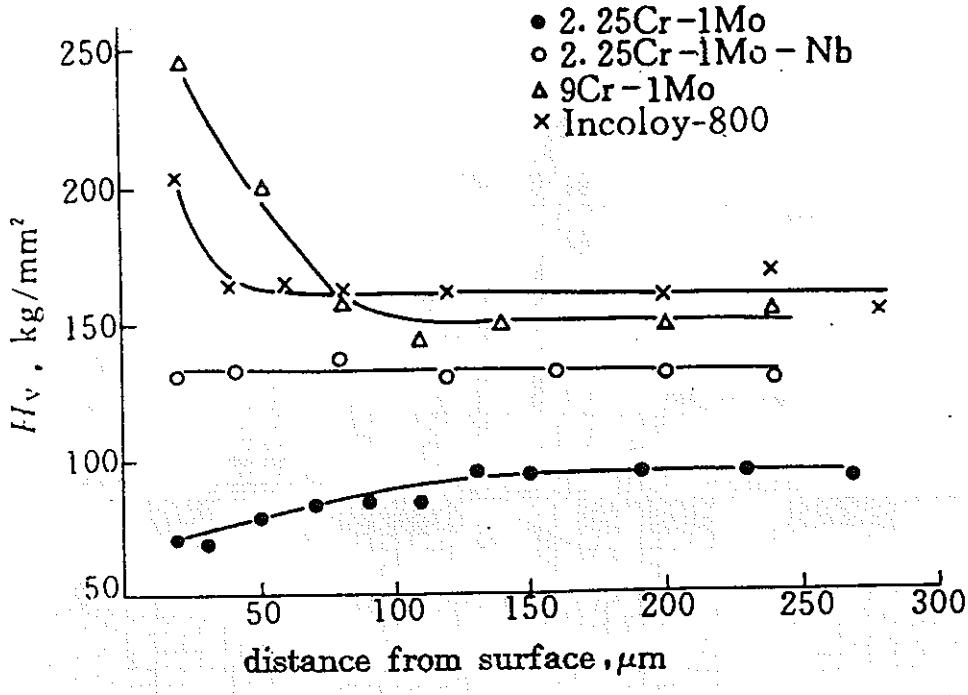


Fig 4 Changes of surface microhardness on different specimens

medium: Na

temperature: 500℃

duration: 1500h

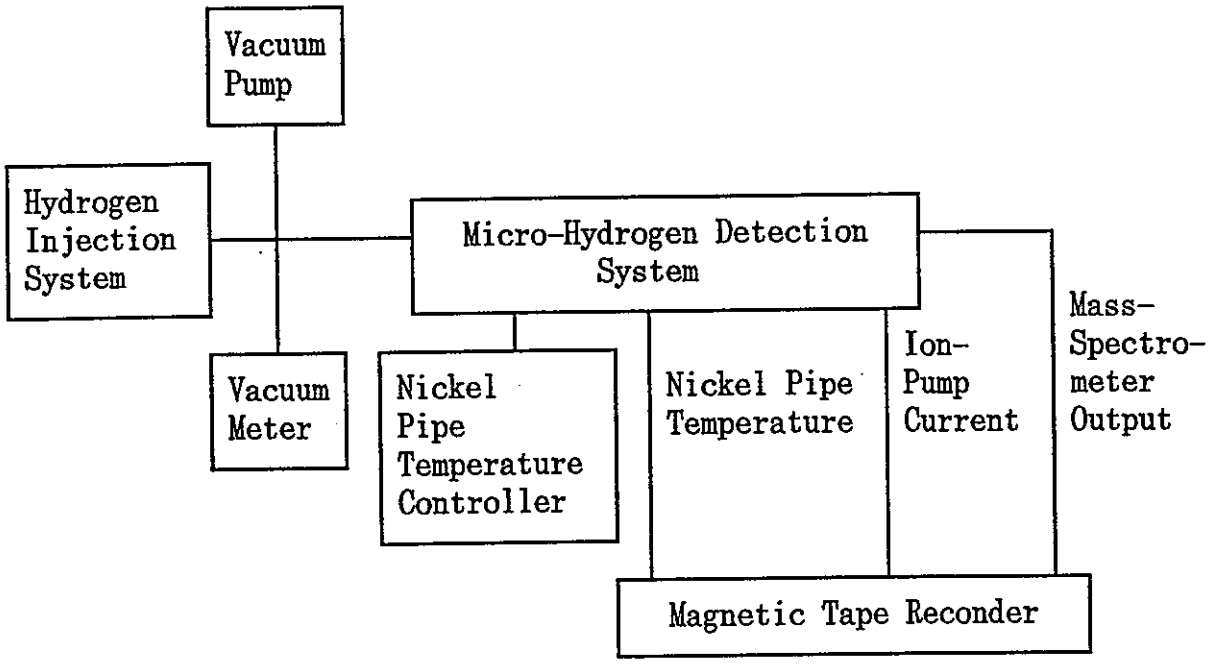


Fig 5 Test System Sketch for Micro-Hydregen Measurement

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The Status of Fast Reactor Technology Development in China

1. Introduction

1991-1995:

The annual average increasing of GNP: 12%.

The total electricity power capacity increasing 75GWe, 15GWe each year.

1995 electricity capacity: 210GWe

Load Factor

NPP		Period	Load Facter
Qingshan-300MWe		1994	68%
		1995	84%
Daya Bay	No. 1	1995	45. 22%
2 × 900MWe	No. 2	1995	77. 92%

NPP Planned

Name	Power (MWe)	Reactor	First concrete	Connected to grid	Commercial operation
Qing Shan phase 2	No. 1 600	PWR	1996-06-01	2002-06	
	No. 2 600	PWR	1997-04-01		
Guangdong Ling Ao phase 1	No. 1 984	PWR	1997-05-15	2002-04-15	2002-07-15
	No. 2 984	PWR	1998-01-15	2002-12-15	2003-05-15
		FRAMATOM 합작			
Qing Shan Phase 3	No. 1 700	HWR			
	No. 2 700	HWR			
		CANDU			
Liaoning	No. 1 1000	PWR			
	No. 2 1000	PWR			
		RUSSIN			

2. CEFR Design

CEFR design Schedule

1988–1990	Design codes preparation Technical Selections
1990–1992. 7	Conceptual design
1992. 7–1993. 12	Confirmation and optimization
1994	Preliminaly design preparation
1995. 2–	Preliminary design

Main Design Parameters

Parameter	Unit	Value
Thermal Power	MW	65.5
Electric Power, net	MW	20
Reactor Core		
Height	cm	50
Diameter Equivalent	cm	58.5
Fuel		$(Pu, U)O_2$
Pu, total	kg	121.6
Pu-239	kg	93.2
U-239 (enrichment)	kg	97.6 (30%)
Linear Power max.	W/cm	430
Neutron Flux	$n/cm^2 \cdot s$	2.97×10^{15}
Burn-up, target max.	MWd/t	100000
Burn-up, first load max.	MWd/t	50000
Inlet Temp of the Core	℃	360
Outlet Temp of the Core	℃	530
Diameter of Main Vessel	m	8.0
Primary Circuit		
Number of Loops		2
Quantity of Sodium	t	300
Flow Rate, total	t/h	1400
Number of IHX per Loop		2
Secondary Circuit		
Number of Loops		2
Quantity of Sodium	t	48.2
Flow Rate	t/h	986.4
Tertiary Circuit		
Steam Temperature	℃	480
Steam Pressure	MPa	10
Flow Rate	t/h	93.5
Plant Life	a	30

DBA

1. leakage on primary purification tube
2. leakage of cover gas
3. fuel subassembly blockage
4. hottest fuel subassembly moved to Ar environment
5. leakage on the decay tank

BDBA

	Power supply	shutdom system	DHRS
1.	failure	failure	O.K.
2.	failure	O.K.	failure

3. guillotine rupture of primary purification tube
4. a regulation rod withdrawl and shutdown system failure
5. two check valves closed
6. leakage on both main and guard vessels

3. R&D

3.1 Materials

2.25 Cr-1Mo
9 Cr-1Mo
Incoloy-800
2.25 Cr-1Mo-Nb

have been tested on the corrosion sodium loop:

test temp.	550℃
diffusion cold trap temp.	120℃
sodium flow rate	6cm/s

O: 15PPm, C: 3PPm

3.2 Instramentation

a Diffusion H-meter has been developed by the Liquid Metal Technology Lab. of Tsinghua Universitgy. 5×10^{-9}, a superior sensibility has been obtaind.

4. CEFR feasibility study

4.1 Reports to be approved

General Report
CEFR Design Report
Site Selection,
Safety Evaluation,
Environment impact Evaluation,
Capital cost Estimation

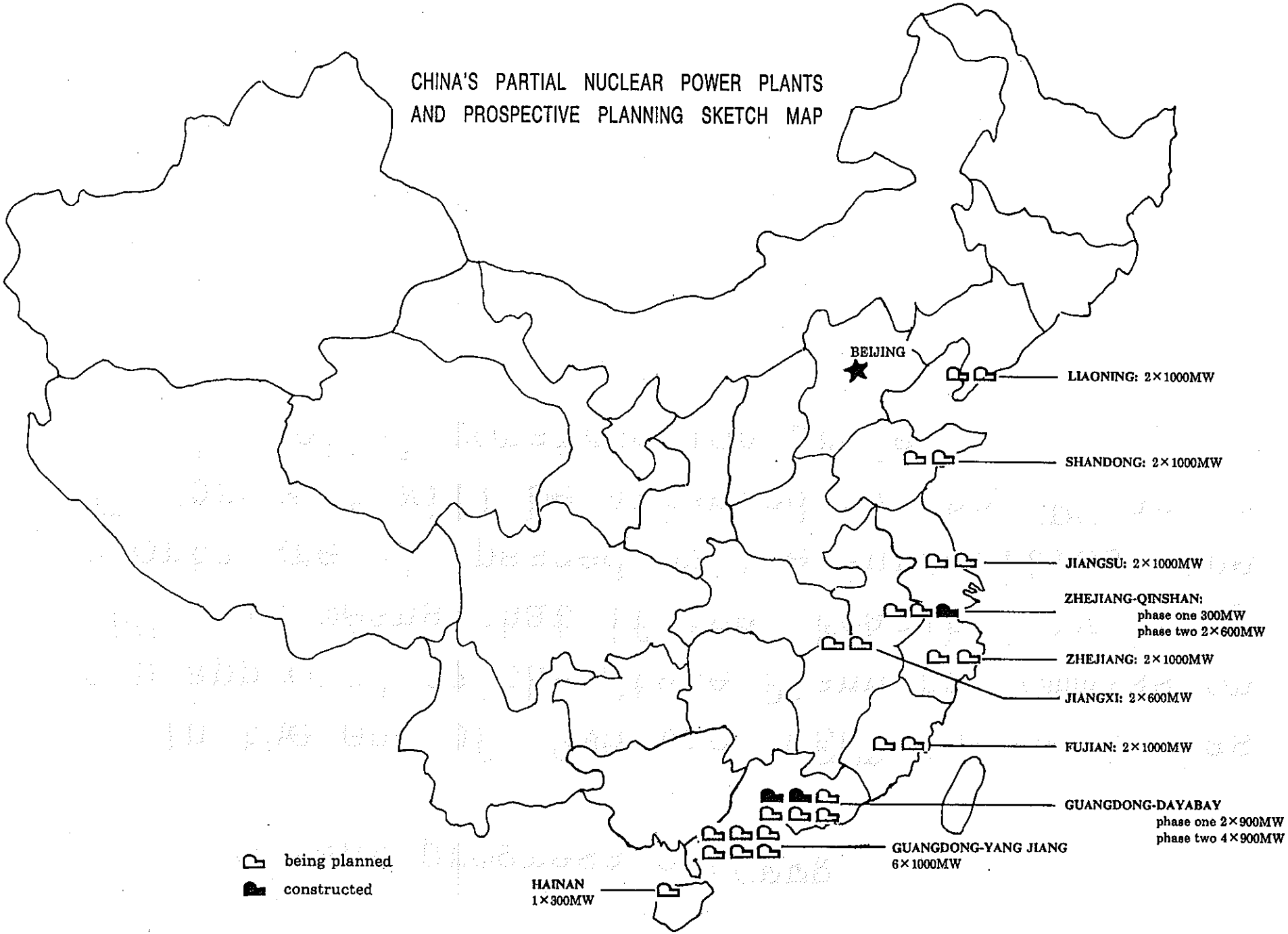
have been prepared ant submitted to the Authorities.

4.2 Site exploration

The design basis vibratory ground motion
SL-1: 107gal
The limited safety Vibratory ground motion
SL-2: 214gal

4.3 Important progress of CEFR

In the end of 1995 the CEFR project has been approved by the State Planning Commission (SPC). It means that if the Feasibility Study Reports are all passed by the Authorities, the CEFR project will be arranged by the SPC as an annual capital construction project.


CHINA'S PARTIAL NUCLEAR POWER PLANTS
AND PROSPECTIVE PLANNING SKETCH MAP
BEIJING
LIAONING: 2×1000MW
SHANDONG: 2×1000MW
JIANGSU: 2×1000MW
ZHEJIANG-QINSHAN:
phase one 300MW
phase two 2×600MW
ZHEJIANG: 2×1000MW
JIANGXI: 2×600MW
FUJIAN: 2×1000MW
GUANGDONG-DAYABAY
phase one 2×900MW
phase two 4×900MW
GUANGDONG-YANG JIANG
6×1000MW
HAINAN
1×300MW
being planned
constructed

FAST REACTOR DEVELOPMENT PROGRAMME IN FRANCE DURING 1995*

prepared by C. LE RIGOLEUR

COMMISSARIAT A L'ENERGIE ATOMIQUE
DEPARTEMENT D'ETUDES DES REACTEURS
CENTRE D'ETUDES DE CADARACHE

With contributions from :

- "EFRUG Board" for the "European Reactor Development Strategy - Position of EFRUG" paragraph
- *B. MESNAGE* (CEA), for the SUPER-PHENIX paragraph,
- *P. ANZIEU* (CEA), *R. DEL BECARRO* (EDF) for the "Knowledge Acquisition Programme" paragraph,
- *J. ROUAULT* (CEA) for the CAPRA paragraph.
- *G. HUBERT* (EFRUG/EDF-SEPTEN), *C.H. MITCHELL* (EFR Associates) for the EFR paragraph.
- *R. DEL NEGRO* (CEA), *P. MARMONIER* (CEA) for the "Complementary Information about the March 31, 1994 accident at CADARACHE".

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ABSTRACT

In 1995, the total amount of electricity produced in France was 471 TWh, out of which 358.2 TWh (76 %) were produced by nuclear power plants, 36.9 TWh (7.8 %) by conventional thermal plants, and 75.5 TWh (16 %) by hydraulic plants. The net electrical power consumption was 368.7 TWh.

At the end of 1995, "Electricité de France" had 54 PWR units in operation. The availability factor for these units was maintained at 81 %. 1995 was marked by a decrease of unexpected shutdowns (1.8 % in 1995 instead of 2.2 % in 1994), a new reduction in programmed shutdown periods, and a good safety level was maintained.

In the field of Fast Reactors, the main events of 1995 were the following.

At the end of December 1994, the PHENIX reactor was authorized to perform its 49th cycle at 350 MW th (143 MWe). This 49 th cycle was completed without any significant problems on April 7, 1995. During the remainder of the year, the reactor had been shut down in order to carry out several tasks within the scope of the ten-year extension of the PHENIX reactor's lifetime.

Concerning the CREYS-MALVILLE plant (SUPER-PHENIX) the first part of the year was devoted to repairing argon leak of one of the IHX. Authorization to restart the reactor was given on August 22. The end of the year was beset by a number of minor incidents. The reactor was restarted at the end of 1995 and reactor power was increased by successive steps (30 % Pn (Nominal Power) up to February 6 1996 ; followed by 50 %...).

The "Décret d'Autorisation de Création" stipulates that because of its prototype character, SUPER PHENIX will have to be operated under conditions explicitly giving priority to safety and knowledge acquisition, with an objective of research and demonstration.

In this context, the so-called "knowledge acquisition" programme designed to prove the capacity of a large FBR to produce electricity on an industrial scale, to test the consumption of plutonium and minor actinides in a large fast reactor, as well as to provide information on the technology of sodium-cooled fast reactors.

The CAPRA programme, initiated in February 1993, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high a rate as possible. The first two-year phase of the CAPRA project studies (1993-1994) was completed. Complementary studies were performed in 1995 and will be presented.

The European Fast Reactor was launched in 1988 and has reached an important stage with the completion of the Concept Validation Phase in 1993. The status of the project will be presented as well as its evolution.

The four main objectives can be summarized as follows and will be presented :

- Good operation of PHENIX and SUPER PHENIX.
- The "knowledge acquisition" programme.
- The CAPRA programme.
- The EFR programme.

The R and D in support of these objectives is presented is also presented.

This R and D must also be placed within the context of the international collaboration.

The above general objectives must be placed in the prospect of a decision to construct a fast reactor, to be made around 2005.

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1 - GENERAL SITUATION

In 1995, the total amount of electricity produced in France was 471 TWh, out of which 358.2 TWh (76 %) were produced by nuclear power plants, 36.9 TWh (7.8 %) by conventional thermal plants, and 75.5 TWh (16 %) by hydraulic plants.

70 TWh were exported, 4 TWh were used for pumping, while line losses came to 28.3 TWh. Consequently, the net electrical power consumption in France was 368.7 TWh, corresponding to an increase of 2.8 % in comparison to the 1994 consumption.

At the end of 1995, "Electricité de France" had 54 PWR units (900 MWe and 1300 MWe) in operation.

The availability factor for these units was maintained at 81 % (81.3 % in 1994) : 79.8 % for the 1300 MWe units and 82 % for the 900 MWe units.

1995 was marked by a decrease of unexpected shutdowns (1.8 % in 1995 instead of 2.2 % in 1994) and a new reduction in programmed shutdown periods.

In 1995, a good safety level was maintained. A level 1 classification was given to 78 events. There were no higher level events.

The problems regarding the penetrations of the vessel heads and steam generator tubes have now been overcome.

In the field of Fast Reactors, the main events of 1995 were the following.

At the end of December 1994, the PHENIX reactor was authorized to perform its 49th cycle at 350 MW th (143 MWe). This 49 th cycle was completed without any significant problems on April 7, 1995. During the remainder of the year, the reactor had been shut down in order to carry out several tasks within the scope of the ten-year extension of the PHENIX reactor's lifetime (see § 3.1).

Concerning the CREYS-MALVILLE plant (SUPER-PHENIX) the first part of the year was devoted to repairing argon leak of one of the IHX. Authorization to restart the reactor was given on August 22. The end of the year was beset by a number of minor incidents (see § 3.2). The reactor was restarted at the end of 1995 and reactor power was increased by successive steps (30 % Pn (Nominal Power) up to February 6 1996 ; followed by 50 %...).

The "Décret d'Autorisation de Création" stipulates that because of its prototype character, SUPER PHENIX will have to be operated under conditions explicitly giving priority to safety and knowledge acquisition, with an objective of research and demonstration.

In this context, the so-called "knowledge acquisition" programme (§ 4), proposed by NERSA, "Electricité de France" and "Commissariat à l'Energie Atomique" is designed to prove the capacity of a large FBR to produce electricity on an industrial scale, to test the consumption of plutonium and minor actinides in a large fast reactor, as well as to provide information on the technology of sodium-cooled fast reactors.

The CAPRA programme, initiated in February 1993, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high a rate as possible. The first two-year phase of the CAPRA project studies (1993-1994) was completed. Complementary studies were performed in 1995 and will be presented in § 5.

The European Fast Reactor was launched in 1988 and has reached an important stage with the completion of the Concept Validation Phase in 1993. The status of the project will be presented in § 6 as well as its evolution.

The four main objectives can be summarized as follows :

- Good operation of PHENIX (§ 3.1) and SUPER PHENIX (§ 3.2).
- The "knowledge acquisition" programme (§ 4)
- The CAPRA programme (§ 5)
- The EFR programme (§ 6)

The R and D in support of these objectives is presented in § 7.

This R and D must also be placed within the context of the international collaboration. Specific agreements exist between Europe and Japan, between France and Russia (CEA-MINTOM agreement in which Germany and the United Kingdom are "Associated Partners"), and between France and China.

The above general objectives must be placed in the prospect of a decision to construct a fast reactor, to be made around 2005 (see B. BARRE'S presentation at EUROFORUM, Paris, January 25-26 th, 1995). This position is in agreement with the European Fast Reactor Utilities Group, "EFRUG", which will be detailed in the next paragraph (§ 2).

2. - EUROPEAN FAST REACTOR DEVELOPMENT STRATEGY-POSITION OF EFRUG

EFRUG Board.

European Fast Reactor Development Strategy
EFRUG position

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European Fast Reactor Development Strategy
EFRUG position

EXECUTIVE SUMMARY

In collaboration with its industrial and R&D partners, EFRUG has reviewed the forward strategy and work programme on Fast Reactor development. The review takes into account the decision to restart Creys Malville and the first conclusions of EFR and CAPRA programmes while recognising the current lessening of the prospects and resources for new Fast Reactor projects.

The approach considers nuclear power development worldwide, trends for the nuclear fuel market in the short and medium term, potential missions, now being evaluated, for Fast Reactors in the back-end of the fuel cycle, and experiences from previous projects and from operating reactors, including successes and difficulties met in the licensing of Creys Malville.

The basic incentives for developing fast reactor technology are unchanged. Although the timescales have extended, the final purpose remains the same : by a gradual introduction of breeder reactors to allow satisfaction of the future energy needs of Europe and to ensure its energy independence(1).

The fundamental ability of an established programme of Fast Reactors to manage plutonium resource in the nuclear fuel cycle has always been recognised. More recently, some possible new missions are proposed in the field of nuclear waste burning and management of plutonium steadily arising from operation of current nuclear facilities. These new missions are already under consideration in a number of programmes, such as CAPRA, the conclusions of which are not expected for several years.

A new EFRUG strategy must therefore consider the following established trends :

- the EFR collaboration has resulted in a highly developed and potentially competitive design which meets current licensing requirements and therefore the collaboration in Europe should continue, aiming at the eventual construction and operation of a demonstration commercial size reactor,
- from present national energy policies, no decision on a demonstration Fast Reactor is likely before 2005-2010 in Western Europe. Even so, provision should be allowed for changes in the conditions governing nuclear energy development such that, if there were renewed interest in Fast Reactors, it would be possible at any time to launch a project. In the medium term (2010-2020) it could be possible to anticipate their introduction into the nuclear generating capacity,
- the choice of mission for the future demonstration plant is undecided, ranging from burning to breeding or a combination of both. Breeding is essential for the long term future of nuclear energy.

While the long term objective remains unchanged, in the current unplanned conditions governing energy markets it is impossible to anticipate a unique way forward for the development of Fast Reactors in the short or medium term. A range of possibilities and timescale must be considered rather than a straightforward

(1) This is also the objective of the Japanese fast reactor programme which aims to limit the total Japanese reliance on uranium to 10% of the known world resources.

development programme based on a single strategic option. This is a major change compared to the previous EFRUG programme based on construction schedules.

The uncertainties concerning the short to medium term role of Fast Reactors in the fuel cycle (as breeders or Pu burners) require the evaluation of variations in the basic reactor design. Fortunately, the differences are very limited because :

- much of the reactor design is common to the different types,
- the principal interests of the Utilities are in the economic generation of electricity from a safe and robust design. Therefore, whichever mission is chosen, the objectives remain to minimise the production costs of electricity and to ensure that the plant can be operated reliably for 40 to 60 years;
- all technological developments must remain compatible with the long term breeding mission.

Therefore EFRUG's proposed approach to sustaining the development of Fast Reactors is based on a "Breeders and Burners" work programme which aims to generate confidence in the Fast Reactor technology and to preserve the specific expertise on Fast Reactor sodium technology and the capability to use this expertise for a Fast Reactor project. It includes :

- R&D studies, notably in connection with the CAPRA programme and continuing progress on generic topics (e.g. safety, ISI & R, integrated fuel cycle studies, codes and standards, international harmonisation),
- acquisition of experience from operational reactors worldwide, both in Europe and elsewhere in the world with Japanese and Russian reactors,
- retaining an up-to-date Reference EFR (two versions : breeder or burner),
- studying innovative design proposals for validation before incorporation into the Reference EFR,
- reviewing past decisions on the potential for more revolutionary design approaches.

In considering a work programme to meet the above objectives, EFR partners see also strong strategic and economic incentives for Europe to cooperate with other nations having significant ongoing fast reactor development programmes such as Japan.

The work programme will need to be reviewed periodically in line with developments in the design assessments, the R&D work and the outcome of moves towards greater international collaboration, particularly on large projects. It will continue until a decision is made either to start a new project phase or to participate in a foreign project or to revise the overall EFRUG strategy.

It is proposed that in the period 1998-2000 a first major review of progress should be undertaken by EFRUG with its industrial and R&D partners to confirm or to modify the orientation of the work. Short term milestones are defined up to this review date.

1. THE LONG TERM PERSPECTIVE

1.1. BREEDERS

EFRUG remain convinced that FRs will play an essential role in the long term future of nuclear energy because of their breeding role. FRs have the better potential for saving uranium resources compared with any other reactor. Progress in the management of the fuel in water reactors (MOX fuels, new concepts such as spectral shift reactors) can delay the need for breeders but FRs remain the only proven technology, that is environmentally friendly, independent from raw material markets and able to satisfy future world energy needs.

Because of the current uranium market and the general situation of nuclear development in the world, the time for the breeding mission is uncertain and belongs to the long term future. For this mission the delay for commercial deployment of FRs is extending. Nevertheless past experience shows that rapid changes in the energy market (on a timescale less than that necessary for reactor development) are always possible and that we have to be prepared to provide appropriate responses; hence a strategy of precautions is advocated.

The time required to establish a reliable reactor technology and to implement it into the nuclear generating capacity means that steps have to be taken to preserve FR technology over the near term, maintaining its state of development up to date (e.g. according to safety standards) and responding to the challenge of demonstrating economic competitiveness compared with other types of advanced reactor.

Consequently EFRUG believe that Europe should maintain a lead position in FR development.

1.2. BURNERS

The delay in commercial deployment of breeders leads to postponing the total closure of the Pu utilisation cycle. This raises the question of whether, in the meantime, Pu should be stored (e.g. in spent MOX fuel) or burned(2). Furthermore concerns have been expressed in some countries about the need to reduce minor actinides in nuclear wastes to decrease their radiotoxicity. These questions lead to a consideration of new missions regarding the management of the back end of the fuel cycle.

In this field the CAPRA studies aim to benefit from the flexibility of FRs to evaluate their potential for Pu and minor actinide burning, developments which EFRUG support.

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(2) Evaluation of the effect on Pu stocks of the implementation of a series of FR burners in the nuclear facility shows that, in the medium term, the burning performance has a small influence. This is due to the fact that FRs retain a large quantity of Pu in their core (about 9 tonnes in EFR CD) and their fuel cycle and this effect is predominant on the stocks. The availability of Pu (and of course the fuel fabrication capacities) is rather a limiting factor for FRs implementation.

The result is that interest in Pu utilisation in FRs beyond MOX recycling in PWRs is, in the medium term, rather independent of the burning or breeding performance.

This also gives the possibility of a specific new mission for FRs which could be not to burn as much Pu as possible but to tie up the largest possible Pu mass in large cores (e.g. one batch style). One could say a "CAPTURE" fast reactor (Concept to Amplify Pu Tied Up in a REactor). Such a mission which could preserve Pu for future breeders could also be consistent with the current concern in Russia about military Pu stocks, considering that Russia does not have a large capacity to absorb MOX fuel in LWRs but has an interest in the development of FRs

For these missions however, new fuel cycle and core management methods in LWRs may provide an alternative to FRs or opportunities from a co-ordinated programme of LWRs and fast reactors. The potential role of FR burners in the back end of the cycle therefore needs to be clarified in comparison with other solutions before any decision on introducing new systems (e.g. one of the advantage of FRs compared with LWRs could be a better efficiency in the reduction of long life wastes per kWh).

It should also be noted that large reductions in the volume of actinides and in the production rates per kWh of high and intermediate level wastes are obtained when using breeders. Also it is possible to couple actinide incineration to Pu breeding so that waste minimisation is not restricted to Pu burners.

In all cases the interest in a series of commercial scale FR for such a mission remains uncertain and not for the immediate future. We are rather in the phase of evaluation studies and their schedule can be linked to current developments on the management of nuclear wastes and the final repositories (e.g. programme launched by the French law of 30 December 1991, which requires a conclusion within 15 years).

2. SHORT AND MEDIUM TERM PERSPECTIVES

The extension of the development timescale for Fast Reactors gives time to improve the technology by taking full benefit of the on going experience of operating reactors, R&D programmes and design studies. Reliable operations of prototypes, demonstration of the inherently forgiving safety characteristics of the concept, fuel improvements, ISIR,...will result in securing the capital cost estimate, the safety demonstration and the full life time operability.

2.1. ACCUMULATION OF TECHNOLOGICAL EXPERIENCE

Operation of current reactors (Phénix, Superphénix, Monju, etc.) allows technological knowledge to be gained on main NSSS systems and components (such as fuel, sodium circuits, steam generators, handling systems), monitoring and In Service Inspection techniques, operating experience. To take full benefit of this knowledge at a worldwide level it is important to organise and to formalise data compilations and to develop international exchanges.

In Europe the Superphénix "knowledge acquisition programme" (a research and test programme over several years) launched in July 1994 will be the main example of getting value from operating experience for FR development.

2.2. TEST AND EXPERIMENTS ON ADVANCED FUELS, PU AND ACTINIDE BURNING

The Superphénix "knowledge acquisition programme" includes feasibility tests of the burning process in the context of a commercial size reactor and, taking benefit from its handling flexibility, Phénix will provide the facility for irradiation experiments and advanced fuel tests. The total accumulated experience will cover both burner and breeder cycles.

For the short term, extending the life of Phénix for 10 additional years in combination with the Superphénix programme is appropriate to cover the need to perform irradiations and to test various core configurations.

To face the possible unavailability of Phénix a strategy of precaution would be to develop the collaboration with programmes in countries having irradiation facilities outside Europe.

Earlier closure of Phénix would made these agreements even more important.

In any case the question of replacing Phénix will appear sooner or later but it is premature to work on such a project until the prospects for FRs have been improved by satisfactory operation of SPX for several years and until other studies (CAPRA, ...) have progressed.

2.3. REACTOR STUDIES

The reference scenario for the orientation of reactor studies is based on the prospect of a decision on a demonstration FR in the years 2005-2010.

As already mentioned, there are uncertainties today on the type of mission (burner or breeder) of a demonstration plant. However a large part of the reactor design is independent of its mission in the fuel cycle, therefore studies and developments must be maintained on the common features of the reactor design(3). From these common features, adaptations to the different missions (burning or breeding) should be developed. A "Breeders & Burners Programme" can be launched.

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(3) This was checked in the first phase of the CAPRA studies however it is probable that each project -breeder or burner- would require its own optimization

These developments include codes and standards which have to be maintained to an up to date level of development, taking into account the evolution of knowledge and possible changes in the requirements of the safety and regulatory authorities International collaboration on codes and standards is important to maximum the potential licensability of European designs worldwide.

The same applies to computer models (e.g. core evaluations), which must be validated by past and current experiments, and be able to make up for a temporary lack of experimental tools.

Without short term obligations it is also possible to open the range of potential designs to be assessed and advanced, including those developed outside Europe, and to make a critical review of the options which have been chosen (such as the choice of sodium as coolant) against operational experience and checking if these options always correspond to present views on the conditions and missions for FRs.

2.4. FUEL CYCLE STUDIES

For the definition of a consistent strategy it is also necessary to deal with fuel cycle studies and to consider full integration and optimisation of the whole system : fuel fabrication, the introduction of different reactor systems, storage and reprocessing, nuclear waste management. These studies are not part of the current EFRUG work programme in itself, which deals mainly with the reactor systems, but different national and international programmes have already been launched to cover these areas.

Due consideration must be given to these studies in evaluating the strategy and the EFRUG work programme maintains the flexibility to accommodate their conclusions.

Exchanges of information in this field should take place in the context of the EFRUG work programme and, possibly, with additional partners. The CAPRA Committee is proposed for such exchanges.

3. THE OPTIONS FOR EUROPE AND THE KEY POINTS FOR A FUTURE DECISION ON A DEMONSTRATION PLANT

3.1. THE OPTIONS FOR EUROPE

In broad terms the options for Europe could be :

(i) Stop fast reactor development completely,

(ii) Stop work on EFR type technology and start development of new revolutionary Fast Reactor technology, e.g. steam, gas or lead coolant or reactor without IHX,...

(iii) Adopt the strategy proposed here which advances technological development by assessing innovative options while also preserving the accumulated knowledge and the capability of the European partners to decide at any time to go forward by building a demonstration plant.

Option (i) corresponds to the trends in a number of organisations and might be seen as consistent with decisions by some governments to stop financial support to R&D. However it is inconsistent with decisions already taken by European organisations and governments involved in the continuation of Creys Malville operation. If such a strategy were to be followed by European Utilities it could impair their future and as explained in section 1.1.1 above it may in fact incur high risk because :

- it is believed that breeders will be essential to the European economy by the mid-21st century, and possibly earlier if there is a major change in NPPs development in the world due to the availability or acceptability of fossil fuels. Such a change might be dictated by environmental concerns, or by a change in world energy markets due for example to increase demand in Asia;
- this option would waste the large investment in Fast Reactor already made in Europe where the commitment will remain with the current operating reactors. It is totally incompatible with the preservation of the expertise necessary for these and would place the European industry at a grave disadvantage in participating in the future expansion of the world-wide nuclear industry.

Option (ii) should be dismissed because there is no need at present to make any significant changes of direction. We have confidence in the current technology and there is no need to throw away the European lead to pursue a high risk, high cost new programme. Nevertheless, alternative technical approaches should be kept under review, particularly in case there is any need for major change in the safety and reliability considerations.

Option (iii) is a low cost, low risk strategy and is preferred. The R&D has largely been completed and the substantial engineering programme has proceeded for enough to give confidence that we have a competitive design licensable throughout Europe.

3.2. CONDITIONS FOR A DECISION ON THE CONSTRUCTION OF A DEMONSTRATION PLANT

The schedule for a decision on construction of a demonstration plant will be influenced by the following considerations :

(i) the outcomes of the R&D programme on Pu and actinides burning. Results are not expected before several years from now (e.g. the French law of 30 December 1991 requires a conclusion within 15 years),

(ii) the results of fuel cycle studies (cf. § 2.4) which must lead to consistent and agreed conclusions on the role of fast reactors as breeders or on a possible interim role as burners,

(iii) the timing when fast reactors will be required to start playing a role (either as burners or breeders according to the previous item) in the European fuel cycle. These could need to prepare and perhaps to anticipate the penetration of FRs in the nuclear power generating capacity because their beneficial effects on fuel stocks and on the market will take time to be felt,

(iv) the development of the necessary confidence in fast reactor technology through successful operation of and feedback of experience from existing plants with appropriate embodiment into future designs,

(v) the need to maintain a viable design team to support current reactors in operation and to preserve expertise for the future,

(vi) the possibility to wider international collaboration, particularly with the Japanese DFBR.

Because the conditions are not fulfilled it would not be appropriate to seek funding for construction of a demonstration plant immediately. However we have to be prepared for a possible change of the situation and, if the central scenario is based on a decision in 10 or 15 years from now, the work programme must also consider a possible shorter commitment.

4. THE EFR WORK PROGRAMME

4.1. GENERAL SPECIFICATIONS

Due to the uncertainties in the schedules of the short and medium term missions, the work programme has to be flexible and prepared to face a variety of conditions. Nevertheless some conditions common to all possible scenarios allow the main points of interest to be identified for the short term orientations.

4.1.1. The prime importance of economics

This requirement was followed when defining the EFR "Consistent Design" which has not been optimised to have the best possible breeding capability but to have better cost and licensability. Further improvement of the CD or more advanced designs remain possible in these domains which give incentives for development of advanced or alternative (including non-European) designs. All these potential improvements will have to be compared against the Reference EFR performance (see 4.2.1 below) keeping the objective of competitiveness with advanced PWR.

The same approach has to be applied for the studies of CAPRA cores : they must not aim at the highest burning capability but at the lowest cost in the general context of a burning mission.

It must be recognised that such an economic optimisation may lead to different reactor designs according to their mission. This leads to reconsidering the idea of reversibility (see 4.1.3 below).

4.1.2. An up to date safety level

The Ad'Hoc Safety Club concluded that, if achieved, the targets adopted for the EFR CD will result in levels of safety comparable with those expected from future PWRs. The AHSC made a number of recommendations to improve the demonstration of the EFR safety case in connection with the remarks formulated by Safety Authorities during the licensing of previous reactors (notably criticism of particular aspects of FR technology expressed in a letter from the French DSIN, in June 1992).

It must be recognised that the safety objectives for future reactors have not been fixed for ever and that their definition is currently under development through continuing international discussions. Nevertheless considering the inherent characteristics of FRs, the AHSC conclusions will remain valid for a long period in the future.

Accordingly it is important to keep the same safety targets for any designs which are now being considered and, first, to check the adequacy for meeting the AHSC recommendations.

Consideration also has to be given to the evolution of international standards for future reactors and the need to check that EFR targets are always valid, notably in the fields which are not reactor dependent such as protection against external hazards.

4.1.3. Reversibility (burner / breeder)

Due to the uncertainties concerning the short and medium term missions of FRs, the ability to satisfy the final aim of a breeding mission has to be kept. In any case, as already mentioned, much of the development is independent of the type of mission.

Reversibility must not be seen as a requirement for a particular plant (it would be incompatible with the economic optimisation mentioned above) but as a need to maintain the compatibility of the technologies and their development. As EFRUG

remains convinced that breeders represent the future for nuclear energy and that the end of this nuclear era has not to be considered (which could be the aim of developments dedicated only to Pu burning), no costly development of new technologies which have no interest for an economic breeding mission should be retained in the EFR programme.

According to this, whatever the design, it will have always some flexibility regarding its mission : this has been shown for EFR CD (breeding gain between 0.8 and 1.2) and it is now applied in Creys Malville with progressive changes from a breeder core to a burner core.

4.1.4. The need for a secure design based on the accumulated knowledge and experience

The stage of development of FRs is already considerable and operating experience will continue to be accumulated in Europe and worldwide. The Creys Malville "knowledge acquisition programme" already mentioned is an important step to take benefit from this substantial experience.

For future development it is important :

- to keep yardsticks of well known and validated designs,
- to review the project designs against the feedback from operating reactors with the participation of international expertise.

Operating experience and knowledge accumulation must also help for improving codes and standards in the scope of a large international collaboration. In order to protect investment, future codes should address reliability as well as safety criteria.

4.1.5. The need for further progress in techniques for In Service Inspection & Repair

Concerns have been expressed about the difficulties of carrying out diagnostic analysis and repairs on the internal structures of current pool reactors and the subsequent need to remove this concern for future LMFRs. This leads to investigate further how to optimise ISI&R with respect to the design of a pool reactor, associated ISI&R techniques and tools, whole core discharge.

It also gives incentives to co-operate in the evaluation of alternative designs such as Top Entry for which promoters claim better ISI&R capabilities.

4.1.6. The interest for international co-operation

Due to :

- the uncertainties concerning the date of implementation of series FRs,
- the necessity to face short term requirements and to share the development cost,
- the trends and incentives to come to worldwide agreement on the safety specifications and technical characteristics of future reactors,

the EFR programme has to be compatible with development schedules outside Europe (notably Japan(4)) and should offer ways for collaborations with Japan, Russia and possibly others.

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(4) Japan has launched its own actinide burning programme (OMEGA) and more recently seems to be interested in CAPRA

4.2. THE WORK PROGRAMME FOR DESIGN STUDIES

The aim of the renewed forward programme, along the proposed lines of development, is for EFRUG to study several reactor designs (probably not all at the same stage of definition) ready for a decision on further development or to enter a construction phase, optimised economically for a dominant mission (breeding or burning).

However as the economics of electricity generation represent the prime parameter and as the basic technology of FR burners must remain compatible with breeding, the fundamental differences in the designs of these reactors will need to be fully analysed and understood.

Therefore the work programme is organised mainly with the hypothesis of a decision for a demonstration plant in 10 - 15 years from now. To keep some flexibility and to take account of possible changes in perspective, the work programme is based on 4 main items :

- a Reference EFR , with two versions : breeder or burner;
- innovative designs and generic studies;
- developments of codes and standards and computer models,
- evaluation or collaboration on non-EFR projects.

Generally these studies are for commercial size reactors (1500 MWe). From previous studies it is known that it is possible to design a reduced size prototype with a penalty on the electricity production cost. The design would be based on the commercial size design and it would be possible to validate its innovative options. Without current interest for such a reduced size prototype, the work programme does not include any specific studies on this item but could be easily reoriented if new perspectives emerged (e.g. if the need to replace Phénix was to be considered).

The role of such a demonstration reactor in the grid would be limited to stable operation : consequently no stringent requirements such as grid following will have to be considered for the basic design, but the capability for a commercial reactor for such load variations will have to be assessed.

Considering the different possible missions of FRs and the fact that economic optimisation could lead to differences in reactor designs, adaptation of each type of reactor to an alternative mission (breeding or burning) will be performed (reversibility is associated with the technology, not with a specific design as already mentioned). Major differences between the designs of these reactors are not expected.

In addition to the above studies, closer collaboration, association and participation of EFR partners in foreign projects, has to be considered, notably :

- the DFBR in Japan : the interest could be that a Top Entry prototype might have advantages regarding ISIR or possibilities to test advanced ISIR techniques. It could also be seen as an intermediate step towards a futuristic design without intermediate sodium circuits.
- a possible new reactor in Russia.

4.2.1. The Reference EFR

The **Reference EFR** is a commercial size reactor (1500 MWe) based on a design which is largely validated.

It is the reference because it is at the same time the most secure and the most economic regarding electricity production.

This secure design will be based on the 1993 Consistent Design but with integration of innovative ideas from the innovative and generic design studies when they have

been demonstrated to be really attractive (cost, safety, robustness, ISIR...) and when they have reached a consistent level of validation compared with other design options. Therefore the design of the Reference EFR results from an evolutionary process, without challenging the basic options which have benefited from previous large R&D programmes and operating experience.

To ensure the design is robust, periodic reviews of the options are made against the worldwide experience of past and current operating reactors notably the Creys Malville "knowledge acquisition programme".

The Reference EFR is used as a yardstick for comparisons. As a consequence an up to date economic evaluation is kept ready for comparison against other FR designs and also against the European reference designs of Advanced LWRs.

Being of a commercial size this reactor is associated with the future breeding mission rather than burning, nevertheless its adaptation to a burning mission is part of the studies.

4.2.2. The innovative designs and generic studies

The 1994 studies have shown benefits and drawbacks of a number of innovative ideas aiming at a better accessibility, economics, robustness, safety. These ideas deal with the fabrication and design of the reactor block but also with alternative containments, lay out, core safety etc.

Other concepts, including those developed in other countries, could be assessed and could lead to different types of innovative reactors.

New ideas include core design and actinide burning capabilities.

Progress is expected in the domains identified in § 4.1 : economy, safety, robustness, in service inspection and repair, flexibility, validation through operating experience and international consensus.

Generic studies (not specific to a given design) are performed in these domains but the innovative options can be assessed through global remontages which must not be seen as reactors to be built but as tools to evaluate the benefit of these new ideas.

To be compared to the Reference EFR, these remontages must be based on commercial size (1500 MWe). Inherent advantage, if any, of some options for reduced size reactors could be identified.

Periodic reviews allow rationalisation of the number of alternatives through a selection based on a clear understanding of drawbacks/benefits. Some of these reviews should be open to partners outside EFR to take benefit of other views and possibly to share the development cost. The most promising options could lead, if necessary, to a programme of validation or possible testing in operating plants.

When an innovative option is seen as advantageous and has reached a stage of sufficient validation it can be decided to implement it in the Reference EFR if it is compatible with other main options.

Some innovative options may be identified as having particular interest while not being compatible with the general options of the Reference EFR or they may require a large programme of validation. These options may be classed as "revolutionary" designs (e.g. steam, gas or lead coolant or reactor without IHXs), in contrast to the evolutionary approach adopted for the Reference EFR.

The decision to launch studies of these revolutionary designs will be taken by EFRUG after consultation of the R&D and industrial partners(5), on the basis of past and prospective studies.

4.2.3. Codes and Standards, computer codes

The development of agreed Codes and Standard is particularly important. Progress is expected in the elaboration of the rules themselves (such as structural mechanics codes), but also in their validation by R&D studies and the largest possible international agreement. The codes should be improved to address reliability as well as safety criteria.

With the same idea, development and use of validated computer codes for analysis of normal and transient conditions must take advantage of the worldwide operating experience with a possible economy on future requirements for experimental tools.

4.2.4. Outside EFR projects.

4.2.4.1. Participation in DFBR

The continuation of the current technical exchanges with Japan in the frame of the Utilities agreements will lead to a better understanding of the respective choices and may achieve common conclusions on the design of some systems. Exchanges of information or participation in the R&D programmes of common interest are also part of the current R&D agreements.

A more ambitious objective could be achieved with an evaluation of the interest for EFR partners to participate more directly to the Top Entry project, notably if the Japanese enter a construction phase, with an equivalent participation of Japan in the EFR programme.

It is noted that a decision on such cross participations is not necessary before the date of commitment to a construction phase (around 2003 according to DFBR schedule). This gives time to evaluate the mutual interest and to further progress current technical exchanges.

4.2.4.2. A reactor in Russia

According to the current R&D agreements a test period of two years (94 -95) will show the possibilities for in depth technical and design exchanges with Russia. Meanwhile participation of OKBM Institutes in some identified tasks of the EFR programme will be tested.

At the end of this test period it may be possible to launch common design oriented studies on either an EFR demonstration or BN 800 or another project.

This could cement the collaboration with the Russians and help to define further R&D collaboration. However the ambiguity of the situation with the Russians must be emphasised while they are looking for support either for their BN 800 project or for the construction of a reactor to burn their military Pu.

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(5) In particular the reasons and justifications for the choice of sodium as a coolant rather than steam or gas will be reviewed.

It is recognized that at an earlier stage of fast breeder development (in the sixties) an evaluation of pros and cons of these competing coolant media had been performed. The choice of sodium was made according to the technical knowledge, experience and expected developments of that time. It has to be checked that this conclusion remains valid and that there is no reason to launch studies on an alternative coolant which might be judged advantageous in the long term.

5. THE R&D PROGRAMME

Supporting the above studies, an important R&D programme is required which can be classified according to the following items :

- support for the reactors in operation,
- CAPRA studies and more generally the development of core and fuel designs and the fuel cycle,
- generic and technological developments,
- safety aspects in relation to core accidents and containment.
- operating experience.

There are strong interrelations between all these items, as reflected by the fact that, for example, international collaboration will include these subjects.

5.1. SUPPORT FOR OPERATING REACTORS

In Europe the support for operating reactors concerns mainly Phénix and Superphénix. For the next decade the main items will be the life extension project for Phénix and the "Knowledge Acquisition Programme" of Superphénix which has strong links with the CAPRA programme.

However it also covers actions in the fields of :

- safety, In service Inspection and Repair,
- computer codes development and experimental validation,
- maintenance of expertise on sodium technology, C&I, fuel and core physics, thermohydraulics, etc.

5.2. CAPRA STUDIES, CORE AND FUEL DEVELOPMENTS

The CAPRA project is led by the R&D organisations. The feasibility of a MOX-fuelled burner has been established, and a programme of fuel irradiation testing has been started. A three year continuation R&D programme has commenced, at the end of which outstanding technical questions on the design of such a core will have been resolved, and first Post Irradiation Examinations results will be available, allowing a first stage of validation of this option. In addition the feasibility of a burner with advanced fuel or improved performance will have been assessed.

Among the developments which are in common to the burning and breeding missions, efforts have to be maintained on the fuel and core developments and validation aiming at high burn up, the search for reduced void coefficient, and the core physic data.

Alternative fuels such as nitride must also be considered and technological awareness on the metallic option could be kept through the knowledge of the American developments on this subject (by maintaining, if possible, collaboration on this item).

Consideration must be given to the complete fuel cycle, including reprocessing regeneration and waste management as well as the fuel and reactor aspects if the ultimate goal of a fully optimised and commercially sound system is to be achieved.

5.3. GENERIC AND TECHNOLOGICAL DEVELOPMENTS

Practically all technological development of EFR components has been stopped to turn to more generic developments such as In Service Inspection and Repair, structural mechanics and materials, sodium technology and prevention of recriticality risks in case of core accidents.

The importance of actions in support to Codes and Standards, In Service Inspection and Repair developments has already been emphasised.

5.4. SAFETY

Previous items include actions which are safety related. In addition there are specific safety actions concerning the containment issues and local and global core accidents.

The containment issues are mainly related to sodium fires. Important developments have been made these last years in support of the licensing of Creys Malville restart; for the next years efforts should swing from the large experiments to computer codes and technological developments (such as leak detection features).

The needs for developments on core accidents will remain for several years with computer codes and experiments in support, notably in Cabri and Scarabée facilities.

5.5. INTERNATIONAL COLLABORATIONS

Because of the important and costly needs for the R&D works, the development of the international collaboration is even more essential than in the past. Agreements between European and Japanese R&D organisations which have been signed on June 17, 1994 and those with MINATOM (Russian organisation) are the main basis for the future developments during the next years.

6. GENERAL SCHEDULE AND MAIN MILESTONES

The programme strategy defines activities for the period of time extending until a decision is made to enter a construction phase or to participate in a foreign project or to revise the global EFRUG strategy. It covers a period about a decade but is flexible enough to be adapted to changes in timescale or in the overall strategy.

It is proposed that at the end of a first period of 3 to 5 year (between 1998 and 2000) a major review of progress will be undertaken by EFRUG to confirm or to modify the orientations. The analysis will include R&D works and fuel cycle studies.

Objectives for the end of this period are given hereafter.

6.1. R&D

Generic actions and those in support of operating reactors correspond to programmes over several years. For each item a schedule is proposed as for the definition of the Knowledge Acquisition Programme of Superphénix now under progress.

Concerning the CAPRA studies, by the end of the 3 years a first level of experimental validation of a MOX-fuelled Pu burning core will have been completed and the feasibility or otherwise of a high performance burner using non uranium fuel will have been established.

As mentioned in § 2.4 the CAPRA Committee will deal also with the role of Fast Reactors in the fuel cycle : during the first 3 years an assessment of the place of FR in the European fuel cycle will be made by some partners and will include discussions on the effect of Fast Reactors (breeders, possibly preceded by burners in the medium term) on Pu and minor actinide quantities, waste arisings, personnel doses and on the cost of electricity.

6.2. DESIGN STUDIES

6.2.1. The Reference EFR

It is useful periodically to check the consistency of the different design evolutions of the Reference EFR. For that purpose a global remontage is foreseen after a minimum period of 3 years (necessary for progress) and a maximum of 5 years (to keep a visible target).

At the end of the 3 to 5 year period (between 1998 and 2000) EFRUG expects an up to date remontage of the Reference EFR and of its economics with the same level of definition and validation achieved with the present EFR CD. Additionally a review of the options against the experience from operating reactors is expected.

6.2.2. The innovative designs and generic studies

Being paper tools the remontages based on innovative options have no identified time target but these remontages are requested to ensure consistency of the options and their possible interest for the Reference EFR.

At the end of the 3 year period EFRUG expect an evaluation of the advanced options and results of the generic studies considering the different aspects : economy, safety, compatibility with the Reference EFR. The evaluation must include identification of the R&D programme which would be necessary for their validation, its schedule and cost.

The general EFRUG review at the end of the 3 to 5 year period will consider also the interest for launching "revolutionary design" studies on the basis of preliminary

assessments of innovative options which may have particular interest but are not compatible with the Reference EFR or which require a large programme of validation (see notably footnote on § 4.2.2).

6.2.3. Codes and Standards, computer codes

Work in this area is expected to continue throughout this period.

For structural analysis, progress is expected in simplified design rules, for example concerning ratcheting prevention, and in fracture mechanics, notably at high temperature. Elaboration of a common material data and design rules using the RCC-MR as a basis should continue.

A design code concerning the "sodium fire" risk is expected and progress for some specific designs such as the "anchored safety vessel" which is the reference option for EFR.

Developments are also expected in computer codes for normal and transient operating conditions, core accidents, sodium fires, containment behaviour,...

Codes and standards, computer codes, have to be strengthened through international benchmarks and when possible with the elaboration of common rules and tools. Reliability as well as safety criteria must be addressed.

6.3. INTERNATIONAL COLLABORATION

Concerning the collaboration with Japan, according to the progress in technical exchanges and the evolution of the Japanese schedule for DFBR, the possibility of more active participation in the respective projects will be examined at the end of the 3 to 5 year period.

Concerning the collaboration with Russia a new step, including common design oriented studies, will be considered in 96 according to progress during the 2 years test period of the CEA / Minatom agreement.

7. CONCLUSIONS

EFRUG recognise the long timescale to commercial deployment of Fast Reactors but believe that they provide the only long term future for nuclear power when uranium resources become scarce.

The final breeding mission for the FR remains unchanged however EFRUG see that the delay in programmes provides the opportunity to develop designs further, to take account of future FR operating experience and developing reactor technology and also to investigate the wider potential of the FR to manage the back end of the nuclear fuel cycle.

Thus, EFRUG recognise the CAPRA studies as part of a larger "Breeders and Burners programme", proposed here, which seeks to cover not a unique, but a range of possible futures for Fast Reactor development. One of its main objectives is to check that the reversibility of the technology (breeder / burner) can be preserved.

It is recognised that the programme could extend for a decade or more before a decision is taken to enter or to participate in a new project phase or to revise the global EFRUG strategy. A plan extending for such a period cannot be defined in detail but short term milestones have been defined. It is proposed that at the end of the first three to five year period (between 1998 and 2000) a major review of progress should be undertaken by EFRUG with its industrial and R&D partners, to report the progress, to confirm or to modify the orientation of the work and the strategy according to the evolution of the conditions.

This ongoing programme will be more effective and will have greater security within the framework of a broader international collaboration that EFRUG is seeking to develop. Furthermore it will have the benefit of preserving sufficient design and development expertise to support the operation of current reactors.

3 - OPERATION OF PHENIX AND SUPER-PHENIX IN 1995

3.1. - PHENIX

3.1.1. - 49th Cycle

As explained in the paper "Fast Reactor Development Programme in France during 1994" presented at the 28 th IWGFR meeting, the secondary loops SL1 and SL3 were available at the end of August 1994.

The plant was ready for operation at the end of September 1994.

After a technical review by the Safety Authority, the authorization to perform the 49 th cycle at 350 MW th (143 MWe) was given on December 21, 1994. This 49 th cycle was completed without any significant problem on April 7, 1995 (**Fig. 3.1**) :

- The reactor was shut down three days in January 1995 to load the CAPRIX 1 experiment (see § 5) and to proceed with minor repairs on the water/steam circuit,
- The reactor was also shutdown two days in February in order to repair a steam leak on a valve,
- A decrease of power in March was necessary to correct a defect on the lubrification system of the turbine.

3.1.2. - 50th cycle

For the reasons explained in the report mentioned above (§ 3.1.1), major decisions were taken in 1994 in order to extend the life time of the reactor another 10 years. A work programme was decided which involves long shutdown periods (see § 3.2.3) Such a period occurred after the 49 th cycle.

At the end of 1995, the secondary loops SL2 and SL3 were available.

Finally, the plant was ready for operation at the end of February 1996.

After technical review meetings, in February 1996, the Safety Authority did not rule on the possibility of extending the lifetime of PHENIX, and did not want to authorize performance of the 50 th cycle.

The Safety Authority has requested safety improvements and technical demonstrations, before it examines the possibility of authorizing a new start-up of PHENIX ; in particular :

- The installation of a Safety Scram System (Complementary Shutdown System). This will be made for the end of 96,

- A demonstration of the possibility of reinforcing the plant against seismic loads (historically maximum seism) taking into account new design rules,

- An improvement of the emergency cooling system in case of earthquake,

- Consideration of the foreseeable soundness of the core support line.

3.1.3. - PHENIX life extension project

Extending the lifetime of the reactor another 10 years has been planned for three purposes :

- To qualify high burn-up oxide fuel,

- To perform the necessary experimental irradiations in support of the CAPRA project (enhanced plutonium consumption) and SPIN (separation and incineration of long-lived wastes), and

- To supplement operating experience feedback, in particular for a better understanding of negative reactivity incidents and the ageing process of the structures.

In order to complete this programme, the reactor needs to operate a further 55 000 hours. After completion of these 55 000 hours, the reactor will have 152 000 hours of operation and this is in excess of the 140 000 hours value in the Safety Report (precisely 110 000 hours at nominal power, 15 000 hours with 4 IHXs out of 6, 15 000 hours with 2 primary pumps out of 3).

As already mentioned, major decisions were taken in 1994 in order to extend the reactor's lifetime, namely :

- The order of three extra IHXs ; they will be delivered no later than spring 1997 ;

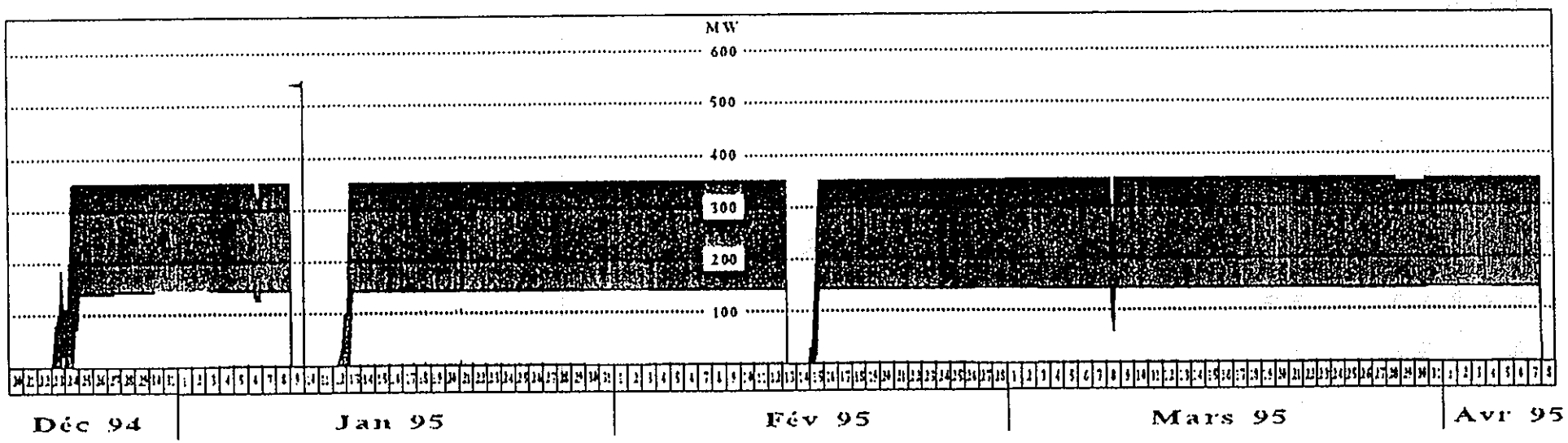
- The replacement of the 321-stainless-steel pipes in the secondary circuits. This work on the SL2 secondary loop was started in 1994 and completed in 1995. The work on SL1 was started in 1995 and will be completed in 1996. The work on the SL3 loop will start in 1996. When the new IHXs are delivered in 1997, all three loops will be available ;

- The installation of insulation covered by steel plates on the floor of the SL1 and SL3 galleries was completed. Similar work is being carried out at present in the SL2 gallery ;

- The improvement of the protection against sodium fires in the steam generator building will be completed in 1997. The main tasks are : protection of the pipe penetrations between the reactor building and the steam generator building, improvement of the ventilation system in order to avoid any connection between the two buildings in the event of sodium fire, heat insulation of the support of the sodium loop in the SG building as well as heat insulation of the iron structuring, protection of certain valves and electrical cabinets, improvement of the alarm origin identification ;

- The reinforcements against seismic loads concern : the steam generator building, the emergency cooling system building, the handling building. These tasks will be performed in 96-97.

ACHEVEMENT du 49ème CYCLE
– Chargements N° 145 et 146 –



PHENIX

FIG. 3.1

3.2 SUPERPHENIX

- *B. MESNAGE* (CEA)

3.2.1 GENERAL VIEW OF 1995

After shutdown of the reactor on 25th December 1994 following deterioration of the argon leak in the bell zone of one of the intermediate heat exchangers (IHX's), two periods typify 1995 :

- a first period covering the first seven months of the year, devoted to the problem of the IHX leak, and concluded after locating the leak and perfecting a process by repair of the IHX in-situ on 28th July ; the solution to replace the component which was,at the outset the most probable, was however continued as an emergency solution ,

- a second period from 22nd August when authorization by the Safety Authorities to take up the programme of Plant resumption was given ; this was marked by two long untimely outages following the triggering of a secondary pump, then a steam leak on a steam generator, but on the other hand heralded, after link-up of the turbogenerator sets on 30th and 31st December, a period of constant Plant operation which saw its power build up step by step to reach 50% of nominal power from February 1996.

At the same time as these operating activities, numerous maintenance operations can be added and the ongoing preparation of the subassembly handling line as a whole with a view to the handling campaigns to come. 1995 allowed us to initiate the Knowledge Acquisition Programme KAP (§ 4) bestowed on Superphenix in 1994 by the French Government, and taken in hand by NERSA, owner of the Plant, in its new agreement between the partners signed in September 1995.

In this new Accord, the partners accept that this programme, financed by EDF and CEA, is applied at Creys-Malville and commits them to assume the financial expenses associated to loans and operation, until 31st December 2000. But in return, because the plant can at any moment adapt or even interrupt its production to enhance its new mission, the partners will receive over the six years to come a guarantee of electricity supply equivalent to the energy potential of the fuel subassemblies at the moment in the reactor (core 1) or already manufactured (core 2) and paid.

In other respects, Creys-Malville continued throughout the year its policy of discussion and transparency. In May 1995 it participated in the Conference between supporters and opponents of Superphenix organized in Zurich by the Swiss Federal Government, and as from September 1995 issues a weekly information bulletin to the regional, national and Swiss press, giving the status of operation and the various events of the past week.

Lastly, those responsible for NERSA and operation of the Plant were interviewed at the end of the year by the Committee instituted by the French Government in September 1995 and chaired by Professor Castaing. This scientific committee of independent experts, which groups French and foreign personalities, has been appointed to appraise, before summer 1996, the aptitude of Superphenix to fulfil its role as a research tool.

3.2.2. REPAIR OF THE IHX LEAK

Two types of solutions had been retained at the beginning of 1995. The IHX replacement solution made possible by the presence on-site of a spare IHX, and the in-situ repair solution which could allow avoiding a considerable operation that replacing an intermediate heat exchanger represents.

In-situ repair solution

Appearing as more and more possible after location on 15th February of the leaktightness defect on the argon supply piping link to the bell, the in-situ repair solution involved, applied in parallel throughout the six months period, not only investigation and diagnosis operations but also work on perfecting the repair process. The main steps were the following :

<u>January - February 1995</u>

- location by tracer gas injection at the beginning of January of the leak in the lower part of the IHX bell argon supply piping, then development and qualification on scale 1 mock-up of :

  - a televisual inspection equipment fitted with fibre optic mounted on a stainless steel hose and introduced into the piping by a device allowing continued bell argon pressure,

  - a cleaning device through hot argon blowing of sodium aerosols deposited in the piping.

- in mid February, location and visualization of the leaktight defect on the welded link of the bell piping : a defect of about 30mm long (half the circumference) situated some few centimetres from the apex of the bell (see figure 1).

March - April 1995

- perfecting the solution chosen for repair by sleeve developed by Stein Industrie allowing preservation of the supply function of the piping, and where leakproofness is realized by expansion under fluid pressure (see figure 2), the development of a plug proposed by Novatome was in other respects pursued as an emergency solution.

Perfecting the sleeve was followed at the beginning of April by a campaign of qualification tests in sodium loop at Cadarache where results led to the decision to replace the organic fluid "Gilotherm" chosen originally, by water, as the Gilotherm had caused rupture of the sleeves by cracking under effect of temperature.

- at the same time, in the context of research on the causes of the defect, making of a dimensional replica of the piping using a specially developed device as well as control by eddy current probe of the piping and its guidance along the IHX ; these two devices were then introduced into the piping using a device developed for televisual inspection.

In addition, inspection of another IHX in April allowed confirmation of the non-generic aspect of the defect in question.

May - June 1995

- validation of the final sleeve design (installation of two capillary tubes for pressurization allowing the draining of water after expansion) and the positioning procedure of the sleeve on the scale 1 mock-up, completed by a second campaign of tests in sodium loop confirming, this time, the perfect behaviour of the sleeves.

- lastly, before repair, verification of the exact location of the defect by a last inspection, by fibre optics and a double measurement head eddy current probe.

After receiving approval from the Safety Authorities on 25th July, implementation of the repair was carried out successfully on July 28th and was followed by leaktightness controls and the reconnection of the normal circuits for maintaining bell pressure.

The IHX replacement solution

The study of the IHX replacement solution, carried through to its completion, was continued in parallel in spite of the choice of the repair solution. In this way, the following were carried out in the course of the first six months :

- the control and conditioning of the spare IHX,

- preparation of the equipment and procedures necessary for cutting and handling operations,

- specifications of assessment and washing the intermediate heat exchanger extracted.

3.2.3 RESUMPTION OF POWER

After criticality on 23rd August and calibration at 345°C of the steam generator leak detection systems, turbogenerator set A was linked to the grid on 3rd September. Plant build-up was soon to be interrupted by a rapid automatic shut-down following the triggering of a secondary pump which occurred on September 4th.

This outage caused by the failure of an electronic board in the pump speed regulation, was prolonged to around the end of the month to deal with two problems which occurred in the course of the transient :

- a leak detection alarm on a secondary sodium loop which involved the draining of the loop to replace a "sandwich" detector where experience showed that it revealed a manufacture defect,

- the untimely triggering of a steam valve following an incorrect operating manoeuvre, which needed detailed analysis to confirm that the incident was of no consequence on the installation.

After approval from the Safety Authorities, resumption was taken up and the turbogenerator set B in its turn was connected to the grid on 28th September, followed by the A set on 30th September. The Plant then was increased to 30% of nominal power, the maximum allowed by the the Safety Authorities' approval granted on 7th November 1994. At this stage all the tests and acquisition scheduled for the preparation of the file requesting operation above 30% of nominal power were carried out. This preparation was interrupted on 23rd October following the shut-down of the reactor following the appearance of a steam leak on the steam header of one of the steam generators.

This new shutdown, caused by the tearing by creep of an spare instrumentation tube showing an anomaly in the type of material (see figure 3), was used advantageously to control all the instrumentation tubes equipping the steam generators and the main steam piping so as to correctly modify the anomalies observed. This shut-down was also prolonged by the necessity to modify certain protections put in place in the context of "sodium fire" works, which prohibited the free displacement of sodium piping and particularly those of the decay heat removal circuits at the point of their penetration through the containment building.

Re-authorized on 19th December to take up its resumption programme again, the Operator reconnected the turbogenerator sets to the grid on 30th and 31st December and regained 30% nominal power operation at the beginning of January 1996.

With a little more than 8 EFPD achieved, the net electricity production for 1995 is approximately 154 million kW hours. As for the two untimely shutdowns, although the repairs were rapid and easy, thorough analysis of the causes and the elaboration of dossiers without any doubt required a great amount of time because of the prototype nature of the installation.

3.2.4 KNOWLEDGE ACQUISITION PROGRAMME (KAP)

The modest operating results of the Plant have not hindered starting the Knowledge Acquisition Programme, and in particular acquisition of technological know-how which is acquired whilst the Plant is operating but also at shutdown during interventions carried out on-site following untimely failures.

In this way, the repair in-situ of the IHX provided experience from a remote-controlled intervention on a large component immersed in sodium. It contributed to development in detection , investigation (televisual inspection, eddy current probes, etc.) and repair (sleeve, plug, etc.) resources.

Resumption at low power furthermore allowed one to :

- confirm the sound behaviour of the core by checking notably the safety criteria,

- follow up the purity of primary sodium and further knowledge on the workings of measurement devices and purification systems,

- improve knowledge of behaviour of hydrogen detection systems in the steam generators,

- check the displacement of sodium piping depending on temperature.

Lastly, in the context of this Knowledge Acquisition Programme, a prototype system of numeric processing of core outlet temperature measurements and their fluctuations, which could later well contribute to the efficiency of core monitoring, has been installed on the Plant in June.

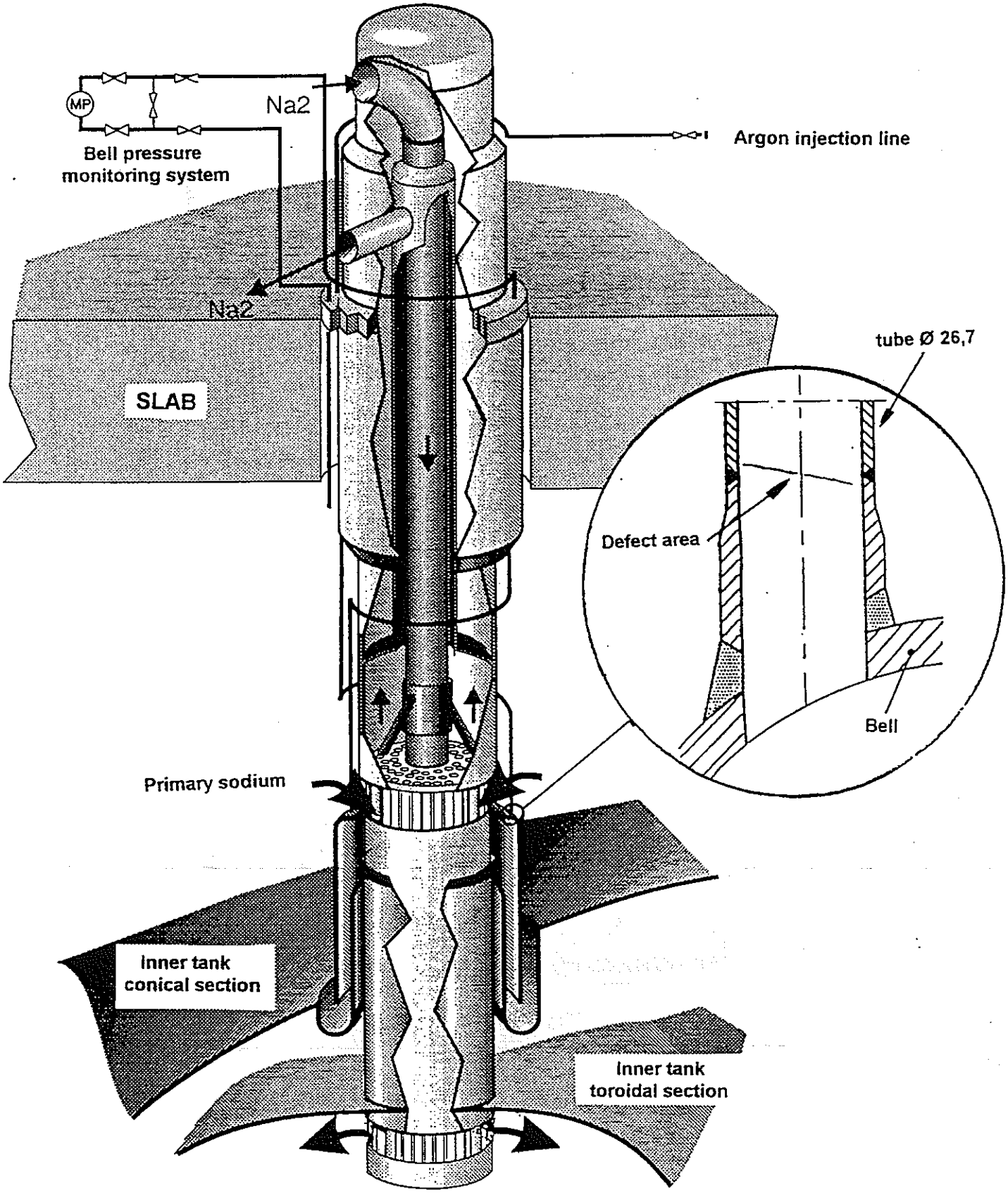


Figure 32 - CREYS-MALVILLE - IHX

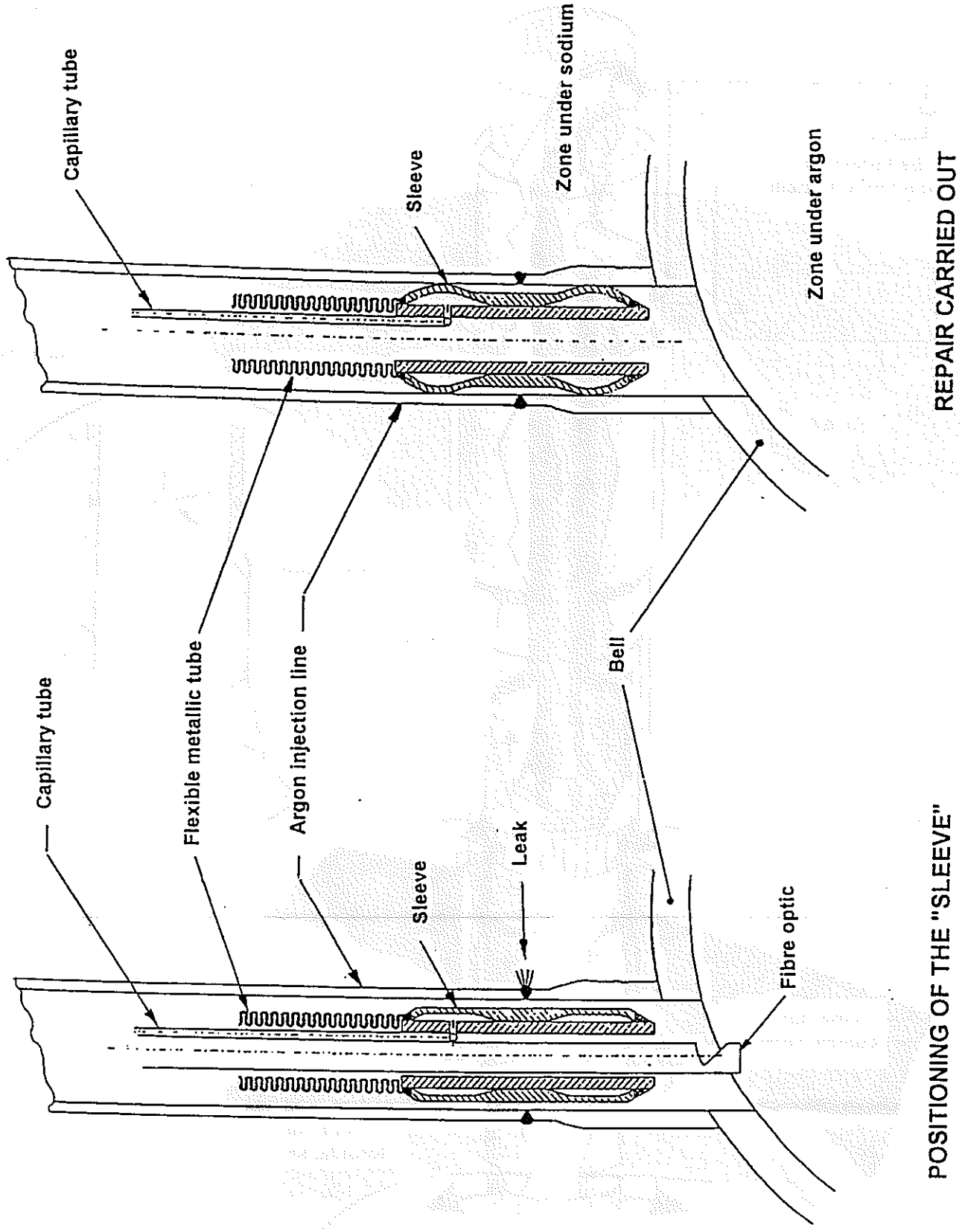


Figure33 - CREYS-MALVILLE - Repair of IHX

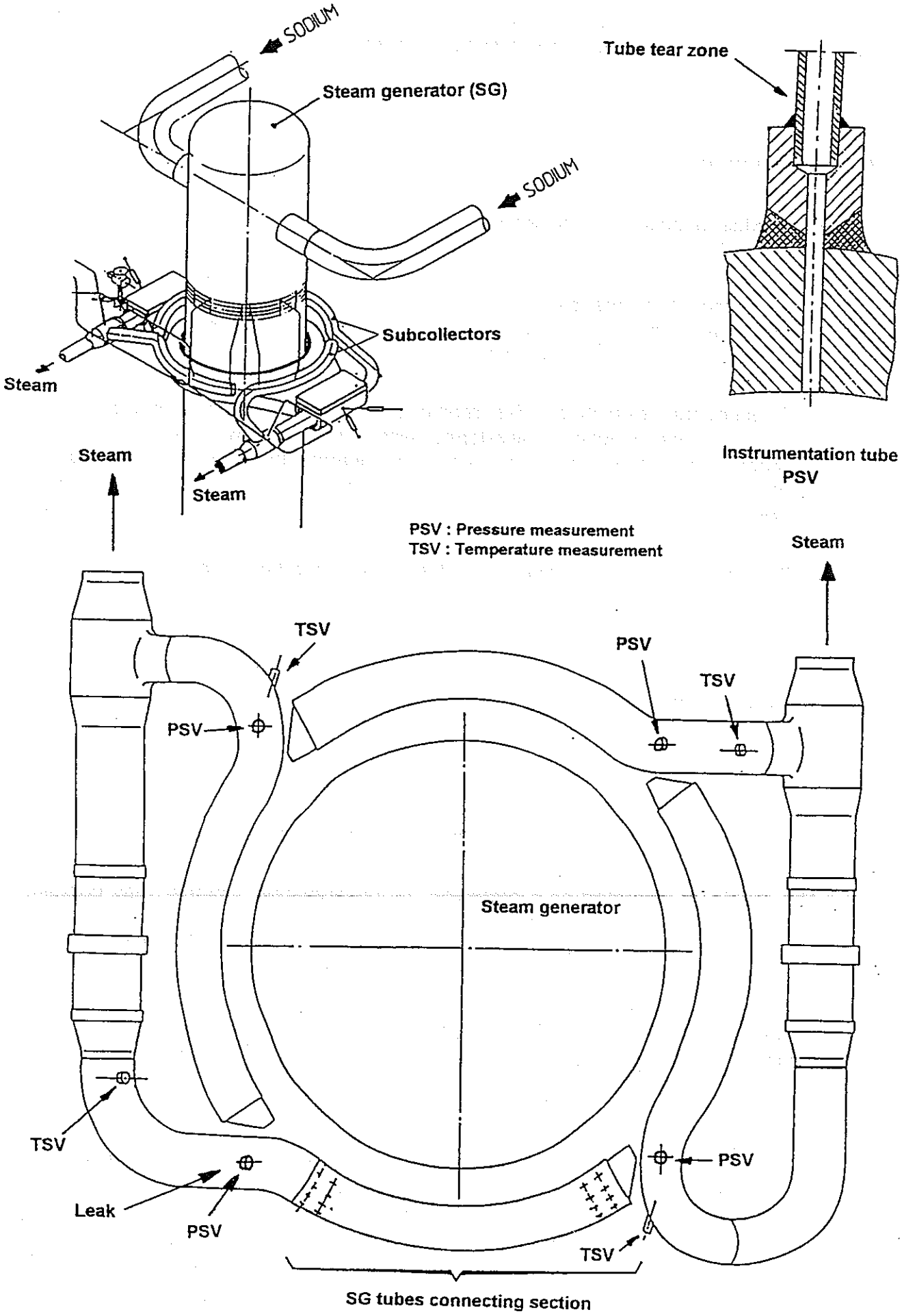


Figure 34 - CREYS-MALVILLE - SG steam collector

4 - KNOWLEDGE ACQUISITION PROGRAMME

P. ANZIEU (CEA), R. DEL BECCARO (EDF)

4.1. - Objectives

The knowledge acquisition programme, necessitating the operation of the Creys-Malville plant (Superphenix), has three major complementary objectives :

- To demonstrate the capacity of a fast reactor to produce electricity on an industrial scale, while contributing to plutonium management and the reduction of long-lived radioactive waste ;

- To study the flexibility of a fast reactor using plutonium as fuel, and to qualify the technical solutions developed within the framework of a research programme, whose major concern is to operate this type of reactor as a plutonium consumer.The CAPRA project is specifically involved in this case (see § 5) ;

- To study the possibilities of destroying long-lived radioactive waste, particularly the minor actinides, americium and neptunium, the framework of the SPIN programme.

4.2. - Research Programme

This programme hinges on the three major research objectives defined above.

4.2.1. - Demonstration of a fast reactor prototype in operation

The reactor is operated as a prototype. All observations, whether they concern normal or abnormal operating conditions of the reactor, are used to make an analysis, to reach conclusions and findings allowing feedback. The specific goal is to prepare the definition of an improved, future generation of fast reactors.

Follow-up on the state of the performances of the different systems and components in the NSSS involves in-situ measurements, periodic inspections, and special examinations during maintenance, along with specific test periods.

This technological knowledge acquisition concerns the major components of the NSSS : fuel, sodium circuits and reactor block, steam generators, handling, and monitoring during operation. For example the ALPES system has been implemented on-line at Superphenix. This system, still in the experimental stage, should improve core thermal surveillance by numeric processing of the thermocouple temperature measurements at the subassembly outlets. This system should improve the detection of inadvertent control rod withdrawal (§ 7.3.2).

4.2.2. - Research on plutonium consumption

In the field of research regarding the use of plutonium, the results expected from the operation of the Superphenix reactor are of two different sorts. They concern :

- First, a gradual conversion of the core from a fast-breeder operating mode to a plutonium-burning operating mode, demonstrating the flexibility of the concept, while providing initial experience of the operation of a fast reactor as a plutonium consumer on an industrial scale ;

- Second, the qualification of technical solutions concerning entire fuel sub-assemblies. Such solutions are being developed within the framework of the CAPRA project in particular, in order to provide decisive elements for a reactor of this type.

4.2.2.1. - Evolution towards a plutonium burning mode

The present core of Superphenix is that of a fast-breeder reactor, essentially owing to the presence of fertile uranium oxide blankets, which are arranged radially with 220 subassemblies and axially with uranium blanckets (Fig. 4.1). The programme suggested for converting the reactor to a plutonium burner involves three phases :

- Removal of the first row of the radial blankets of the present core (Scheduled in 1997, Fig 4.1 and Fig. 4.2). This blanket will be replaced by steel subassemblies (Fig. 4.3) The 72 steel subassemblies are being fabricated and will be delivered in the autumn of 1996. Beginning with a core that produces 36 kg/TWh in total plutonium, this stage reduces production to 29 kg/TWh. On the planned cycle, up to the complete burning of the present core, in other words after about one year of operation, total plutonium production will be reduced by 50 kg. Evolution of the plutonium composition will be measured after unloading.

- The complete removal of the radial blankets with the loading of core 2. Planned for the end of 1999, the loading of this core will bring about a plutonium production totalling approximately 15 kg/TWh instead of 36 kg in presence of the radial blanket. This blanket will be replaced by steel subassemblies. The neutronic characterization of this core without a radial blanket will be carried out in order to acquire unique experimental data for the qualification of computer codes on core physics.

- Specific measurements of radial distribution of the neutron flux, particularly at the edge of the core, will be carried out at the loading of this core, much in the same way that such measurements were carried out in 1986 at the time of the first core power increase.

- The perfecting of a core 3 globally adapted to a total plutonium consumption of about 15 kg/TWh (Table 4.1). Such a performance will be accomplished by completely removing all the blankets of the core, radial and axial, and by an increasing initial plutonium enrichment.

SUPERPHENIX	Present core	Core 1 after modification	Core 2	Core 3
Total Pu production rate (kg/TWhe)	36	29	15	-15

TABLE 4.1 : PLUTONIUM BURNING RATE

The experimental programme associated with the planned core modifications mentioned above also involves the following measurements and examinations :

- Reactivity measurements of certain subassemblies and of the core,
- Precise determination of the power delivered by each subassembly,
- Determination of the overall core reactivity coefficients,
- Measurement of the evolution of the decay heat in the core versus time,
- Measurement of the anti-reactivity of the absorber subassemblies,
- Special examinations after irradiation of core elements.

4.2.2.2. - Tests conducted on subassemblies

Studies carried out in the framework of the CAPRA project led to the definition of new fuel elements. They differ from those used in the breeder version in their geometry and their composition, and their plutonium content is much higher.

These new fuels must be able to be used with different isotopic qualities of plutonium, and possible additions of americium and neptunium.

The demonstration programme in Superphenix is aiming at an industrial pre-qualification of solutions which might be selected for CAPRA type reactors. This programme comprises : (Table 4.2)

SERIES	1	2	3
loading	core 1 in 96	core 2 in 99	core 3 in 2003
High burning rate Pu	CAPRA-1A CAPRA-1B (31 % Pu)	2 CAPRA-2 (35 % Pu)	U free Pu pins CAPRA-3 zone (40 % Pu, 19/S/A.) Incl. 1-4 diluents,
Np transmutation	NACRE-1 → (1.9 Kg Np)	CAPRIN-2A (30 % Pu, 3% Np) CAPRIN-2B (+ moderator) NACRE-1 bis (extension)	and 2-8 CAPRIN-3 (35% Pu, 3% Np) with moderator
Am transmutation	ACRE-1 → (1% Am)	ACRE-1 Bis (extension) 1 Am pin loaded 2002	ACRE-3 (Am pins with inert matrix)

TABLE 4.2 : TEST MATRIX

- At the end of 1996, the loading into the reactor of two special subassemblies (Table 4.2) characteristic of the mixed oxide solution with a high plutonium content (31 %)and corresponding dilution of the fuel. Each of these subassemblies was manufactured with a plutonium having a special isotopic composition. The first sub-assembly, CAPRA 1A, contains plutonium from a standard PWR fuel with uranium (Burn-up 33000 MWd/t) ; the second subassembly, CAPRA 1B, was manufactured with plutonium coming from the re-processing of MOX fuel. (This plutonium contains more Pu 238, Pu 240, Pu 242 than the plutonium of CAPRA-1A).

  These two sub-assemblies will be delivered in autumn 1996 with the following basic characteristics :

- A 31 % enrichment in plutonium

- A dilution of the fuel obtained by a reduction in size of the pins, their number being increased from 271 (standard subassembly) to 397, and also by the introduction among these pins of one third of pins made of steel.

- The pins have a PHENIX pin diameter.

- The fuel pellets are hollow pellets.

- The removal of the axial uranium blankets and their replacement by hollow stainless steel blankets.

The programme might also involve :

- At the time of core 2 loading (Table 4.2), the introduction of special new subassemblies destined for a long irradiation period and having characteristics similar to those of the CAPRA 1 subassemblies :

namely

- an enrichement in plutonium of about 35 %,

- a pin geometry closer to the solutions envisioned for CAPRA type reactors, taking into account present day manufacturing limitations,

The programme might also involve (Table 4.2) :

- At the time of core 3 loading, a zone test of several sub-assemblies representing a core with a very high plutonium consumption.

4.2.3. - Research on the destruction of long-lived waste

- Research, especially in the framework of the SPIN programme, has revealed the particular interest that fast reactors present in destroying minor actinides, americium and neptunium, through the fission process, which leads to their transformation, for the most part, into short-lived waste.

The long-awaited demonstrations in Superphenix have a three-fold goal : to confirm the capacity of a fast reactor to satisfy the need to specify the performance of this solution and to provide indispensable elements for the compatibility of the two processes, plutonium consumption and actinide destruction, carried out simultaneoustly in the same reactor.

4.2.3.1. - Incineration (transmutation) of neptunium

The solution best suited to the incineration (transmutation) of neptunium, without creating great penalties, appears to be homogeneous reprocessing in fuel. The calendar of demontrations must be adapted to the possibility of having corresponding quantities of this element and also to the possibility of manufacturing the experimental elements. The proposed programme successively involves the following : (Table 4.2)

- The NACRE 1 sub-assembly will be loaded into the present standard core and will contain 1.9 kilogram of neptunium in a homogeneous dilution (2 % in max), that is to say, uniformly distributed in the fuel of the subassembly. This subassembly, which is being manufactured at the COGEMA in Cadarache, will be loaded into the core at the end of 1996 at the same time as the two CAPRA 1 subassemblies. It will be unloaded with the present core, in other words, after about two years of irradiation. The expected rate of eliminated neptunium in this subassembly is about 20 % during this irradiation time (~.400 g).

- Two special subassemblies CAPRIN-2A (30 % Pu, 3 % Np) and CAPRIN-2B (30 % Pu, 3 % Np + moderator) will be manufactured and will be loaded at the end of 1999. The role of the moderator for a core completely loaded with such SAs would be to increase the Doppler effect.

- An extension of the irradiation of NACRE 1 (NACRE 1 Bis) may be decided upon (Table 4.2)

- Possibly greater quantities in the loading of core 3, a plutonium burner, under conditions defined according to the results obtained from experiments carried out in the Phenix reactor and in during the two preceding stages (Table 4.2).

4.2.3.2. - Incineration of americium

In the case of americium, the situation is more complex due to the different ways of formation of its major isotopes, to difficulties in extracting it when reprocessing and also to its gamma activity which complicates manufacturing operations. For its destruction, the choice of a heterogeneous recycling solution has been selected for the initial stage. The Superphenix programme has a twofold mission :

- To take advantage of the ageing of the fuel elements, and thus the formation of americium 241 through the radioactive decay of plutonium, in order to acquire data on the destruction of this isotope in conditions representative of homogeneous recycling. Before the restart of Superphenix mid-1995, its core contained an average of 0.5 % americium (mass content brought by heavy nuclei) with a maximum content slightly higher than 1 % in 5 of its subassemblies. The restart of the core will bring precious information thanks to a certain number of examinations of one subassembly (named ACRE-1) after unloading of the core in 1999. A second stage might be decided upon with an extension of the irradiation of ACRE 1 (ACRE 1 Bis).

- The fabrication of a few pins specially enriched in americium which may be introduced into a demonstration subassembly loaded in core 2 in 2002 (Table 4.2).

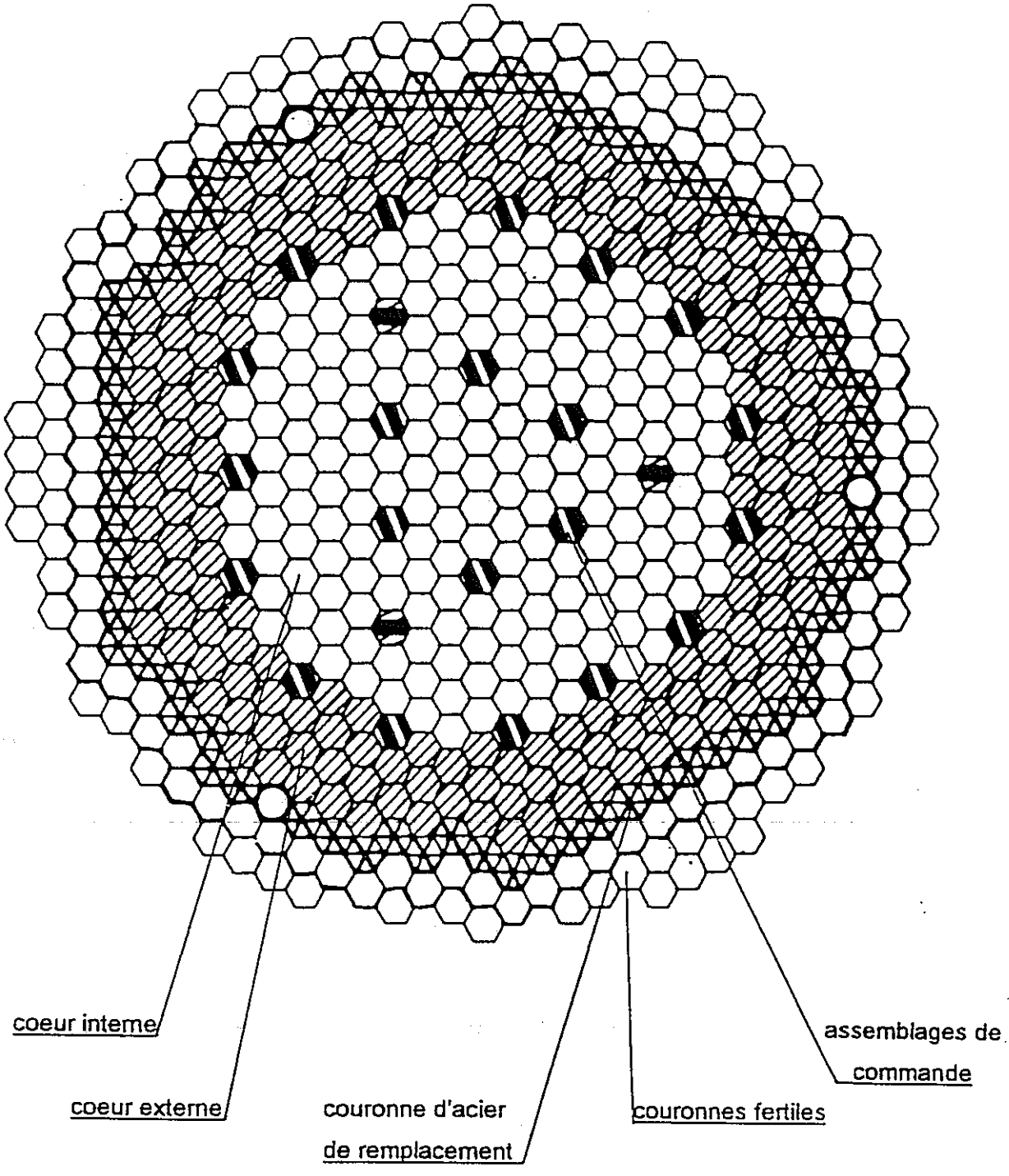


Fig. 4.1. : Present core of Super-Phénix

UP TO NET PLUTONIUM CONSUMPTION

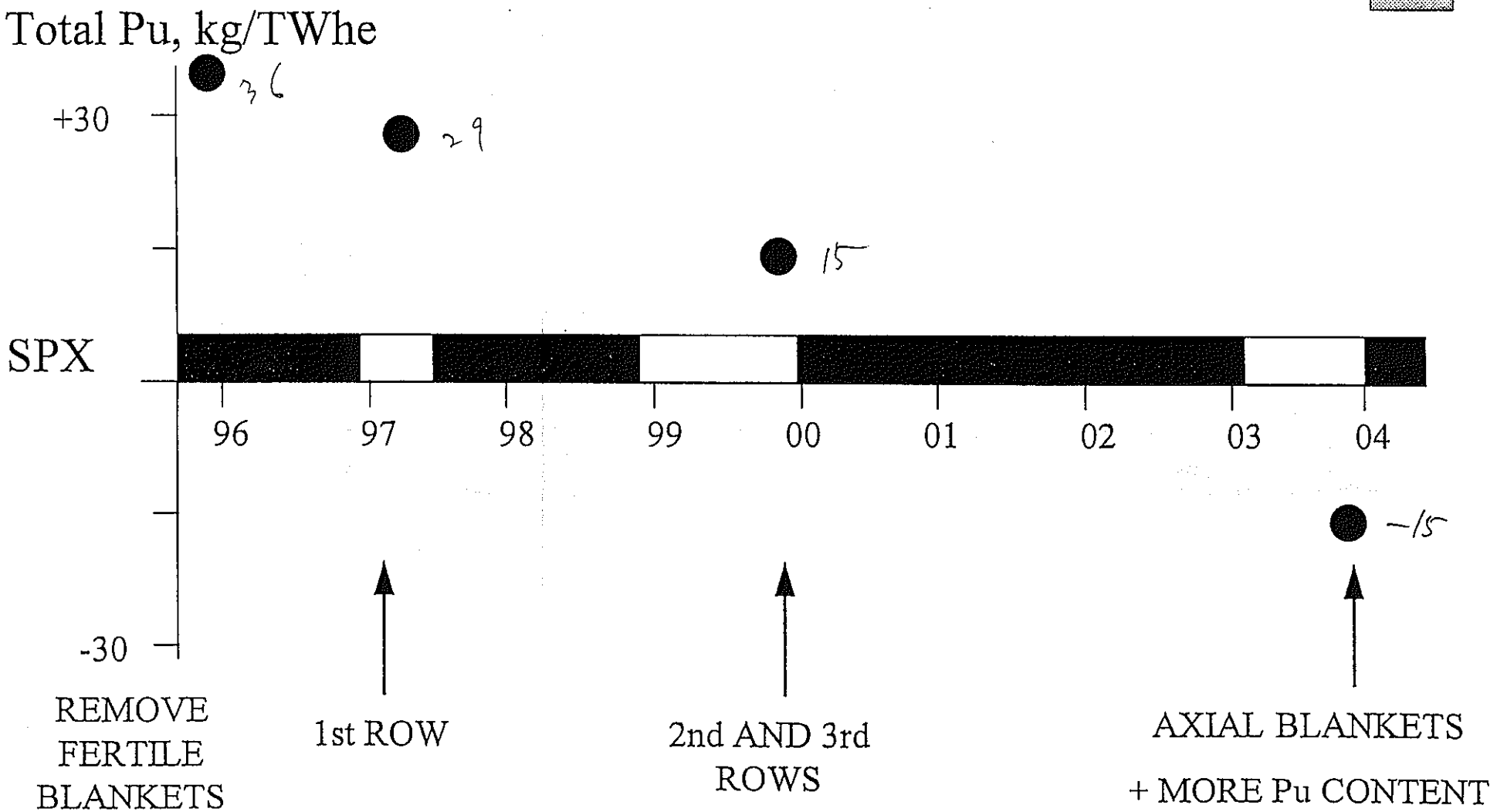


Figure 42

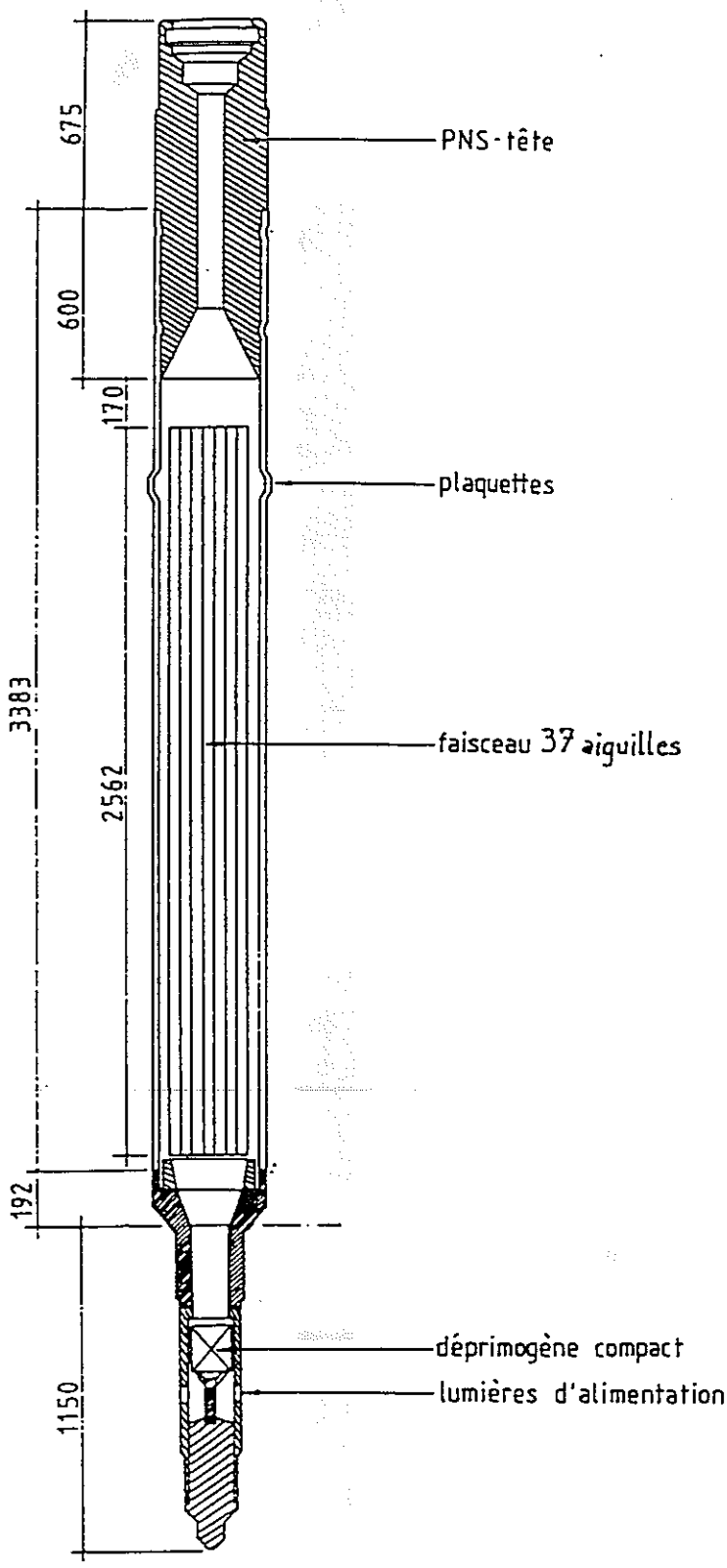


FIG. 4.3. : Steel Subassembly

5 - THE CAPRA PROGRAMME

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This synthesis note summarises the work performed in 1995 in the frame of the European CAPRA Project. It therefore gathers contributions coming from the various project partners BNFL, NNC, AEA-T, FZK, NOVATOME and CEA.

First, let us recall what were the 95 objectives : achievement of the feasibility demonstration of the reference CAPRA 4/94 oxide core; assessment of the possibility of fuel residence time increase; pursuing investigations on U-free cores in view of the characterization of a first "remontage". It was also the year of launching of a new activity on scenario studies. What is today the situation with respect to those objectives :

1 Achievement of the feasibility assessment on the 4/94 reference oxide core

In the 94 feasibility assessment, there were two main points on which the answers were not satisfactorily : some shutdown criteria were not fulfilled i.e. the feasibility of the 4/94 core reactivity management was not demonstrated, and the criticality risk assessment after local melting events was not sufficiently evaluated.

For the first problem, it has been demonstrated that a modification of the diluent S/A implementation associated to a slight adaptation of the relative duration of the two sub-cycles and of the B_4C content in the heavy diluent at beginning of cycle made it possible to fulfil the shutdown criteria with equivalent reactivity margins as for the EFR. The modifications made in the core did not alter its main performances, so that at the present stage of the studies, it does not seem useful to repeat the whole set of characterisations. This new core version is called 4/94m, m for modified.

The second question, the criticality risk, is a generic FRs issue to which CAPRA type reactors are expected to be more sensitive due to the high fuel Pu content. The studies were performed using as initiator of local melting the hypothetical Total Instantaneous Blockage accident. Nevertheless, the conclusions can apply to other possible types of initiators. It has been shown that :

- The melting down of one fuel S/A induces reactivity effects, largely below the βeff, but high enough to be detected by the reactivity-meter if the S/A is located in the inner core. It therefore brings some redundancy in the detection and a better level of prevention. Nevertheless, this does not bring a global answer to the problem so that the melting propagation to the six neighbouring S/As have to be considered.
- In that situation, it has been demonstrated that the reactivity insertion associated to the degraded configuration of seven melt S/As could be counterbalanced by the residual efficiency of the control system.

As a conclusion, complementary studies are still necessary in this area for example the coolability of the degraded configurations, but it is clearly not a feasibility issue for the CAPRA 4/94 reference core. Furthermore, a specific optimisation of the diluent S/A implementation and design could further reduced the criticality risk.

2 Fuel residence time increase

The analysis of the various possible means to increase the fuel residence time (moderating and poisoning, de-rating the linear power by either increasing the fissile height or the number of pins in the fuel S/A) made us conclude that the unique valuable solution is the so-called (moderated) high burn-up core version (in which the important margins on the peak damage dose of the 4/94 are used to increase the fuel burn-up; note that a similar approach is not possible on the EFR). It offers a potential for a significant increase of the fuel residence time (+60% when compared to the 4/94). This core version is now considered as the starting point for a CAPRA core optimized for an increased residence time. Nevertheless, it must be noticed that this optimisation will be made at the expense of the core flexibility to accommodate either a very degraded Pu quality or Minor Actinides into the fuel.

3 Conclusion on oxide cores

One can now consider that the residual pending questions are solved and that the preliminary feasibility assessment of the 4/94 CAPRA reference core is achieved.

In view of the final core "remontages", two directions are worth to be followed in parallel : on one side, a 4/94 type core flexible enough to accommodate a large variety of actinides but with a reduced fuel residence time, and, on the other side, another core, less flexible but optimized for an increased fuel residence time. In both cases, the feedback of the R&D programme becomes essential in order to confirm the basic CAPRA options (fuel Pu content in particular) and to validate and adapt the calculation tools to the particular CAPRA designs.

4 U-free cores

An overall assessment of the possibility to conceive a large U-free core have been made in 95 covering fuel design, physics, core management, transient analysis and the energetic potential in a whole core accident. For that, a particular remontage around an innovative and probably very questionable fuel element design have been used. The general design remains the 4/94 one with diluent S/As and an heterogeneous fuel bundle with fuel-free pins. Even if it is clear that no definitive conclusions could be drawn from that work, the possibility to conceive a large U-free core seems demonstrated. Among the more noticeable results or trends gained so far, we can inoicate :

- the interest to specialise U-free cores in the use of very degraded Pu qualities : the difference in the levels of Pu consumption between an oxide burner (75 kg / TWhe) and a U free burner (115 kg / TWhe) is not so important and can hardly justify by itself the study of Pu without U burners. The idea was therefore to specialise U free burners in the use of those degraded Pu qualities. Physics studies demonstrated that, with this degraded Pu, a near conventional Doppler effect could be recovered when also associated with the introduction of a $^{11}B_4C$ moderator in the empty pins of the fuel S/A (enhancement of Pu 240 and Pu 242 contributions).
- an optimised refuelling scheme based on S/A shuffling has been proposed in order to limit the out-let temperature differences between adjacent S/As below 80°c.
- the choice of the nitride fuel, in a form that still remain widely open, seems to be a necessity if one remains in a logic of a PUREX type reprocessing.

- The innovative annular fuel element (spherical particles of pure PuN are poured in the annular space between an external cladding and an inner tube) must be seen as one possibility knowing that much more conventional fuel concepts can also be considered. Nevertheless, the essential part of the conclusions (fuel management scheme, cores performances, acceptable behaviour under transients,...) remains valid and of a generic character. The innovative fuel element has the merit to show that, when considering U-free fuels, the introduction of an inert matrix is not an absolute necessity and that solutions can also be found in S/A and fuel pin designs.

5 Scenario studies

According to the initial workplan, scenario studies started by the consideration of very basic situations in order to derive, on the basis of pure physics consideration, the relative merits of the various means of actinides recycling. Results gained today only concern the Pu multi-recycling (UOX Pu of first generation) in an oxide CAPRA type reactor and in a 100% MOX PWR with a conventional moderation ratio. Feasibility issues forgotten, the advantage of multi-recycling the Pu in a CAPRA type reactor appears essentially when considering its reduced level of minor actinides (americium, curium) production. Nevertheless, it is clear that further investigation of the physics of PWRs is called for.

6 R&D programme

The irradiation programme has entered its active phase :

- IFOP1 short time irradiation in the SILOE reactor has been terminated end of 94. In 95 examination of this experiment (short fissile column made of 45% Pu content oxide annular fuel) showed a conventional behaviour. Samples are now to be sent for solubility testings.
- CAPRIX1 (2 PHENIX pins containing a 45% Pu content annular oxide fuel) has been irradiated in PHENIX during the first 95 trimester. It is waiting for another cycle of irradiation (96).

- TRABANT1 is under irradiation in the HFR reactor. This experiment contains three short pins : oxide pin similar to the IFOP1, oxide pin with both a high Pu content (40%) and Np content (5%), Pu without U pin filled with the (Pu,Ce)O2 solid solution.
- CAPRA1A and B S/As to be irradiated in 97 in SUPER-PHENIX are under manufacturing.

The first part of the CIRANO programme devoted to the impact on the core characteristics of the removal of the fertile blankets and their replacement with sodium/steel reflector assemblies have been realised. It showes that the newly developed European calculation scheme makes possible the use of improved treatment which are found necessary and well adapted for the calculation of steel reflected configurations.

6 - THE EFR PROGRAMME ALIGNED ON THE EFRUG STRATEGY (STATUS ON APRIL 1996)

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THE EFRUG STRATEGY

During 1995 the strategy for EFR was reviewed with the European Utilities Group (EFRUG) taking the lead with the preparation of a strategy statement which was finally issued in May 1995.

This strategy recognises the success of the previous stages of the EFR studies which resulted end of 1993 in the definition of the "Consistent Design". The economic objectives have been reached and this design was reviewed by a "Ad Hoc Safety Club" (group of European experts).

But considering the general conditions for the development of FRs in Europe, EFRUG decided not to enter a construction phase as previously intended. Actually it is now considered that such a decision would likely only be possible after 2005-2010.

This places effectively the project on hold but in the meantime a EFR Programme has been launched. This continuing EFR programme aims to concentrate on the key aspects of the fast reactor design and technology with the motivations to preserve it for the future and to assess the whole range of its abilities for contributing to the management of the fuel cycle.

If past FRs developments have mainly be oriented on the breeding mission (which remains for EFRUG a long term objective) current European R&D efforts are now mainly dedicated to possible new missions in the back end of the fuel cycle (e.g. CAPRA and minor actinides burning programmes). These developments are integrated into the EFR programme considering that, in any case, the technology is common to the different missions.

In addition to the clarification of the possible roles of FRs in the fuel cycle, a future decision on a demonstration FR in Europe requires :

- several years of successful operation of Superphenix : restart decided in 1994 with the "Knowledge Acquisition Programme" gives the opportunity to accumulate useful operating experience and feedback to future design,
- the restoration of the confidence in the technology and the demonstration that it is possible to progress further in key issues such as In Service Inspection and Repair,
- the preservation of the expertise for the future, notably thanks to a wide international collaboration.

The EFR programme contributes to these objectives.

EFR PROGRAMME: OVERVIEW

This programme is centred on the EFR Reference Design developed from the Consistent Design which was the outcome of the EFR Phase 2 (Dec. 1993) by progressively incorporating technically and economically effective design changes. There is also flexibility in the programme so that in parallel to this development of the Reference Design a broad range of alternative and innovative concepts will be

considered which give the possibilities to assess new ideas or which have a longer term potential.

Flexibility will be retained to accommodate a range of core designs with the minimum of adaptation of the reactor design for breeder and burner cores. The strategy recognises that there is no longer a need for high breeding gain and that in the medium term there may be interest in plutonium consumption (in addition to possible interest in minor actinide burning, whatever the Pu management). But the most probable future extends from moderate Pu consumption to moderate breeding and the ultimate interest of the utilities remains for a breeder reactor which has to be an economic power generator. It is therefore important that all technological developments maintain compatibility with this breeder mission.

An objective of the programme is to promote international collaboration through a programme of broad interest which includes elements aimed specifically at developing the collaboration.

THE WORK PROGRAMME

There are six main design areas for the programme:

A) Reference Design,

B) Innovative Concept studies,

C) Generic studies,

D) Specifications, Codes and Standards,

E) Feedback from Operating Experience,

F) International collaboration.

The basic studies under the heading "operating experience", "generic" and "innovative design" produce results which will be progressively introduced in the Reference Design. The generic studies correspond to areas of broad interest so have elements which form part of the international collaboration. *Specifications, Codes and Standards* and computer models developments are also of interest for a wide range of applications.

The progressive evolution of the Reference Design by integrating the outcome of the Generic and Innovative studies will be judged by the success in improving the economics, safety, robustness or inspectability / reparability of the design.

It is also important to learn the lessons from the operating fast reactors so the programme includes a systematic review of the design of each system and component in the light of the European Reactors experience *(Feedback from Operating Experience)*. Initially particular emphasis is being put on Phénix, PFR and SPX to make sure the information is available for the systematic review. It is intended to extend the reviews by considering further other operating reactors worldwide.

During 1995 the first systems which were reviewed were the absorber rod drive mechanisms and this will be followed by the IHX's and main pumps and then the sodium and argon cover gas auxiliary circuits. Each review consists of :

- detailed comparison of the design features of EFR with that of past FRs,
- record and analysis of significant events and operational difficulties which called for effective or desirable design changes on these reactors, or on the contrary successful design and operation of systems,

- technical synthesis with recommendations of feedback on the EFR design for improved performance.

Integration of the CAPRA core may lead to two variants of the *Reference Design* (breeder and burner) so as to avoid compromise for either mission as an economic power generator. Finally the programme also includes consolidation of the novel features of the Consistent Design notably concerning the solid reactor roof and the vault.

The practice of the EFR project continues to be pursued, i.e. to have a secure reference design with the introduction of well validated improvements. However, the *Innovative Concept Studies* were introduced to ensure that the scope of the whole programme was not inhibited by the extent of validation needed or the timescale for introduction of new features in the Reference Design. The scope of this work is twofold:

(i) development of the EFR concept retaining the same essential features, i.e., sodium cooled pool type fast reactor,

(ii) evaluation of alternative basic concepts, i.e. other sodium cooled FRs and FRs using alternative coolants.

For the first, innovative design features of EFR are examined, a number of them under the general theme of the 'Machined Reactor'. Some features, individually or as a package, have the potential to be incorporated in the Reference Design in the medium term, e.g.,

- high NPSH pump / bottom axial inlet pump,
- core support by conical shell / conical redan / pipework feeding the diagrid.

For the second, the basic concept options of fast reactors will be reviewed in the light of current priorities, e.g. great importance was put on achieving a high breeding gain, and now the emphasis is on flexibility of fuel utilisation and inspectability / reparability. The scope of work includes :

- a review of the sodium cooled reactor options : bottom supported pool type, conventional loop type, top entry loop type, reactor without intermediate heat transfer system, and others
- a review of alternative coolants: lead, gas, water steam, supercritical water.

The *Generic Studies* address fundamental fast reactor concerns and are of interest generally to sodium cooled fast reactors. The objective is to propose improvements and/or to increase confidence which can be applied to the EFR Reference Design. The tasks being performed are also of relevance to international collaboration and particular aspects are being promoted as joint tasks. The tasks include :

- Core safety: WCA prevention and computing tools,
- Containment enhancement,
- Sodium fire prevention and mitigation,
- DHR function optimisation,
- Development of design & construction codes and computing tools,
- ISI & Repair capability improvement.

Due to :

- the uncertainties concerning the date of implementation of series FRs,
- the necessity to face short term requirements and to share the development cost,
- the trends and incentives to come to worldwide agreement on the safety specifications and technical characteristics of future reactors,

the development of the *International Collaboration* is even more essential than in the past. Agreements between European and Japanese R&D organisations and Utilities, those with MINATOM (Russian organisation), are the main basis for the future developments during the next years.

MAIN MILESTONES

The outcomes of the Innovative Concept and Generic Studies and the feedback of operating experience will be introduced in the EFR Reference Design as they become available. A review of the basic concept options for fast reactors against current conditions and priorities will be completed by 1998. Recommendations on the innovative options will be made by 1998 on the basis that either they can be introduced in the EFR Reference Design or are worthy of being pursued further. The decisions on the Innovative Concepts together with the outcome of the Generic Studies allow a comprehensive update of the EFR Reference Design to be completed two years later - at around the year 2000.

7 - R AND D.

7.1. Fuel and Absorber Elements, Core Materials

Irradiation progress.

As a result of the start-up of PHENIX on December 24, 1994, for its 49th cycle, the leading subassembly with reference cladding (CW 15-15 Ti) and an EM10 wrapper, reached 161 dpa .

Fuel modelling.

Certain models of the thermomechanical GERMINAL computer code were improved on the basis of PHENIX pin results and some specific CABRI tests :

- Improvement of the "fission product layer between fuel and cladding" (JOG),
- Improvement of transient swelling phenomena ; these phenomena may induce a reduction of the diameter of the central hole of some pellets and an increase of fuel porosity and consequently the degradation of fuel thermal conductivity (see § 7.3.3),
- Improvement of the radial redistribution of the Pu.

CAPRA fuel.

Theoretical and experimental studies concerning the hydraulics and thermalhydraulics of CAPRA subassemblies were carried out with respect to the design of these subassemblies.

The characterization of mixed fuel with a high Pu content is in good progress. Fuel fabrication in the (40-45 % Pu) range is established, as well as fuel solubility in nitric acid. Also good progress was made in the fabrication and the characterization of CERCER fuel MgO/PuO2 (fuel without uranium).

The irradiation programme has entered its active phase :

- The IFOP1 experiment in the SILOE reactor was completed in 1994 with a 1.15 at % BU. In 1995 examination of this experiment, short fissile column, annular UPuO2 (45 % Pu fuel), showed conventional behaviour. Samples are now to be sent for solubility testing.

- CAPRIX 1 (2 PHENIX annular (UPu)O2 pins containing a 45 % Pu) were irradiated during the 49th cycle of PHENIX. This experiment requires an additional irradiation cycle.

- TRABANT 1 is under irradiation in the HFR reactor. This experiment contains three short pins : one oxide pin similar to the IFOP1 - one oxide pin with a high Pu content (40 %) and a 5 % Np content, one Pu without U pin filled with the (Pu, Ce)O2 solution.
- CAPRA 1A and 1B S/As, to be irradiated in 97 in SUPER-PHENIX, are being manufactured (see § 4.2.2.2).

7.2. - Core physics

7.2.1. - Nuclear data and the "unified code system" ERANOS

- The current ERANOS nuclear data library has been built from the JEF-2.2 evaluated microscopic data library. Results of measurements in MASURCA have been selected to assess the performance of this unadjusted library - The overall results of the comparison are satisfactory - However, better agreement is expected of adjusted data. In addition, application of the adjusted technique may reveal the existence of compensating errors. An adjustment method was used in the past and has given satisfactory results. Consequently it was decided to acquire an adjusted library (ERALIB 1) using this process in 1995. The final adjustment of ERALIB 1 has been delayed due to difficulties during the final validation operation.

- The development of the "unified code system" ERANOS has been satisfactory. Version 1.2 should be released in 1996.

  The overall numerical qualification of ERANOS was started, using comparison with MONTE-CARLO calculation methods.

7.2.2. - The CIRANO programme

The CIRANO programme is one of the various programmes carried out to demonstrate the feasibility of using a fast reactor to burn as much plutonium as possible.

In order to reach this goal, the U^{238} content must be as low as possible. Consequently, with respect to breeder cores , this requires :

- Removing the fertile blankets,

- Increasing the plutonium content,

- Substituting inert material for the U^{238}.

The CIRANO programme is conducted at the MASURCA critical facility, the first issue currently being studied.

Only a few experiments were performed without fertile blankets and stainless steel reflectors and it seems that the discrepancies between the experiments and calculations increase as the thickness of the blanket decreases. Therefore, the first set of studies consists in gradually removing blankets from ZONA 2A (classical UO2 + PuO2 Core) as in shown is the following schedule :

April 1994	ZONA2-A	reference core (with axial and radial blankets) (Fig. 7.1)
September 1994	ZONA2-A3	radial blanket removed and replaced by sodium-steel elements (Fig. 7.2.1)
February 1995	ZONA2-B	core without blanket (Fig. 7.2.2)
July 1995	ZONA2-B.SI	core without blanket and with internal storage (SI) : without B4C axial shielding on half the core and with an extension of the UO2/Na radial reflector on the same half of the core (Fig. 7.3).
October 1995	ZONA2-B.SI1	Same configuration as ZONA-2B-SI, but fuel subassemblies set up in the internal storage area (12 tubes).

December 1995 ZONA-2B.SI2 Similar to ZONA 2-B SI.1 but with a protection of B4C SA between the core and SI location.

- From 3/96 to late 96, a series of substitutions will be performed in the central part of the core. Plutonium enrichment will vary from 25 to 50 %, and Pu isotopics will cover a wide range of values. (The Pu-240 content will be varied from 8 to 35 %).
- In 1997, a mock-up of a CAPRA "dilution" core will be constructed (ZONA5) The bare core will first be fully characterized. Subsequently, all heterogeneity's described earlier will be studied : control rods, diluants, heterogeneous sub-assemblies.
- Finally, starting in 1998, a large core will be simulated.

Fig. 7.1.

Cellule Combustible ZONA2 (PuO2/UO2/Na)

Na	PuO2	Na	PuO2
PuO2	Na	PuO2	Na
Na	PuO2	Na	PuO2
PuO2	Na	PuO2	Na

Na = Réglette Sodium

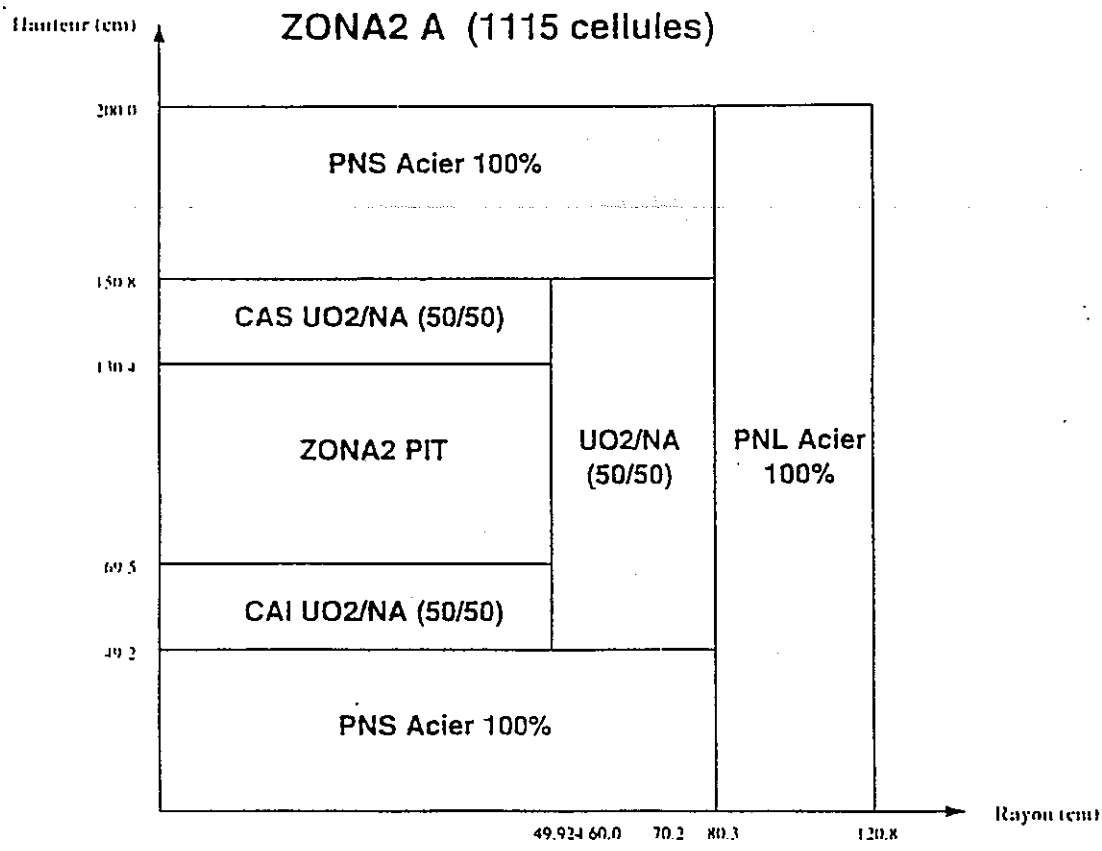
PuO2 = Réglette Plutonium Uranium Oxyde (E=25%)

Cellule Couverture UO2 Sodium (UO2/Na)

Na	UO2	Na	UO2
UO2	Na	UO2	Na
Na	UO2	Na	UO2
UO2	Na	UO2	Na

Na = Réglette Sodium

UO2 = Réglette Uranium Oxyde



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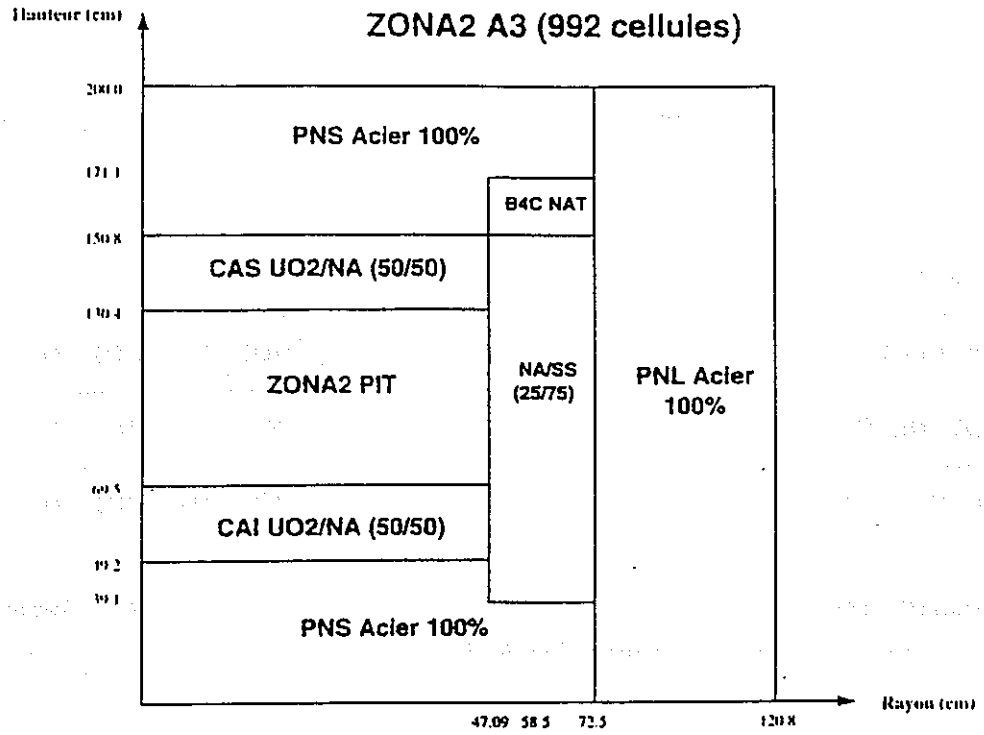


Fig. 7.2.1

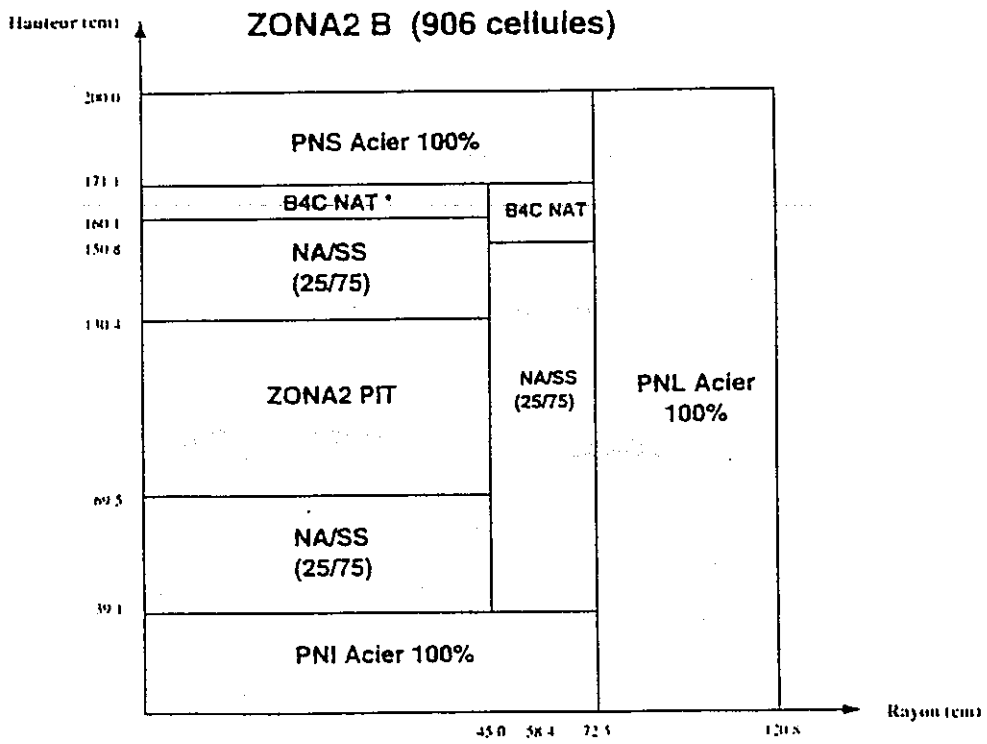
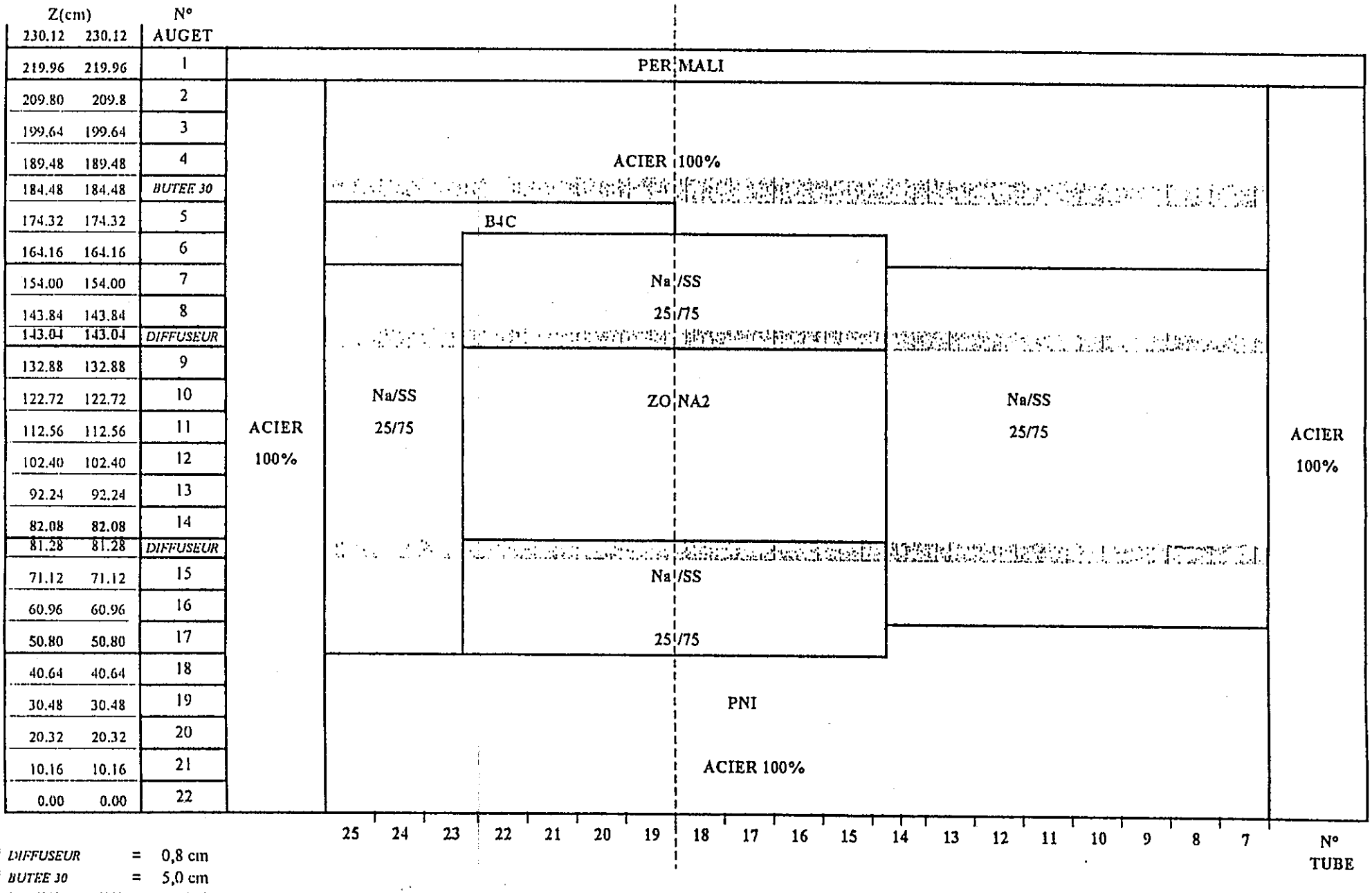


Fig. 7.2.2



* *DIFFUSEUR* = 0,8 cm

* *BUTEE 30* = 5,0 cm

* les diffu. les diffuseurs et la butée 30 sont présents partout sauf dans l'écran radial.

Fig. 7.3. : CIRANO ZONA2B
Internal Storage REF

In the CIRANO tests, the following measurements were performed :

- Critical masses : the experimental uncertainty on the critical mass is given as 300 pcm (at 1)
- Fission rate traverses : they are performed for the fissile isotopes : U235, U238, Pu239 and Np237 using fission chambers or activation detectors for two radial traverses and 20 axial traverses distributed throughout the cores.
- Spectral indexes : using the relative fission rate measurements described above and an absolute index measurement determined as the ratio to the U235 fission rate performed at the centre of each traverse. In addition, the indixes for the higher Pu isotopes, as well as Np237, Am241 and Am 243 isotopes were measured in a central position using fission chambers placed in removable pins. The experimental uncertainties are given to be between 25 % (at 1).
- In addition, shielding traverses using Au, Mn, In and Rh detectors, gamma heating traverses using ionisation chambers and TLD detectors, and sodium void reactivity have been performed.

Detailed interpretation of the data is in progress.

The main results obtained when gradually removing blankets from the ZONA-2A reference core show :

- An increase in the reflection of lower energy neutrons back into the core. This has an effect on the critical mass, rod worth, and power distributions,

- An increase in the neutron flux in the shield, internal storage subassemblies, and beyond (heat exchanger). This in turn leads to increased damage to certain components, increased secondary sodium activation, and increased heat deposition in the shield and internal storage regions.

7.3. - Safety

7.3.1. - Reactor dynamics in normal and design basis accident conditions

The development of the OASIS project "An Interactive Simulation Case" began with the idea that, within the framework of the Fast Reactor Programme, there was a definite need to have an analytical tool assisting personnel in coming to a decision in a situation of potential crisis. Such a tool would be specially designed for the team in charge of crisis situation management. This team would have to decide the procedures to follow in the case of a severe accident in a nuclear reactor. Therefore the tool must be able to accurately analyse a situation and to give its probable evolution

During the tool specification stage it soon became obvious that, with the emergence of the "work stations", it would be quite possible to produce simulators that could not only meet the specific needs of the crisis situation management but also satisfy the needs of engineers working either on project design or safety design. These considerations led to the final orientation of the OASIS project (OASIS standing for "Simulator tool for safety analysis") Its goals are to study the behaviour of either existing reactors in various design basis accident conditions or the first stages of hypothetical accident conditions (before loss of core geometry) and to help engineers in the design phase of reactors.

OASIS is today qualified on the SUPER-PHENIX configuration and is used by all French partners working on this plant : CEA, EDF, FRAMATOME.

At present the work consists in completing this work on other reactor configurations : PHENIX and EFR.

Moreover, improvements are still necessary, mainly regarding the water-steam circuit.

7.3.2. - Core surveillance

The ALPES system has been implemented on-line at SUPER-PHENIX. This system although, still in the experimental stage, should improve core thermal surveillance by the numeric processing of the thermocouple temperature measurements at subassembly outlets. Furthermoore, this system should improve the detection of inadvertent control-rod withdrawal. In order to determine the performance of the ALPES system with respect to the detection of this accident, control-rod insertion tests will be performed in SUPER-PHENIX in 1996.

7.3.3. - Fuel pin safety tests in CABRI

The CABRI-FAST international programme, CEA-EDF, PNC (Japan), has been completed. In this programme the behaviour of annular fuel cladded with the improved (low swelling) clad material 15-15 Ti ε (CW 15-15 Ti) was studied under slow ramp and TUCOP conditions.

The test matrix and characteristics of the pins are shown in the following tables .

	1992	1993	1994	1995
CABRI-FAST TEST MATRIX	BCF1 (12/92)	PF2 (3/93) PF1 (12/93)	LT2 (3/94) EFM1 (5/94)	PFX (3/95) MF2 (11/95) LT1 (12/95)

TEST	OBJECTIVES	PIN
<u>Slow ramp rate tests</u>		
PF1	Pre-failure fuel behaviour (1 % Pn/s)	SCARABIX
PFX	Determination of power to melt (1 % Pn/s)	"
MF2	Margin to failure (1 % Pn/s)	SCARABIX
BCF1	Beyond clad failure - Ramp effect (3 % Pn/s)	VIGGEN - 4
<u>Initiation phase test (CDA)</u>		
PF2	Pre-failure behaviour - Fuel squirting	QUASAR
LT2	TOP. Fuel squirting	QUASAR
LT1	LOF-TOP. Fuel squirting	QUASAR
EFM1	Extended fuel motion	SCARABIX

Pn means Nominal Power

NAME	TYPE	CLAD	D_{out} mm	Bu at %
VIGGEN - 4	PX homogeneous solid (no UAB)	15-15 Ti	6.55	11.7
QUASAR	PX homogeneous annular (no UAB)	15-15 Ti	6.55	11.8
SCARABIX	SPX homogeneous annular (UAB)	15-15 Ti	8.5	6.4

Three tests were carried out in the CABRI reactor in 1995 : PFX, MF2, LT1.

PFX test

In the event of a slow power ramp the fuel can swell, due to an increase in the macroscopic porosity of the fuel, with, as a result, a reduction of the central hole and degradation of fuel thermal conductivity. These phenomena lead to a power to melt lower than predicted with modelling which does not take these phenomena into account.

These phenomena have already been observed in previous tests : E9 and E9 Bis (CABRI II programme) fuel, and PF1. Nevertheless, at the end of these tests there was a rather significant melting extension (particularly E9 and E9 Bis), which did not allow precise determination of the power to melt.

In order to determine the power to melt in the CABRI-reactor, the PFX test was performed on a SCARABIX pin (annular fuel at 6.4 %, 15-15 Ti clad) under slow ramp rate conditions with a maximum power calculated to reach fuel melting. A very small molten fuel fraction was observed in the post-test examinations, which allowed precise determination of power to melt.

MF2

This test was performed in order to determine the margin to failure of a SCARABIX pin under a slow power ramp simulating a control rod withdrawal (≃ 1 % Pn/s). No clad rupture was observed at the end of the transient (≃ 1240 W/cm, which is the maximum CABRI performance for this type of fuel pin and for this transient). Internal fuel motion was observed with the hodoscope. This test confirms that hollow pellets can withstand a large melt fraction without failure.

LT1 Test

The goal of this test was to study internal fuel motion (fuel squirting) during a TOP.

The conditions of this test were similar to those of the LT2 test regarding the pin (QUASAR pin with no UAB) and the energy of the TOP (1.23 kg/g), but the initial conditions of the TOP were different :

for LT1 : initiation of the TOP during Na flow reduction when the sodium temperature was ≃ 890°C at the top of fissile column,

for LT2 : initiation of the TOP with steady state sodium flow and a sodium temperature of ≃ 580°C at the top of fissile column.

7.3.4. - Future CABRI programme

This programme named RAFT (Reactivity Accident Fuel Tests) will be performed in collaboration : CEA-EDF, (France) and PNC (Japan). The objectives of this programme are :

- Determination of the power to melt of high burn-up annular fuel,
- Determination of the margin to failure of high burn-up annular fuel,
- Study of the possibilities of fuel ejection under the following conditions : the induction of clad failure during a slow ramp rate when a low melt fraction is obtained,
- Study of the melting behaviour of an irradiated fuel-pin bundle,
- Study of the advanced fuel under accidental conditions.

7.3.5. - Hypothetical Core Descriptive Accidents

7.3.5.1. - Initiative Phase : SAS4A Code

In September 1995, FZK (Germany)the SAS 4A "code manager" issued the SAS 4A Ref. 94-R1 release. Now the "code manager" is CEA - IPSN. This code manager is in charge of maintenance, Quality Assurance and the introduction of improvements for the benefit of users. SAS 4A is still being developed in close collaboration with FZK, PNC and CEA.

The modelling of in-pin fuel motion was introduced in SAS 4A Ref. 96.R.O issued at the beginning of 1996. This model is being validated on the basis of the CABRI tests : AI3, BI3 and PF2, LT1, LT2.

Other modelling improvements concern :

- Radial O/M redistribution (FZK)
- Plenum gas blow-out (PNC)
- Boiling model (FZK)
- Freezing model improvement (PNC+CEA)

Validation of SAS 4A is still in progress on the basis of previous CABRI tests.

7.3.5.2. - Transition Phase

A transition phase state-of-the-art was carried out. The main phenomena were described and analysed :

- Fission gas influence

- Behaviour of a of molten steel and fuel mixture

- Material discharge.

Regarding the SIMMER III code, being developed by PNC (Japan), the Europeans are participating in the qualification of the code.

7.3.5.3. - Expansion phase

An expansion phase state-of-the-art was carried out and expansion phase calculations are being performed with the SIMMER 2 code and the "Advanced Fluid Dynamics Model" (AFDM code). Results obtained with the two codes are in good agreement.

7.3.6. - Source term and radiological safety analysis

The transport of the fuel aerosols and fission products from the cover gas to the various cells of the reactor building and to the environment is calculated with the CONTAIN-LMR code. In 1995 a modelled exhaust device was implemented in the code. FRAMATOME used the code for FBR evaluation of the source term to the environment and they found that specific models had to be developed and implemented.

7.3.7. - Sodium fires

Experimental work on sodium fire was completed in 1993.

Interpretation of the experimental observations is still in progress using the computer codes PULSAR and FEUMIX.

In order to acquire data to extrapolate the sodium experimental results for large leak rates (a few tons/s) the AIRBUS programme is still in progress. The goal of this programme is to measure the Sih coefficient (interface area multiplied by a heat transfer coefficient) which is used in the modelling of the FEUMIX 3 code. The AIRBUS programme is being carried out in water and air and in similarity with sodium conditions.

7.4. - Technology

7.4.1. Complementary information about the March 31, 1994 accident at Cadarache. *(R. DEL NEGRO and P. MARMONIER, CEA)*

The first information on this accident was given at the meeting of May 1995.

It should be recalled that the accident took place in a basement next to the leaktight containment building of the reactor, at the end of dismantling operations. An explosion occurred during the destruction of 100 to 150 kg of sodium that remained at the bottom of a 55 m^3 tank. This destruction was performed by successive introductions of 50 l of ethylcarbitol. A member of the CEA team was killed and four other people were injured.

In addition to the administrative and judicial inquiries, the CEA set up an internal Inquiry Commission on July 94. The latter concluded on :

- ⇨ The pneumatic explosion of the tank following unknown chemical reactions that occurred in the heterogeneous and multi-phase physico-chemical mixture that had formed during the washing operations, that had been suspended for several days, and during the heating of the system ;

- ⇨ The need to decree : design rules for equipment in order to facilitate its dismantling, organization principles in order to preserve the memory of the risks in the facility, and procedures for implementing dangerous products ;

- ⇨ The suspension of the ethylcarbitol cleaning technique ;

- ⇨ The need to pursue investigations in order to understand what happened and, if possible, to draw up principles to be respected in the future.

The investigation programmes were performed as defined, and provided the first results summarized below :

- ⇨ The accident area was cleared and the information gathered confirmed that there was pneumatic explosion of the tank, with a release of 200 MJ of energy (that is to say the equivalent of about fifty kilograms of TNT) ;

- ⇨ The chemical reactions and the fireball corresponding to the ignition of the gases released into the air did not constitute a significant energetic input ;

- ⇨ Metallurgical analyses, althrough not yet completed, have permitted the tearing lines in the tank's metal plate to be located and their origin situated ;

- ⇨ The chemical analyses focussed on :

    - Analysing samples taken just after accident in the residue, and from the venting and ventilation circuits ;

- Attempting to identify the reactions that may have occurred ;

- Identifying and quantifying the gases produced when ethylcarbitol, ethylcarbitolate, a mixture of these two products, or a mixture of the products with the decomposition products generated by the various reactions, is heated. The abundant production of hydrogen was confirmed, in addition to the hydrogen expected from the reaction of the ethylcarbitol to the sodium, besides concentrations of carbon, methane, and carbon dioxide that increase when the temperature increases ;

- Defining the energies released as a function of overheating conditions. As long as the temperature remains below 220 or 230° C, exothermic reactions are slow, beyond this temperature, temperature increases of several hundred degrees in very short periods of time were measured ;

- Identifying the residues produced as a function of the temperature. These residues are of variable aspect and consistency, depending on the temperatures reached. They contain numerous, not yet identified intermediate products (up to 50 % for reaction temperatures ranging from 220 to 390°C), oxalate, carbonate, and amorphous carbon in varying proportions.

It was demonstred that isothermal exposures of between 210 and 230°C can lead to exothermic reactions with an incubation time that increases as the temperature decreases. It was also recorded that some reactions could occur at room temperatures.

Future tests are in progress, focussed in particular on :

⇨The initiating cause of the runaway chemical reactions,

⇨The catalytic reactions that may occur.

Although these results confirm the initial conclusions of the internal Inquiry Commission, it has been shown that the chemical reactions at the origin of the accident are numerous and complex. They had not been explicitly described before 1994 and even today not all have been clearly defined. The conditions and causes of the runaway reactions that may occur even starting at ambient temperature, have to be specified.

The follow-up of the programme should provide results that will enable these phenomena to be quantified and thus clarify the feedback from this accident and the knowledge acquired since that time.

One should, however, particularly stress the need to maintain the suspension of the implementation of ethylcarbitol cleaning until definite conclusions have been drawn from both the analyses now in progress and future studies.

7.4.2. Sodium technology.

7.4.2.1. - Sodium leak detection - Stainless-steel corrosion mechanisms

Activities on sodium leak detection and corrosion involve :

- The FUTUNA experimental programme, and,
- Based on the FUTUNA results, mechanisms of the transport of sodium products from the leak and pipe stainless steel corrosion are being studied.

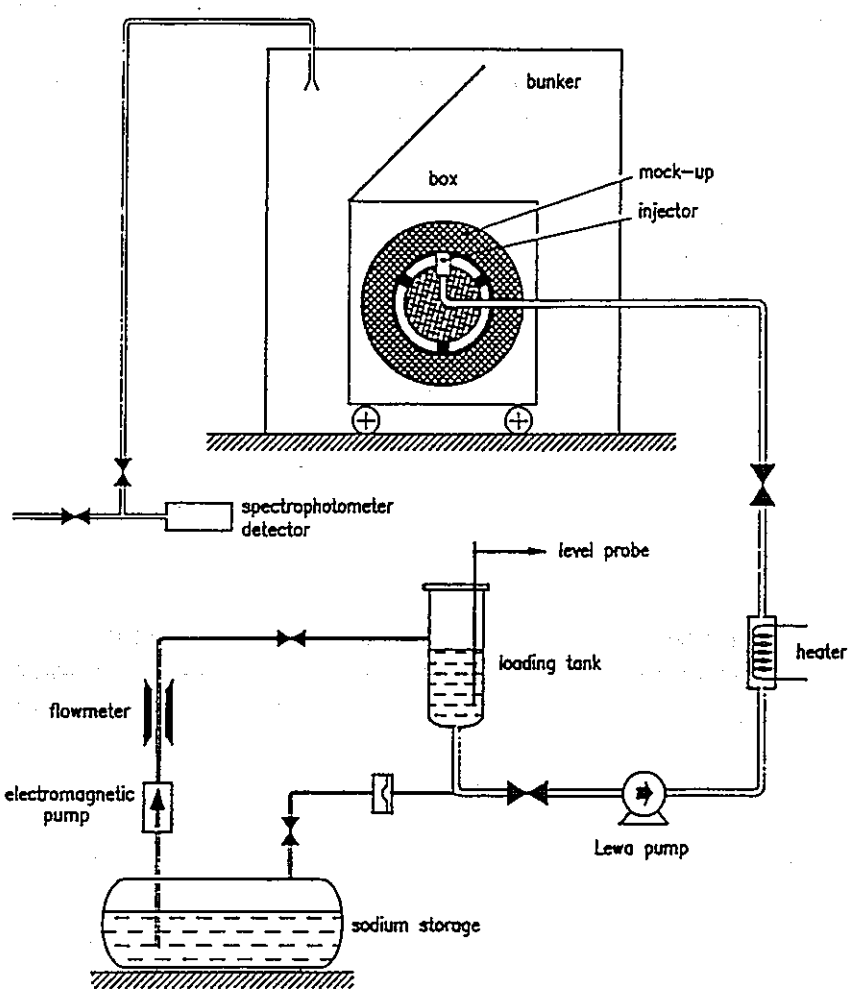
The FUTUNA experimental programme

More than thirty FUTUNA tests have been carried out since 1989. The aim of this programme was to provide data on sodium leak detection. Classical leak detection systems (beaded wires, short circuit on trace heaters, flame spectrophotometer), "sandwich" detectors (metallic foil placed between two perforated silicoaluminate layers), and aerosol detection, were tested on pipe sections.

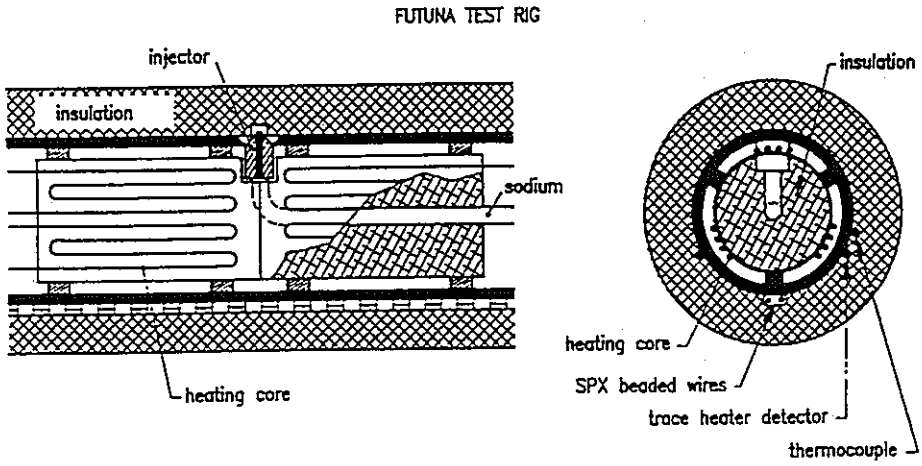
Moreover, specific tests on flat plates were carried out mainly to provide data on sodium product transport from the leak to the corrosion area and data on the corrosion phenomena.

A sodium leak of constant and known flow was produced either on the top of a pipe (Fig.7.4) or on the upper flat plate surface, both covered with mineral wool insulation. The tests were carried out with sodium leaks of less than 1.5 cm^3/min and are representative of small leaks. The wall temperature ranged from 180°C to 550°C ; most of the tests were carried out at 550°C and the duration of a test did not exceed 550 hours. The pipes or plates were heated by trace heaters with a temperature regulation (Fig. 7.4).

The pipes were representative of SUPER-PHENIX secondary loops. The 316 stainless-steel plates were 1 m x 1 m, thickness 10 mm. Pipes and plates were covered with six layers of mineral wool insulating material (total thickness 250 mm).



The FUTUNA tests facility



Sodium pipe

Fig. 7.4

Experimental results show that the corrosion is limited to a peripheral annular area which extends outwards in concentric waves. The diameter of this corrosion area is practically constant. Furthemore, the tests show that the sodium does not flow directly onto the metallic surface. The sodium is sucked through the mineral wool insulation and divided into droplets due to capilarity forces. Chemical reactions between silica fibers, occluded oxygen and water vapour occur during diffusion transport of the sodium. Finally, there is deposition of sodium droplets charged in oxygen.

The study allows the electrochemical nature of the corrosion to be proved. The droplets charged in oxygen induce an anodic dissolution mechanism by differential aeration. This phenomenon explains the random microscopic and macroscopic aspects of corrosion.

7.4.2.2. Other items

- Experimental and theoretical studies with respect to the optimization of the sulfophosphoric decontamination process, especially for IHXs, were completed. The aim was to lower the phosphoric acid content ; this acid prevents intergrannular corrosion, but leads to undesirable phosphate in the waste. The process is at present being used for the IHXs in PHENIX.

- The study of the behaviour of impurities in sodium is still in progress. An experimental programme concerning chromium and iron impurities is being completed. It will provide information regarding the identification of corrosion products, their transfer mode and behaviour according to oxigen content and temperature.

7.4.3. In-service inspection and repair.

7.4.3.1. - Main vessel inspection.

The MIR remote-controlled inspection device (Module d'Inspection des Reacteurs Rapides) (Fig. 7.5) was developed to monitor, throughout the lifetime of SUPER-PHENIX, the surface and internal defects on the main vessel of the reactor. Improvement of the MIR device is still in progress :

- Development of transducers with a piezo-composite material which should improve the control for thick (up to 60 mm) butt welds,

- Improvement of analysis method in order to determine the characteristics of the defects in such butt welds (data were obtained on a mock-up with the transducers used in the 1988 inspection of the main vessel of Superphenix),

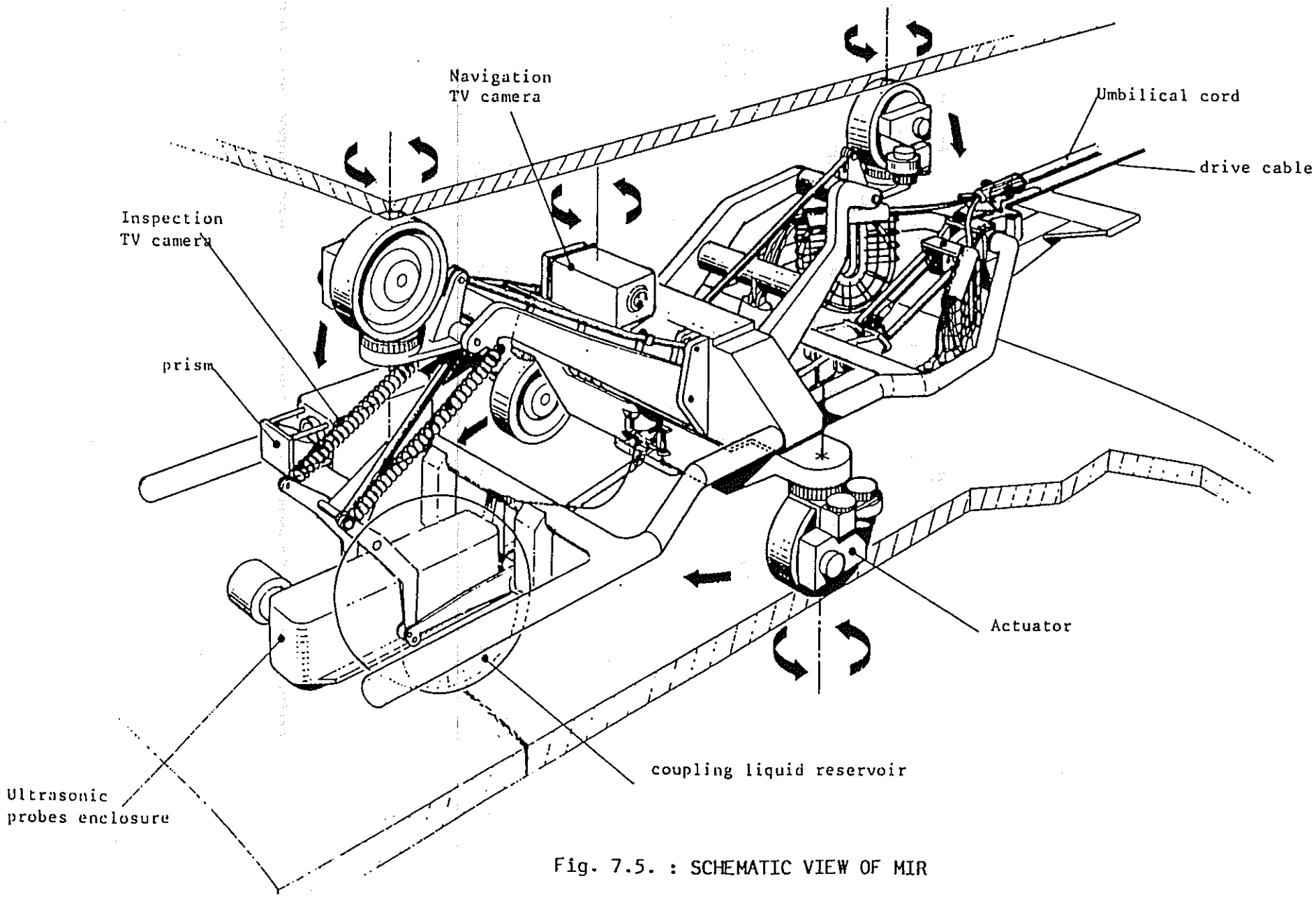


Fig. 7.5. : SCHEMATIC VIEW OF MIR

- Improvement for the inspection of the "triple point" (where the core support structure rests on the main vessel) with respect to surface (emerging) defects. The procedure consists firstly in detecting the defect with an eddy current probe, then in dimensioning the defect with U.S transducers.

- Coupling with current specifications (inspection of all welds, inspection of internal structures through the main vessel) has led to increased requirements for ultrasonic transducers, emphasing the interest of array transducers. The first application of such transducers is being performed on mock-ups.

7.4.3.2. - Structural monitoring in sodium

- Within the scope of non-destructive control in sodium, a few materials were selected after compatibility tests in sodium (T = 200°C). These materials will be used in the design of U.S transducers generating narrow pulses in sodium ($\simeq$ 1µs). A review of electromagnetic ultrasound transducers (EMA) had shown that, up to now, they were not used in sodium. Work is being performed to determine their possibilities in this field of application.

- Ultrasonic telemetry in sodium is still in progress, mainly using signal processing. A numerical model from the literature (LAHMERY, published in the Journal of Acoustic Society in America) was applied to increase the accuracy of sound velocity measurements using small targets (a few millimetre in diameter) at a known distance ($\simeq$ 50 cm). Experimental verification will be carried out in 1996.

7.4.3.3. - In-gas telemetry.

- Laser telemetry studies in air, including thermal perturbations representative of the plena above a sodium pool, showed that the accuracy of the measurements is better than $\pm$ 3 mm on distances of a few meters.

  Experiments above a sodium pool will be performed in 1996.

7.4.3.4. -. Above-sodium repair.

- The MIRSA facility ("Maquette pour Interventions et Reparations Sous Argon") which is a facility (Fig. 7.6) for Intervention and Repair in Argon above a sodium pool was used to test welding techniques. A special tool was designed to study "Tungsten Inert Gas" welding engineering. In a first series of tests in argon at 130°C, a bead was deposited on sheet steels that were either clean or wet with sodium.

- Tests in a glove box were carried out to clean sodium-wetted surfaces using hot argon (180°C) blowing. A special tool is being designed to use this process in the MIRSA facility (in 1996).

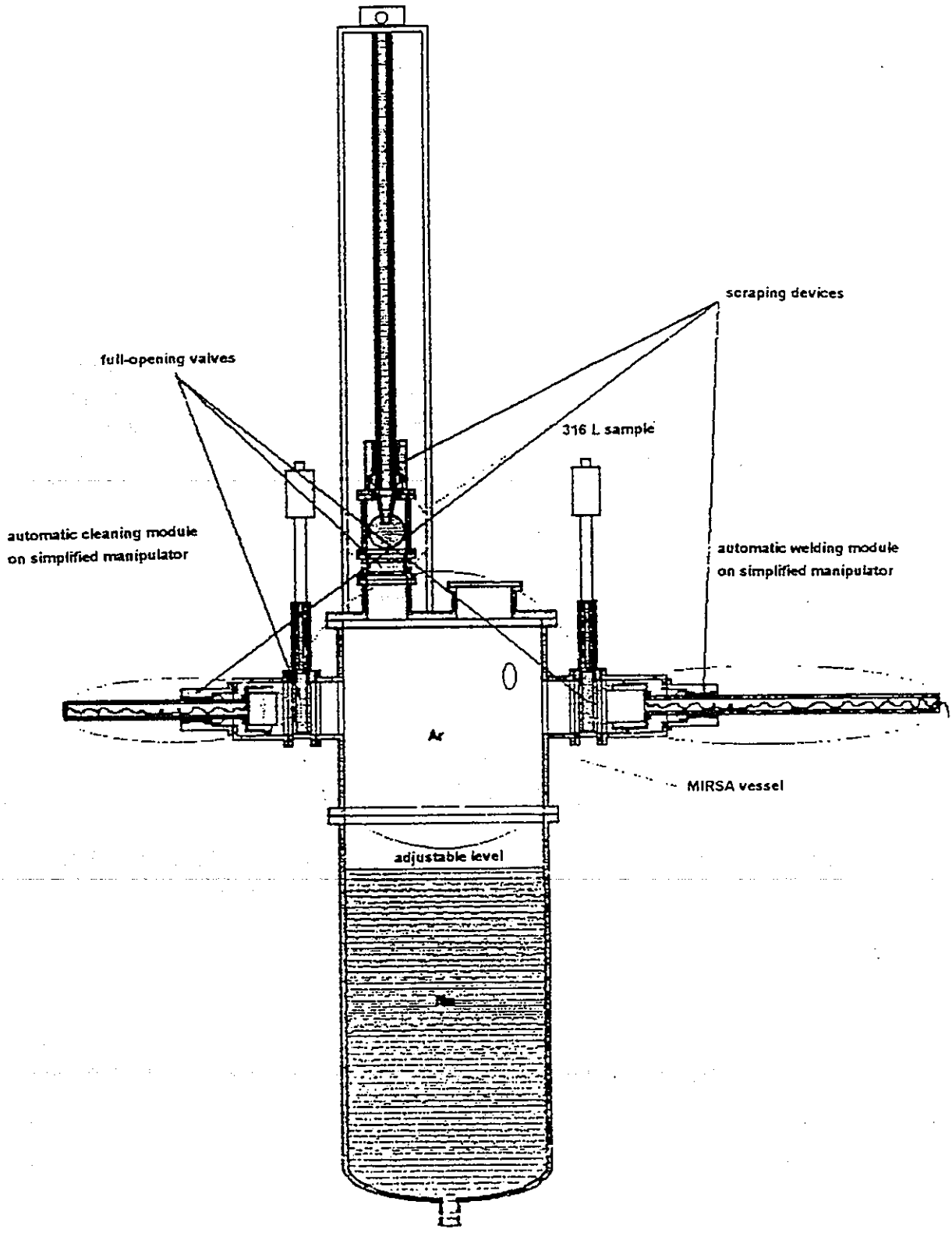


Fig. 7.6. : MIRSA facility

- A laser process to clean sodium-wetted surfaces sodium was tested in a glove box under a nitrogen atmosphere. In this process a residual layer of 0.035 μm is obtained (half the thickness obtained with hot argon blowing) and the steel surfaces remain intact.

- Cuttings with a YAG 1 kW pulse laser were performed in one cut on steel sheets up to 25 mm thick. High quality cuts were obtained, but the recovery of debris is difficult. In order to avoid this problem a multicut process is being tested.

7.4.4. - Steam generators.

7.4.4.1. - ATLAS monotubular rig (Fig.7.7)

The ATLAS monotubular rig, with a straight tube, was designed to obtained detailed data on heat transfer, pressure loss, dry-out correlations and flow instabilities in view of validation of the thermalhydraulics case of the EFR steam-generator units. The ATLAS rig was connected to the steam-water and sodium supply system of the SET facility at CEA-CADARACHE. The programme was successfully performed in 1995 with a series of 34 tests. Sensitivity studies on vapour pressure, flowrate and inlet temperature of sodium and water around nominal conditions (vapour pressure (185 bar)), in a large power range (20, 50, 80 and 100 % Nominal Power) were carried out. In addition, a series of experiments was performed at "low pressure" (60, 80 and 120 bars).

Analysis of the data is in progress.

7.4.4.2. - Sodium-Water and leak Detection

- A review of data available worldwide combined with the CEA data base resulted in the development of a new version of the wastage laws with modelling of the various characteristics of the wastage impact profile (with determination of central wastage speed, and toroïdal wastage speed). These wastage laws were introduced into the PROPANA code, which calculated wastage profiles and temperature fields.

- When a defect leak rate is in the intermediate leak range (approximately greater than 100 g/s) tube bursting due to overheating in the reaction area becomes a more important effect than wastage alone. The thermomechanical MECTUB code calculates the time to rupture under these conditions. The validation of this code is still in progress. In 1995 data previously obtained in the SUPER-NOAH Rig (AEA T) were used for this validation.

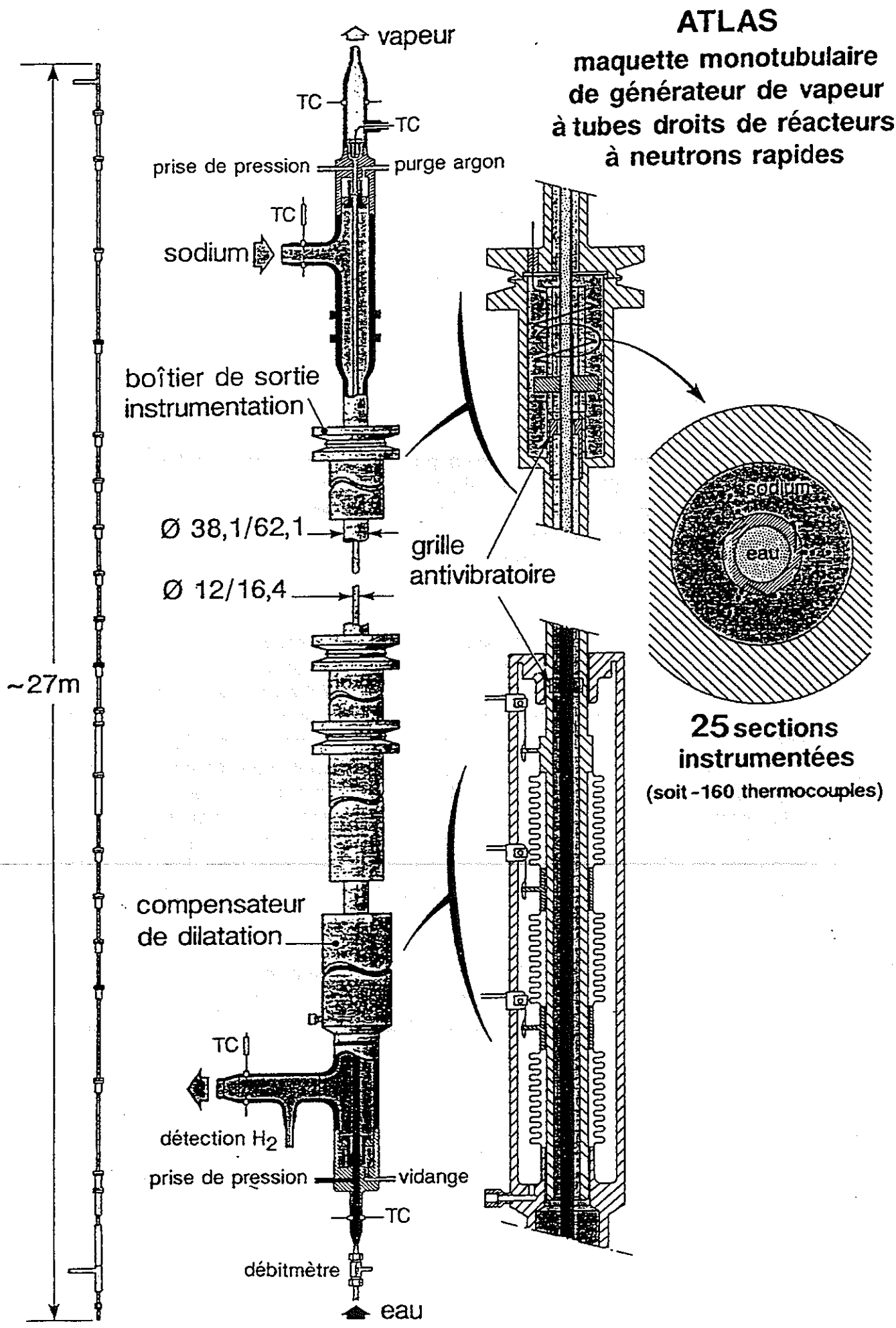

ATLAS
maquette monotubulaire
de générateur de vapeur
à tubes droits de réacteurs
à neutrons rapides
vapeur
TC
TC
prise de pression
purge argon
TC
sodium
boîtier de sortie
instrumentation
Ø 38,1/62,1
grille
antivibratoire
Ø 12/16,4
~27m
sodium
eau
25 sections
instrumentées
(soit ~160 thermocouples)
compensateur
de dilatation
TC
détection H_2
prise de pression
vidange
TC
débitmètre
eau

Fig. 7.7.

7.5. - Materials and mechanics

7.5.1. - Materials

7.5.1.1. - Steam generators - Alloy 800

Due to its exceptionally high temperature corrosion and creep resistance, Alloy 800 is used for steam generator tubing in SUPER PHENIX. Modified 9 Cr 1 Mo steel has been selected for steam generator tubing in EFR, but Alloy 800 remains the back-up material.

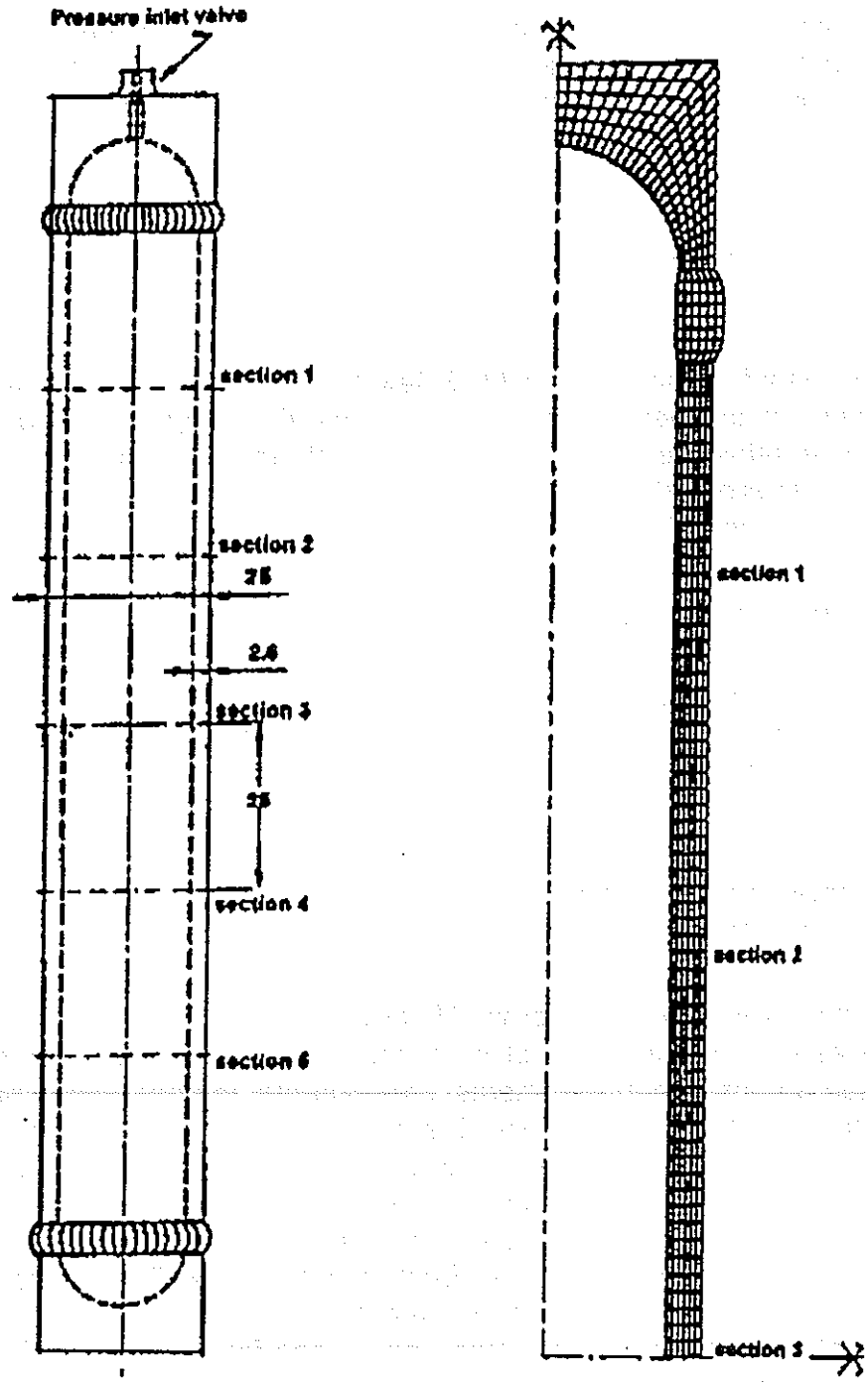
- Multiaxial creep behaviour for INCOLOY 800 tubes under internal pressure

A considerable amount of mechanical properties design data on Alloy 800 has already been produced for SPX1, but certain additional work, especially long creep tests were necessary for confirmation. These long term creep/stress rupture tests include the effects of titanium and aluminium additions and welding evaluations. Work has been carried out mainly on pre-pressurised tubes tested at temperature 550°C (a series of over one hundreds tests were performed).

The design of the Incoloy 800 specimen tubes is shown in Fig. 7.8. The specimen tubes were cut from the as-received steam generator tube material and were sealed by electron beam welding of the end caps. One of the end caps was fitted with a high pressure inlet valve to allow tube pressurization prior to testing. The length of the specimens was about 200 mm, the external diameter was 25 mm and the wall thickness was 2.6 mm. After pressurization the tubes were positioned inside a filled-argon protection vessel which was then placed in a furnace at 550°C. Tube failure was registered by a manometric technique. The tubes were removed from the furnace for inspection every 2.000 hours in order to make tube measurements at room temperature. Outer diameters were measured sections (see Fig. 7.8). Thiknesses were measured by ultrasound at every 90° of circumferential angle on cross sections 2, 3 and 4.

During the creep test, when one of the tubes cracked (of burst), the rupture time was registered and the furnace heat was automatically stopped by the manometer. The failed tube removed, the heat was re-started and the experiment continued until the next measurement inspection time. Tube leakage could be detected only by weighing. The manometer was not sensitive to slight gas leaks, so that the tube leakage time could be controlled to within every 2.000 hours.

Stresses and temperature (550°C) were choosen superior to those indicated in normal operating conditions because the average time of each test did not exceed 20000 hours.



Specimen design.
(Dimensions in mm)

Finite element
model.

Fig. 7.8. : Incoloy 800 Tubes and Finite
Element Mesh used for Analysis

Thin-walled tube approximation was used to estimate the stresses across the tube wall thickness required for given internal tube pressures. A refined numerical modeling of stresses and strains in the pressurized test specimens was developped taking into account : internal pressure changes, nonlinear geometrical effects and creep deformation during all the test procedure cycles. Comparison of the numerical results with the experimental measurements shows a good agreement which validates the multiaxial generalisation of the RCC-MR strain creep laws which were used in the calculations. A multiaxial creep rupture criterion was proposed.

- Crack initiation and growth under creep and fatigue loading for the Incoloy 800 alloy in the SPX steam generator.

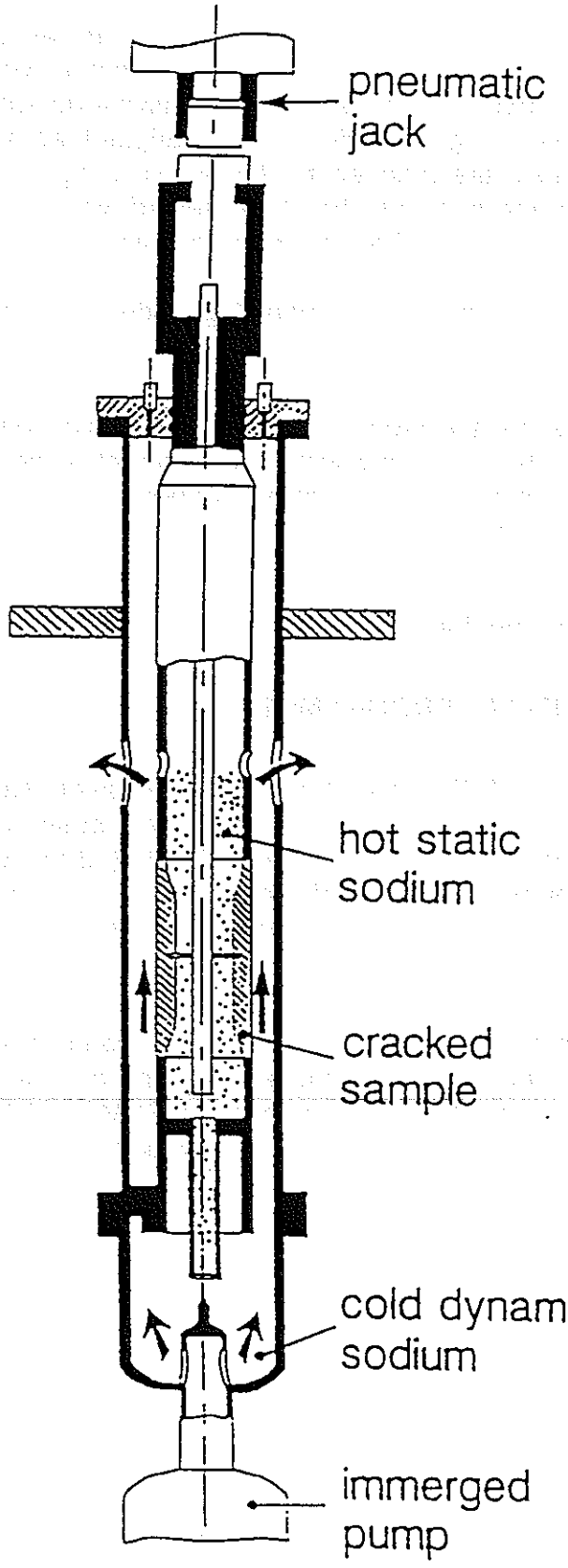
A new identification of the PARIS law for the Incoloy 800 in a very representative parameter range was proposed. At the same time fatigue crack growth experiments for a representative geometry under complex thermal loading have been made. A Finite Element interpretation with a sensitivity analysis for several fracture mechanic parameters was proposed.

- Corrosion studies in INCOLOY 800

Corrosion studies on pre-pressurized tubes with butt welds in both 0.1 and 0.001 NaOH solutions were carried out up to 100 000 hours. No cracking were observed on the 11 tubes tested (0.1 % Na OH).

7.5.1.2. - Cracks initiation and growth in stainless steel 316 L at high temperature for F.B.R.'s components

In 95 the experimental program TERFIS (Fig. 7.9) (carried out on a Sodium bench test) was intended to the analyse of the influence of a residual stress field on crack initiation and growth in weldments. Experimental interpretations have been made with simplified methode and Finite Element Analysis, both were based on a global approach of fracture mechanic. Concerning simplified methods, new analytical limit loads for 3D flows under complex loading have been proposed with a lower bound formulation. On the other hand, the local approach with an intergranular creep damage model, proposed by the "Ecole des Mines de Paris", has been validated on a wide range of high temperature data for crack initiation and growth on CT and Axisymetric specimens under creep and creep-fatigue loading. The damage model is now partially coupled with the Finite Element analysis and an automatic remeshing algorithm.



GENERAL VIEW OF TERFIS
TEST DEVICE

Fig. 7.9. : TERFIS Testing bench

7.5.1.3. - The Leak Before Break concept (L.B.B)

In order to validate the extension of the Leak Before Break concept to very long internal surface cracks an experimental data base is made with CHARFIS program (Fig. 7.10). The tubular specimen has an internal axisymetric surface crack located in a weldment, which initiate and grow under a creep-fatigue loading. Last year two more tests were made (six tests are planned for the whole program). The influence of a thermal gradient load in the thickness has been tested with the fourth campaign. For interpretations a Line Spring Finite Element approach was used.

7.5.1.4. - Initiation of welded joint design discontinuities under mixed mode creep-fatigue loading.

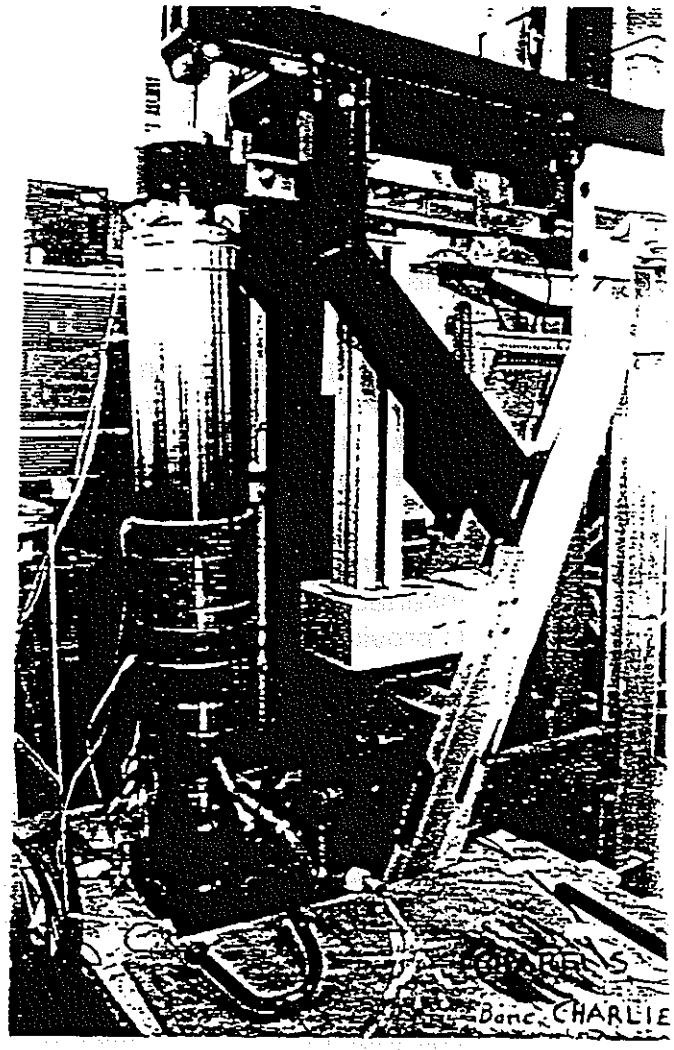
A new experimental test for the study of welded join design discontinuities was pre dimensioned last year. The specimen, with a representative geometry, will be tested under a creep-fatigue mixed mode loading, generated with a complex thermal gradient, on a Sodium test device.

7.5.2. - Dynamic core mechanics

SYMPHONY EXPERIMENTAL PROGRAMME

The seismic response of LMFBR cores has been extensively studied during the past few years in France. The modelizations were developed, based on the computer code CORALIE at the beginning and later on the CASTEM 2000 code was used. The experimental programme was based on the observation of the seismic response of a RAPSODIE core mock-up which includes 91 dummy subassemblies and 180 neutronic shield elements.

In order to observe the dynamic response of LMFBR cores at a larger scale, it was decided to start the construction of two mock-ups using dummy PHENIX assemblies. This experimental programme has been named the SYMPHONY programme. Part of this programme is henceforth performed within a cooperation contract between CEA and PNC (Japan).



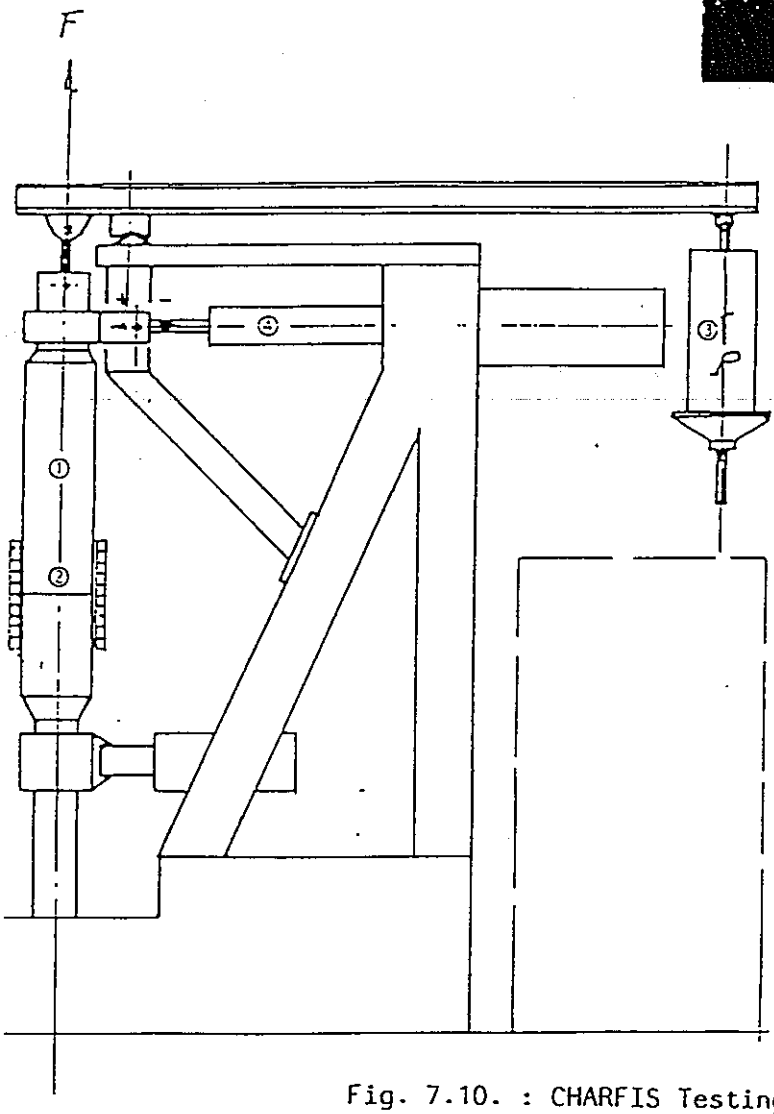


Fig. 7.10. : CHARFIS Testing bench

Two experimental mock-ups have been designed and manufactured.

- 3 row mock-up (Fig. 7.11, Fig. 7.12) :

This mock-up includes a central row and adjacent rows.

The central row accomodates 24 assemblies and 22 neutronic shield elements (11 at each end). Each of the two adjacent rows include 23 fuel assemblies and 22 neutronic shield elements. All together this mock-up can be equipped with 70 fuel assemblies and 66 neutronics shield elements.

The diagrid is composed of 2 plates connected by chandels. In these chandels are inserted the spikes of the fuel assemblies or the neutronic shield elements. Some gap is provided for the introduction.

The diagrid is 6.3 m long and is bolted to the shaking table (Fig. 7.11). A watertight vessel is connected to the diagrid to perform tests in water. Stiffening beams prevent lateral vibration of the vessel.

Two types of bundle configuration can be tested : free standing core and barrel restrained core. In order to achieve this latter configuration rigid stops can be anchored to the vessel at the pad and top level.

The total mass of this mock-up, including the fuel assemblies and the neutronic shield is 58 T.

- Hexagonal mock-up (Fig. 7.13)

This mock-up has exactly the same geometry as the RAPSODIE mock-up which has been previously tested but the dummy fuel assemblies have the PHENIX dimensions. It is composed of 91 fuel assemblies (a central assembly and 5 surrounding crowns) and 180 neutronic shield elements (4 crowns).

The diagrid is circular and is bolted to the shaking table. The bundle is surrounded by a watertight vessel to perform test in water. Similarly to the 3 rows mock-up, the bundle can be tested in a free standing and core restrained configuration. Removable rigid stops can be placed on the vessel at the pas and top levels.

- Fuel assemblies (Fig. 7.14)

91 dummy fuel assemblies are available. They have exactly the PHENIX dimensions. There is only one pad level (1435 mm from the top) where the gap is 0.25 mm between two adjacent assemblies.

Their weight is 182.5 kg. Steel cylinders located inside the hexcan are simulating the pin bundle without adding rigidity to the assembly. Water can be introduced in the hexcan from the top. The pitch of the bundle is 127.2 mm.

- Neutronic shield (Fig. 7.15)

180 neutronic shield elements have been specially designed for the mock-ups. They are made of a steel bar which have been machined to obtain a spike at the bottom and a gap of 3.2 mm at the top. Their rigidity has been designed to provide a stiffening effect relative to the flexibility of the fuel assemblies. Functional gap in the chandel allows for a small rotationnal rigid body motion of the neutronic shield element.

TEST PROGRAMME

The test programme is divided in two principal phases corresponding to each mock-up. Presently only the three rows mock-up have been tested with only the central row in place. It can be considered that only preliminary tests have been performed up-to now.

The different phases of the tests were the following :

- verification of the dynamic performances of the test vessel ;
- test on a single fuel assembly
  * Static test
  * white noise excitation at different levels to measure the assembly frequencies and damping. These tests were performed in air and water at different locations within the diagrid and with different subassemblies :
- test on a single neutronic shield element. The same test programme as for the fuel assembly was performed ;
- single row test with 24 fuel assemblies :
  * white noise excitation at different levels in air and water
  * seismic excitation at different levels with two time histories. One corresponding to a reactor directly founded on the ground and an other one with base isolation,
- single row test with 24 fuel assemblies and 22 neutronic shield elements. The test programme was the same as above : white noise and seismic tests.

First results were published at the SMIRT 95.

The preliminary tests on the SYMPHONY three rows mock-up have shown that the fuel assemblies are moving as a lumped mass.

Due to the gap within the neutronic shield element chandels, the single neutronic shield element exibits a strong non linear behaviour. It was decided to reduce this gap. This modification was carried out in 1995 and new single neutronic shield element tests were performed.

Instrumentation was also improved and a series of tests on the three row configurations with modified neutronic shield elements spikes was started at the end of 1995. Tests on hexagonal bundle are planned in 1996.

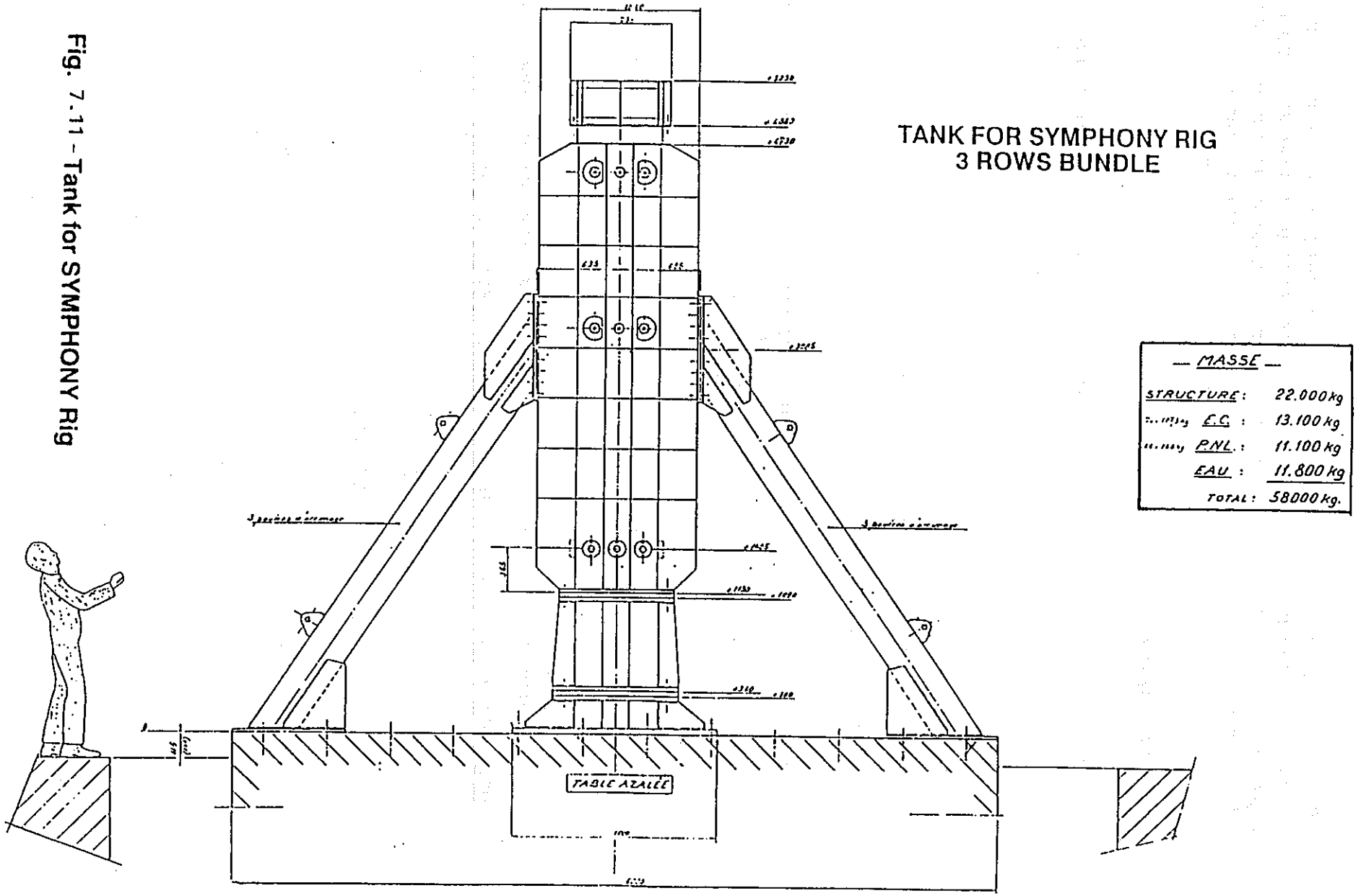


Fig. 7.11 - Tank for SYMPHONY Rig

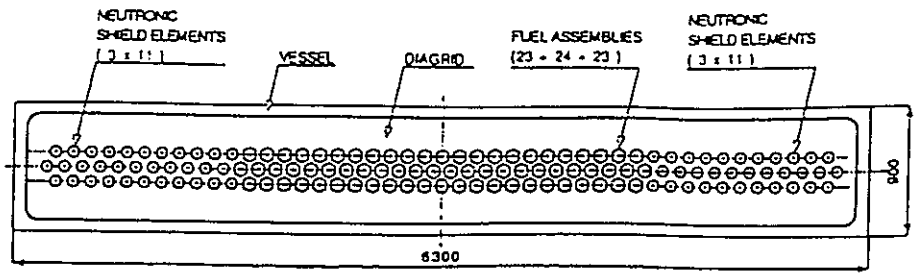
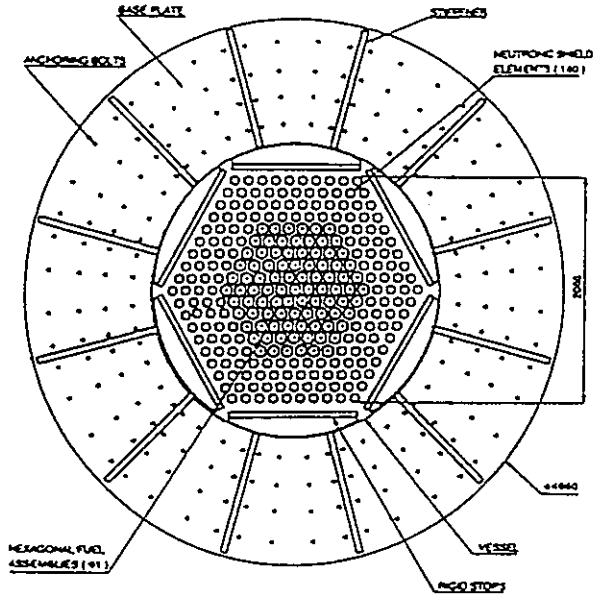


Fig. 7. 12 SYMPHONY 3 ROWS MOCK-UP
TOP VIEW



SYMPHONY HEXAGONAL BUNDLE
Fig. 7.13 TOP VIEW

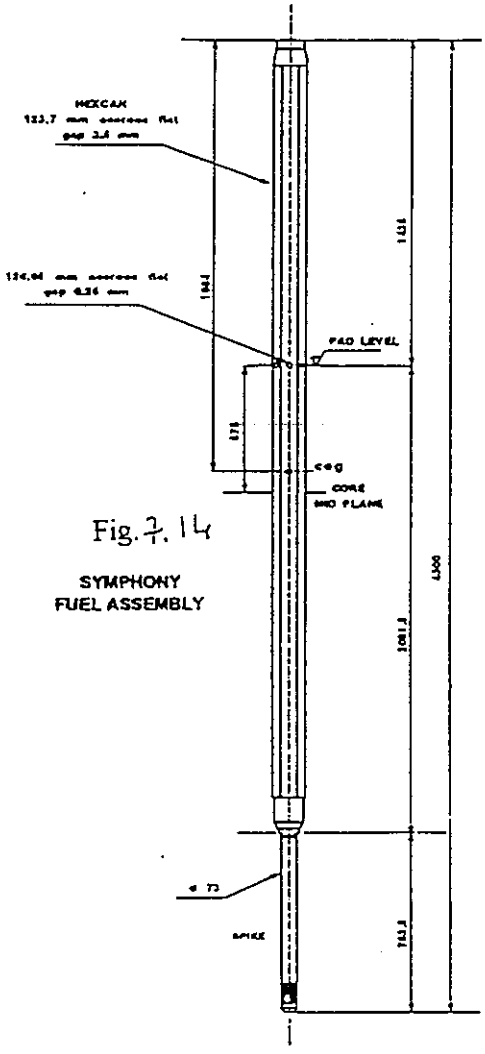


Fig. 7. 14

SYMPHONY
FUEL ASSEMBLY

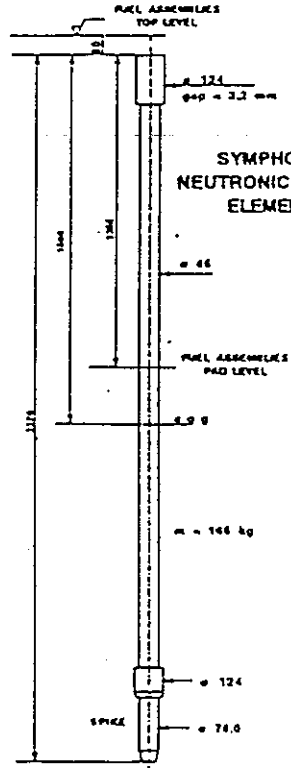


Fig. 7.15

European Fast Reactor :

The EFRUG* strategy and the aligned EFR programme

*** European Fast Reactor Utilities Group**

presented by G. HUBERT (EDF)

General context

- **In 1995 outcome of an EFRUG review of the forward strategy in collaboration with the European R&D and EFR Associates partners.**
- **Context :**
  - success of the EFR collaboration : "EFR Consistent Design" end of 1993, positive outcomes concerning notably the safety level (Ad Hoc Safety Club) and economics.
  - but in the same time general conditions lead to the EFRUG decision to not enter in a construction phase and lessening of resources for FR projects....
  - SPX restart with consideration of possible new missions for FRs in the back end of the fuel cycle, launching of CAPRA studies.
- **From present national policies : no decision on a new project of a demonstration Fast Reactor is likely in Western Europe before 2005-2010.**
- **In the meantime the *"EFR programme"*, replacing the previous EFR project, aims to improve and to adapt the concept to the new conditions, with 2 main motivations :**
  - to demonstrate the ability of FRs to participate to the industrial management of the whole fuel cycle,
  - to preserve the technology in view of a possible future hardening of the fuel market or of the requirements on nuclear waste management.

Which strategy for the Pu cycle (beyond PWR MOX)?

- **EFRUG remain convinced that FRs have an essential role in the future of nuclear energy notably because of the breeding mission.**
- **But the time for a commercial deployment of FRs for this mission is extending and belongs to the long term future.**
- **The delay in commercial deployment of breeders leads to postponing the total closure of the Pu cycle beyond one or two MOX recycling in LWR.**
  - In the meantime : Pu to be stored (e.g. in spent MOX fuel) or used (e.g. in dedicated reactors)? Different competing solutions are to be considered.
  - In addition consideration to a mission of reduction of minor actinides and radiotoxicity of nuclear wastes.
- **Evaluating Pu consumption and M.A. burning missions :**
  - Past FRs developments have mainly been oriented on the breeding mission : this has to be preserved and remain the long term objective but currently there is an emphasis on the developments on the Pu consumption and M.A. burning.
  - Current European R&D efforts are now mainly dedicated to these new missions.

    e.g. the CAPRA (Pu consumption in a FR) programme and the KAP (knowledge acquisition programme) in SPX.

- **CAPRA studies and minor actinide burning are considered in the EFR programme, whereas:**
    1) for these missions FRs are in competition with LWRs : role of FR burners to be clarified before any system introduction,
    2) actinide minimisation might be coupled to various Pu management (burning/neutral/breeding),
    3) interest in commercial FR burners remains uncertain and not for the immediate future : we are in a phase of evaluation studies for the next decade (e.g. the French law of 30 December 1991 requires a conclusion on nuclear waste management issues within 15 years)
- **At the end the debate concerning the FR missions is surpassed, important parameters for their present development are :**
    - ability of FRs to cope with a large variety of Pu origins and different possible fuel cycle management,
    - much of the FR design is common to the different missions. Most probable future extending from moderate breeding to moderate burning,
    - there is no possibility to consider a new FR project before significant cumulating of operating experience of SPX.

Key points for a future decision on a demo plant

(i) **clarification of the role of fast reactors :** the outcomes of the R&D programme on Pu consumption and actinides burning and fuel cycle studies,

(ii) **fuel market :** the possible need to anticipate the penetration of FRs in the nuclear power generating capacity (nuclear fuel market hardening),

(iii) **reliability :** development of the necessary confidence in fast reactor technology through successful operation of and feedback of experience from existing plants with appropriate embodiment into future designs,

(iv) **economics :** whatever the mission in the fuel cycle : confidence in the possibility of an advanced FR economically competitive for electricity production,

(v) **expertise :** the need to maintain a viable design team to support current reactors in operation and to preserve expertise for the future,

(vi) the possibility to wider **international collaboration.**

Some, but not all of these conditions are currently fulfilled,

The central scenario is based on a possible new project phase in 10 or 15 years from now.

The EFR work programme

6 main lines of activities :

A) *EFR Reference* **design (evolution of the 1993 "Consistent design"),**

B) *Innovative Concept* **studies,**

C) *Generic Topics*,

D) *Specifications, Codes and Standards*,

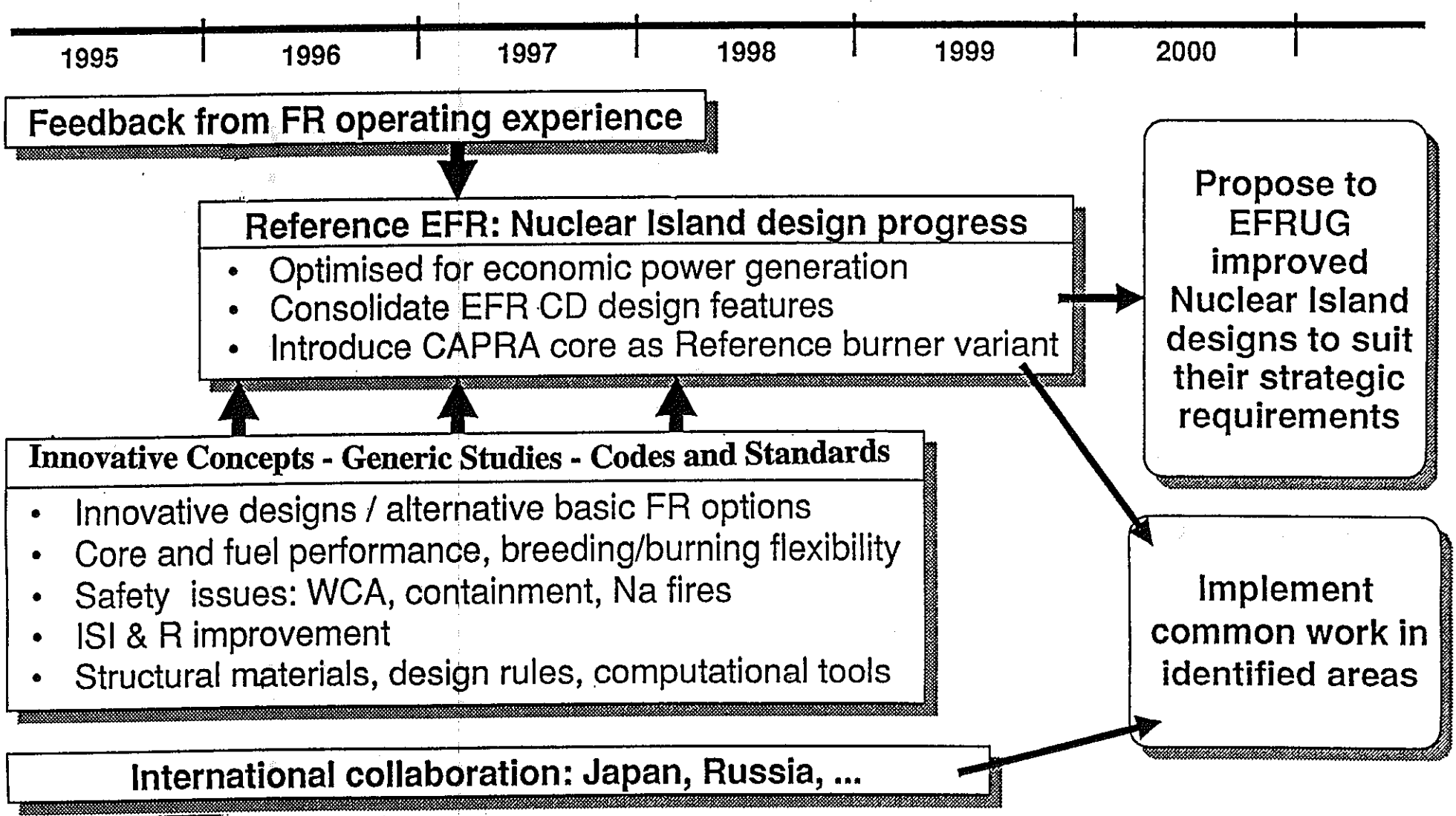
E) **Feedback from** *Operating Experience*,

F) *International Collaboration.*

These activities benefit from :

- *R&D tasks* **: support to operating reactors, generic issues and CAPRA, Pu consumption and actinides burning, irradiations test and experiments on advanced fuels and materials, safety developments, ISIR techniques....**
- *acquisition of experience* **from operational reactors worldwide (NSSS systems and components, sodium technology, monitoring and ISIR, operating experience.....),**
  e.g. KAP : *"knowledge acquisition programme"* **of SPX**

EFR work programme outline



EFR programme : design / R&D interfaces

EFR Associates' Programme

GENERIC ACTIVITIES

- Core design for Pu breeding / burning & M.A. transmutation
- Development of alternative fuels (e.g., nitride)
- Improved core safety (WCA prevention & DBAs)

- Improved core safety (BDBAs)
- Enhanced containment

- Improved reactor vault design / DHR diversity
- Improvement of Na fire prevention & mitigation

- Improvement of ISI&R capabilities

R & D Organisations' Programme

CAPRA PROGRAMME

R&D GENERIC PROGRAMME

- Core recriticality

- Sodium hazards

- Development of ISI&R methods & equipment

GENERAL TASKS

- DCRC
- Feedback from FR operational experience in design

EFR Reference Design Update

Innovative Concept Studies

International Cooperation

FAST REACTOR OPERATION

A) EFR Reference design

Progressive evolution of the Reference Design from the C.D. (1993) and completion of the validation of some of its options (solid roof, reactor vault,...).

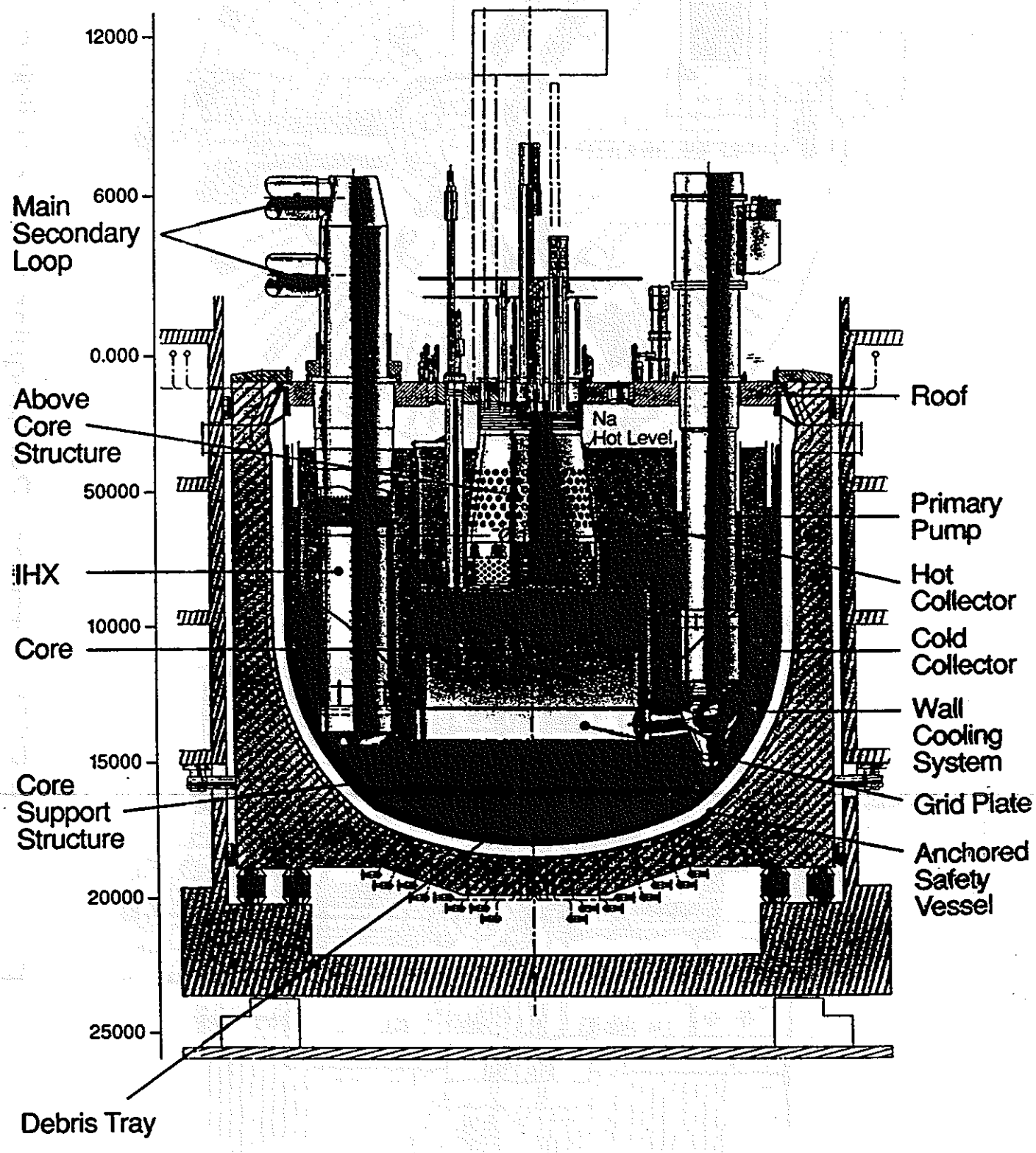
- **Initial priority is put on the basic studies under the heading "Generic" and "Innovative" studies. As these produce results they will be introduced in the Reference Design.**
- **Conditions for integration of an option into the Reference EFR :**
  - improvements which bring demonstrated benefits to the economics, safety, robustness, inspectability / reparability,
  - compatibility with the general options of the Reference Design,
  - state of validation consistent with the Reference.
- **Feedback from the existing fast reactors realised by comprehensive reviews of the design of each system and component in the light of reactor operating experience (see E).**
- **Integration of the CAPRA core may lead to two variants of the Reference Design (breeder/burner)**
- **Milestone : comprehensive update of the EFR Reference Design in 2000**

Examples of developments in the frame of the Reference EFR :

- **Integration of Capra core: e.g. optimisation of the diluent S/A layout and management, investigation to increase the fuel residence time and eliminating sub cycles, etc.**
- **Rectangular reactor building option** *(figure)*
- **Anchored Safety Vessel or Hung Safety Vessel + floating liner** *(figure)*
- **Sodium fire mitigation : demonstration of the effectiveness of the partially closed above roof area ("polar table area") as a mean of reducing loads induced by postulated sodium fires on the roof** *(figure)*.
- **Inner vessel structural mechanics : 3D inelastic analysis of the redan shows that creep and fatigue damage is acceptable.**

EFR EUROPEAN FAST REACTOR ASSOCIATES

REFERENCE EFR



EFR Primary System, Cross Section

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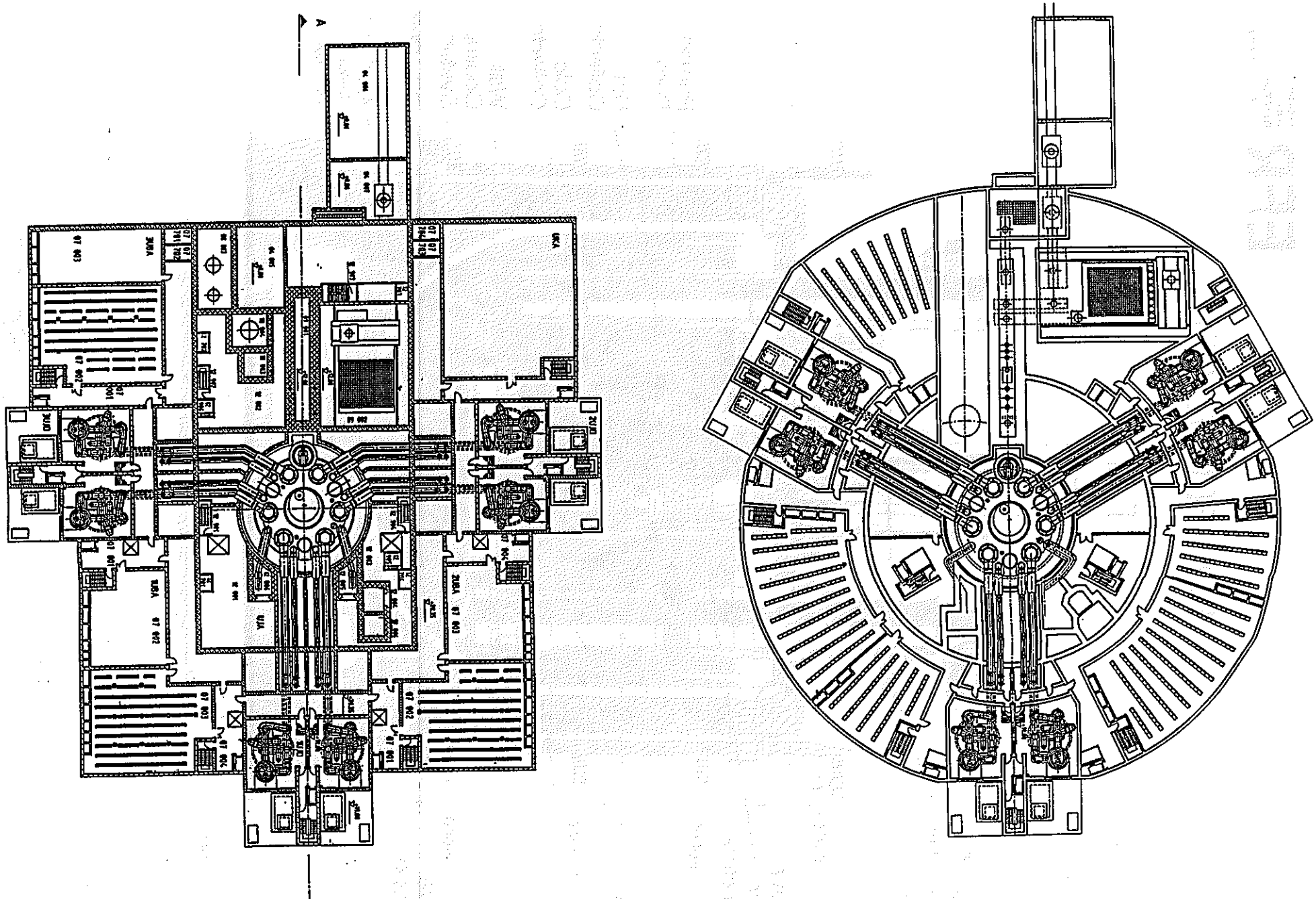
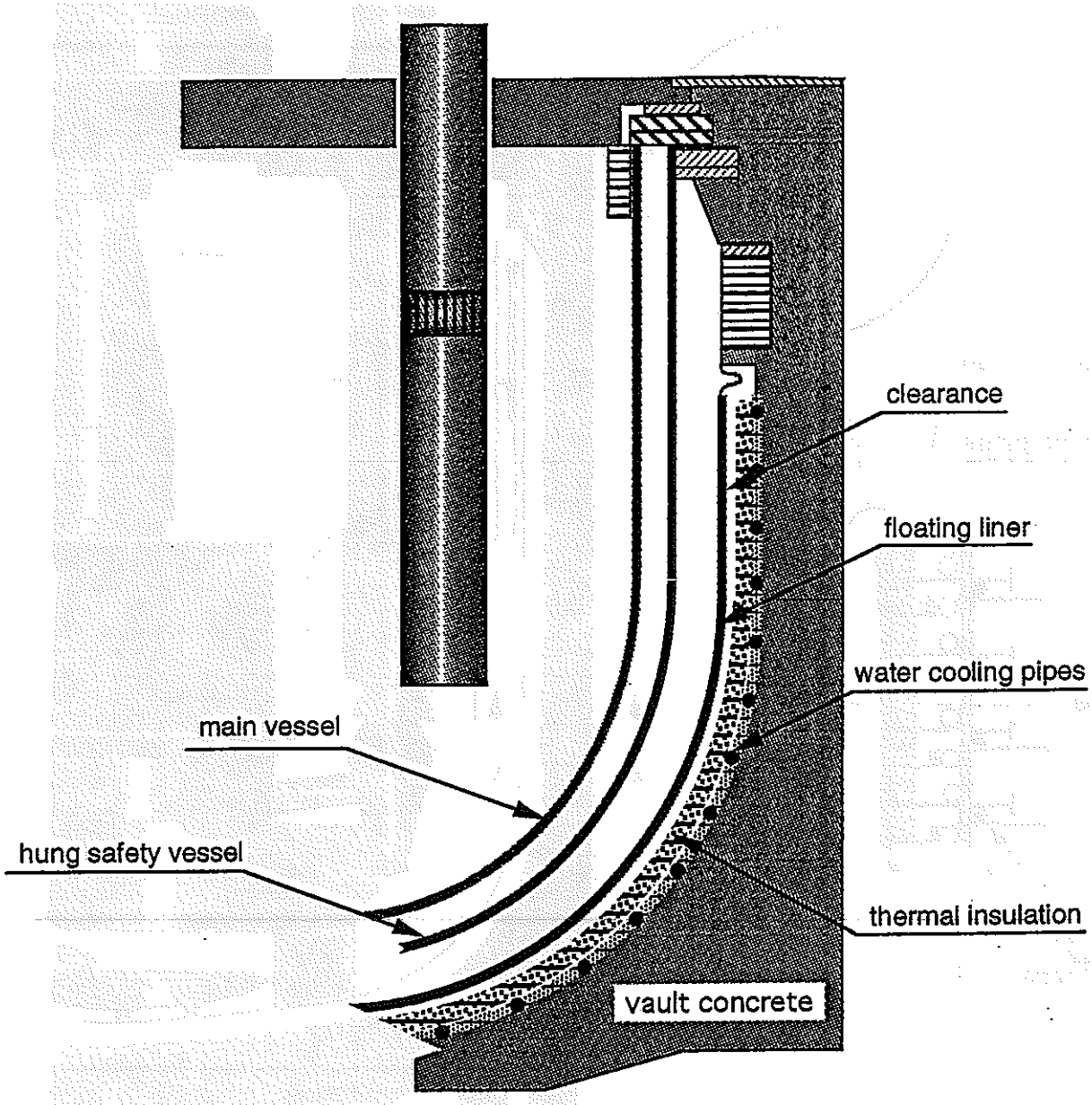
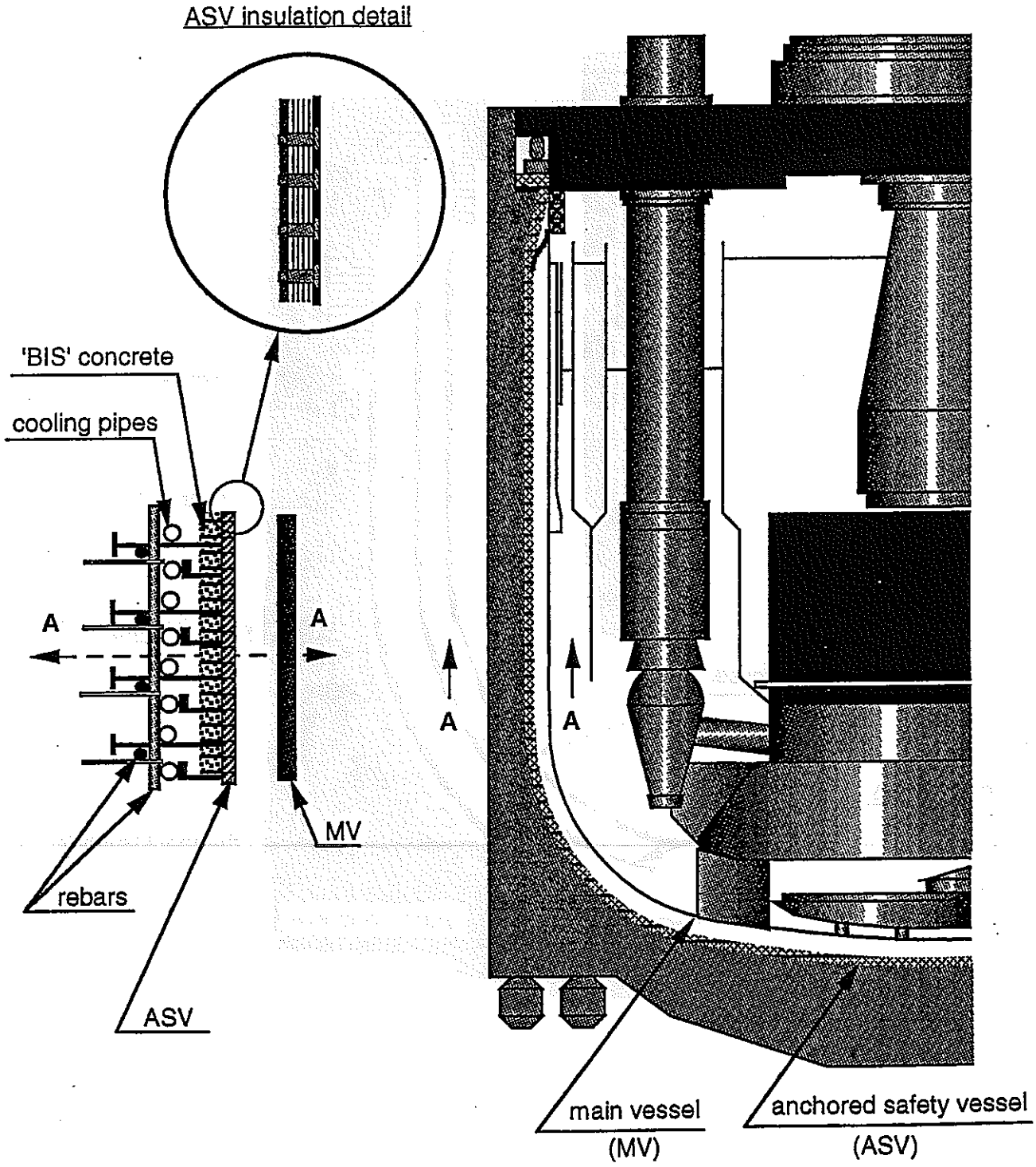


REFERENCE EFR : Circular / Rectangular reactor building

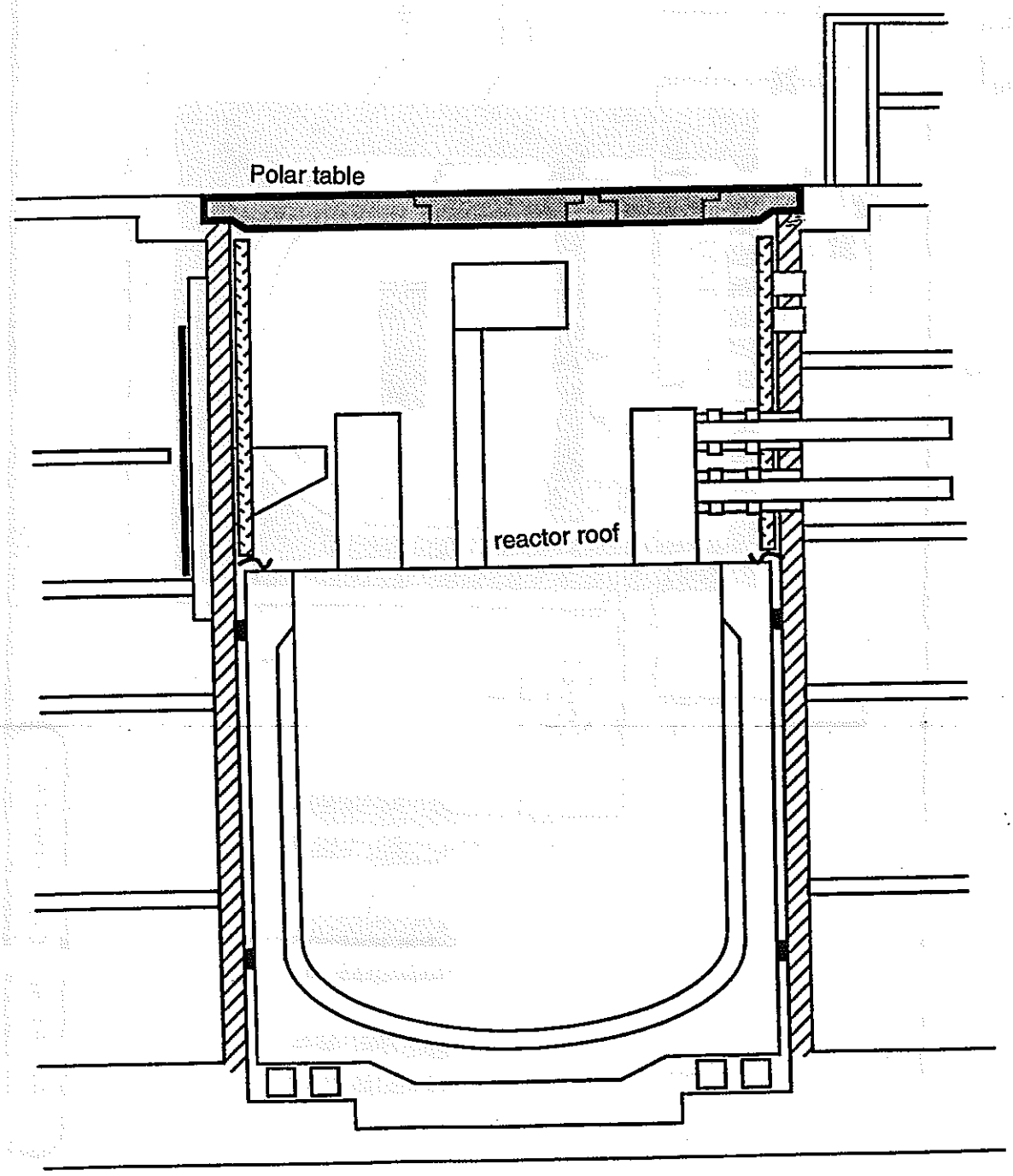


HUNG SAFETY VESSEL + FLOATING LINER



ANCHORED SAFETY VESSEL

The "polar table" area



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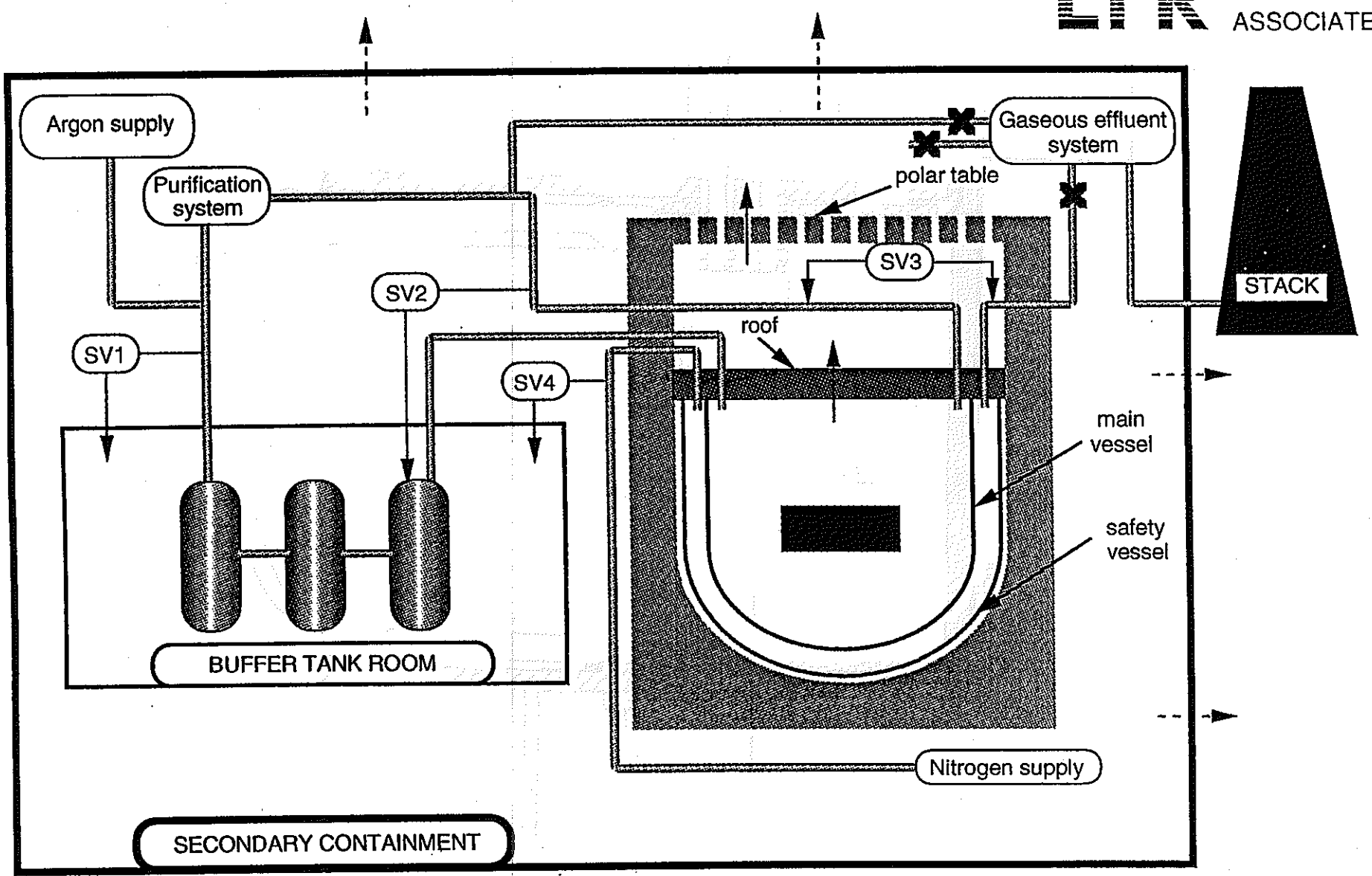
EFR EUROPEAN FAST REACTOR ASSOCIATES



RELEASE PATHS IN CASE OF WCA

B) Innovative Concept studies

- **A review of fast reactor concepts is undertaken to evaluate their potential to improve the Reference Design in economics, safety, robustness, etc. The scope is not inhibited by the extend of validation needed or the timescale for introduction.**
- **The work is twofold :**

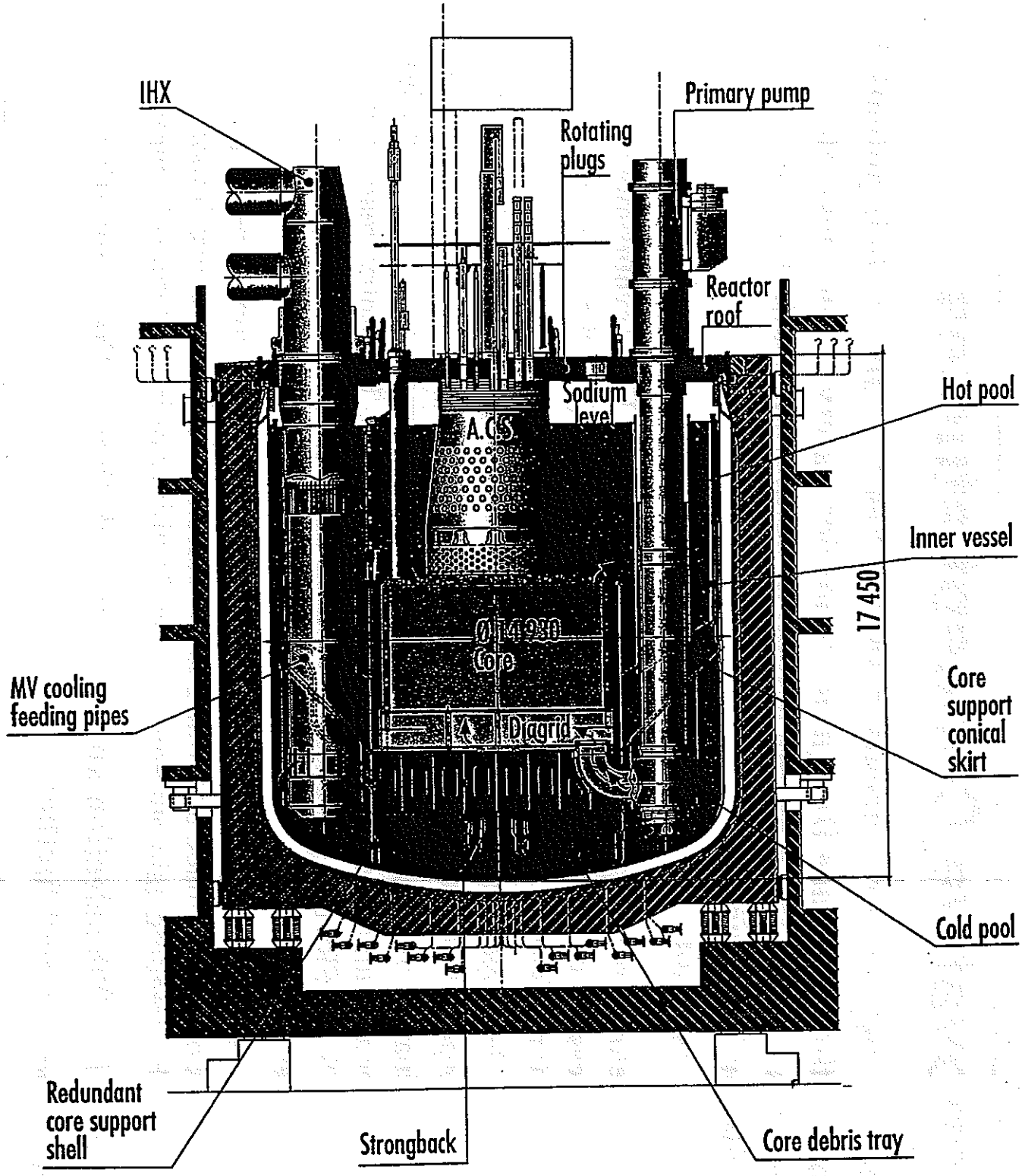
  a) developments keeping the basic general options of EFR (**evolutionary process**), i.e. sodium cooled pool type fast reactor (e.g. the "machined reactor": *figure*)

  The developed features, individually or as a package, have the potential to be incorporated in the Reference Design in the medium term.

  b) evaluation of **alternative basic concepts**, i.e., alternative fuels (e.g. nitride), other sodium cooled FRs (loop type, modular type, Top Entry, reactor without IHTS and FRs using alternative coolants (lead, gas, steam or supercritical water...) : *figure*

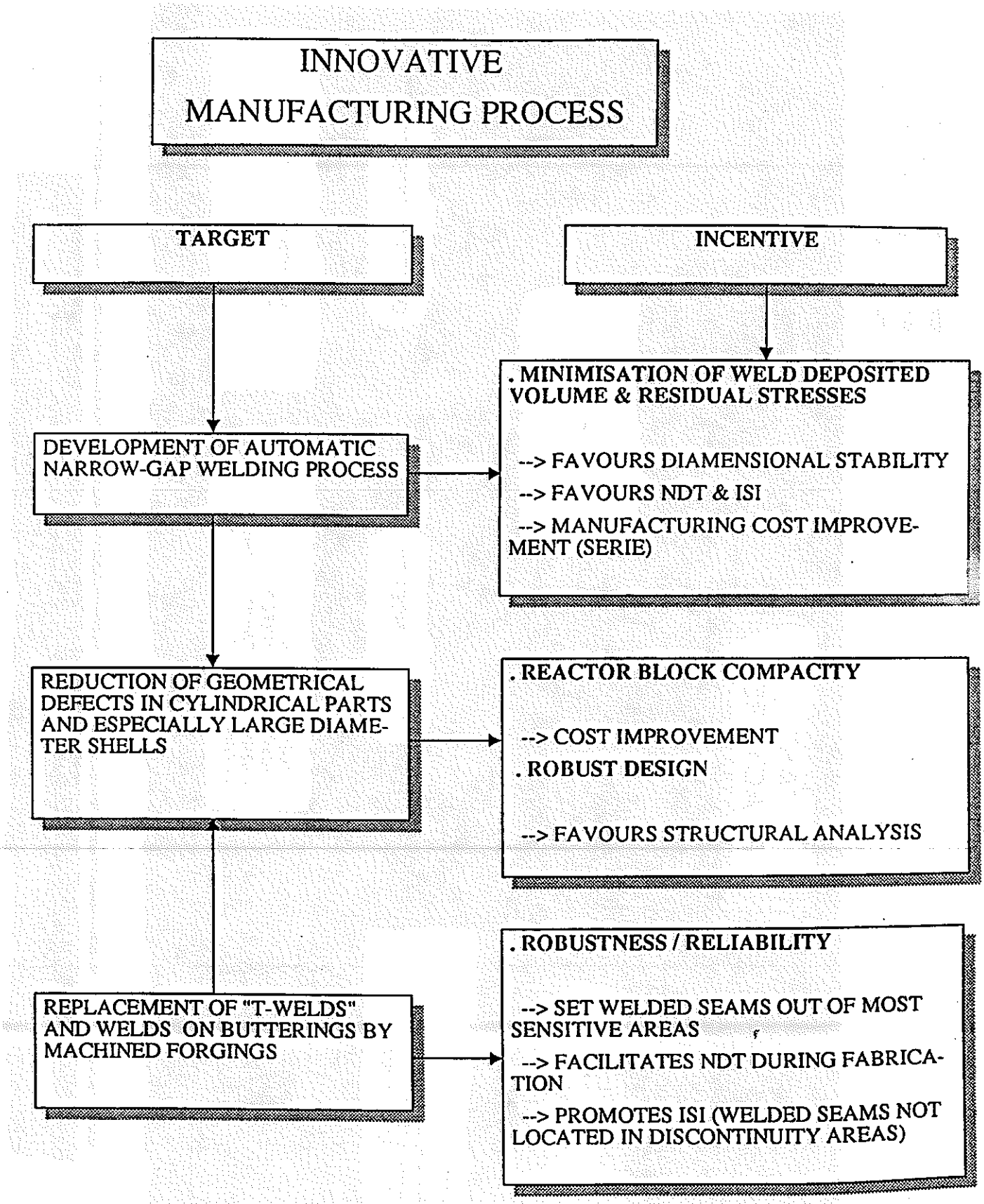
  The review of the concept options (some of them were considered in the early days of FRs) is made according to the current priorities, e.g. less importance on the breeding gain but emphasis on flexibility or inspectability/reparability.

- **Milestone : by 1998 selection and recommendation on the innovative options to be pursued further.**

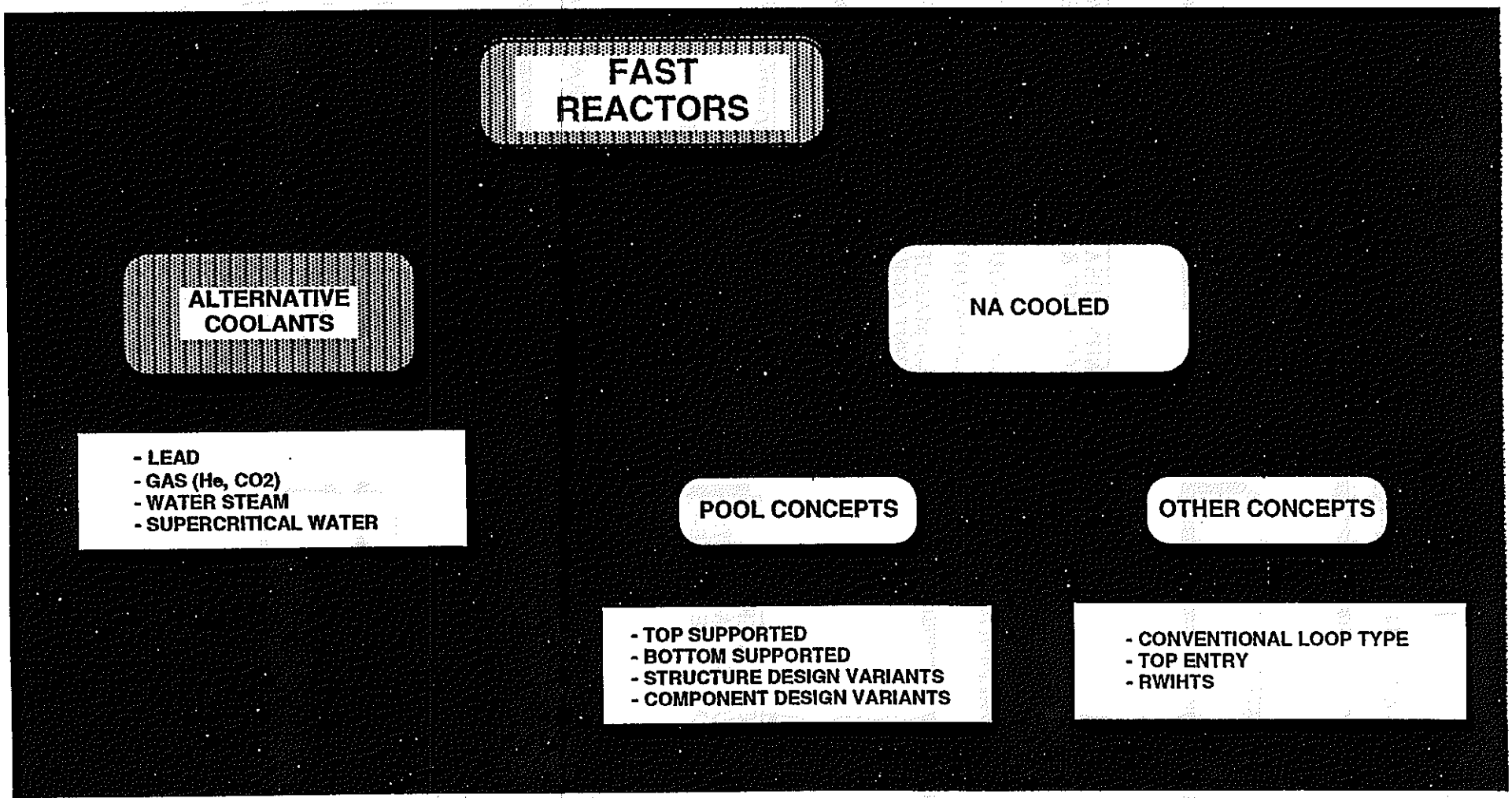


Innovative concepts : the Machined Reactor

Innovative concepts : the Machined Reactor



EFR EUROPEAN FAST REACTOR ASSOCIATES



INNOVATIVE CONCEPT STUDIES : Fast Reactor Family Tree

1.General Review (2)

- EXPERIMENTAL FAST REACTOR "JOYO": 51,200 hours OPERATION SINCE 1977.(as of March 1996)

- PROTOTYPE FBR "MONJU": FIRST CONNECTION TO GRID WAS ACCOMPLISHED IN AUG. 1995.
  SODIUM LEAK INCIDENT OCCURED IN DEC. 8 1995.

  PRESENTLY MONJU IS UNDER THE COLD SHUTDOWN STATE AND MULTIPLE CAUSE INVESTIGATIONS OF SODIUM LEAK ARE BEING PERFORMED.

1. General Review(3)

- DEMONSTRATION FBR (DFBR): DESIGN STUDY FOR PLANT OPTIMIZATION IS IN PROGRESS BY JAPC WITH MAJOR MANUFACTURERS.
- DFBR RELATED R&Ds ARE IN PROGRESS BY 4 ORGANIZATIONS, PNC,JAPC, CRIEPI and JAERI UNDER THE LEADERSHIP OF FBR R&D STEERING COMMITTEE. R&Ds FOR FUTURE FBRs ARE IN PROGRESS IN PNC.
- PROGRAM OF THE DESIGN STUDY AND THE RELATED R&DS FOR DFBR ARE BEING DISCUSSED IN THE SUBCOMMITTEE ON FBR DEVELOPMENT PROGRAM OF JAPAN ATOMIC ENERGY COMMISSION.

D) specifications, codes and standards

- **Comparison and updating of the "EFRUG technical specifications" aiming at consistency with the EUR (European Utilities Requirements) -** *task led by EFRUG-.*

- **Sodium fire prevention and mitigation : writing of a "sodium fire manual".**

- **Development and application of FR specific design & construction rules (task incl. R&D and international collaboration) : e.g. RCC-MR and DCRC (Design Codes and Rules Committee = European experts) tasks**
  - elastic fatigue and creep fatigue rules,
  - shakedown design rules for fast reactor application,
  - design methods for tubeplates, design properties of modified 9Cr 1Mo steel ,
  - corrosion cracking checklist,
  - guidance document on dissimilar metal weldments,
  - etc.,

- **computing tools :generic studies provide the direction and incentive for further development and validation of computing tools (operating code, safety codes : SAS 4A, SIMMER3, CONTAIN....)**

E) Feedback from Operating Experience

Methodology :

- **a systematic review of all EFR plant areas and components in the light of FR operating experience (with possible extension to the construction and pre operational tests),**
- **use of feedback from European reactors (incl. shutdown reactors such as PFR), experience from beyond Europe is used where available**
- **first reviews : absorber rod drive mechanisms, IHXs, sodium pumps, auxiliary argon and sodium circuits,...**
- **detailed comparison of the design features of EFR (Reference Design) with that of past FRs,**
- **record and analysis of significant events and operational difficulties or satisfactory operation which called for desirable design changes in EFR,**
- **technical synthesis with recommendations of feedback on the EFR design for improved performance (main application on the Reference Design).**

F) International Collaboration

A broader International Collaboration is one of the important item defined in the EFRUG Strategy with the objective :

to cement the links between the FR community when a number of countries (Europe incl.) have drastically reduced their support to FR development due to the long term objective,

to open the developments to innovative ideas and to share the development cost,

to keep the FR development in the main stream of worldwide agreements on safety and technical characteristics of future reactors.

Japan and Russia are our main partners on FR development outside Europe,

(others : China (CEA/CNNC), US (planned CEA/GE agreement))

JAPAN/EUROPE :

European R&D/PNC-JAERI-CRIEPI-JAPCO : 7 technical groups

EFRUG/FEPC, EDF/FEPC : technical exchanges and common developments (incl. EFR Associates and Japanese design companies)

CEA IPSN/PNC : agreement on SURA experiments,

EDF/PNC (SPX/MONJU) : operating experience exchanges

RUSSIA / EUROPE :

CEA(rep European R&D)/MINATOM : 8 technical groups (incl. EDF and EFR Associates)

participation of Russian Institutes to some EFR studies,

agreements between operators (SPX/BN600)

TACIS actions led by the EC (also for Kasak BN350)

Conclusions

Probable long timescale before commercial deployment of FRs,

The interest in breeding remains unchanged for the long term future of nuclear energy.

The delay provides the opportunity :

- **to investigate the wide potential of the FR for participating in the management of the back end of the nuclear fuel cycle.**
- **to develop the technology and designs further,**
- **to take account of FR operating experience**

The EFR programme launched by EFRUG aims to keep the FR option open (common technology for breeders and burners) and :

- **seeks to cover not a unique, but a range of possible futures for Fast Reactor development.**
- **preserves the reversibility of the technology (burner / breeder)**
- **covers a period of 10 to 15 years until a decision of a new project phase or to revise the strategy .**
- **preserves sufficient design and development expertise to support operation of current reactors**
- **and will be better achieved and secured within the framework of a broader international collaboration.**

Energy Policy in Germany

- In 1995 nuclear power contributed the largest share (34 %) of all energy sources (lignite 29 %, hard coal 26 %).

  The 19 power stations (Würgassen and Mülheim-Kärlich didn't operate) produced 154 TWh, which is a 2 % increase compared to 1994.

- The energy consensus discussions (resumed in 1994 to achieve a basic agreement) were stopped without reaching a solution or a success. It is now endeavoured to reach a part consensus at least about the back-end of the fuel cycle.

- Finishing the dismantling campaign of the KKN reactor at Niederaichbach in 1995 was the first example of a "restoration of the original state". FZK acted as project coordinator for this work.

- Based on the decision by SIEMENS and German utilities not to operate the MOX facility plant at Hanau the first activities for the dismantling started already.

Remaining R&D activities in the area of Fast Reactors

Some research projects connected with former fast reactor R&D and studies for separation/transmutation of Minor Actinides are performed in international cooperation.

These are:

1. Status of HCDA code development
   - 1.1 SIMMER III development and validation
   - 1.2 Status of SAS 4A development
2. Qualification of the SAS 4A model development
3. Transmutation of Minor Actinides
   - 3.1 Chemical separation
   - 3.2 Neutron physics
   - 3.3 CAPRA core safety problems
   - 3.4 Irradiation experiments

Remaining R&D activities in the area of Fast Reactors

1. Status of HCDA code development

At FZK SIMMER-III and SAS 4A are developed in co-operation with European partners and PNC.

1.1 SIMMER-III development and validation

- Integration of advanced neutronic methods into the code (based on TWODANT).
- Code validation with the QUEOS premixing experiment.
- Code validation with the KATS melt spread experiment.

Remaining R&D activities in the area of Fast Reactors

1.2 SAS 4A development

- Release of the SAS 4A, Ref. 96.Rel 0 code version (Common development of IPSN, OEC and FZK)
  - revised COMMON-block structure
  - correction of coding errors
  - implementation of the EJECT model developed by IPSN
  - new mechanical property correlations of the clad material.
- Improvement of existing models (Rel. 0.MOD FZK)
  - basic revision of the two-phase flow model
  - new approach to calculate solid fiss. product swelling
  - improvement of the JOG model
  - revised approach for fission gas induced fuel swelling
  - implementation of the IPPE clad swelling law.

2. Qualification of the SAS4A-model development

- Experiments with Rig-1 and Rig-2 fuel pins (solid pellet design with 6.2 and 7.5% as fabricated pellet porosity) of the CABRI-1 programme with low target burn-ups of 1 and 2.8 at% respectively
  - steady state fuel pin characterization and rapid thermal calibration tests (no failure tests)
  - rapid power transients with high energy releases
  - nominal cooling conditions up to extended two-phase flow conditions

- Experiments with VIGGEN-4 and OPHELIE-6 fuel pins (solid and hollow pellet design with 4.5% as fabricated pellet porosity) of the CABRI-2 programme with high and medium target burn-ups of 11.2 and 5.8 at% respectively
  - steady state fuel pin characterization and thermal calibration tests (no failure tests)
  - structured power transients with high and medium energy releases
  - nominal cooling conditions up to extended two-phase flow conditions
  - slow power ramps (0,1% P_N/s)

- Experiments with SCARABIX and QUASAR fuel pins (hollow pellet design with 4.0 and 4.5% as fabricated porosity) of the CABRI-FAST programme with medium to high target burn-ups of 6.4 and 12 at% respectively
  - steady state fuel pin characterization and thermal calibration tests (no failure tests)
  - structured power transients with high and medium energy releases
  - nominal cooling conditions up to extended two-phase flow conditions
  - mainly slow power ramps (0.1% P_N/s)

Remaining R&D activities in the area of Fast Reactors

3.1 Chemical separation process

Objectives:

- Separation of long-lived actinides (Am + Cm) from fission "lanthanides" (La + Eu) in the HAW of the Purex process.
- Development of a flowsheet for an aqueous solvent extraction.

Results:

- First findings (1992-94) showed that extractants containing only sulphur atoms are not suitable, also compounds with mixed nitrogen and sulphur atoms could not distinguish between Am(III) and Eu(III).
- Best results could be gained using extractants which contain only nitrogen as donor atoms. In this case these extractants, which must be fabricated

Remaining R&D activities in the area of Fast Reactors

3.1 Chemical separation process (continued)

separately, are compounds between organic diamines and Pyridine derivates.

- The work is partially sponsored by the EU and performed in close cooperation with many European partners.

Remaining R&D activities in the area of Fast Reactors

3.2 Neutron physical studies

General:

- Nearly all our studies in the field of transmutation are directly connected with CAPRA and SPIN programmes, therefore concentration on very specific items only.

Main tasks:

- Refined neutronic core calculations for safety related input data (Doppler, Na-void, k_{eff}, power distribution, etc.) considering
  - core loading strategies
  - light/heavy diluent insertion.

Remaining R&D activities in the area of Fast Reactors

3.2 Neutron physical studies (continued)

- Studies to define the tolerable amounts of MA in CAPRA cores with homogeneous fuels. First results indicated that larger quantities of these elements additionally require the presence of "moderator" materials.
- Starting conditions for scenario studies for Pu-burning were defined, based on various LWR types (Biblis, N-4) followed by multiple recycling in a CAPRA burner core.

  These scenario studies are part of an EU-sponsored programme.

3.3 Safety Analyses for Comparison of the Behaviour of Conventional and Advanced Core Designs

- Consequence analyses of unprotected loss-of-flow transients in conventional core designs
  - EFR Core design (three batch loading scheme with a high burn-up target of up to 20 at%)
  - SPX-like core design (two batch loading scheme with a medium target burn-up up to 8.4 at%)

- Activities in the framework of the CAPRA-project
  - Investigation of the power distribution and of reactivity worth distributions of sodium, steel and fuel at end-of-equilibrium cycle (EOEC) conditions dependent on core loading strategies with subcycling of heavy and light diluents
  - Steady state core characterization at EOEC-conditions (three batch loading scheme and hollow pellet design with high target burn-up up to 20 at%)
  - consequence analyses of an unprotected loss-of-flow accident up to first shut-down in the initiation phase with SAS4A-REF94

- Participation in the IAEA/CEC sponsored comparative exercise for ULOF-analyses of a BN-800 like core design with a nearly zero sodium void effect

  - Review of the case set-up specifications as provided by IPPE, Obninsk Russia

  - finalization of the steady state core characterization at EOEC-conditions (three batch loading scheme with a target burn-up up to 8.2 at%)

  - Preboiling and boiling transient analyses up to first fuel pin failure taking into account structure feedbacks as control rod drive line expansion and parametrically radial core expansion

  - Stage V of the exercise (post failure analyses) will be performed from mid of july 1996 onward.

Remaining R&D activities in the area of Fast Reactors

3.4 Irradiation experiments in HFR

Objective:

- Besides SILOE and PHENIX, the HFR irradiation experiments strongly support the development of new CAPRA fuel types.
  Main point is to study the irradiation behaviour at medium and high burn-ups (≈ 11 at.%), but also very specific properties like solubility in HNO_3, transmutation rate of actinides (Pu + Np), diluent suitability.

Status:

- First irradiation series (TRABANT 1) started in 1994, 3 different fuel types:
  mixed oxide with 45 at.% PuO_2 (burn-up 9.5 %)
  mixed oxide with 40 at.% PuO_2 + 5 % NpO_2 (b.u. 4.4 %)
  uranium free fuel type $(Pu,Ce)O_2$ (b.u. 3.9 %).

Remaining R&D activities in the area of Fast Reactors

3.4 Irradiation experiments in HFR (continued)

- Second series under preparation,
  containing two different fuel types
  mixed oxide with 40 at.% PuO_2 and differ. O/M ratios
  uranium free fuel types based on PuN, i.e. pure PuN
  or PuN + diluent "nitrides".

सत्यमेव जयते

GOVERNMENT OF INDIA
DEPARTMENT OF ATOMIC ENERGY
INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH
KALPAKKAM

STATUS OF FAST REACTOR DEVELOPMENT IN INDIA

APRIL 1995 - MARCH 1996

S.B. Bhoje
Director, Reactor Group

INTERNATIONAL ATOMIC ENERGY AGENCY
INTERNATIONAL WORKING GROUP ON FAST REACTORS
29th ANNUAL MEETING, AKTAU, 14-17 MAY 1996

STATUS OF FAST BREEDER REACTOR DEVELOPMENT IN INDIA

April 1995 - March 1996

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IGCAR, Kalpakkam, India

ABSTRACT

The energy scenario and economic conditions in India are presented. India needs considerable energy for its rapid industrialisation with the liberal economic policy. Nuclear energy with FBR is the only large scale energy resource other than coal, available in the country. The present economic constraints have delayed the construction of new NPPs. The performance of operating reactors has improved considerably during the year. Operating experience of FBTR has been detailed particularly the reactivity incident and its investigations. Updated design of 500 MWe PFBR is presented. Various R&D works in support of FBR in the engineering, metallurgy, chemistry, reprocessing, safety etc. are detailed out.

1.0 BACKGROUND

India has a population of about 900 million which is currently growing at an annual rate of about 2%, and is expected to cross the billion mark by the turn of the century. India supports 16% world's population on 2.3% of land area. Traditional biomass fuels such as firewood, agricultural and animal wastes, about 250 million tonnes a year, provide about 40% of total energy needs. Coal provides 40% of total energy requirements. Remaining 20% is contributed by oil, gas, hydro, nuclear, solar, wind etc. The percapita average consumption of electricity is very low, only 420 kWh per annum. About one-fourth of the total power generated is consumed in the rural areas, even though they constitute three-fourths of the population. 85% of the villages are deemed to have been electrified, but only about 28% of the households use electricity. 71 million rural households still do not use electricity. About 86,000 villages, in the remote areas, are yet to be electrified.

The year 1996 was a one of mixed fortunes on different fronts. On the economic front the country continued to maintain steady growth. GDP growth was sustained at 6.2%. The industrial production continued to rise at 12%. The export boom continued with growth of 24%. Though the import also grew at 29%, the deficit was restricted to 1.5% of GDP. The inflation was brought down this year below 5%. The rupee value which fell for a short while raising alarms rallied later to stabilise. Forex reserves continue to be good, touching 21 billion dollars. However, the liberalised economic reforms have not helped on the electrical energy front. The total electrical energy capacity addition during the current eighth five year plan (1992-97) will be less than that added during the seventh five year plan. This has happened because government financing for power sector did not go up while the much sought after private investment in power sector is going through a gestation period. Here again, there does not seem to be a clear-cut

policy over sector-wise growth. The need of the country is power source to bridge the peak demand for which the hydro power is the best suited while the private investment is totally fossil fuel based. Another area of concern is revamping and expanding the transmission and distribution system.

Total installed electricity generation capacity is 83.3 GWe at the end of March 1996. Out of this, 60.1 GWe is thermal, 21 GWe is hydro and 2.2 GWe is nuclear. Capacity addition during last year was only 2120 MWe. Total electricity generated by the utilities during the year was 379.3 TWh, working out to a load factor of about 51%. Sectorwise electricity generation during 1995-96 is given below :

- Fossil Fuel - 299.5 TWh

Coal based	-	250.2	TWh
Lignite based	-	012.9	TWh
Gas based	-	025.8	TWh
Oil fired	-	001.6	TWh
Multi fuel	-	009.0	TWh

- Hydro - 072.4 TWh
- Nuclear - 007.9 TWh

Total - 379.8 TWh

Construction works of 4 x 220 MWe PHWRs are nearing completion and are planned to be commissioned during 1998-99. Financial sanctions for additional 220 and 500 MWe PHWRs could not be obtained due to paucity of funds. Construction of the 100 tpa capacity Kalpakkam Reprocessing Plant (KARP) is completed and cold commissioning trials were started by end March 96. Regular operation is expected to start by Dec'96.

There is a bright outlook for wind energy and 366 MWe was added last year making up to a total of 733 MWe. Solar energy is also being pursued vigorously. Solar heaters and photovoltaics are being popularised for various domestic applications and irrigation in remote areas. An ambitious capacity addition is being planned for the coming years to raise the total capacity to around 400 GWe by A.D.2020. This requires rate of addition of 9 GWe/a during the next ninth five year plan , 12 GWe/a in the tenth plan and 15 GWe/a in the later period. Apart from financial resource mobilisation, the planning requires considerations of energy security and diversity. Nuclear energy continues to be recognised as one of the main sources of power for the country. The cut in government funding has affected addition of new capacity over past few years; but a major thrust in the nuclear energy front is bound to come sooner than later. The lull in the current period is being utilised by the Department of Atomic Energy to consolidate on the technology front and improve the load factors and safety of the operating plants. The present target for nuclear energy is 20 GWe by 2020 A.D.

Electricity generated in 1995 by nuclear plants increased substantially, inspite of no additional capacity and RAPS not being available during the year, to 7.98 TWh from 5.6 TWh last year, an increase of 40 %. Except for TAPS 2 and MAPS 2, an average load factor of 70 % has been achieved in all the other reactors. The performance figures for the reactors are shown in the following table.

Out of the 20 GWe capacity planned for 2020 A D from nuclear energy, 2.5 GWe is proposed to be from 5 x 500 MWe FBR. Towards this end, early commissioning of the 500 MWe Prototype Fast Breeder Reactor (PFBR) is sought so that after evaluating the performance of PFBR for a few years, 4 more 500 MWe reactors of same design could be commissioned by 2020. The present plan is to start construction of PFBR by 1998.

OPERATING PERFORMANCE OF NPP 1995-96

Unit	Generation (TWh)		Availability factor (%)	Capacity factor (%)
	Target	Actual		
TAPS-1	0.960	1.108	90.14	78.89
TAPS-2	0.680	0.445*	35.60	31.65
RAPS-1	shutdown for rectification of PRD on calendria			
RAPS-2	shutdown for replacement of coolant channel			
MAPS-1	0.905	1.136	79.59	76.07
MAPS-2	0.995	0.274*	08.79	18.38
NAPS-1	1.222	1.295	75.31	66.99
NAPS-2	1.226	1.457	81.07	75.40
KAPS-1	1.225	1.115	73.47	57.07
KAPS-2	0.737	1.152**	84.70	70.02
Total/ average	7.950	7.982	67.33	60.00

* Shortages in target are on account of longer shutdowns required for regulatory inspection.

** KAPS-2 commenced operation from 1.9.1995. The figure includes 0.365 TWh generated during the infirm power period.

2.0 FAST BREEDER TEST REACTOR (FBTR)

FBTR is a 40 MWt/13.2 MWe, mixed carbide fuelled, sodium cooled, loop type reactor. It has been provided with two once- through serpentine type Steam Generators (SG) in each of the two secondary loops. The reactor has 100% steam dump capacity, in order to continue reactor operation when Turbine Generator (TG) is not available. The reactor achieved its first criticality in Oct'85. Since then it is operated at various power levels in stages upto 10.5 MWt. The small carbide core with 25 fuel subassemblies (Fig 1) has been licensed to operate upto 10.5 MWt with a peak 320 W/cm linear heat rating of fuel. Commissioning of TG was started in March 96.

2.1 Reactivity Incidents

In Nov'94, when the reactor was operating at 10.1 MWt, power was found to be slowly increasing. Though the control rods were lowered one by one, the power continued to rise and reached 10.4 MWt in about a minute. The control rods were cumulatively lowered by 7.6 mm and the power was brought down to 10.1 MWt. The reactivity meter registered a spike of 3 pcm during the incident. Reactivity before and after the incident did not reveal any measurable permanent reactivity gain.

In April'95, during a startup, when the reactor was at 7.1 MWt, power increased sharply by 450 KWt in 7 s (Fig 2). No control rod movement was made at this time. Reactor underwent a scram on high positive reactivity and the recorder indicated a spike of +10 pcm. Criticality measurements before and immediately after the incident revealed a reactivity gain of about 24 pcm. However, a reduction of this value to 14 pcm was observed upon subsequent measurements. The measurement error itself is of the order of ± 10 pcm.

19 postulates were studied to find out the probable cause for the transient and 13 of them were tested during reactor operation at low power as well as at high power.

a. **Primary sodium flow change(test done at 9.5 MWt):** When the primary sodium flow was increased by 4%, the power went up by about 100 KWt and came down by the same amount when the flow was restored to the original value. Changes in sodium flow, inlet & outlet temperatures were observed on recorders during the test, but no such change was observed during the transient. A flow increase of about 17% is required to cause similar transient and is very unlikely.

b. **Secondary sodium flow change(test done at 9.5 MWt):** When speed of one of the secondary pumps was increased by 9.5%, the power went up by 350 KWt and then came down when the pump speed was lowered. The flow changes were perceptible on the flow recorder. The recorders response to fast changes in speed were also confirmed. However, there was no flow variation recorded during the transients and hence this postulate was ruled out.

c. **Feed water flow change(test done at 9.5 MWt):** When feed water flow increased by 13.4%, power started raising after about 90 s and a net increase of 200 KWt was recorded 200s after the perturbation. Flow variation and a dip in steam temperature were observed on recorders during the test. During the transients, no changes in feed water flow & steam temperature were observed.

d. **Steam pressure change(test done at 9.8 MWt)** : The change in steam pressure from 114.7 kg/sq.cm to 97.4 kg/sq.cm, resulted in flow increase, decrease in steam temperature & fall in sodium inlet temperature and all were noticed on the recorders. Vernier power started going up about 60 s after the drop of steam pressure and power started raising fast after about 5 min upto 500 KWt. Combining this test with the test of feed water flow change, it may be inferred that a power raise of 350 to 450 KWt warrants small fluctuations in steam pressure /feed water flow to persist for more than 3.5 min and such changes are recordable. At the time of the incidents, no abrupt change in the recorders of steam pressure, feed water flow, steam temperature & primary sodium inlet temperature were observed.

e. **Operator error**: Possibility of inadvertent raising of control rod by the operator was studied. Calculations, tests and analysis revealed that it is unlikely that operator has raised the control rods just before the incident.

f. **Stuck contacts:** Any contactor in the CRDM logic raising circuit getting stuck would have resulted in continuous raising of control rod and the power profile in such a case would have been steep and do not confirm to the profile during the transient.

g. **Sodium voiding**: If the reactor is operated with sodium in a subassembly boiling, the sudden collapse of this void and immediate resumption of boiling can lead to a transient rise in reactivity and power. For the transient of this order to occur, more than 16% of the volume in a subassembly is required to boil and this should have resulted in scram by plugging detection subroutine of central data processing system.

h. **Argon entrainment**: If the reactor is normally operated with 41 cc of gas continuously present in the core, the exit of this gas from the core and its appearance can explain the transient. This is possible by cover gas entrainment, if the operating level in IHX fluctuates about the safe level. Several tests conducted with the IHX sodium level changes by ± 100 mm, revealed no perceptible reactivity changes.

i. **Geometrical deformation of core due to CCPM**: It was postulated that the weight of the CCPM, if resting on three subassemblies for the accurate measurement of outlet temperatures of subassemblies would have resulted in slight geometric changes in core, which might have got reversed during the transient. Experiment at 10 KWt revealed that a negative reactivity of 3 pcm is added when CCPM rests on the core and reactivity become zero as soon as CCPM is lifted off the SA heads. The experiment was repeated at 10 MWt, but no changes in reactivity were seen. This could be due to the rigidity of the core when the buttons touch one another at high power.

j. **Hydrogen from cover gas(He/Ar):** For the reactivity to persist for a minute at the sodium flow rate of 360 cu m/h, the quantity of H2 required works out to 255 gm. This again works out to 9.2 ppm of H2 after full dissolution in primary sodium which would have resulted in a plugging temperature of 290 deg C. Plugging temperature of primary sodium measured after the incident was less than 105 deg C.

k. **Hydrides from cold trap**: To test this hypothesis, the cold point temperature of primary cold trap was raised to 130 deg C and maintained for 12 h. No reactivity effects were seen. It may be noted that the cold trap mainly traps oxides of sodium and hence hydride leaching from the cold trap is a remote possibility.

l. **Moderator ingress**: A micro filter steel subassembly was loaded in the core and about 520 times the hold up of primary sodium was circulated through the subassembly to trap sodium-oil reaction product (due to any oil ingress from sodium pump), if any, in the primary sodium. The filtrate was analysed for carbon content and found to be very small (300 mg), considered not adequate to cause the transient.

m. **Hydride deposits from cover gas space**: The ingress of hydride deposits from a pump or IHX can cause a transient. However, the plugging temperature should have gone up to 290 deg C, as against the observed value of less than 105 deg C. The postulate was hence ruled out.

n. **Fuel restructuring:** Any fuel restructuring leading to axial contraction can add positive reactivity. The fuel had seen a max burnup of more than 13,000 MWd/t and average burnup of more than 10,000 MWd/t at the time of the incidents. Gap closure due to initial restructuring by pellet cracking should have been completed long back. Further closing of gap will come about by fuel swelling, which is a slow process. The rate of closure averaged over a large number of pellets in the core, is not expected to vary suddenly, due to the power distribution in the core. Also since fuel swelling is an irreversible phenomenon, the resultant reactivity changes will be permanent, which has not been observed.

o. **Melting of source pin**: Since antimony is a neutron absorber, the removal of antimony from the core will lead to positive reactivity. Almost total inventory of all the pellets in source pin has to melt, breach its double containment and get entrained in sodium for the transients of this order to occur. Such a scenario leads to permanent reactivity gain and also the shut down count rates before & after the transients do not give any indication of antimony loss.

p. **Improper gripping of control rods**: The possibility of reactivity change due to upward movement of control rod relative to CRDM by improper gripping of control rods was checked experimentally and it was confirmed that there was no anomaly in gripping of control rods which could have caused the transient.

q. **Sodium voiding in control rods**: Detailed analysis revealed that sodium voiding by helium generation and subsequent displacement by sodium has to take place simultaneously in all six control rods for the transient to occur. This being highly improbable, was hence ruled out as the cause for the transient.

r. **Change in control rod & 25th S/A reactivity worths:** Remeasurement of control rod worths and the 25th fuel subassembly worth compared well with the earlier values.

s. **Subcritical operation:** Reactor was kept in subcritical state at 180° C and 400° C for about two weeks under observation and no transient was observed.

The results of the various investigations carried out did not indicate the cause for the transient. Since the transients had occurred during high power operation, it was decided to operate the reactor after increasing the reactivity scram threshold (+ or - 30 pcm) and augmenting the data acquisition system to gather sufficient data for analysis, if the incident recurs. The reactor operated in Feb & Mar'96 at 10.5 MWt for about 600 h and the transient did not recur.

2.2 Malfunctioning of CCPM

CCPM is a 6 m long, 82.5 mm diameter command tube with fixed and mobile core cover plates at the bottom and is centrally located in control plug (Fig 3). The fixed cover plate houses the thermowells for the 84 pairs of thermocouples and the mobile cover plate houses the sleeves to direct the jet of sodium from the top of the subassemblies to the thermowells. The mobile plate has three positions viz., fuel handling position (75 mm above the SA heads), normal working position (15 mm above the SA heads) and bottom position (resting over the SA heads). The up & down translation is done manually through a ball screw assembly. The primary and secondary leak tight barriers are provided by SS and neoprene bellows respectively and the interseal space between them is supplied with fresh argon.

During normalisation of pile after fuel handling operations in July'95, it was observed that CCPM could be lowered only upto 7 mm above the normal working position. Subsequently, during one of the trials to check for the movement, CCPM got stuck at 6 mm above the fuel handling position, while checking for the actuation of the top limit switch. Various investigations were carried out. Ball screw assembly was dismantled and found to be in good working condition. Leak testing of the interseal space was carried out and found that primary metallic bellows is leaktight. The interspace between the bottom of the mobile cover plate and the subassembly heads was scanned using under sodium ultrasonic scanner and confirmed no physical obstruction below CCPM. Based on safe load analysis, a jacking down force was applied and CCPM got released from top position at 780 kg and lowered upto the normal working position with progressively reducing force. Subsequently, movement of CCPM was repeatedly checked for its entire travel and found to be smooth. The precise cause for the malfunctioning of CCPM could not be found out and may be due to presence of foreign materials (burr, chip etc) in the interseal space, where minimum radial gap is 1 mm.

2.3 Reactor Operation

During this period, sustained operation of reactor for 17 d at 10.5 MWt was achieved. Operational run with 25 SA will be completed shortly.

Reactor operated in April '95, Feb. & March '96 and logged 962 h of operation, out of which 600 h at high power (Fig 4). Reactor remained shut down for the remaining period for the investigations of reactivity scram incident and the malfunctioning of CCPM. During this period, reactor operated at low power intermittently for remeasurement of control rod worths, 25th fuel subassembly worth etc and in subcritical state towards the investigation of the incident.

FBTR has logged 14212 h of cumulative operating time, out of which 2500 h has been at high power. The fuel has seen a peak burn up of 20,400 MWd/t without any failure. (Burnup 22,500 MWd/t upto 19.4.96). During the year reactor tripped 8 times (4 scrams and 4 LORs) and there were 2 forced manual shutdowns.

Major works like commissioning of new 3 cu m/h DM plant, connection & commissioning of underground concrete channel for enhanced condenser cooling water return flow from cooling tower, installation of fire stoppers at cable penetrations, RCB leak test, various modification works for the improved performance of different systems, major preventive maintenance works and surveillance activities were carried out during the shut down period.

Reactor operated at 10.5 MWt continuously for 17 d in March'96. The performance of reactor systems, sodium systems, control rod drive mechanisms and other safety related systems and auxiliary systems were generally satisfactory. The primary and secondary sodium purity was maintained below the plugging temperature of 105 deg C. The four sodium pumps and their drives are operating well and have logged 76750 h, 63440 h, 82230 h & 62000 h operation.

The heat transport parameters achieved during 10.5 MWt operation are as follows:

- Reactor inlet/outlet temperatures : 323/410 deg C
- Primary sodium flow through core : 374 cu.m/h
- Central FSA sodium outlet temperature : 470 deg C

- Sodium temperature at SG inlet/outlet : 405/295 deg C
- Secondary sodium flow (2 loops) : 282 cu.m/h
- Feed water temperature : 185 deg C
- Feed water flow : 15.5 t/h
- Steam temperature : 400 deg C
- Steam pressure : 118 kg/sq.cm

2.4 First rolling of TG

After completing all improvements including seal welding works for the main condenser, vacuum pulling was carried out and a vacuum of 625 mm of Hg was obtained which was adequate for rolling the turbine. Nuclear steam from the reactor was valved into the turbine for the first time and the first rolling of turbine was achieved on 12.3.96. The turbine was rolled upto 1680 rpm. Further commissioning of turbine upto the rated speed of 3000 rpm is in progress.

3.0 PROTOTYPE FAST BREEDER REACTOR - DESIGN

3.1 Conceptual Design

3.1.1 Main options

Selection of main options for a large FBR power plant is an important phase of the project as it gives a definite shape to it. The satisfactory plant operation depends on correct selection of the concepts. The following main options have been decided for PFBR after the cost reduction studies carried out during the last 2 years. Fig 5 shows the flow sheet of PFBR.

- Power - 500 MWe
- Thermal power - 1250 MWt
- Type - Pool
- Number of PSP - 2
- Number of IHX - 4
- Number of sec loops - 2
- Number of SG - 8
- Number of TG - 1
- Na inlet temp to core - 670 K (397° C)
- Na outlet temp from core - 820 K (547° C)
- Water inlet to SG - 508 K (235° C)
- TSV condition - 763 K (490° C) , 16.6 MPa
- Fuel - PuO_2-UO_2
- Fuel pin diameter - 6.6 mm
- No of fuel pins/SA - 217
- Core height - 1 m
- Peak fuel burnup - 100,000 MWd/t
- Core component handling - 2 rotatable plugs and 1 transfer arm
- Irradiated fuel storage - Ex-vessel storage in water
- Design life - $2x10^5$ h (30 y with 75% LF)
- Site - Kalpakkam

3.1.2 Design bases

Various plant design data, viz. plant life, design basis events, safety classification, seismic categorisation, applicable design and construction codes, computer codes, materials data and site data have been compiled.

3.1.3 Core

The core plan is decided taking into account number of in-vessel storage locations and additional shielding required for in-vessel transfer position. Static neutronics calculations for the 181 fuel subassembly 2 enrichment zones core are completed. Enrichment, neutron flux, power distribution and danger coefficients are evaluated for beginning of cycle and for equilibrium cycle.

The neutron irradiation dose (dpa) and helium production in grid plate and core cover plate, inner vessel and main vessel (MV), effect of fuel pin failure of internal storage on the sensitivity of DND system and neutron source and spectra in fresh and irradiated fuel assemblies are completed.

Effect of reducing the number of blanket rows from 3 to 2 on the breeding , economics and shielding is being studied.

A preliminary analysis with 'row-model' for the dynamic response of core SA under S1 (OBE) seismic excitations indicates that the maximum peak to peak displacement of the fuel SA is about 8 mm which is not of concern (Fig 8). 3 D analysis is in progress with an in-house code 'CORE-SEIS'.

3.1.4 Reactor assembly

The overall dimensions of reactor assembly components, particularly for the MV have been worked out. The MV diameter is 12.9 m. Detailing out of individual components(shape optimisation, thickness) is in progress. Elastomer seals are opted in place of liquid metal seals for rotating plugs to facilitate reduction in LRP and MV diameter and ease in construction.

The temperature of MV with the weir cooling arrangement is analysed. The contribution of the mixed convection in the annular space between the weir shell and MV is estimated as about 10 K and the maximum MV temperature is about 700 K which is still less than the acceptable value (725 K).

Structural analyses are repeated for the components (MV, IV, grid plate, control plug. etc.) for 2 loop concept. Detailed elastoplastic buckling analysis of inner vessel for 2 loop concept is completed including the shape optimisation and effects of thermal gradients, creep deformation and dynamic pressure under seismic accelerations. The optimised shape has potential up to 4.5 m (Na head) of pressure difference, to satisfy the RCC-MR (1993) buckling rules.

Study on the effects of one of the primary pump discharge pipes (2 pipes/pump) break indicates that it is possible to detect the pipe break by 2 diverse signals, viz. reactivity and high pump flow of the operating pump. It is a level D event.

3.1.5 Sodium circuits

DHR system consisting of 4 independent loops has been selected. Each loop comprises of one Na-Na heat exchanger dipped in reator hot pool and one Na-air heat exchanger with motorised dampers in airside.

The conceptual design of IHX has been finalised with the tube size of 19 mm OD & 0.8 mm thickness. A mechanical seal arrangement with piston ring has been opted instead of argon packet. For SG, 15 mm OD and 2.2 mm thickness tube (T91 material) is selected. The revised conceptual design of the primary compact pump for the modified design of 2 loop reactor is completed.

Various incidents viz. total power failure, one primary sodium pump (PSP) trip, seizure of one PSP, one secondary sodium pump (SSP) trip, seizure of one SSP are analysed. For PSP, the flow halving time of 8 s has been recommended in order to respect the hotspot clad limit 1073 K for upset events. No non-return valve is provided in the PSP.

A criteria for the design basis leak (DBL) for SG has been established. Accordingly, the leak rate that ensures rupture of both the rupture discs in the inlet and outlet of SG module is considered to be DBL. The instantaneous double ended gullotine failure of 3 tubes at the top of SG is taken as design basis accident.

Transient analysis of secondary sodium circuit with pump in the cold leg at lower elevation has shown some concern for flooding of pump vessel which may in turn cause damage to the mechanical seal. Alternative arrangements, viz. piston ring instead of mechanical seal and pump in the hot leg are being studied.

3.1.6 Steam water system

Conceptual design of steam water system has been updated and it is essentially the same as that for the coal fired station of same capacity (Fig 6). The standard 500 MWe modular concept turbine manufactured in India has been adopted and reheating by live main steam is used. The only difference is that the main steam and reheat steam temperatures are 763 K (490° C) and 613 K (340° C) as against 811 K (538° C) in case of conventional thermal power stations.

The steam cycle is of conventional type with regenerative feed water heating based on steam bled from HP and LP sections of the turbine. The LP feed water heaters (3 nos) are arranged in single train and the HP feed water heaters (2 nos) are arranged in 2 train configuration.

3 x 50% capacity condensate extraction pumps are provided to deliver the condensate from the condenser hot well to deaerator storage tank through the three low pressure heaters. 3 x 50% capacity boiler feed pumps (one electric motor driven and two steam driven) are provided to deliver the feed water from the deaerator storage tank to SG.

Condenser cooling is accomplished by sea water in once through mode and the temperature rise is 10 K in the condenser. 3 x 50% capacity pumps are provided for circulating the sea water through the system. The circulating water system is designed for heat load of ~755 MWt. The total cooling water requirement is ~21m^3/ s.

3.1.7 Electrical power supply system

Conceptual design of electrical power supply has been worked out. Fig 7 shows the single line diagram. PFBR is connected to the southern regional grid to transmit the power generated and these connections also provide off-site supply to the station. A 220 KV substation with six transmission lines is provided for power evacuation. The power supply to various station loads required for start-up, operation and shut down of the unit is divided in to four classes as follows:

- Class IV, Normal power supply forms the main source of power to all station loads.
- Class III, Emergency power supply for safety related loads, which can withstand power supply interruption of the order of a few minutes, are fed from class III buses. Various I & C system equipment are provided with this class of power at different voltage levels, viz. 48 V DC, 220 V DC and 220 V AC, 50 Hz.
- Class II, Uninterrupted power supply, (220 V, 50 Hz, 1 Ph) is provided for station computers, distributed degital controllers and all other instrumentation and control equipment requiring AC input power.
- Class I, 48 V DC & 220 V DC Suppliers are provided for I & C equipment (eg. associated with reactor safety etc.) requiring DC power at lower voltage levels and TG control & switch gear control respectively.

3.2 Design Basis Cyclone (DBC)

Post monsoon survey of sea coast near PFBR site at Kalpakkam was conducted. Based on these observations, the final report on DBC has been submitted by the Department of Oceanography of Indian Institute of Technology, Madras. The estimated water level is +7 m above mean sea level. The level of present approah road is +4.6 m. Hence it would be necessary to increase the ground level by ~3 m.

3.3 Site Seismicity

The Kalpakkam site seismicity values have been defined based on the updated earthquake data supplied by Geological Survey of India and guidelines issued by Atomic Energy Regulatory Board (AERB) to derive the site seismicity values. The site meets the AERB's site selection criteria (wrt seismic considerations) that no active fault lies within 5 km. The site seismicity values are as follows.

Design Basis	Peak Ground Accelerations (g)	
	Horizontal	Vertical
S1 (OBE)	0.078	0.052
S2 (SSE)	0.156	0.104

Incorporating all the above features, the conceptual design report has been prepared.

3.4 Estimation of Unit Energy Cost (UEC)

A systematic method is being evolved to calculate the investment cost and unit energy cost and comparison is also being made with coal fired plant and PHWR of 500 Mwe capacity.

3.5 Technology Development

Technology developments of main vessel , inner vessel, steam generator and primary sodium pump are progressing at manufacturers works. Forming procedure for the qualification of petals for the main vessel has been completed and crown and 3 petals have been pressed. Welding of crown with 1 petal has been completed. one Dies for technology development of inner vessel are ready. For SG reheater, drilling, spigot machining, tube-tubesheet welding developments have been completed and for the evaporator, tubesheet drilling is completed. Spigot machining and tube bending operations are qualified. Full scale hydraulic mock up of the PSP has been manufactured in SS and arrangements are being made for its testing.

For the fabrication of control rods, stellite hardfacing and hard chrome plating development is in progress. The raw materials, SS 316, SS 316 LN and D9, required for initial hydraulic test SA were procured indigenously. Clad tubes and other hardware required for the test fuel asssembly have been manufactured. Efforts are on hand to develop the process for manufacture of hexagonal tubes. The equipment required for various operations towards fabrication of assembly have been designed and manufactured indigenously. Process parameters have been established for the fabrication and testing.

3.6 Seminars on PFBR Design

In order to finalise the designs involving interdisciplinary aspects, a series of seminars are planned. 4 seminars are already conducted. The first one was on "Operating Experience of FBTR", the second on "Operating Experience of other FBR", third one on "Seals for Rotatable Plugs" and the recent one on "Inherent and Engineered Safety Features of PFBR Design". Further seminars on "Constructability of PFBR" and ISI are planned for the year.

An IAEA-IWGFR TCM on **'Conceptual Designs of Advanced Fast Power Reactors'** was held at Kalpakkam during Oct 3-6, 1995.

3.7 Collaborations with other R&D Institutions

With conceptual design nearing completion, it is time now to involve other R&D institutions within the country including DAE family for speeding up of PFBR R&D. Totally 10 R& D institutions are identified for component design, thermal hydraulics and structural mechanics.

4.0 RESEARCH AND DEVELOPMENT

4.1 Reactor Physics

A new 121-group (100 neutron, 21 gamma) coupled neutron-gamma cross section library for fast reactor shield calculations is generated and validated against a number of experimental and theoretical benchmarks available. A 25-group lumped fission product cross section set is generated using the ENDF/B-VI basic nuclear data library.

To assess the accuracy of the results of internal radial shield optimisation calculations carried out with 1-D transport codes and DLC-37 cross section library, a scaled mock-up of radial shield has been designed and fabricated to carry out shielding experiments with neutron source from a swimming-pool reactor, APSARA. A report giving details of the shielding models and experiments to be performed has been prepared and submitted to the Safety Committee for a review.

Work on the BN-800, IAEA/EC benchmark problem to study the effect of near zero sodium void coefficient on the severity of an accident initiated by unprotected loss of flow(ULOF), was carried out.

4.2 Engineering Development

4.2.1 Large Component Test Rig (LCTR)

The LCTR with a sodium inventory of 84 t is being operated at the required experimental conditions (sodium temperature 823 K) and the temperature distribution in the test vessel from the free sodium surface to the top closure has been obtained. Presently the effect of cover gas height (400 to 800 mm) has been studied on the total heat transfer and the total heat flux is found to decrease slightly with increased cover gas plenum. In order to simulate the roof slab cooling system, a nitrogen cooling circuit has been commissioned.

An indigenously designed and fabricated electromagnetic pump viz. Annular Linear Induction Pump (ALIP) of 5 cu.m/h and 0.5 MPa rating is being tested in the LCTR for checking its performance in high temperature sodium upto 773.K.

4.2.2 Sodium Pump

A sodium pump development test rig is under construction to test the dynamic performance of the pump rotor assembly. The 150 mm dia test rig piping has been erected and the main test section is expected to be completed shortly. The study will focus on the performance of the pump upper compliant support and the hydrostatic bearing would be studied. In regard to hydraulic development of primary pump, manufacture of full size hydraulic parts of 4 PSP design was completed and pump assembled for testing at the supplier's works.

4.2.3 Testing of CSRDM

A full scale model of the dash pot required for the CSRDM has been fabricated. As a better alternative for the bellows type sealing system of the CRDM, a sealing system comprising silicon rubber 'V' seals and oil seals have been fabricated. Fabrication of a test set-up for the dash pot and the seal has been completed.

4.2.4 Fuel subassembly tests in water

FIV measurements and pressure drop studies on a 19 pin bundle model of fuel subassembly were carried out in a hydraulic test rig for different pin spacer-wire pitches. A significant reduction in pressure drop has been obtained for an increase in spacer wire pitch from 150-200 mm without significantly increasing pin vibration.

The FSA test facility to measure flow and pressure drop has been commissioned. Some preliminary measurements have been made.

4.2.5 Instrumentation

A bypass flowmeter system for measuring sodium flow at primary pump discharge was manufactured and calibrated in a water loop. The system will be tested for designed flow during the tests on primary pump hydraulics in water.

A discrete (ON-OFF type) level detector for sodium based on the mutual-inductance principle has been developed and tested in sodium upto a temperature of 823 K.

4.2.6 Thermal hydraulics

Experiments on a semi-transparent 1/3 scale. 90 degree sector model to study the temperature uniformity indicates that the flow uniformity over the weir shell in MV is affected significantly by the number of cooling tubes. The gas entrainment and sloshing are varying with 9 cooling tubes (36 total for the prototype) as compared to 6 tubes. Less number of tubes is favourable. However, design modifications to destroy the vertical flow of the jet from each tube are necessary to make the flow over weir independent of number of tubes.

Experiments on 1/3 scale model of grid plate with baffle plate have been completed at Fluid Control Research Institute , Palghat. Flow uniformity in the SA is ensured during all the pumps in operation and also under 3 by 4 and 2 by 4 pump operations.

4.2.7 5 MWt SG test facility

Preliminary design has been completed on 5 MW SG facility to simulate 19 tube bundle of SG. This facility will be utilised for validating the thermal & mechanical design and endurance test of Indian built sodium heated SG. The construction of this facility will be done in coming years.

4.3 Structural Mechanics

4.3.1 Theoretical studies

Detailed creep buckling analysis of inner vessel, IHX and SG tubes has indicates that creep buckling failure mode is not of concern for PFBR.

LBB analysis is made on the primary pump discharge pipe by postulating cracks at the critical junction with the grid plate as per RCC-MR A 16 proposal. In order to satisfy the LBB justification criteria, the thickness of the pipe needs an increase from 10 mm to 20 mm locally in the vicinity of the junction.

For generating floor response spectra by random analysis method, equivalent sectional properties for the 'stick model' have been obtained using detailed 3-D analysis (Fig 9). In the model soil-structure interaction effects are studied by varying shear wave velocity in the range 1200-1500 m/s.

4.3.2 Experimental studies

Two buckling experiments on 1/40th scale models of main vessel under shear load were carried out. Comparison with theoretical prediction of critical buckling loads and mode shapes is satisfactory.

Natural vibration frequencies for a single FBTR subassembly vibrating in air and impact loadings at the loading pad have been measured using experiments with modal hammer. The code CORE-SEIS predicts well the experimental data.

A 15 t capacity high temperature component model test facility has been designed and erected. Commissioning activities are in progress. Fabrication of supporting structures and models for conducting fatigue and fracture experiments on benchmark geometries, viz. ORNL plate, C-clamp, is completed. Fabrication of 100 t capacity loading frame for static and buckling tests on large size components are nearing completion. Modal analyser and LVDT have been procured.

The test loop for simulating thermal shock on the control plug mockup has been commissioned at Indian Institute of Technology, Madras. The required thermal shock has been simulated on the thermocouple mockup.

At Structural Engineering Research Centre, Madras, dynamic experiments on the empty and water filled aluminium vessel under base excitations have been competed. Significant contributions of shell mode during horizontal excitations are noted. Theoretical analysis with CASTEM 2000 code to understand the same is in progress.

4.4 Metallurgy

4.4.1 Low cycle fatigue and creep-fatigue interaction behaviour

A comparative evaluation of low cycle fatigue (LCF) and creep-fatigue interaction behaviour of SS 316L(N) base metal, SS 316 weld metal and 316L(N)/316 weld joint has been carried out at 873 K. The LCF and creep-fatigue resistance of 316 SS weld metal is found to be better than that of 316L(N) SS base metal. 316L(N)/316 weld joints displays the least fatigue resistance. Fatigue life is observed to be lower in tensile hold as compared to compression hold conditions and the endurance decreased with increase in the duration of hold due to creep and oxidation. Based on the experimental data, fatigue and creep damage fractions have been computed and compared with the bi-linear creep-fatigue interaction diagram provided in RCC-MR. It has been observed that, in general, that RCC-MR curve is conservative.

Strain rate effects on high temperature LCF behaviour of 316 SS weld metal have been evaluated. Fully reversed strain controlled fatigue tests were conducted at a constant strain amplitude of ±0.6% at 773, 823, 873 and 923 K and strain rates of $3x10^{-2}$, $3x10^{-3}$, $3x10^{-4}$ and $3x10^{-5} s^{-1}$. At 773 K, fatigue life showed a peak at an intermediate strain rate of $3x10^{-3}$ whereas at other temperatures, life decreased continuously with decrease in strain rate. The weld metal generally exhibited a small initial hardening followed by gradual softening that continued upto the rapid load drop leading to failure. Further with decrease in strain rate, the cyclic stress response was found to decrease at all temperatures. The softening observed in the weld metal was attributed to the annihilation of dislocations during cyclic straining.

Towards studying the effect of Ti/C ratio for optimum creep-fatigue interaction properties of indigenously developed Alloy D9, baseline LCF data is generated at 823 K and 923 K with a Ti/C ratio of 4. The alloy is found to obey the Coffin-Manson relationship at both these temperatures. Evidences for the occurrence of dynamic strain ageing, such as increased cyclic hardening have been observed at 823 K.

4.4.2 Fracture and damage mechanics

The location of crack-initiation point on the load-time traces is problematic during J_{Id} testing using instrumented precracked Charpy impact tests. The stretch zone height (SZH) values determined from the fracture surfaces using scanning electron microscope (SEM) help locate the crack-initiation point and estimate J_{Id}. Based on the analysis of the strech zoner results and their relationship to load-time traces, some empirical relations have been obtained which enable determination of crack-initiation point on the load-time traces. This has promise in J_{Id} testing of irradiated specimens.

Fatigue crack growth (FCG) relation for single R-ratio (load ratio) proposed in the literature is modified for correlating ambient temperature three-stage FCG results generated at different R-ratios. The proposed equation was found to fit data generated on 316 SS at multiple R-ratios. Applicability of this analysis to high temperature FCG results generated on 316L(N) SS and T91 steels is being explored.

4.4.3 Creep rupture properties

A novel semi-empirical parametric correlation for stress rupture that incorporates heat-to-heat variations has been proposed. In this, two heat-sensitive material parameters have been used for scaling respectively stress and rupture life, both of which are also subjected to temperature scaling through temperature independent apparent activation energies. This multiple heat correlation has superiority over single heat correlationsl in simultaneously correlating stress rupture data from literature on eleven heats of 9Cr-1Mo steel.

4.4.4 Weldability

Alloy 718 in the aluminized condition has been selected as the material for tube bundle support for steam generator of the PFBR. Weldability tests are conducted to evaluate the hot cracking resistance by the Moving Torch Varestraint test. This alloy has shown much higher fusion and HAZ cracking susceptibility than Alloy 600 (reference). The higher heat input welds exhibits superior fusion zone cracking resistance than the lower heat input welds. However, the HAZ cracking results show the opposite trend. Extensive backfilling decreases the HAZ cracking because of a wide partially melted zone. Studies further shows that the weldability is not affected significantly by aluminizing.

Diffusible hydrogen measurements are carried out in the weldments of 0.5Cr-0.5Mo, 2.25Cr-1Mo and 9Cr-1Mo ferritic steels. Under identical conditions of welding, the diffusible hydrogen content is the highest in 0.5Cr-0.5Mo and the lowest in 9Cr-1Mo thus indicating that the alloying composition plays a significant role in determining the hydrogen diffusion. However, 9Cr-1Mo is the most susceptible to hydrogen assisted cracking.

The repair weldments of thermally aged 9Cr-1Mo and 2.25Cr-1Mo ferritic steel using both

half-bead and butter-bead temper-bead repair-welding methods are characterised based on hardness profiles and tensile test results. It is observed that the latter repair welding method yields lower peak hardness and better ductility. Amongst the two varieties of steels, low Cr ferritic steel responds more efficiently in terms of tempering behaviour with these repair-welding methods.

The effect of PHT on the interfacial microstructure and tensile properties of as welded Inconel 182 and PWHTed type 316LN/C-Steel joint is investigated. Martensite is formed in the heat affected zone of C-steel with single pass welding. However, with multiple thermal cycle experienced during multi-pass welding, the martensite seems to be changing into an equilibrium structure of ferrite and carbides. The tensile properties of transverse weld specimens carried out before and after PWHT shows only marginal improvement in ductility.

4.4.5 Hot workability and process maps

The constitutive flow behaviour of alloy D9 is investigated for various Ti/C ratios (0, 4, 6 & 8), temperatures (1073 to 1523K) and strain rate (0.001 to 100 s^{-1}) to establish hot working domain. The technique adopted for this study is through generation of processing and instability maps based on dynamic materials modelling. The processing maps revealed dynamic recrystallisation as a favourable mechanism for all the alloys. Based on the instability map and microstructural examination, it is concluded that flow localisation has the microstructural instability at lower temperatures and higher strain rates.

4.4.6 Effect of flowing sodium on ASS

Exposure of SS 316 to flowing sodium in a mass transfer loop (at 823 K) results in changes in tensile properties and chemical composition of the steel. The increase in yield and ultimate tensile strengths and decrease in ductility of the sodium-exposed steel are attributed to the combined effects of thermal ageing and sodium exposure. Analysis of the sodium exposed surface shows a decrease in concentration of Cr, Mo and Ni. Bulk chemical analysis of a thin stainless steel foil exposed to flowing sodium indicated an increase in the carbon content.

4.4.7 Corrosion of CW SS 316

Laser surface melting of CW SS 316 significantly improves its integranular corrosion resistance. Some important high temperature thermodynamic data, such as the standard Gibbs' energies of formation of alkali chromium oxides of the types, $M_2Cr_2O_7$ (where M = Na, K, Rb, Cs), M'_3CrO_4, (M' = Rb or S) , M'_2CrO_4 and $M''Sb_2O_4$ (M" = Fe, Co, Ni) have been determined by the use of precise and accurate solid electrolyte emf measurements. These data help in characterising the products formed during interaction of the fission product with the clad.

4.4.8 Sensitisation studies on SS 316 LN

The time-temperature-sensitisation and the continuous cooling sensitisation diagrams are established for SS 316 LN. These diagrams help in predicting the occurrence of sensitisation during service or welding respectively. The extent of chromium depletion along the grain boundaries is also measured by the elctrochemical potentio-kinetic reactivation (EPR) technique. The nitrogen content of the steel improved its crevice corrosion resistance in acidified chloride media compared to SS 316. Smaller grain size improved the pitting corrosion resistance

of nitrogen-added types 304, 316 and 317 stainless steels. On isothermal ageing of the weldmetal of these steels, the precipitation kinetics and morphology of chromium nitride and sigma phases influence the pitting corrosion resistance.

4.4.9 SCC studies on Alloy D9

Stress corrosion cracking (SCC) property is studied in boiling acidified chloride solution. The higher Ni content in D9 (vis-a-vis SS 316) improves SCC resistance. Also, increasing the Ti/C ratio, and use of fabrication procedures which involve lower strain rates, such as rolling impart better SCC resistance.

4.4.10 Development of techniques and testing facilities

The knowledge base and expertise in the plasma nitriding process has been developed and utilised for achieving optimum surface treatment of valve internals in the steam circuit of FBTR.

In order to establish the chemical constituents, especially carbon and oxygen, of microscopic particles obtained by digesting the sodium used in FBTR, analytical resolution of light elements at the spatial resolution of these sub-micron particles is accomplished by evolving suitable procedures using the electron probe micro-analyser.

4.4.11 Fabrication of $(_{0.8}Pu_{0.2})O_2$ fuel pellets by powder pellet route

Flowsheet has been developed for the fabrication of $(_{0.8}Pu_{0.2})O_2$ fuel pellets of around 85% TD. containing uniformly distributed thermally stable pores produced by controlled addition of pore formers like methyl cellulose. This fuel will be suitable for use as driver fuel for the PFBR. Both the low temperature oxidative sintering (LTS) under (N_2 - air) mixture and the conventional high temperature reducing atmosphere sintering (HTS) under Ar - 8% H_2) mixture have been standardised for fabrication of the pellets. Density, average pore size, pore size distribution etc. of the sintered pellets can be controlled as required by proper adjustment of sintering temperature and time and by choosing proper particle size, amount and mixing technique of the pore former.

4.5 Post-Irradiation Examination of FBTR Fuel Pins

Three experimental fuel pins (pellets in the two pins have present FBTR fuel composition and size and pellets in the one pin have the Mark II core fuel composition and size) have been irradiated in the fifth row of FBTR for durations ranging from 16 to 26 equivalent full power days, at a reactor power of 10.1 MWt. The aim of the experiments is to study the evolution of fuel structure from the beginning of the life of the fuel in the reactor. All the fuel pins have been filled with a mixture of argon and helium as the bond gas between the pellets and the clad to enable generation of higher temperatures at the fuel centreline corresponding to irradiation at higher linear power. The equivalent linear power generated due to this is about 270 W/cm (the actual linear power at the location of irradiation is about 180 W/cm). These fuel pins are received in the hot cell facility at Radiometallurgy laboratory for post irradiation examination. Fuel pins have undergone visual examination, X-radiography, eddy current testing and dimensional measurements.

Sodium deposition in the nozzle portion and other narrow passages of the first carrier SA during the handling caused some difficulties to remove the fuel pins by normal unlocking

procedure. This inturn called for removal of fuel pins by cutting the SA. However for the subsequent SA problems have been solved by modified handling procedure. The fuel pins are retrieved from the fuel pellets by cutting open in the dismantling machine. The fuel pins are further cleaned in an alcohol bath. Visual examination of fuel pins did not reveal any damage or deterioration.

4.5.1 Radiography

For the radiography examination, a specially designed carriage mechanism and film cassette retrieval mechanism are used. The carriage also has tungsten rods of known diameter for calibration of dimensional measurements, embedded at known distance. No crack like indications have been observed. Interpellet gaps at few locations have been observed in the fissile region. Pellet-clad gap has been observed clearly on the radiographs, indicating that the gap has not closed. Diameter measurements of the fissile portion are carried out using a non-contact measurement system, on the radiographic images. These values are compared with similar measurements of the pins taken before irradiation. The measured values are corrected for geometric factors. It is observed that the diameter increase is only marginal and no appreciable amount of swelling has taken place.

4.5.2 Eddy current inspection

Eddy current inspection of the three experimental fuel pins are also carried out. A differential encircling probe mounted on the vertical profilometer bench placed inside the cell was used for this testing. The system which is equipped with sophisticated data logging system has been developed indigenously at IGCAR. A detailed investigation with this equipment has revealed that no indication comparable to that of the artificial defects used in the calibration standard has been observed in any of the pins.

4.5.3 Dimensional measurements

Diameter measurements of the irradiated experimental fuel pins are carried out using a profilometer inside the hot cell. From the profilometry results it is seen that the diameter of the fuel pins shows a slight increase of the order of 10 to 40μ. Since the fuel pins have been irradiated only for a short period, the increase in diameter observed is not due to swelling of the clad, but is likely due to thin sodium/sodium oxide layer adhering to the outside surface of the fuel pin.

4.6 Pre-Service and In-Service Inspection Techniques

4.6.1 Residual stress measurements on the spare CRDM translation tube of FBTR

Measurements by 'hole drilling method' have shown that after annealing, the residual stresses have come down to 0 + 50 MPa as compared to 200 + 100 MPa before treatment. This indicates that heat treatment mentioned above is capable of removing most of the residual stresses and also bringing down the scatter in the stresses. Residual stress measurements enabled to ensure the adequacy of adopted heat treatment schedule for obtaining the component with minimum sensitization and residual stress.

4.6.2 Development of leak simulator for sodium-water reaction studies in FBTR-SG

By introducing a known leak in the experimental loop, steam water reaction studies are to be conducted to understand the response of the material to the leak as a function of leak rate, time, temperature, steam pressure and the size of the microhole. Leak simulators being developed are aimed to serve this purpose. The adopted design for leak simulators for the self wastage as well as the impingement wastage is based on a thin rupture disc element welded to the leaky tube at the location of the microhole which would rupture at a set pressure of 15 MPa to allow steam to leak through the microhole (90 to 250 μ) at known rate towards the sodium.

4.6.3 Projected interferometry and optical profilometry of FBTR cladding tubes

Projected interferometry (PIF) is used to study the profile of the cladding tubes along the axis. Microscopic profilometry is done (using microscopic profile projector) to supplement and corroborate the results of PIF. Variation in wall thickness along the axis is studied by microscopic profiling of cross sections. This technique is used to characterise and compare the cladding tubes manufactured indigenously. The study indicates that the deviations of the outer profile from the nominal plane are higher in the indigenous cladding tube when compared with those of the imported cladding tube. The study also brings out a strong correlation between the external profile of the cladding tube and its wall thickness variations. The wall thickness variations are seen to occur over a short range of few millimeters and found to be higher in the indigenous tube as compared to the imported tube.

4.6.4 Radiography of tube-tubesheet welds for PFBR-SG

About 100 trial tube to tubesheet welds of evaporator and reheater are radiographed using the microfocal radiography unit. The experience gained is useful in suggesting many modifications of practical significance to be incorporated in the technology development procedures.

4.7 Instrumentation and Electronics

Trip units are developed for processing bipolar neutronic signal of FBTR. Fail safe and modular design has been adopted. Charge/discharge flask position measurement system is developed for FBTR by processing the signals from synchro transmitters of the flask. SMART annunciator for fire fighting system is developed for FBTR. The instrument is provided with inbuilt feature to detect the healthiness of the sensor. Instrumentation system for measurement of carbon and hydrogen in argon cover gas is developed with self test feature. For analysing the reactivity transients in FBTR, Fast data acquisition is developed. Safety related signals are scanned every 100 ms. Whenever neutronic power crosses the set limit, the memorised process signals are graphically represented for the analysis of power transient.

4.8 Chemistry

4.8.1 Pyrochemical processing

Studies on techniques for pyrochemical reprocessing of FBR fuels are continued. An air atmosphere glove box is attached to the existing glove box train housing the laboratory scale

facility for pyrochemical reprocessing studies and a cadmium distillation set up is commissioned inside the air atmosphere box. An injection casting set up is assembled inside one of the argon atmosphere glove boxes in the facility and injection casting of pure uranium, uranium-aluminum and uranium-aluminum-molybdenum alloys is successfully carried out. The cast rods have been found to be free of pores.

4.8.2 Thermochemistry

Enthalpy increments of $Gd_2Hf_2O_7$ and $La_2Hf_2O_7$ which are of interest in high level nuclear waste management are measured in the temperature range of 1000-1700 K and from the temperature dependence of these enthalpy increments, other thermodynamic functions such as heat capacity, entropy and free energy functions are derived. Enthalpies of formation of the intermetallic compounds, UNi_5, UNi_2, UFe_2, $CeNi_2$ and $CeNi_5$ at 298 K are determined by using liquid aluminum solution calorimetry. The thermodynamic functions of these intermetallic compounds are of interest for understanding the fuel-clad interaction mechanisms during the irradiation of (U, Pu, Zr) fuel.

To measure the peritectic melting temperature of fuel materials, an experimental facility, based on the "Spot Technique" has been set up. Preliminary tests have been conducted successfully by melting noble metals in a Knudsen cell at a vacuum of $1x10^{-9}$ MPa. The temperature of the cell is measured by an optical pyrometer. Tests are in progress with other metal samples.

A new method has been developed for the measurement of the free energy of formation of carbides. In this method, the oxide is reacted with carbon and the measurement of partial pressure of CO evolved during the carbothermic reduction reaction leads to the calculation of the free energy of formation of the carbide. After validating this technique through studies on the Al-C system, the free energy of formation of selected chromium carbides such as $Cr_{23}C_6$ are measured using the new technique.

4.8.3 Th-U separation by Tri-sec-butyl Phosphate

For separation of ^{233}U from irradiated thorium, the use of Tri-sec-butyl phosphate (TsBP) as an extractant is expected to have advantages over Tri-n-butyl phosphate (TBP) which is presently used. A systematic study of the extraction of Th(IV) and U(VI) from nitric acid medium using TsBP and TBP has been initiated. The distribution coefficients for the extraction of Th (IV) and U(VI) have been measured at various nitric acid concentrations. The data suggest that the use of TsBP will result in much higher U-Th separation factors. TsBP has also been found to have better solubility for the thorium solvate, than TBP.

4.8.4 Development of laser induced vaporization mass spectrometry facility

An experimental facility based on laser induced vaporization combined with mass spectrometry has been developed in-house to generate the data on vapour pressures of fuel materials in the temperature range 2500-5000 K. Using this facility vapour pressure measurements are done on UO_2 sample. The UO_2 surface temperature is calculated as 4300+200K and a total vapour pressure of 0.38 MPa has been determined. The total vapour pressure obtained is also in agreement with the values calculated with the modified principle of corresponding states. This is of interest in the energy release calculations under CDA.

4.8.5 Matrix isolation infrared spectroscopy

This is an experimental technique that has been set up to study weak interactions in phosphate-diluent system with a view to understanding the extraction chemistry of the organic phosphates in fuel reprocessing. The conventional effusive beam technique used in matrix isolation spectroscopy pose problems in the study of large molecule spectroscopy, as in the case of organic phosphates. By introducing a supersonic expansion sampling to the matrix isolation technique, these problems can be circumvented. This is the first successful demonstration of the supersonic expansion-matrix isolation technique.

4.8.6 Fiberoptic based remote sampling system for fluorescence spectroscopy

Remote analysis of samples has an obvious relevance in reprocessing. The utility of remote fiber fluorimetry by coupling a fiber optic system to a commercial fluorimeter for the analysis of uranium solutions has been demonstrated. The detection limits need be improved by increasing the number of fibers in the bundle and by improved light gathering optics.

4.8.7 Solid state studies on nuclear waste disposal materials

Oxide pyrochlores that can incorporate rare earth elements are considered as promising ceramic materials for radioactive waste immobilization. Two series of such compounds $R_2M_2O_7$ [R = La - Gd; M = Zr, Hf] are synthesised by high temperature solid state reactions. Thermal expansion measurements on these compounds as a function of temperature are carried out by high temperature x-ray powder diffractometry. The observed thermal expansion behaviour has been correlated to the variation in the ionic radii of the rare earth elements and thereby to the distortions in the ionic coordination polyhedra RO_8 and MO_6. The oxide ion conductivity of some doped and undoped rare earth pyrochlore zirconates is also studied. Enhancement of conductivity is observed in strontium-doped neodymium and gadolinium zirconates.

4.8.8 Particles in sodium circuit

Particulates sampled by passing primary sodium of FBTR through a filter assembly (25μ pore size) are analysed for its carbon content with a view to ascertain the possibility of oil leak leading to formation of large amounts of carbonaceous product (which could have led to the positive reactivity incidents). The water and acid insoluble fraction of the particulate collected is 130 mg in weight and it is found to contain only 32 mg of carbon.

4.8.9 Assay of filter unit obtained from FBTR

The filter unit is first assayed by gamma spectrometry to identify the radioactive nuclides present. The active particles are isolated by ultrasonic agitation of the filter unit in methanol medium. The particulate matter obtained is assayed for radionuclides. The major nuclides are found to be Mn-54 and Co-60. Most of the radioactivity has gone into solution when the material is treated with hydrochloric acid. Analysis of the residue obtained after HCl treatment indicates the presence of carbon, oxides of Si, Mg and Al, and traces of Ti and Cr carbides. The carbides are suspected to have originated from the filter material.

4.8.10 Electrochemical carbon meters

Electrochemical carbon meters for use in sodium circuits of FBTR are tested for their long term performance in static as well as in flowing sodium circuits at an operating temperature of 873 K. The meters are calibrated by adopting stainless steel foil equilibration technique. The measured temperature coefficient of the meter output studied in sodium loop indicates that carbon is present as dissolved acetylide at the prevailing carbon activities (~0.1) in the loop sodium.

4.8.11 Sodium-oil reaction

With a view to evaluate the type of carbon produced in the event of an oil leak into sodium, reactions are carried out between Servoprime-76 oil and sodium at temperatures ranging from 473 K to 723 K. The reaction is found to initiate at temperatures as low as 713 K and vigorous at ~573 K. The reaction product obtained after equilibrating for 64 h is amorphous at 598 K, flaky and nonsticky at 673 K and partly graphitic at 698 K. When the equilibration is carried out at 748 K, the product obtained is found to be completely graphitic.

4.8.12 Particle release behaviour of reticulated vitreous carbon

Reticulated Vitreous carbon (RVC) which is used as trap material for Cs-137 radionuclide can release particles during the operation of trap. This particle release can increase the carbon content in sodium and cause carburisation of the structural material. The amount of particles getting released from RVC during the operation of the trap is determined following simple experimental procedure. From the results, it is found that it is necessary to provide a filter element with high surface area in order to reduce the carbon release from RVC into sodium system as well as to keep the pressure drop across the filter element low.

4.8.13 Radionuclide trap for Cs-137

Preliminary design of Cs-137 trap for FBTR has been made to incorporate a filter assembly with 25 μm pore size and surface area of 0.4 m^2 at the outlet line of the trap to avoid the release of particles from RVC into sodium.

4.9 Fast Reactor Fuel Reprocessing Plant

The design of all process equipment like thermosyphon and pot type evaporators, condensers, reheaters, acid recovery columns, high active storage tanks etc., have been completed. The fabrication drawings have been prepared for inactive solution tanks, feed clarification centrifuges, etc. Process vessels fabrication is in progress. The inactive solution preparation and pumping stations have been designed and undergoing safety review. Instrumentation system design is nearing completion. A remote sampling system is being designed. The safety implication of nonavailability of cooling water system for high active waste storage tanks has been assessed. Manufacture of fume hoods, free standing gloveboxes and glovebox trains is in full swing.

Development of solvent extraction code SIMPSEX is in progress. Work is in progress to develop empirical models for the limiting organic concentrations of U(IV), PU(III) and Th(IV) and also for characterising the third phase itself for the metal and HNO_3 concentrations and their

volumes. Five, 16-stage banks of Centrifugal Extractors (nominal capacity of 10 lit/h) are at various stages of fabrication. Two 20-stage low holdup ejector mixer settler banks have been fabricated and their performance tested with acid-TBP runs and found to confirm the advantages of high efficiency and individual stage behaviour.

A demonstration facility for processing FBTR fuel (1 kg/d) is under erection. This facility basically consists of two co-decontamination cycles followed by a extraction step, an in-situ reduction unit and one U strip unit. This will initially process the (70%PuC + 30%UC) fuel after diluting to 30%Pu and 70%U and progressively Pu content will be increased to 55%.

4.10 Safety Research and Health Physics

4.10.1 Activity transport

Release and deposition behaviour of radionuclides are critically reviewed. The most appropriate approach for modelling the corrosion product behaviour has been identified. A static facility has been assembled for studying the release and deposition behaviour under equilibrium conditions. In view of depth profiling of deposited radionuclides on primary system components, electroleaching trials are conducted in aqueous medium for controlled progressive dissolution of SS layers.

4.10.2 Sodium fire

In the pool fire code SOFIRE II and the spray fire code NACOM, modifications are incorporated to account for production of Na_2O and Na_2O_2 in a more realistic manner. For the sodium aerosol source term calculation, the pool temperature dependent evaporation rate equation has been incorporated. The modified versions give good results when compared with thirty two experimental runs conducted at AI and KfK.

4.10.3 Sodium-concrete interactions

The small scale sodium-concrete contact facility is refurbished to enable prolonged exposure of limestone concrete to larger quantities of hot sodium. The on-line instrumentation is also augmented.

4.10.4 Sodium-oil interaction

The nonvolatile reaction products which remain in the sodium circulated through the core are of concern. Such products were generated in a static experimental facility by contacting a representative pump oil with hot sodium.

4.10.5 Aerosol studies

For studies on aerosol characterisation under simulated reactor accidents, scoping experiments are carried out with copper by fast condenser discharge technique. Mass aerodynamic size measurements show bimodal distribution indicating the types of particles. X-ray diffraction measurement carried out on the sample reveals that the aerosols constitute CU20 and CUO conforming the bimodal distribution.

4.10.6 Fuel-coolant interaction

Simulated small scale MFCI experiments with different systems like tin-water, zinc-water, bismuth-water and lead-water have been carried out and the fines generated during fragmentation are analysed based on heat transfer and fuel physical properties.

4.10.7 Atmospheric dispersion studies

Sea breeze model SBLBML (Sea Breeze Land Breeze with Mixing Length parameterisation) developed for modelling coastal air circulation has been evaluated using a bench-mark data obtained from Lake Michigan field experiment. The model performance has been tested for the Kalpakkam coastal conditions by comparing the calculations against the SODAR data on wind field for a complete 24 h period.

Acknowledgement

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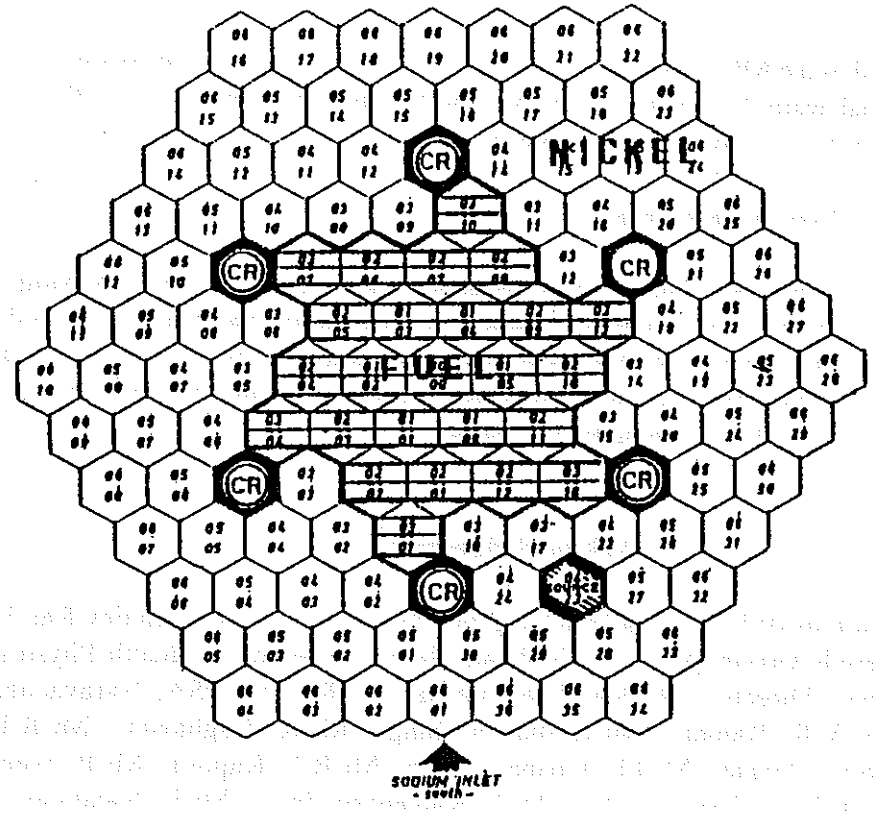
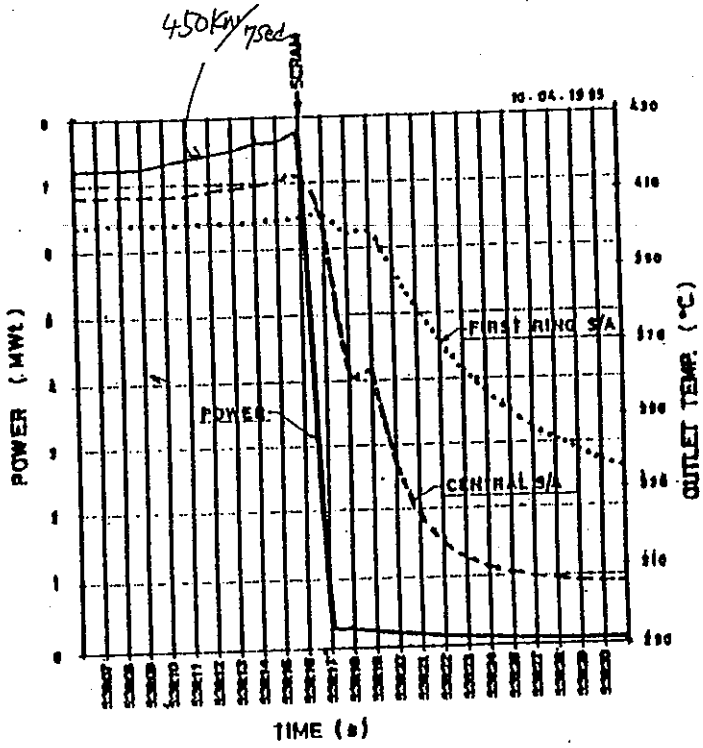


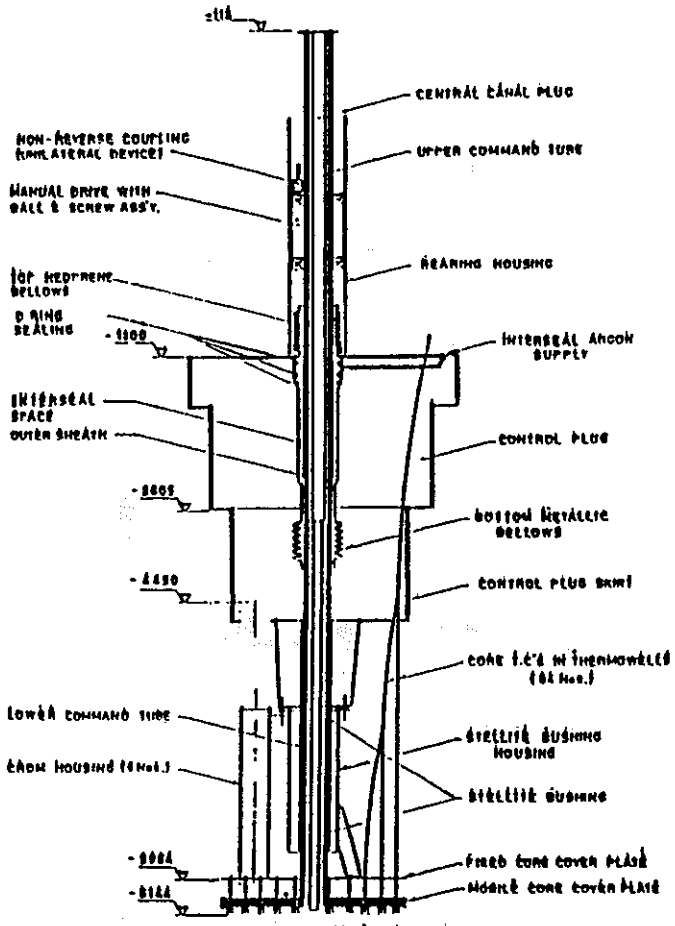
CORE PLAN WITH 25 FUEL S/As

Fig 1



REACTOR POWER & OUTLET TEMP

Fig 2



SCHEMATIC OF CCPM ASSEMBLY

Fig 3

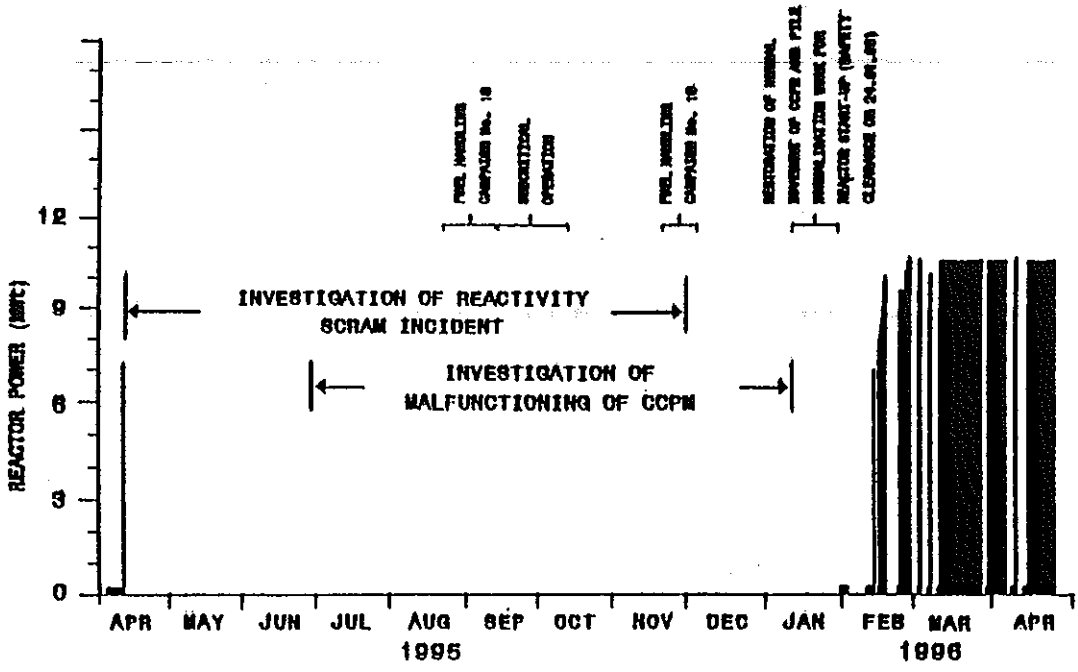
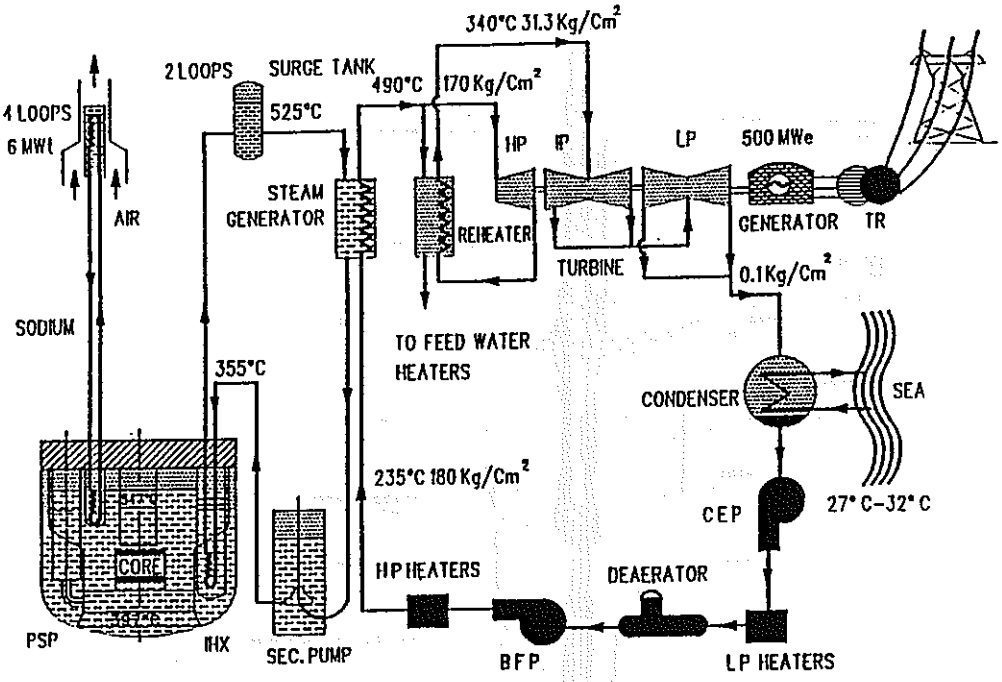
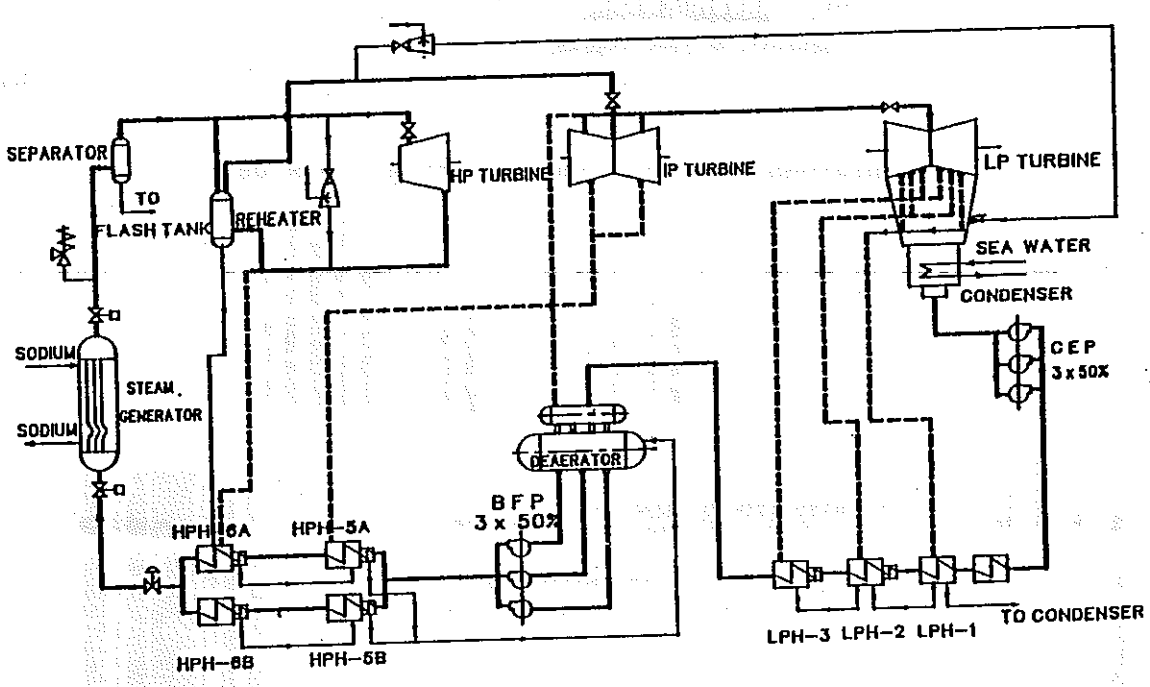


Fig 4



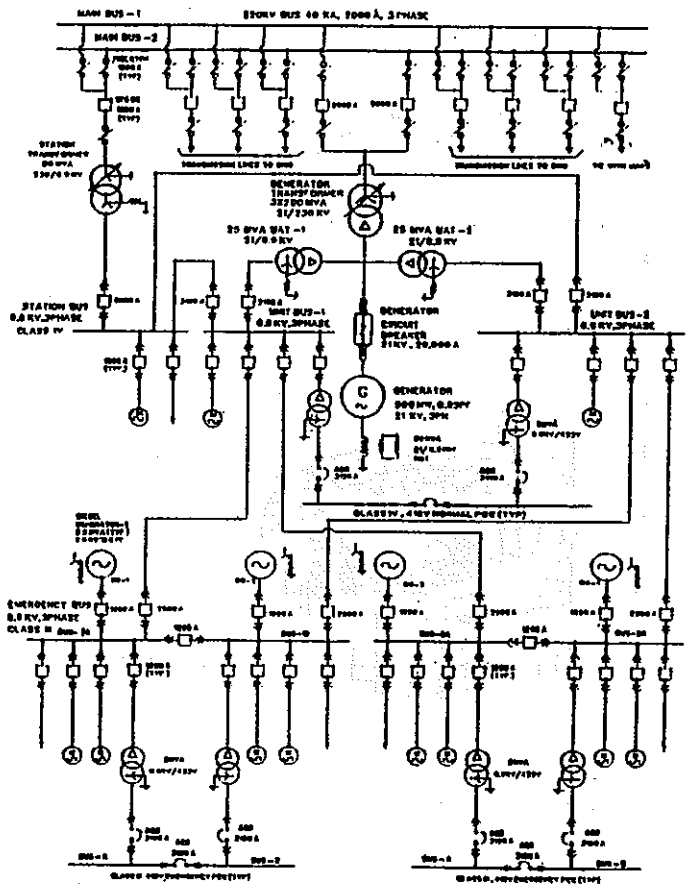
Heat transport flow sheet

Fig 5



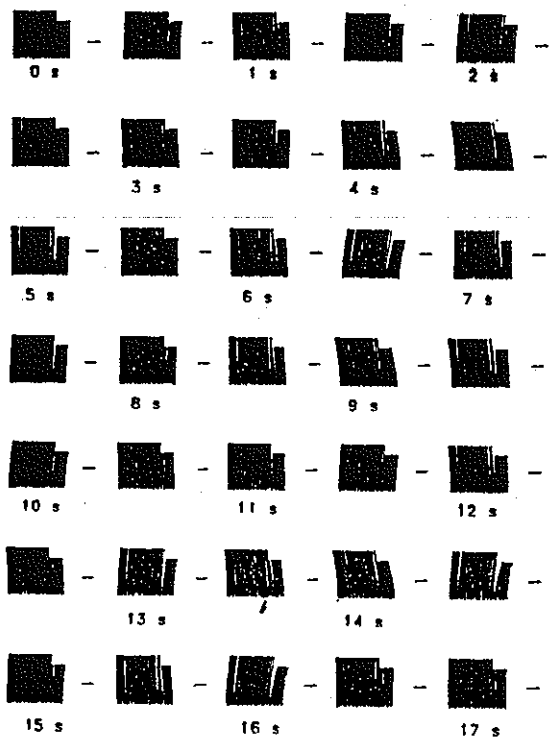
Steam-Water system flow sheet

Fig 6



GRID CONNECTION AND STATION POWER SUPPLY

Fig 7



Dynamic motion of core subassemblies
during seismic event (0.078 g DBE)
Maximum tip displacement = 8 mm

Fig 8

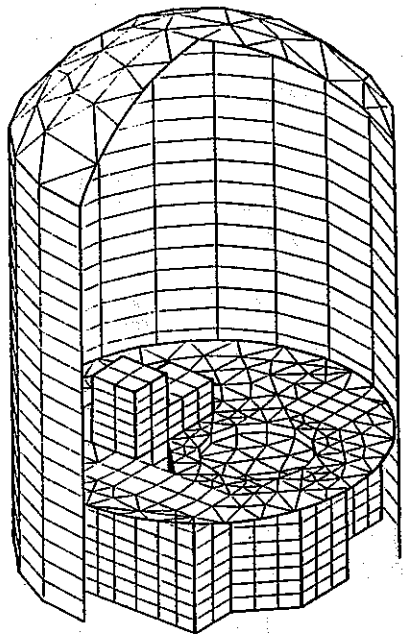


Fig 9: 3-D FEM mesh developed for seismic analysis of RCB

A REVIEW OF FAST REACTOR ACTIVITIES IN ITALY

Edited by: R. Tavoni

Presented at the 29th Annual Meeting of the

International Working Group on Fast Reactors - IAEA

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Abstract

In this paper, Italian activities on liquid metal fast reactors are shown for the period May 1995 - April 1996. During this period the ENEA collaboration with General Electric on ALMR came to an end as a consequence of the reduced effort on the design development. Nevertheless ENEA completed the studies on the PRISM Mod B oxide burner core, the neutronic configuration of which was presented in last year meeting. Some results of the dynamic calculations are shown.

ENEA participated in the IAEA/EC benchmark on the comparative calculations for severe accident in BN-800 reactor. A complete neutronic evaluation has been made including power distribution, Doppler, sodium void and material coefficients.

Activities on sismic isolation are also outlined.

The Italian contribution to SPX restart and operation is described; some information about the complementary convention of the Nersa society is given, together with the Italian industry participation in the SPX restart.

1. ELECTRICITY PRODUCTION AND DEMAND

In 1995 the increase in the energy demand was 3% in comparison to the previous year. The increase has been greater in the Mid-North of Italy (+3.5%), less in the Mid-South (+2.5%) and Sicily (+1.5%) and stagnating in Sardegna (+0.2%).
All the increase has been obtained with a correspondent increase in the national production with a negligible reduction in electricity import.
Oil consumption increased by 3.7% while coal registered an important increase (+21.1%) representing only a partial restoration of the maximum coal production reached in 1990. Natural gas has also presented an important increase (+15.2%). The industrial consumption, increased in 1994 by 4.2%, has presented during 1995 a 4.5% increase in comparison to the previous year. Services consumption increased by 2.4%, domestic consumptions were stagnating (+0.5%), while agricultural consumptions decreased (-3.3%).

TABLE I - Electricity production and demand in Italy during 1994 and 1995/1/

	1994		1995	
	TWh	%	TWh	%
Thermoelectric Energy	180.4	77.9	196.0	81.2
- Oil derivatives	116.3	50.3	120.7	50.0
- Natural gas	40.4	17.4	46.6	19.3
- Solid fuels/2/	23.7	10.2	28.7	11.9
Primary Electricity	51.3	22.1	45.4	18.8
- Hydro	47.6	20.5	41.7	17.3
- Geo	3.7	1.6	3.7	1.5
- Nuclear	--	--	--	--
Total Gross Production	231.7	100.0	241.4	100.0
Net Electricity Import	37.6		37.4	
Total Availability	269.3		278.8	
Energy used by auxiliary production services and by pumping	- 15.7		- 17.7	
Network demand	253.6		261.1	

/1/ Provisional data
/2/ Coal, lignite and others

2. NUCLEAR RESEARCH AND DEVELOPMENT IN ITALY

- Advanced USA Passive LWR designs:

  Both the technical cooperation agreements among Enel, Enea, Ansaldo and Westinghouse on one side and General Electric on another side, that expired at the end of 1995, have been renewed till the end of the century.
  The most important tests that have been performed since the last meeting, were on the Isolation Condenser prototype of SBWR reactor at the Panthers facility of the Siet laboratory in Piacenza. The tests were completed by the end of 1995 and they concluded the overall experimental campaign that was conducted in Italy in support of the SBWR and AP-600 certification.

- Advanced European PWRs:

  . The European Passive Pressurized reactor (EPP, 1000 Mwe) is a design made in cooperation by eight European electric utilities and Westinghouse and its European industrial partners. During the first phase, a reference configuration will be developed to be submitted to the Safety Authority. Enel has been given the chairmanship of the project. The main differences with the AP600 design are: three loops, a double concrete containment boundary, greater limitations of environmental impact in any conceivable accident event, greater margins on the primary containment design pressure, plant designed to cope with external events, both natural and man made.

  . The new European Pressurized water Reactor (EPR, 1400 MWe) is a design for which a cooperation agreement between Enel and EdF was signed in February 1996.

3. DYNAMIC CALCULATIONS ON THE PRISM MOD B OXIDE CORE

The transient calculations performed on the oxide fueled burner core of the 840MWt Prism reactor (MOD-B) concluded an Enea-GE agreement, that has been effective for several years.

Two standard severe accidents, ULOF and UTOP, were calculated using the SAS-3D/PEC code developed by ANL.

- The ULOF case was studied in three different conditions.
  In the first one with coastdown of all pumps, the transient is completely controlled by the intervention of the 18 GEMs.
  In the second condition with four failed pumps the outlet sodium temperature shows a peak of 800°C at 50 second. But it is only in the third configuration with 4 failed pumps and 9 failed GEMs that the outlet sodium

temperature reaches a value very close to the boiling temperature (about 900°C at 40 seconds, see fig. 1).

- The UTOP case was studied with a reactivity ramp of 0.5 $ in 25 seconds and shows a final raise in power approximately 26% over the nominal power of the reactor.

Observing the results of these cases, it can be concluded that the dynamic behaviour of the core is largely satisfactory from the safety point of view.

4. PARTICIPATION IN THE IAEA/EC COMPARATIVE CALCULATIONS FOR SEVERE ACCIDENTS IN BN-800 REACTOR.

Enea is participating in the IAEA benchmark exercise on the BN-800 reactor.

Till now the Enea contribution has been on the neutronic calculations analysing the BN800 heterogeneous reactor core with 3 enrichment zones, proposed by IPPE, at the End Of Cycle (EOC). The geometric model used in the analysis is a 3D (Triang. or Hex, Z) scheme, considering six-meshes plant-points per assembly and 90 axial meshes-points, reaching 62x31x90 total meshes points for the diffusion-theory analysis.

All the diffusion-theory calculations are carried out by the Citation code (finite differences diffusion code) using a 22-groups cross-section library originated from ENDF/B-V.II file and 930 zones (830 fuel zones), while for the sodium void effect evaluations, the results are obtained out by the MCNP4A code (Monte Carlo transport code) using a continuous spectrum library originated from JEF-1 file.

In the diffusion analysis the periodic condition at 120° sector can be reduced to a periodic condition at 60° sector, if we consider that the atomic densities of the "passive" and "scram" control rods are identical.

In the transport analysis, 1.000.000 and 3.000.000 neutron histories are used respectively in homogeneous and heterogeneous keff calculations. The transport analysis has been repeated with and without the lambed fission products in the core, in order to evaluate its influence on the sodium void effect-value.

5. SEISMIC ISOLATION

Seismic isolation (SI) is one of the most significant earthquake engineering developments in the last decade. In Italy since 1988 large efforts have been devoted to the development, validation and application of SI and other innovative antiseismic techniques (e.g. passive energy dissipation and shock transmitters) for civil and industrial structures. This work is being performed in

the framework of both collaborative activities of the Italian Working Group on Seismic Isolation (GLIS) and international cooperative projects with other European and U.S. partners. Several activities are being performed in Italy in the framework of the BE7010 project funded by the European Commission, on the optimization of the High Damping Rubber Bearings (HDRBs). The main studies are outlined below:

- Tests on the individual HDRBs: to date the activities of the Italian partners on this topic have concerned the execution of experiments on non-aged (virgin) HDRBs. Such tests are performed at room temperature at ISMES and the Nuclear Engineering Laboratory (LIN) of the University of Bologna (Enea tests) and at various temperatures at Ansaldo-Ricerche (ARI) (ALGA tests).
  In order to optimize the test quality, Enea significantly improved SISTEM (Seismic ISolator TEst Machine) which is located at ISMES, and designed and manufactured a second equipment (CAT - Creep and Ageing Test) for long duration creep tests, which was installed at LIN.

- Tests on isolated structure mock-ups: In the framework of the BE7010 project, two isolated structure mock-ups are being tested: the first, with a flexible superstructure, is being subjected to shake table tests, while the second, with a rigid superstructure, will be subjected to free-vibration tests. In addition, the use of some HDRBs developed in this project has been planned for pseudodynamic tests on large scale isolated structures at the Joint Research Center of Ispra of the EC.

- Numerical models of the HDRBs: A new simplified model of the HDRBs that can be directly implemented in ABAQUS was developed by Enea and Enel. This model (SEPB) consists of a combination of a non-linear spring with an elastic-plastic beam. By appropriately defining the physical parameters of the system (spring stiffness, Young's modulus and yield point of the beam) it is possible to reproduce the hysteresis cycle of a HDRB, including hardening and/or yielding and hysteretic nature of damping. Work was performed by Enel to define some identification criteria. Enea had already developed SEPB models for the HDRBs used in shake table tests at Ismes. The models have already been used by Enea for the pre-test calculation of shake table experiments.
  According to the excellent results which have already been obtained, SEPB will be the model which will be used by Enea and Enel in the analyses of isolated structures.

- General remarks: The studies described above are considerably extending the knowledge previously acquired in Italy on the behaviour of HDRBs and will make optimized isolators available at reasonable costs. The results obtained so far confirm the SI capability of considerably enhancing the seismic protection of civil and industrial structures, including nuclear plants. They have also provided important input for the development of the standards for seismic isolators and design guidelines for isolated structures.

6. SPX

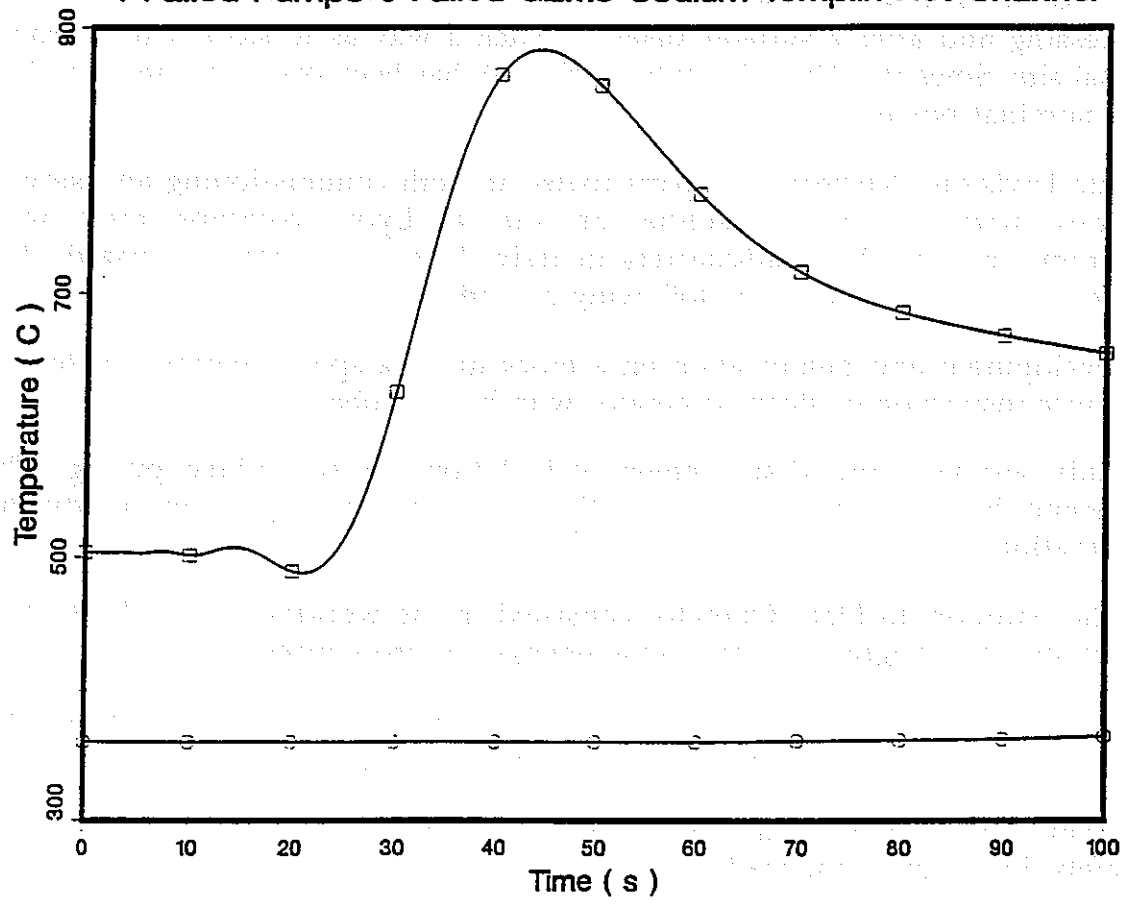
The Nersa society, in which Enel holds a 33% share, made some important decisions last year. A complementary convention has been underwritten by EdF and Enel on 27/2/1995 and by SBK on 27/9/1995, establishing the relative engagements of the three partners till the end of the year 2000. In this convention, the matter of the knowledge acquisition plan that was imposed in July 1994 in the new operation authorization order, is settled.

A further modification of previous agreements concerns the exausted fuel; present agreements allow Enel to take its share of exausted fuel elements without reprocessing and after a suitable delay, in such a way as to store it in a proper disposal site. Since the 22nd December 1995 SPX has been operating again at 30% of the nominal power.

Ansaldo Divisione Nucleare has participated in both commissioning and service activities, with its own personnel on site, at Lyon (common team with Framatome) and at the headquarters in Italy. In such a context, Ansaldo has developed or contributed to the following activities:

- Development and qualification on a mock-up of a special equipment for in-service inspection of steam generator with helical tubes.
- Study analysis and displacement calculations of secondary piping. The method has been used as a real-time tool for control of piping during operation.
- Other support to Plant Operator consisted in the working-out of the system periodic-test programmes and some emergency procedures.
- Design of minor modifications on secondary and auxiliary circuits (re-setting of safety valves, realization of sodium-aerosol traps).
- Preliminary feasibility analysis and initial organization of a Probabilistic Safety Study under the leadership of EdF.
- Updating of the safety analysis of the Fuel Hadling Route.

Fig.1 - Prism MOD-B - ULOF/BOEC
4 Failed Pumps 9 Failed GEMs-Sodium Temp.in Hot Channel



□ = Outlet
○ = Inlet

PNC TN1410 96-026

A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

April 1996

Power Reactor and Nuclear Fuel Development Corporation

This document was prepared for the IAEA Technical Committee Meeting(May 14 - 17, 1996, Aktau, Kazakhstan) on "Progress in Fast Reactors Development - 29th Annual Meeting of the International Working Group on Fast Reactors(IWGFR)" by Power Reactor and Nuclear Fuel Development Corporation(PNC), with contribution from The Japan Atomic Power Company(JAPC).

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Abstract

The main R&D results of Japanese activities are summarized as follows: (1) the experimental 140 MW(th) sodium cooled fast reactor "Joyo" provided abundant experimental data and excellent operational records, attaining more than 50,000 hours of operation since its first criticality in 1977; (2) the prototype 280 MW(e) fast reactor "Monju" reached initial criticality on 5 April 1994, presently Monju is under the cold shutdown state because of secondary sodium leak on 8 December 1995, and multiple cause investigations of the sodium leak are being performed; (3) the Japan Atomic Power Company is promoting design studies for demonstration fast reactor (DFBR) with a power output of 600 MW(e) and R&D for DFBR are being conducted under the cooperation of governmental and private sectors.

1. General Review

1) The Japan Atomic Energy Commission (JAEC) promulgated Japanese "Long-term Program for Development and Utilization of Nuclear Energy" in June 1994. In the program, followings were concluded:

   a) because of the scarcity of energy resources, plutonium utilization is the key development for long-term energy supply and energy security of Japan; Japan intends to steadily and systematically carry forward development of fast breeder reactors,

   b) the research and development for demonstration FBRs (DFBRs) should be done with the cooperation of governmental and private sectors,

   c) utilities should play the major role in design, construction and operation of the DFBR aiming at the commercialization by the year about 2030 through construction of two DFBRs with a step-by-step improvement of technologies and economics; the start of construction of DFBR-1 is expected in the early 2000's in the program.

   d) Power Reactor and Nuclear Fuel Development Corporation(PNC) should play the central role in technological development over the entire period to commercial commissioning of FBRs through continuous long-term independent R&D efforts aiming at the establishment of technological system particular to FBRs.

2) The experimental reactor Joyo, located in the O-arai Engineering Center(OEC) of PNC, has provided abundant experimental data and excellent operational records attaining 51,200 hours operation in total by the end of March 1996, since the first criticality in 1977.

3) On the prototype reactor Monju, the initial criticality was achieved in April 1994. The nuclear heating test was started in February 1995, and the first connection to the grid was

succeeded in August 1995. In the power -up test, however, a sodium leak occurred in the C-loop of the secondary heat transport system on December 8, 1995. Presently Monju is under the cold shutdown state and multiple cause investigations of the sodium leak are being performed.

4) As for the demonstration fast breeder reactor (DFBR) of Japan, the Japan Atomic Power Company (JAPC) conducted conceptual design studies for the past several years, and confirmed the feasibility of top entry loop type reactor concept. Based on the results of the design studies, the Federation of Electric Power Companies (FEPC) decided in January 1994 to start construction of the DFBR plant at the beginning of the 2000's. FEPC also decided the basic specifications of the DFBR plant.

   The related research and development (R&D) works are underway at several organizations under the discussion and coordination of the Japanese FBR R&D Steering Committee, which was established by JAPC, PNC, Japan Atomic Energy Research Institute (JAERI) and Central Research Institute of Electric Power Industry (CRIEPI).

   Progress of the design study and the related R&D are reported to the Subcommittee on FBR Development Program of JAEC.

5) Recent major emphases on the PNC's R&D are placed on the development of technologies specific to FBRs as well as the integrated feedback of all existing R&D results and experiences to the development of DFBR.

   Furthermore, the overall functional and performance tests of Monju, is another important key role to attain further excellency of FBR technology, with full efficient usage of the test results.

6) R&D on following tasks are also in progress for development of DFBR, for excellent technology to attain FBR commercialization, and for technological breakthrough.

   a) improvement of reactor core safety
   b) improvement of plant safety
   c) development of measures for probabilistic safety assessment
   d) development of high performance core and fuel
   e) study for various fast reactor core
   f) development of structure and material for high temperature system
   g) improvement of sodium technology
   h) synthesis assessment of Monju data
   i) synthesis assessment and improvement of plant technology using Monju
   j) design study of plant system

7) In addition to the MOX fuel fabrication at the Plutonium Fuel Fabrication Facility for Joyo, Fugen (ATR), and BWRs in Japan, a new Plutonium Fuel Production Facility (PFPF) was constructed at Tokai Works of PNC. PFPF started production of initial core fuel of Monju in October 1989 and completed in January 1994.

8) On the FBR fuel recycling, adding to the experiences at the Tokai Reprocessing Plant, R&Ds are underway at three Engineering Demonstration Facilities (EDF-I, II, III) and Chemical Processing Facility (CPF), integrating the results to the design of Recycling Equipment Test Facility (RETF) and future FBR Fuel Recycling Pilot Plant. The construction of RETF was initiated in January 1995.

9) Following the national program on waste management, PNC is also actively contributing to the area of vitrification of high level liquid waste, geological disposal of it, and low level transuranium bearing waste treatment, and promotion of construction of a storage engineering center in Hokkaido.

10) Aiming to the age of future FBR commercialization, further extensive and effective collaboration with foreign institutions will also have to play an important role.

2. Experimental Fast Reactor Joyo

2.1 General Status

This report covers the activities of Joyo from April 1995 through March 1996.

Various tests of nuclear and plant characteristics of the MK-II core were carried out successfully during this period. Figure 2.1 shows the core configuration for nuclear and plant characteristics test. Major items of the experiments are as follows :

1) Power reactivity coefficient
2) Isothermal reactivity coefficient
3) Heat transfer characteristics of dump heat exchangers

After this test operation, the 11th periodical inspection has begun in May 1995.

The operating history of Joyo is illustrated in Fig.2.2. The total operation time as of March 31, 1996 is about 51,200 hours since the initial criticality in 1977, and also the accumulated thermal power is about 4,160 GWh.

A large number of irradiation tests, such as for the nitride fuel and the carbide fuel, are now in progress. The creep test of fuel cladding materials under irradiation with MARICO* is also in progress.

* Materials Testing Rig with Temperature Control

2.2 Upgrading Project of Joyo(MK-III project)

The Joyo modification project named the MK-III project is planned to upgrade its irradiation capability. The safety review for the MK-III project by government has been completed in September 1995. The plant modification schedule will be discussed because of the influence of the Monju secondary sodium leak accident.

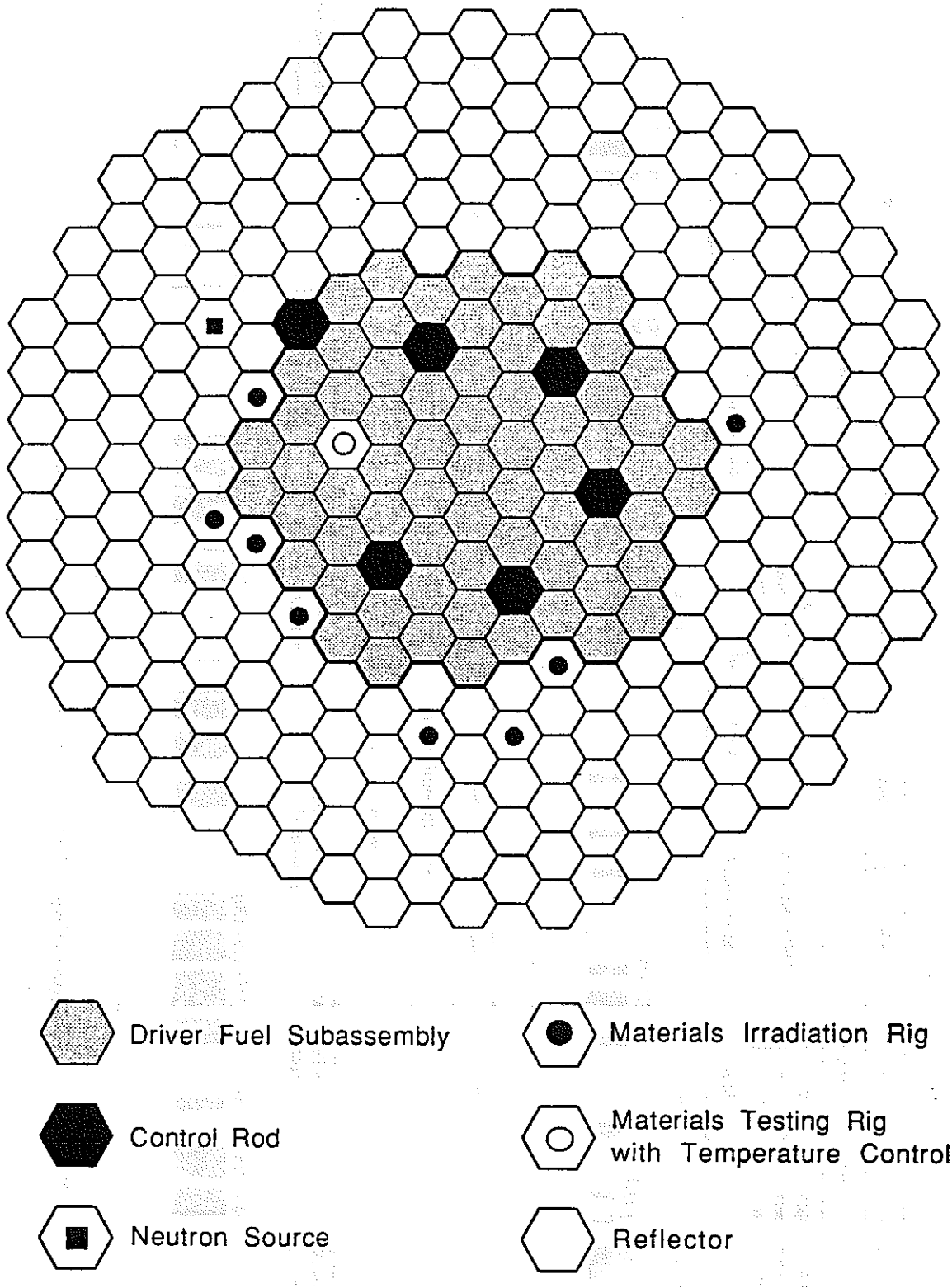


Fig.2.1 Core Configuration for Nuclear and Plant Characteristics Test

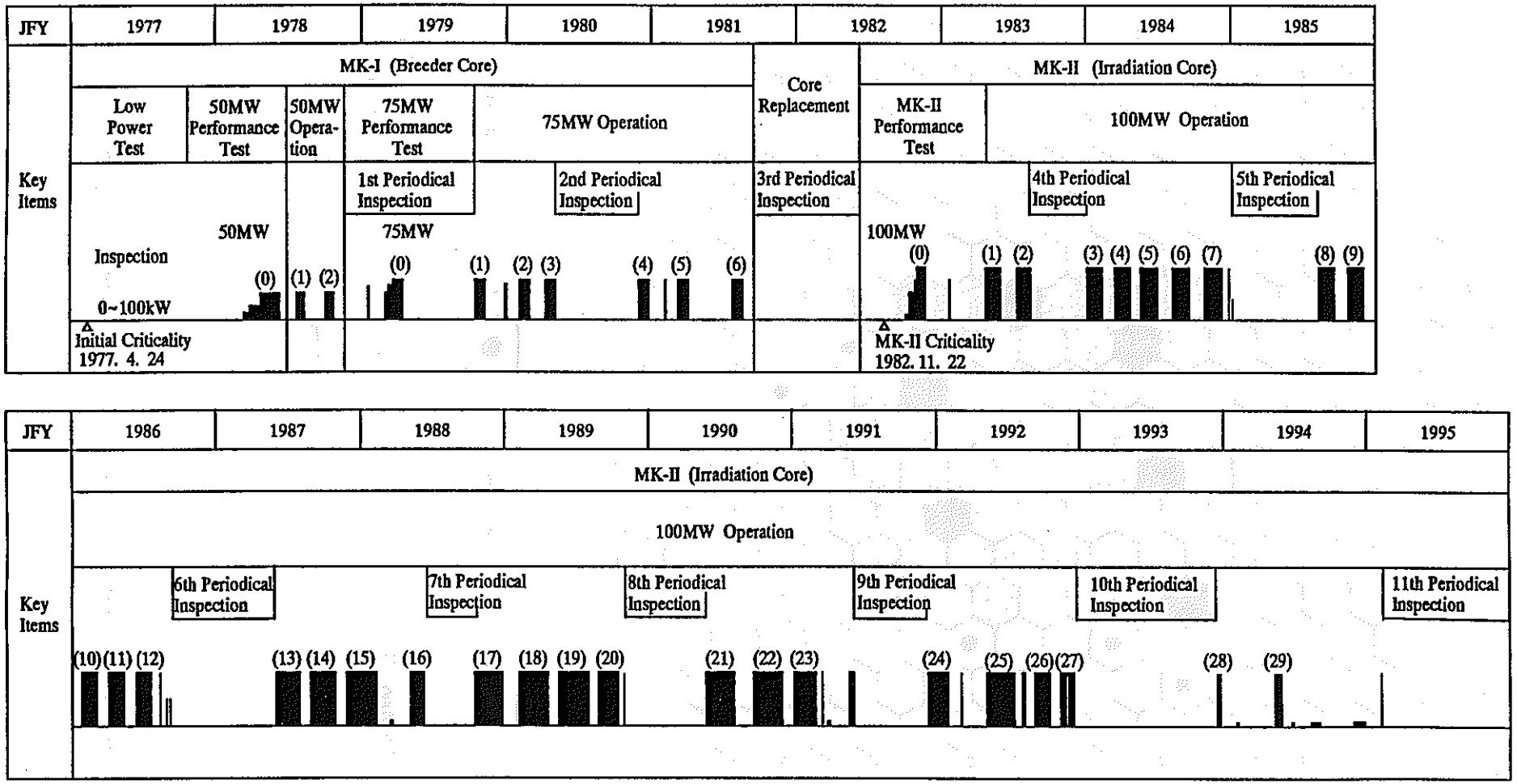


Fig. 2.2 Operating History of JOYO (As of March, 1996)

3. Prototype FBR, Monju

3.1 Construction Schedule

The Monju site is located on the north side of the Tsuruga Peninsula in the central Japan, facing the Sea of Japan and is surrounded by mountains of approximately 300-700m high. Since the plant is located inside the Wakasa Bay Quasi-national Park, its construction works have been carried out with special attention to the environment.

Major milestones of the construction schedule (shown in Fig. 3.1) are as follows;

Oct. 1985	Start of Construction
Apr. 1987	Completion of Construction of the Reactor Containment Vessel
Oct. 1988	Installation of Reactor Vessel
Apr. 1991	Completion of Construction

3.2 Present Status

The equipment installation was finished and the construction was completed in Apr. 1991. The function tests were performed from May 1991 to Dec. 1992. After that, the start-up tests was performed. The reactor power was increased gradually from Feb. 1995. Generating electricity and connecting to the grid was started in Jul. 1995. During 40% start-up test, the reactor was manually shut down due to the sodium leak in the secondary heat transport system (SHTS), Loop C in Dec. 1995.

3.3 Pre-operational Tests

Pre-operational tests were composed of Function Tests and Start-up. The schedule in Fig. 3.2 shows the general outline of the Pre-operational tests.

1) Function Tests

Function tests were carried out from May 1991 to December 1992.

Function tests were conducted to confirm the function and performance of the plant systems, and were composed of various tests and inspections during fabrication and installation of the components in Monju.

2) Start-up Tests

The purpose of the start-up tests is to confirm and to evaluate the performance of the core, the plant systems and the components.

The start-up tests consist of criticality test, reactor physics test, power-up test, etc.

First of all, the pre-performance test of plant system was conducted to examine plant characteristics under high temperature condition without nuclear heating.

Initial criticality was achieved with 168 fuel assemblies in Apr. 1994.

After the criticality test, the reactor physics test was performed and core reactivity worth, core reaction rate distribution , core flow rate distribution, etc. were measured.

The nuclear heating test was started in Feb. 1995 and the reactor power was increased gradually. Monju was connected to the grid in Aug. and then power-up test was started. In the power-up test, plant characteristics under power operation and transient condition was confirmed up to 40% power operation.

When power was being increased for the plant trip test, as a part of 40% power test, a sodium leak occurred in SHTS, Loop-C in Dec. 1995. The leak took place near the outlet of the secondary side of the intermediate heat exchanger (IHX). (Fig. 3.3)

The reactor was manually shut down to cold shutdown state, and the sodium in the primary and the secondary loops of the affected Loop C was drained.

Presently the examinations of the leak part of the temperature sensor have been performed, and also multiple investigations of the cause of the accident are being performed.

Table 3-1 **Principal Monju Plant Design Characteristics**

Reactor Type		Sodium cooled FBR, loop-type
Thermal Power		714 MW
Gross Electrical Power		280 MW
Core	Equivalent Diameter	Approx. 180 cm
	Height	93 cm
	Volume	2340 lits.
Fuel		PuO_2 - UO_2
Pu Enrichment	(Pu fissile)	(Inner core/outer core)
	Initial Core	15/20%
	Equilibrium Core	16/21%
Fuel Inventory	Core(U+Pu metal)	5.9 ton
	Blanket (U metal)	17.5 ton
Average Burn-up		Approx. 80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Permissible Cladding Temperature (middle of thickness)		675 °C
Power Density		275KW/lit
Blanket Thickness		Upper 30 cm
		Lower 35 cm
		Radial 30 cm
Breeding Ratio		Approx. 1.2
Reactor inlet/outlet Sodium Temperature		397/529 °C
Secondary Sodium Temperature (IHX inlet/outlet)		325/505 °C
Reactor Vessel (height/diameter)		18/7m
Number of Loops		3
Pump Position (Primary and Secondary Loop)		Cold Leg
Type of Steam Generator		Helical Coil, once-thorough Unit Type
Steam Pressure (Turbine Inlet)		127 kg/cm^2 g
Steam Temperature (Turbine Inlet)		483 °C
Refueling System		Single Rotating Plug with Fixed Arm FHM
Refueling Interval		Approx. 6 Months

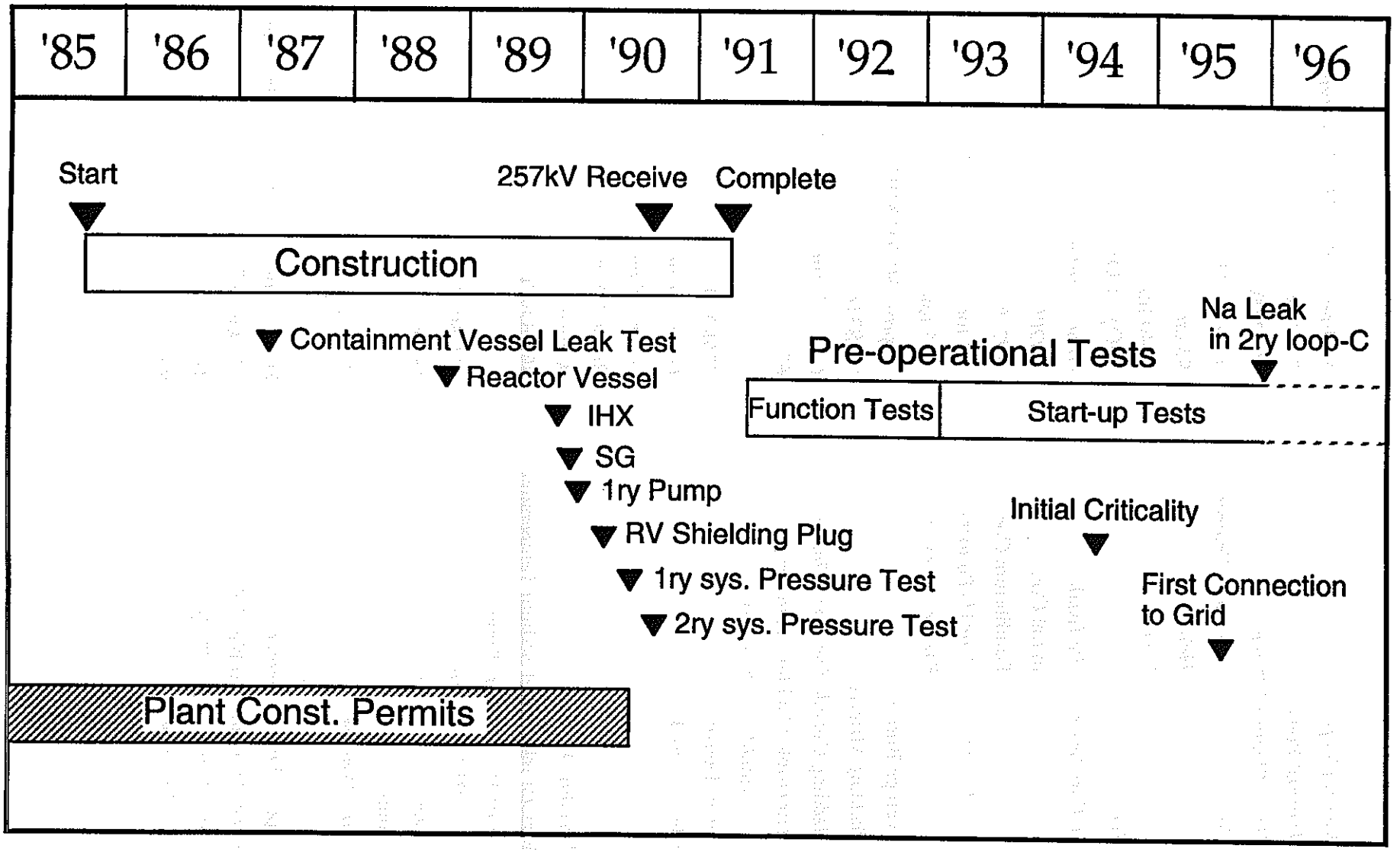


Fig. 3.1 Monju Construction & Tests Schedule

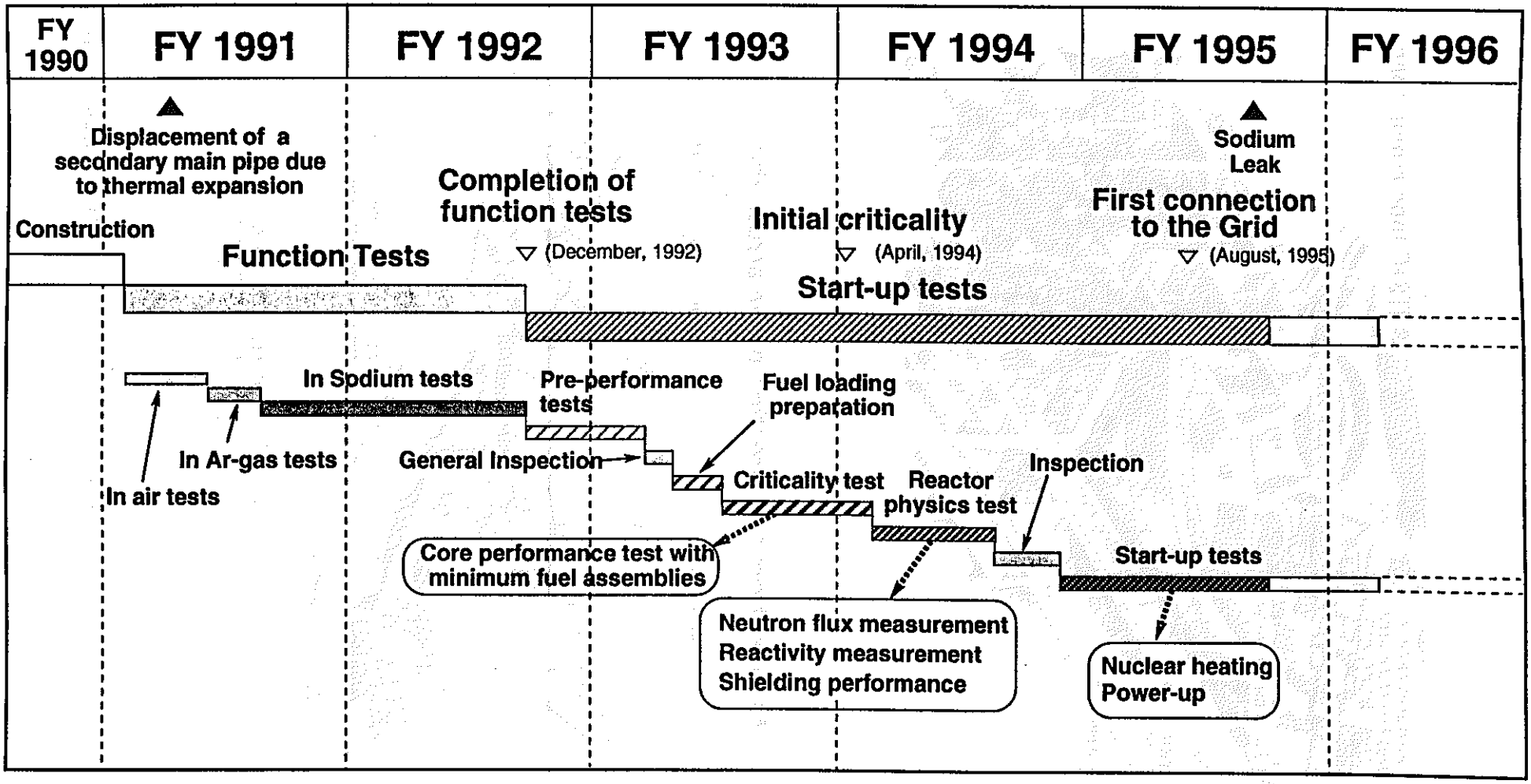
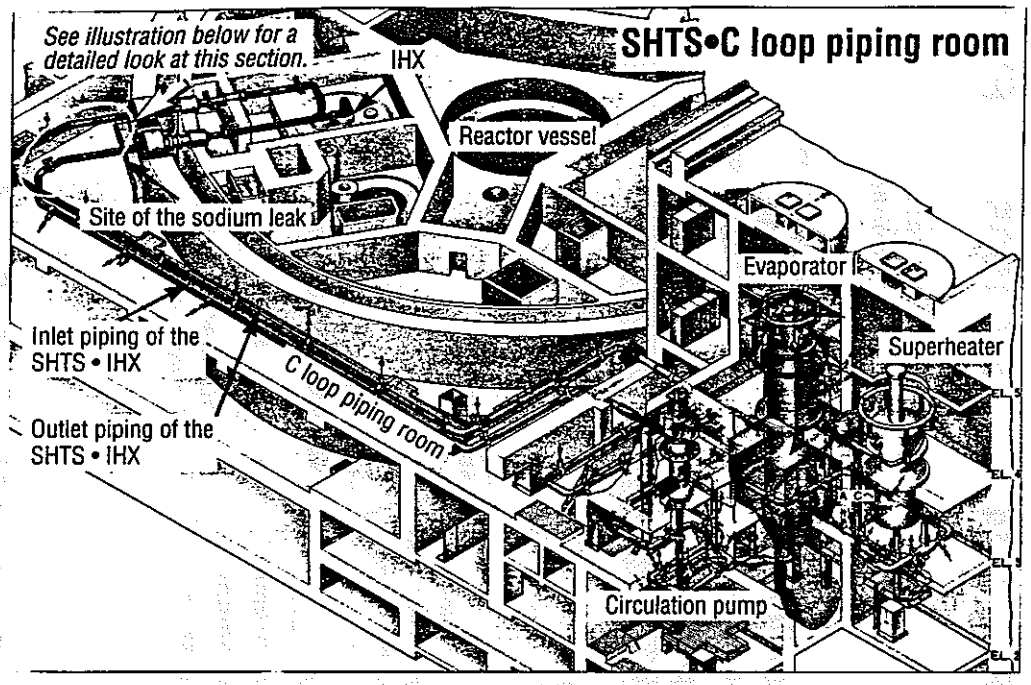


Fig 3.2 Schedule of Monju Pre-operational Tests



As seen from the direction of the arrow above

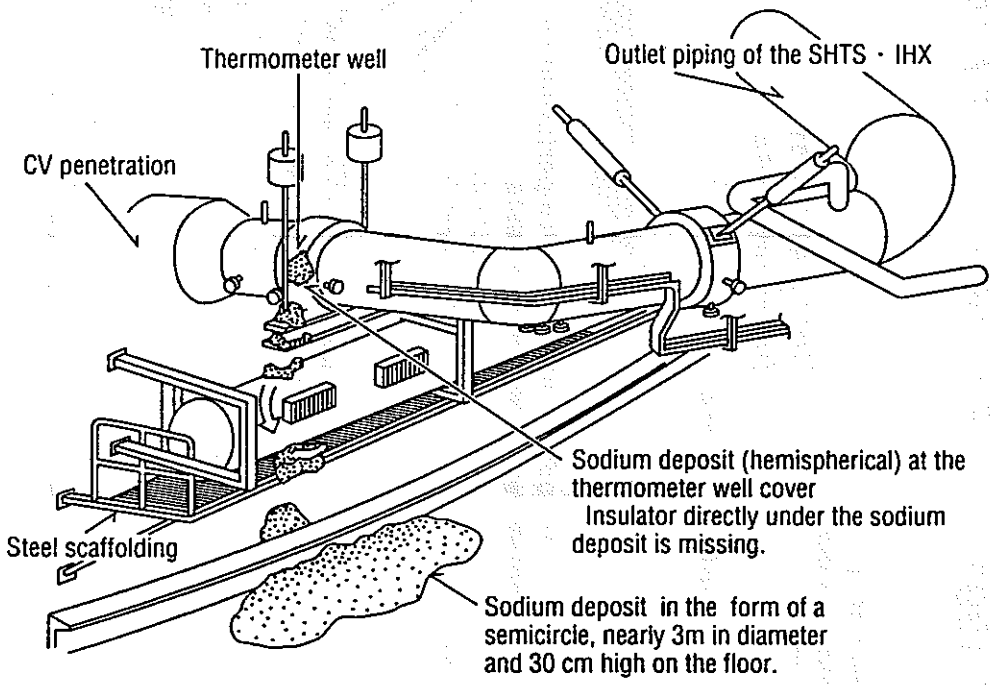


Fig 3.3 Schematic view of a sodium leak in SHTS C loop piping room

4. DFBR and PNC's Design Study

4.1 Overview

The Japan Atomic Energy Commission (JAEC) promulgated Japanese "Long-term Program for Development and Utilization of Nuclear Energy" in June 1994. In the program, it was concluded that the research and development for demonstration FBRs (DFBRs) should be done with the cooperation of governmental and private sectors, and that utilities should play the major role in design, construction and operation of the DFBR aiming at the commercialization by the year about 2030 through construction of two DFBRs with a step-by-step improvement of technologies and economics.

The start of construction of DFBR-1 is expected in the early 2000's in the program.

4.2 Design Study of DFBR

The Japan Atomic Power Company (JAPC) has conducted for the past several years conceptual design studies of the Demonstration Fast Breeder Reactor (DFBR) in accordance with the basic policy of the Federation of Electric Power Companies (FEPC) and has confirmed the feasibility of top entry loop type reactor.

Based on the result of these design studies, FEPC decided in January 1994 to start the construction of the DFBR plant in the early 2000's, and the major specifications.

The optimization study of the DFBR has been conducted by JAPC for three years since FY1994, based on the major specifications and those design studies. These studies have been focused on core safety enhancement including the application of gas expansion modules (GEM), increase of design margins for the structural tolerance of the containment facilities, feasibility and licensability of seismic isolation plant for introduction to the DFBR. Through these studies, the goal to make the design of the whole DFBR plant harmonious with both safety and economic viability, and to prove preparation for the basic design.

4.3 PNC's Design Study

In 1988, PNC started plant design study applying key technologies such as reactor vessel head access concept and performed plant construction cost evaluation.

PNC conducted design study on a 600MWe-size plant for 1990 - 1991. In 1992, JAPC and PNC discussed on the mutual design results to improve the demonstration reactor design study. Design study on a 1300MWe-size plant featured with passive safety using nitride fuel was completed at PNC in 1995.

PNC is now studying advanced recycling system using new type fuel (nitride, metal) and minor actinide, and is conducting conceptual design study of Recycle Test

Reactor focusing on core design in order to study FBR core variety such as plutonium burning, minor actinide burning and use of new type fuel. Necessities of the Recycle Test Reactor are being discussed at the Subcommittee on Advanced Recycling Program of JAEC.

5. Reactor Physics

5.1 Development of Analytical Method

The effort to develop a neutron transport code based on an improved nodal method has been continued in order to treat the Hex-Z geometry of FBR cores more accurately. In the previous version of the nodal code, the axial leakage from a node was approximated by a second order polynomial expansion of the flux, which occasionally leads to the difficulty of numerical convergence, especially in the case of control rod inserted core. A method to evaluate the boundary flux distribution interpolating the fluxes of axially adjacent nodes was newly developed to overcome such steep flux-gradient cases. A benchmark test of an FBR core showed the new treatment of axial leakage could obtain the flux convergence within the error of o.1%dk for keff, even in the case where the previous method failed.

The continuous-energy Monte Calro method, which has no analytical modeling errors in principle, has become more realistic in Japan because of both the release of an efficiently vectorized code and the rapid improvement of computer hardware. PNC is now utilizing the Monte Carlo method as a tool to verify the standard deterministic analytical system which uses ABBN-type 70-group constant sets and three-dimensional diffusion and/or transport codes. A comparison between the Monte Carlo method and the standard deterministic method agreed well within the difference of 0.1%dk for the keff of ZPPR-9 FBR benchmark core.

5.2 Critical Experiment and Analysis

In order to improve the design method and accuracy of large FBRs, extensive work has been performed to accumulate and evaluate many kinds of experimental and analytical results from the past fast reactor physics study. As a part of efforts to develop the standard database for FBR core nuclear design, a latest version of nuclear data library in Japan, JENDL-3.2, was applied to the large FBR core critical experiment JUPITER, and the results were compared with those of the previous library, JENDL-2. One particular feature of JENDL-3.2 is the good prediction of sodium void reactivity within 10%, on the other hand, the former JENDL-2 largely overestimated it by 20-40%. From

a sensitivity study, it is found that sodium inelastic and Pu-239 capture reactions, which had not been considered important before, mainly contributed to the great improvement of JENDL-3.2, although many other nuclide-reactions had opposite effects and canceled with each other.

5.3 Cross-Section Adjustment

Compared with critical experiments, the power reactors have some special characteristics such as changes of fuel compositions with core burnup and increase of fuel temperature due to heat generation. About the burnup characteristics, PNC has successfully developed a code system to calculate the cross-section sensitivity of those, based on a generalized perturbation theory including burnup chain reactions. On the other hand, the present cross-section adjustment method cannot treat the Doppler reactivity which is dominated by resonance-peak broadening of cross-sections. An effort has been continued to extend the applicability of the cross-section adjustment and design accuracy evaluation system to the Doppler effect. This year a cross-section set including self-shielding factors was preliminarily adjusted by Doppler data of ZPPR experiment as well as other general core parameters. As a result, the C/E (Calculation/Experiment) values of the Doppler experiment were found to become close to unity after the adjustment without any harm to prediction of other core characteristics.

5.4 Shielding Experiment and Analysis

Post-analysis of Japanese-American Shielding Program for Experimental Research (JASPER) was almost completed. Analytical results of JASPER were effectively utilized in the Demonstration FBR shielding design review, which was conducted in the joint study of Japan Atomic Power Company.

As the next step of shielding study, a new work was initiated to develop a standard database for FBR shielding design this year. In the database, the experimental and analytical information from JASPER and other shielding experiments will be evaluated and compiled in a systematic and consistent manner. As a part of the efforts, typical radial shielding experiments of JASPER is now being re-analyzed with the latest version of nuclear data library, JENDL-3.2, and compared with the results of the former JENDL-2.

6. Systems and Components

6.1 Reactor Shutdown System

Research on the self-actuated shut-down system (SASS) by using a curie point magnet is in progress. A system performance test on SASS by use of Joyo is under planning based on various out pile characteristic tests conducted in air and sodium environment.

6.2 Process Instrumentation

A development of ISI test equipment for Monju was completed in O-arai engineering center(OEC). Mock-up test rigs and ISI test equipments at OEC were disassembled and transferred to Monju from OEC.

A development of the visualization technique of heat transfer tube inner side by using Laser is in progress.

6.3 Steam Generator

PNC precedes a conceptual design study for a future FBR plant having a steam generator in the primary heat transport system. The studies on a double-wall tube(DWT) steam generator have been in progress regarding structural integrity and thermal hydraulic performance of the DWT steam generator.

The 1MW scale DWT steam generator test was completed and post test examination is underway.

6.4 Sodium Component

An in-sodium performance test of a magnetic flux concentration type electro magnetic pump (EMP) was completed. An evaluation of the EMP was completed.

6.5 Thermal Hydraulics

A verification of plant dynamics simulation code (Super-COPD) by using Monju functional tests was carried out and the accuracy of the models was validated. Predictions for the Monju start-up test were carried out beforehand and the verification by using the test is planned.

7. Fuels and Materials

7.1 Fuel Fabrication

The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems started fabricating Joyo and Monju fuels from October 1988.

7.2 Fuel Pin Performance

Fuel pin performance codes of MOX type pellet for analyzing the steady state and transient conditions have been improved since 1984, with the data of PNC/DOE operational reliability testing program in EBR-II, etc. The modification of codes for annular MOX fuel is emphasized with the irradiation data base. A three dimensional FEM code for an analysis of defected fuel pin behavior has been modified with PIE results of Run Beyond Cladding Breach (RBCB) pin in EBR-II. Development of nitride fuel performance code is also in progress.

7.3 Core Material

PNC1520 advanced austenitic steel was developed for high burnup FBR fuels. Improved performance than current PNC316 steel is demonstrated by the out-of-reacter and material irradiation testing programs. Design base standard of PNC1520 is completed and evaluation study is in progress to apply Monju and DFBR cores.

7.4 Irradiation Experiments

1) Joyo

Fuel pin irradiation continues with advanced austenitic cladding fuel pins and high strength ferritic cladding fuel pins. Large diameter annular pellet fuel pins for Monju high burnup and DFBR cores are also included in the test.

The fuel subassembly with CEA austenitic stainless steel cladding tubes has also been irradiated since 1988 and reached the final stage at 125GWd/t.

2) Foreign Reactors

Phase-II program of PNC/DOE collaborative operational reliability testing in EBR-II has been successfully completed by the end of 1995. Large amount of data base on TOP (transient over power) and RBCB (run beyond cladding breach) tests is summerized for typical MOX fuel pins.

The PIE of fuel subassemblies of PNC316 steel and PNC1520 advanced austenitic stainless steel irradiated in FFTF is in progress. The shipment of irradiated fuel materials to PNC O-arai was continued to take a further investigation of high burnup fuel behavior.

7.5 Development of Advanced Fuels

Feasibility study of advanced fuels (nitride, metal, carbide) has been conducted since 1986. PNC/JAERI irradiation test of nitride and carbide pins has been conducted in Joyo since 1994.

Mixed carbide fuel pins were also irradiated using the thermal reactor JRR-2 and JMTR of JAERI.

7.6 Post Irradiation Examination

Construction of PIE facility of Fuels Monitoring Facility (FMF-2) adjacent to existing FMF-1 at OEC has completed to handle the large fuel assemblies irradiated in FBR Monju and ATR Fugen. Hot operation of FMF-2 will start from late 1997.

8. Structural Design and Materials

8.1 Development of Structural Design Method

1) FINAS nonlinear structural analysis program

Effort to entire the capavility of the general purpose nonlinear structural analysis program FINAS is continued, particularly with respect to adoptive mesh generation based oann r-method and h-method. FINAS was mounted as the solver in CAE systems such as CADAS, ATLAS, FEMAP and so on. FINAS is currently used by many engineers at about 40 sites including fabricators and universities. The latest version, V12.0, was translated into English.

Personnal computer version of FINAS was developed.

2) Improvement of Elevated Temperature Structural Design Guide

The following rules are investigated to improve and extend the current Elevated Temperature Structural Design Guide.

i) Creep-fatigue design methods based on elastic analysis

A new creep-fatigue method, which is based on the concept of a general elastic follow-up model, is being developed. The elastic follow-up equations to predict strain magnification and creep relaxation for structural discontinuities are established.

ii) Design rules for weldment

A new design approach, taking into account the metallurgical and geometrical discontinuities inherent in weldment, is being pursued.

iii) Strain limit critera

A ratchetting criteria for multiaxial stress state, which are not provided explicitly in the design guide, was developed, and is being examined the applicability of the criteria to general components by FE Analysis.

8.2 Structural Test and Evaluation

Structural tests are being performed to improve strength prediction methods, to evaluate the adequacy of elevated temperature design rules, and also to verify advanced nonlinear structural analysis methods.

1) Thermal creep-fatigue test with small sodium loops (STST)

A thermal creep-fatigue test, whose specimen was a new test model "a cylindrical shell with cross-section gradually step-changing" is started and is continued with use of the test facility (STST).

2) Thermal transient tests in large sodium loop (TTS)

A thermal transient test of a vessel with fillets model is now under way. A new test model, made of FBR grade 316 stainless steel, was fabricated.

3) Distortion tests of fuel sub-assembly duct (CMDT, WFT)

A new R&D, which is to investigate the distortion behaviors of internal structures, was started. Firstly a fundamental distortion tests of ducts of a fuel sub-assembly were performed with use of new equipment, CMDT and WFT, and then a distortion tests of several fuel sub-assemblies will be stared in near future.

8.3 Seismic Test and Analysis

1) A conceptual study on the vertical seismic isolation system for FBR components is underway. A series of shaking table tests and analytical works are included to assess the feasibility of the system.

2) Seismic analysis method development and verification on FBR core in the framework of "IAEA/IWGFR Coordinated Research Program on Intercomparison of LMFBR Seismic Analysis Codes" has completed in 1995.

8.4 Fracture Mechanics Tests and Evaluation

Both deterministic and probabilistic fracture mechanics methodologies are being development for the integrity assessment of flawed or cracked structure.

Computer codes developed at PNC as CANIS-J for calculation of fracture mechanics parameters, CANIS-G for simplified crack propagation analysis were modified further.

Crack propagation test with a cylinder with an axial temperature gradient is continued.

8.5 Structural Material Tests and Evaluation

Structural material tests in air, in sodium, in water/steam, and under post-irradiation condition have been conducted to revise the Monju Material Strength Standard and to prepare a new version for DFBR.

The test program in air and in sodium environment is called "Capella" program and the step-1 program(1985-1987) , the step-2 program (1988-1990) and the step-3(1991-1993) were already completed. The step-4 are currently underway with emphasis on long-term extrapolation.

The post-neutron irradiation tests are underway within the scope of neutron irradiation program "Spica".

1) Tests in Air

The present Capella step-4 program includes following subjects;

- Validation of long-term extrapolation of a new criterion for creep-fatigue failure strength of weldment, inelastic constitutive equations on new materials(modified 9Cr-1Mo steel and FBR grade 316 stainless steel)
- Improvement of LBB evaluation method for FBR plant.
- Development of the material strength standard for modified 9Cr-1Mo steel and FBR grade 316 stainless steel

A new concept creep-fatigue failure criterion was proposed using secondary creep basis ductility exhaustion for stainless steels. Applicability of the present criterion to modified 9Cr-1Mo steel was validated by the mechanical and metallurgical studies. The equations of cyclic plasticity of FBR grade 316 stainless steel were revised based on the data-base consisted with low cyclic fatigue and tensile test data.

2) Test in Sodium and Water/steam

Mechanical strength (tensile, creep fatigue, creep fatigue) tests on 316FR (nitrogen controlled) in sodium are still continued in the program to evaluate the carbon and nitrogen transfer effects. Low-cycle fatigue and creep fatigue tests on modified 9Cr-1Mo steel in water/steam environment were conducted.

3) Tests in Irradiation Environments

Surveillance tests for the Class 1 components of Joyo were conducted to confirm the integrity of the reactor by evaluating irradiation effects of the same materials.

The test data were used for the planning of Joyo operating program.

Tests for the Class 1 components of Monju to evaluate irradiation effects on the mechanical properties up to the end of design life and to evaluate irradiation effects on the Material Strength Standard for Monju are also in progress.

Both forged and rolled SUS304 steels were irradiated in Joyo using SMIR (Structural Material Irradiation Rig).

Another test for DFBR was conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition, grain size and production process.

Several post irradiation material tests and in-pile creep tests on FBR grade 316 stainless steel were continued in Joyo and JMTR (Japan Material Test Reactor of JAERI) in accordance with the R&D program Spica step-2.

4) Data Processing System

Material data are complied using specific data coding sheets, and the data inputs to the computer data processing system SMAT are still continued.

Entry data in SMAT are currently more than 12,000 data points on 11 different kinds of mechanical tests (including tensile, low cycle fatigue, creep) for 10 kinds of FBR structural steels.

9. Safety

9.1 Safety Evaluation for Normal and Abnormal Events

Safety evaluation studies have been conducted for confirming the physical phenomena and integrity of the fuel subassemblies, the core internal structures and the heat transport systems during the normal operation, scram transients and the early stage of postulated accidents. On this account, thermohydraulic experiments related to the decay heat removal by natural circulation have been carried out, and the development and validation of the thermohydraulic safety analysis codes is also in progress.

1) Thermohydraulic Experiments

An integral sodium experiment has been carried out with a partial core model composed of seven subassemblies, inter-wrapper gaps, an upper plenum and dipped cooler. The tests for inter subassembly heat transfer and core-plenum interaction were completed. A series of tests is under way focusing inter-wrapper flow under conditions of natural circulation decay heat removal.

2) Development and Validation of Analysis Codes

Subchannel analysis codes, ASFRE for single phase flow and SABENA for two-phase flow, have been developed for the purpose of predicting fuel element temperatures and thermohydraulic characteristics in the FBR fuel subassembly. ASFRE has the detailed wire-spacer model called distributed flow resistance model, which calculates the effect of wire-spacer on thermohydraulics, and the planer and porous blockage models are also implemented in ASFRE to evaluate the accident in the fuel subassembly. In this reporting period, the improvement of the thermal conduction model was carried out to be able to estimate the azimuthal temperature distribution on the fuel cladding. Furthermore, the analyses of the 37-pin and 61-pin bundle data obtained from the sodium experiments were conducted as a code validation study.

SABENA is the two-phase flow analysis code based on the two-fluid model. The correlation equations of wall friction and two-phase friction corresponding to the change of the two-phase flow pattern were implemented in SABENA, and the effects of improvement were confirmed by the validation analyses using sodium boiling test data at decay heat level.

Concerning about multi-dimensional thermohydraulic codes, the improvement of the turbulence model in AQUA was carried out for simulating temperature fluctuations by thermal striping at the core exit region. The peak-to-peak levels of temperature fluctuations were well simulated by the turbulence model based on an algebraic stress model (ASM).

A fluid-structure thermal interaction analysis code FLUSH have been developed to evaluate the temperature distribution in the shield plug. As for the pre-test analysis of the MONJU performance test, the temperature distribution in the shield plug was successfully predicted by the interactive calculation between structural components and cover gas fluid regions, and the applicability of FLUSH to practical systems have been identified.

A three-dimensional arbitrary-Lagrangian-Eulerian (ALE) finite element code SPLASH-ALE has been modified and applied to a simulation of free surface behavior. A capillarity model based on the cubic Spline function has been developed and incorporated in SPLASH-ALE. Free surface sloshing and capillarity phenomena analyses have been performed to validate the model. The simulation results are in excellent agreement with the theoretical results regarding sloshing

frequencies and the shape of the free surface. It was confirmed that capillarity should be adequately considered in the free surface dynamics analysis.

Regarding the whole plant simulation, a module type plant dynamics code Super-COPD has been developed and validated using MONJU plant data. In this reporting period, a cover gas module has been implemented in Super-COPD. The cover gas module is considered to be quite useful to evaluate the effect of cover gas pressure fluctuation on coolant surface levels in plant components. The performance test analysis of MONJU is in progress to validate the effectiveness of the model to practical problems.

A space-dependent plant dymamics code system has been developed to apply to the performance evaluation of passive safety features of FBRs. Three-dimensional calculations on neutronic characteristics have been perfomed to investigate the local power distorsion that is expected to occure in the case of asymmetric functioning of passive shut-down devices. The development of interface program between neutronic and thermohydraulic calculations is sheduled to be started.

9.2 Degraded Core Research

The degraded core research at PNC addresses: the fuel failure propagation during local-fault accidents, and physical phenomena during core disruptive accidents (CDAs).

The local fault studies include out-of-pile experiments on local coolant blockages by porous media with water to confirm fuel pin integrity through analyses using a detailed sub-channel code, ASFRE. A new series of sodium experiments is also planned. To focus on an intra-subassembly failure propagation behavior and to establish a termination scenario, a synthesis study is in progress by reviewing and interpreting the past in- and out-of-pile experimental data base, such as SCARABEE, MOL 7C, SLSF and TREAT.

The current out-of-pile experimental program at PNC consists of various simulant melt experiments using the MELT-II facility. A series of experiments to investigate the erosion behavior of solid structures by a high-temperature molten jet was completed. Experiments to study thermal interactions between a molten jet and coolant are in progress. A low-temperature series of tests with Woods metal and water has been completed. From the experiments, four distinct modes of interaction behaviors were observed. Conditions necessary for energetic interactions were identified, based on consideration of the minimum film boiling temperature. From extrapolation to reactor materials, it is predicted that such conditions are unlikely to be met. High-temperature experiments with steel and and alumina are planned for the next few years.

On the in-pile experiments jointly conducted with French CEA, all the planned tests have completed for the CABRI-FAST program, in which slow and fast transient tests have been conducted, mainly with high burnup, annular fuel pins. A joint synthesis work is underway. Starting in 1996, a next joint in-pile test program, CABRI-RAFT, is inihated, where the total of 8 tests are planned in the CABRI and SCARABEE reactors through 2000.

The CDA analysis code development and validation studies have continued extensively: SAS4A for CDA's initiating phase and SIMMER-III for the transition phase. For SAS4A, under collaboration with FZK and CEA, a reference code version REF-96 has been developed for common use in the future. A special single channel code PAPAS-2S, which models detailed fuel pin mechanics, is also being elaborated through CABRI analyses. Version 2 of SIMMER-III has almost completed with coupling with the neutronics model. A code assessment program is jointly participated by FZK and CEA. Also underway is the development of a new interface code SAME-II that couples SAS4A and SIMMER-III. The system of new-generation codes are being applied to CDA studies of D-FBR and future fast reactors.

Finally, a long-term research program SERAPH (Safety Experimental Reactor for Accident Phenomenology), has been undertaken over the last several years at PNC. This program aims at identifying long-term research needs of integral in-pile safety experiments, which are essential to further advance safety technologies towards FBR commercialization in the next century. The areas of primary interest include: eliminating recriticality concerns during CDAs, demonstrating advanced fuel design and types, and establishing local-fault scenarios. A feasibility study with related R&D's on elementary technologies is in progress to define required features and performance of the in-pile test facility, and to establish a basic design concept of a reference core.

9.3 Plant Accident Research

FBR plant accident research consists of two major activities. One is a study on a non-radiological sodium fire caused by sodium leakage from the intermediate heat transport system (IHTS), and the other is a study on the radiological source term, with emphasis being placed on quantifying various mitigation factors of fission product (FP) release and transport from failed fuels to the environment. The latter study also includes an integrity assessment study of the reactor containment with respect to FP leakage during a severe accident.

Most topical activity in FY1995 is the Monju sodium leak reproduction test and analysis. A series of prototypical sodium fire experiments were scheduled to clarify the

sequence of the Monju SHTS sodium leak accident on December 8th, 1995 by use of the sodium fire test facilities such as SAPFIRE. Two tests were conducted so far to evaluate the sodium leak rate in the accident. Monju accidental analysis as well as pretest and post test analyses are under way using sodium fire code such as ASSCOPS, SOLFAS and CONTAIN.

In the source term study, a new experimental rig has been constructed for investigation of FP release behavior from high temperature fuel under an accidental condition. Various performance tests are in progress to validate experimental procedures and measurements. Experiments using irradiated fuels of Joyo are planned in near future. The FP bubble behavior in sodium and the FP release behavior from sodium pool surface are studied with the SABER and the START rigs, respectively. The TRACER code is under development to describe various FP behavior in an in-vessel sodium system. For an ex-vessel study, the hydrogen burn experiments are continued in atmospheric conditions containing sodium aerosols or mists. The CONTAIN-LMR code is continuously improved for analyses of overall severe accident progression in a containment.

9.4 Steam Generator Safety Research

Current steam generator (SG) safety researches consist of two major activities. The one is for improvement of the sodium-water reaction evaluation method for the large-scale demonstration FBR plant. For this purpose, an overheating failure propagation process is studied in detail because this may potentially determine the design base leak (DBL) of future plant SG's. A series of simulation experiments in the TRUST rig has been conducted to examine structural performance of high-temperature tube under accident conditions. Obtained data are analyzed using the computer code, FINAS, and resonable agreement was found between the data and the code. Present TRUST rig which simulates the failure of the heat transfer tube in the nitrogen gas pressurization is remodeled and TRUST-2 rig is newly manufactured in order to simulate the failure of the heat transfer tube in the superheated steam flow condition. Since these TRUST experiments simulate reaction heat by means of induction heating, large scale sodium-water reaction experiments are planned in the modified SWAT-3 facility. Effects such as sodium flow and tube cooling by water system blowdown will be simulated in the large scale experiments.

Another study for the SG safety is aimed for the FBR design concept which eliminates the secondary sodium loop. To support this conceptual feasibility study, double-wall tube SG design is proposed and examined. Since a reliable leak detection

system is essential for this design, detailed investigation of tube failure detectability is performed. A inert gas system in the gap of double-wall tube is used for detection. In case of an inner tube failure, steam in the inert gas system is detected. On the contrary, inert gas contents in sodium is detected in case of an outer tube failure. Both experimental studies and code development are in progress to validate this detection system.

9.5 Research on Probabilistic Safety Assessment

PNC has been performing the research on Probabilistic Safety Assessment (PSA) for more than ten years as part of the R&D of a fast reactor.

The purpose of this research is to construct probabilistic safety models for a typical loop-type FBR plant so that an overall safety assessment can be performed. It is expected that (1) a systematic evaluation on the plant safety is conducted based on the quantitative analysis, (2) the insights on measures to enhance system reliability and safety are provided, (3) the operation and maintenance procedures are established based on a risk-based consideration, and (4) useful information is given to the development of basic policy on safety design and evaluation of a large LMFBR.

PNC has been improving the systems analysis code network which is able to perform a level-1 PSA. Recent efforts have focused on the development of a system configuration control program that evaluates the risk during the shutdown, a dynamic reliability analysis program (DYANA) dealing with the Emergency Operating Procedures, and a Living PSA System (LIPSAS). The LIPSAS has been installed at the site of Monju plant to examine the applicability of the system to safety management of a real plant. Furthermore, PNC has been developing a new software SAGE (Severe Accident Guidance using Expertise) which is able to quantify the consequence of core damage quickly with the expert system that models accident phenomena based on the experience of full scope PSA.

Efforts are being made to develop a new relational data base system of LMFBR component reliability data on an engineering workstation. The system is based on CREDO (Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, which ended in 1992. As part of the data analysis, failure rates of external leakage mode were quantitatively estimated for various valves in sodium system. Additionally, risk-related data on energy production systems such as solar photo voltaic energy system and LMFBR nuclear fuel cycle have been collected and health risk to the public is compared among various energy production systems.

A detailed shutdown PSA is underway. The reliability of the decay heat removal system (DHRS) during the plant shutdown has been evaluated based on the latest information on the inspection schedule. PSA is being applied to a large LMFBR with three IRACS loops, one DRACS loop, and a water steam system as the DHRS. The importance measures of each DHRS to the PLOHS frequency were evaluated. Also the study on the classification of safety evaluation events (SEEs) is ongoing. The failure probability of PS (prevention system) and MS (mitigation system) for a typical large LMFBR was examined and the occurrence frequency of the SEEs were quantified taking into consideration the multiple failure of PS and MS.

Level-2 PSA tasks (consequence analysis) are underway. The current effort includes a preliminary analysis of an E/T of ULOF accident in a large LMFBR particularly reflecting recent experimental and analytical knowledge. For example, it has been becoming apparent that the molten fuel could be discharged from the core region through an escape path with a large hydraulic diameter such as control rod guide tubes. Since this phenomenon could be highly effective to mitigate an energetic event in the transition phase, it is anticipated based on the preliminary assessment that the most probable sequence of ULOF is a mild one with neutronic shutdown due to fuel escape and thus non-mechanical energy release. For the ex-vessel physical process, an analysis of source term has been continued postulating a reactor vessel melt-through event. A debris-concrete interaction has been mainly investigated by means of a sensitivity analysis of the relation between the initial conditions such as debris mass and temperature and the quantity of radiological nuclides to be released.

10. Fuel Cycle

10.1 Mox Fuel Fabrication

1) Construction and Fuel Fabrication

R&D on fabrication of uranium-plutonium mixed oxide (MOX) fuel has been carried out since 1965 at the Plutonium Fuel Development Facility (PFDF) in Tokai works of PNC.

The Plutonium Fuel Fabrication Facility (PFFF), which started operation in 1972, has two fuel fabrication lines for Advanced Thermal Reactor (ATR) (10 ton MOX/year) and FBR (1 ton MOX/year). It has supplied the fuel necessary for the operation of ATR Fugen and FBR Joyo.

In parallel with the construction of Monju, construction of the Plutonium Fuel Production Facility (PFPF) (FBR line; 5 tons MOX/year) started in July 1982. It was aimed to develop fuel fabrication technologies with larger scale and automated equipment as well as to fabricate fuels for Monju and Joyo. The construction was completed in October 1987. After testing operation, production of Joyo fuel started in October 1988 as the first production campaign at PFPF. Production of the initial core fuel of Monju started in October 1989 and was completed in January 1994 at PFPF.

The PFPF is currently fabricating fuels for Joyo.

PNC is planning to extend the FBR line at PFPF so as to meet a Monju improving program.

About 145 tons of MOX fuel have been fabricated by the end of March 1996.

2) R&D on MOX Fuel Fabrication

Remotely controlled operation technology is one of the most important key element to achieve a large scale production of MOX fuel.

PFPF equipments including material transfer system are designed and manufactured so as to realize the fully automated operation.

Through the operation of PFPF FBR line so far, PNC has been accumulating experience for it.

10.2 Plutonium and Uranium Conversion

PNC developed a co-conversion technology utilizing the microwave heating direct denitration process (MH method) which converts plutonium nitrate and uranyl nitrate solution to MOX powder. Compared with the conventional method, it is a simple process and generates less liquid waste.

The Plutonium Conversion Development Facility (PCDF) (conversion capacity: 10 kg MOX/d), designed for demonstration of the co-conversion technology by MH method, was completed in February 1983. By the end of March 1996, it produced 12.6 tons of MOX powder containing about 5.2 tons of plutonium. The converted MOX powder were transported to PFFF and PFPF, in addition to about 1.8 tons of MOX powder processed at PFPF, and are being used for fabrication of MOX fuel for Fugen, Joyo and Monju.

Since depleted uranium recovered from reprocessing of spent fuel has generally higher U235 concentration compared to natural uranium, our country has decided to try

to use it as LWR fuel by re-enriching and mixing it with other enriched uranium and by mixing with plutonium as fuels for ATR, etc..

Reconstruction of pilot scale reprocessed uranium conversion facility was completed in June, 1994.

Conversion test for reprocessed uranium was started in August, 1994.

11. FBR Fuel Recycling

In the area of FBR fuel reprocessing, PNC has developed process and equipment with remote handling technique, through large scale cold mock-up tests at the three Engineering Demonstration Facilities (EDFs) and laboratory scale hot tests at the Chemical Processing Facility (CPF) in Tokai Works, taking into account of accumulated experience in the Tokai reprocessing plant for LWR fuels.

The construction of Recycle Equipment Test Facility (RETF), where advanced process and equipment tests in engineering scale under hot conditions are conducted in order to enhance the technology and economical efficiency, is initiated in January, 1995.

Almost all of R&D activities are oriented towards the RETF project.

11.1 Process Research and Development

1) Head End process

In order to remove the hexagonal wrapper tube efficiently prior to fuel chopping, a disassembly system with CO_2 laser has been developed and tested. A reference cutting scenario has been established through tests with dummy fuel assemblies. A prototype test equipment of geometrically safe continuous rotary dissolover was fabricated and tested.

Test results on both equipment have been reflected on the RETF design.

2) Chemical Separation Process

Major effort of solvent extraction contactor development is paid on centrifugal contactor. The design of the prototype contactor for RETF has been completed successfully.

In order to eliminate the generation of secondary salt-bearing waste in the purex process, studies and tests on solvent cleanup with salt-free reagents and electro-reoxidation process for Pu have been continued.

3) Common Technology

To establish remote maintenance concept with rack system, key technologies such as bilateral servo manipulator (BSM), roll-in type rack, remote connector bank, and remote sampling system have been developed.

Materials of process equipment and advanced analytical system have been developed.

4) Hot Tests at CPF

Irradiated fuel from Joyo with burnup up to 99,800 MWD/T have been used for hot tests at CPF.

Through these hot tests, information of dissolution characteristics and nuclides behavior in the solvent extraction process have been obtained. Especially, PNC has developed the Np co-extraction process with U and Pu to enhance non-proliferation aspect.

11.2 Plant Construction and Design of Recycling Facilities

1) Recycle Equipment Test Facility (RETF)

Verification of high availability and economical prospects of FBR fuel reprocessing are essential for deployment of FBR and its fuel cycle. In order to accomplish them at future pilot plant, hot engineering demonstrations of important process and equipment are necessary in advance. From this viewpoint, PNC has planned the RETF project to provide a test bed for advanced equipment and process.

RETF features a large remote cell which accommodates both head-end and chemical process equipment test areas. Most of the chemical processes will be mounted on the racks installed along either cell wall. The maintenance of these chemical process equipment as well mechanical components will be conducted by using overhead crane and bilateral servo-manipulator (BSM).

The construction of the main test building was initiated in January 1995 and it is scheduled to start hot tests in the beginning of 2000s.

It is expected that important data and experience for designing of future FBR fuel reprocessing plants will be compiled through the RETF operation.

2) FBR Fuel Recycling Pilot Plant

PNC has a plan to construct the FBR Fuel Recycling Pilot Plant to verify the whole plant system and it will be the first step for achieving the cost requirement.

The operation will be initiated in the middle of 2010s.

A REVIEW OF FBR DEVELOPMENT IN JAPAN

Presented to the 29th Annual Meeting of the IAEA/IWGFR
Aktau, Kazakhstan, 14-17 May 1996

POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
JAPAN

A REVIEW OF FBR DEVELOPMENT IN JAPAN

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   Safty,Fuels&Core,HighTemp.Structural System etc.
6. ADVANCED NUCLEAR FUEL RECYCLE TECHNOLOGY
7. INTERNATIONAL COOPERATION

1. General Review(1)

- JAPAN'S FBR POLICY IS SHOWN IN THE "LONG-TERM PROGRAM FOR DEVELOPMENT AND UTILIZATION OF NUCLEAR ENERGY", IN 1994 EDITION.

- Pu UTILIZATION IS KEY DEVELOPMENT FOR LONG-TERM ENERGY SUPPLY & ENERGY SECURITY OF JAPAN.

- PNC SHOULD PLAY THE CENTRAL ROLE IN THE ESTABLISHMENT OF TECHNOLOGICAL SYSTEM PATICULAR TO FBRs.

1.General Review (2)

- EXPERIMENTAL FAST REACTOR "JOYO": 51,200 hours OPERATION SINCE 1977.(as of March 1996)

- PROTOTYPE FBR "MONJU": FIRST CONNECTION TO GRID WAS ACCOMPLISHED IN AUG. 1995.
  SODIUM LEAK INCIDENT OCCURED IN DEC. 8 1995.

PRESENTLY MONJU IS UNDER THE COLD SHUTDOWN STATE AND MULTIPLE CAUSE INVESTIGATIONS OF SODIUM LEAK ARE BEING PERFORMED.

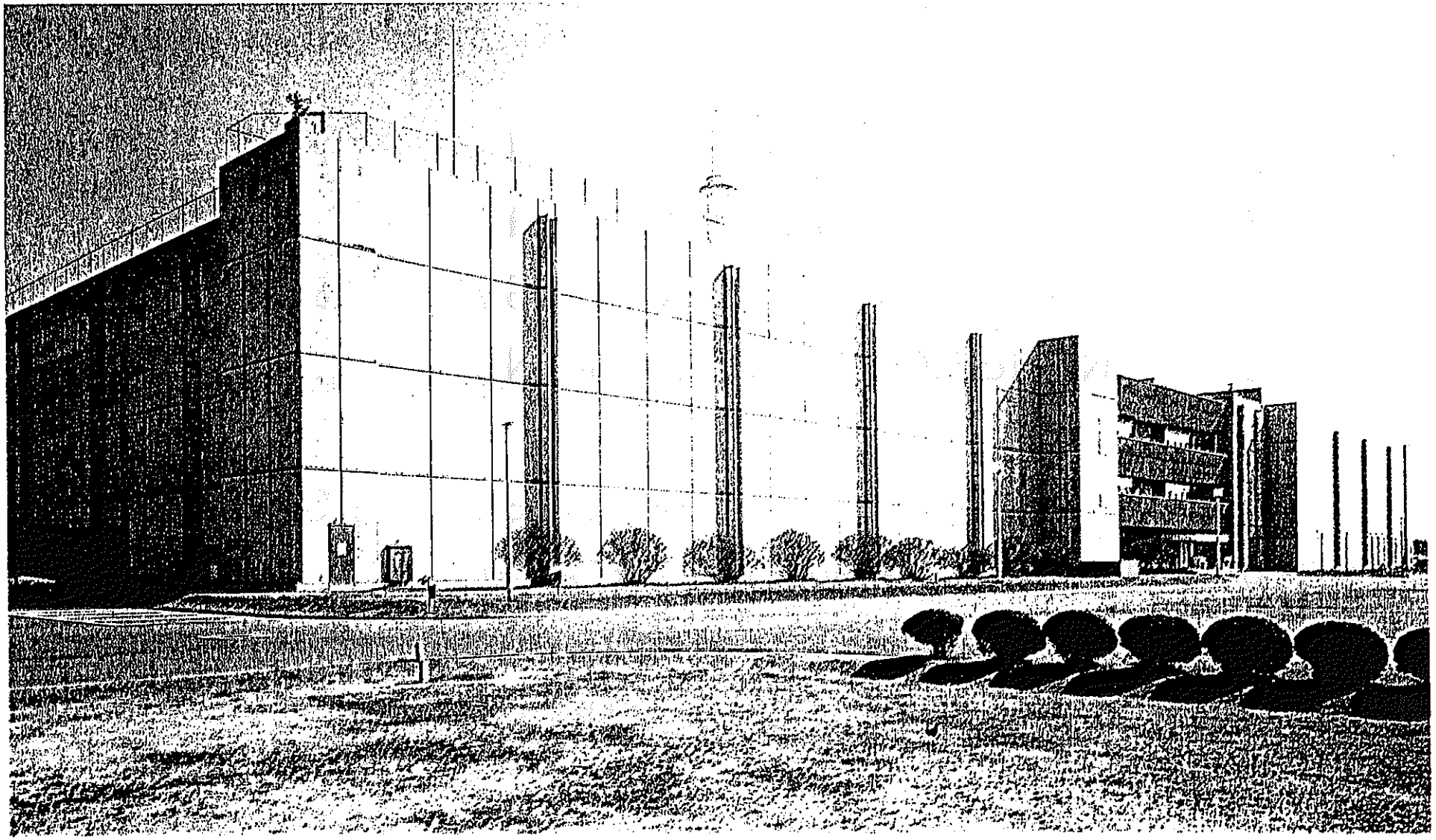
1. General Review(3)

- DEMONSTRATION FBR (DFBR): DESIGN STUDY FOR PLANT OPTIMIZATION IS IN PROGRESS BY JAPC WITH MAJOR MANUFACTURERS.
- DFBR RELATED R&Ds ARE IN PROGRESS BY 4 ORGANIZATIONS, PNC,JAPC, CRIEPI and JAERI UNDER THE LEADERSHIP OF FBR R&D STEERING COMMITTEE. R&Ds FOR FUTURE FBRs ARE IN PROGRESS IN PNC.
- PROGRAM OF THE DESIGN STUDY AND THE RELATED R&DS FOR DFBR ARE BEING DISCUSSED IN THE SUBCOMMITTEE ON FBR DEVELOPMENT PROGRAM OF JAPAN ATOMIC ENERGY COMMISSION.

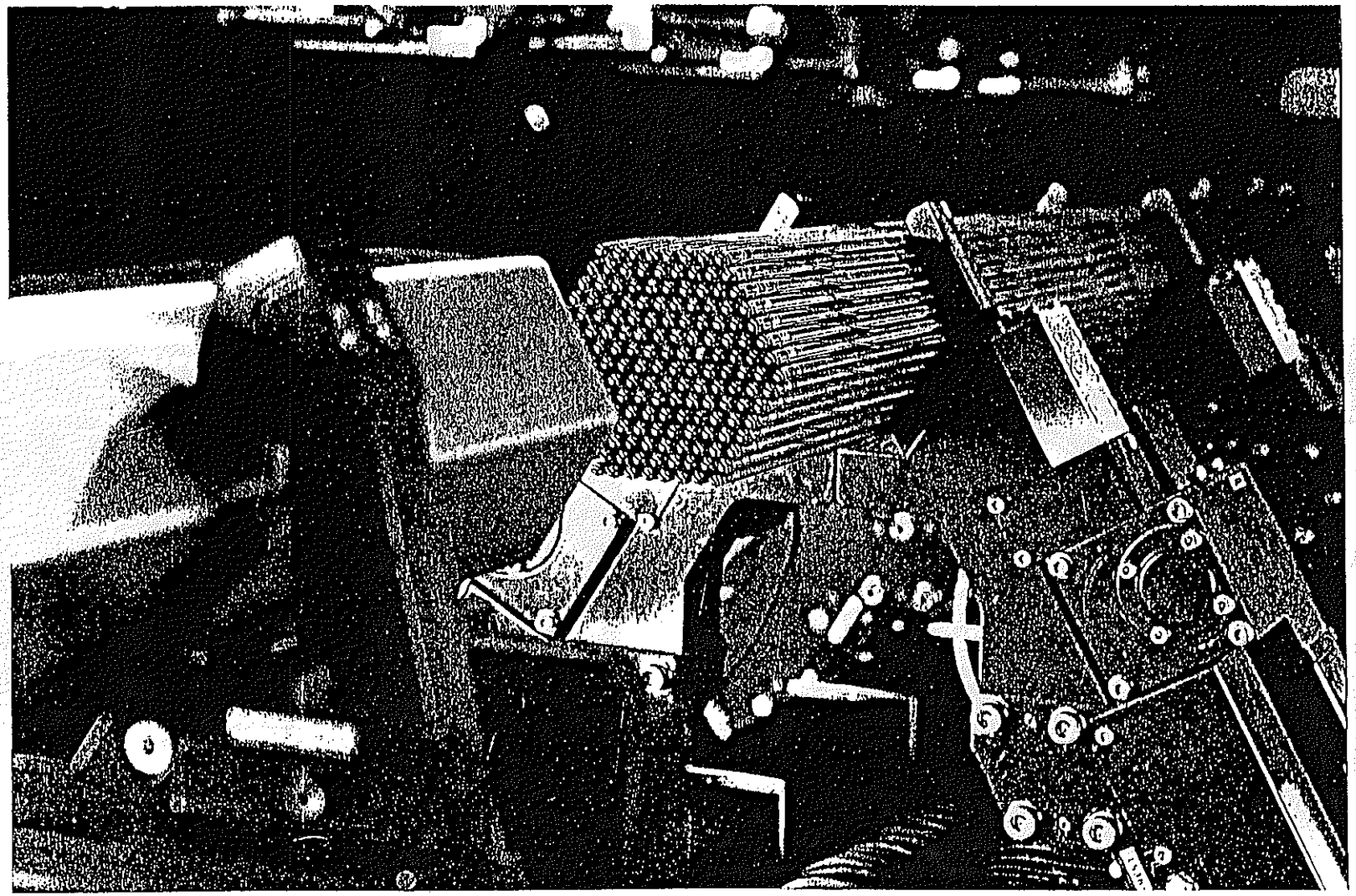
1. General Review (4)

- MOX FUEL FABRICATION : PFPF(Pu Fuel Production Facility) IS IN OPERATION AT PNC TOKAI WORKS.
  ABOUT 145 TONs OF MOX FUEL HAVE BEEN FABRICATED BY THE END OF MARCH 1996.

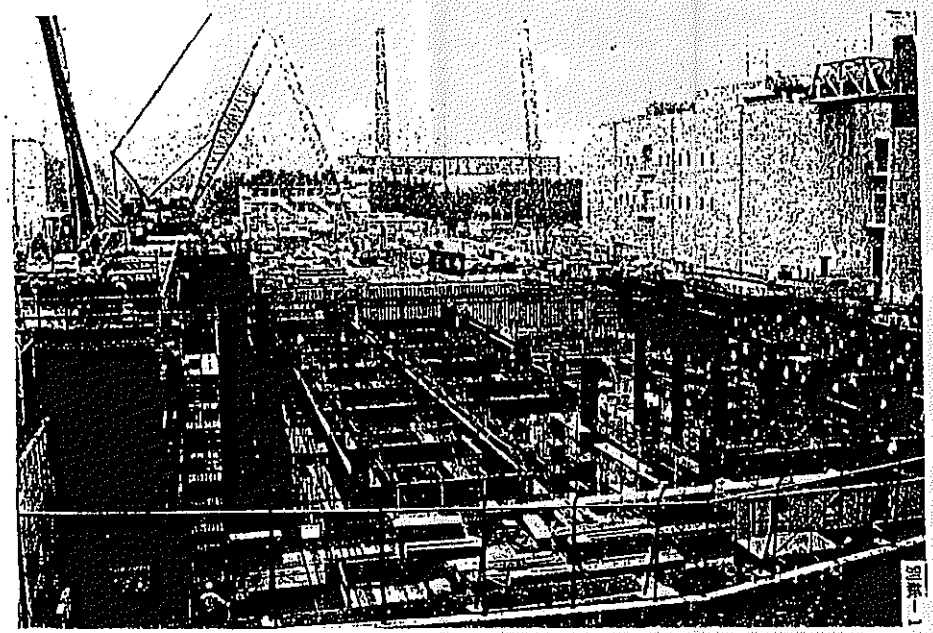
- THE CONSTRUCTION OF RECYCLE EQUIPMENT TEST FACILITY(RETF) WAS INITIATED IN JAN. 1995.
  ALMOST OF FBR FUEL RECYCLING R&D ACTIVITIES ARE ORIENTED TOWARDS THE RETF PROJECT .

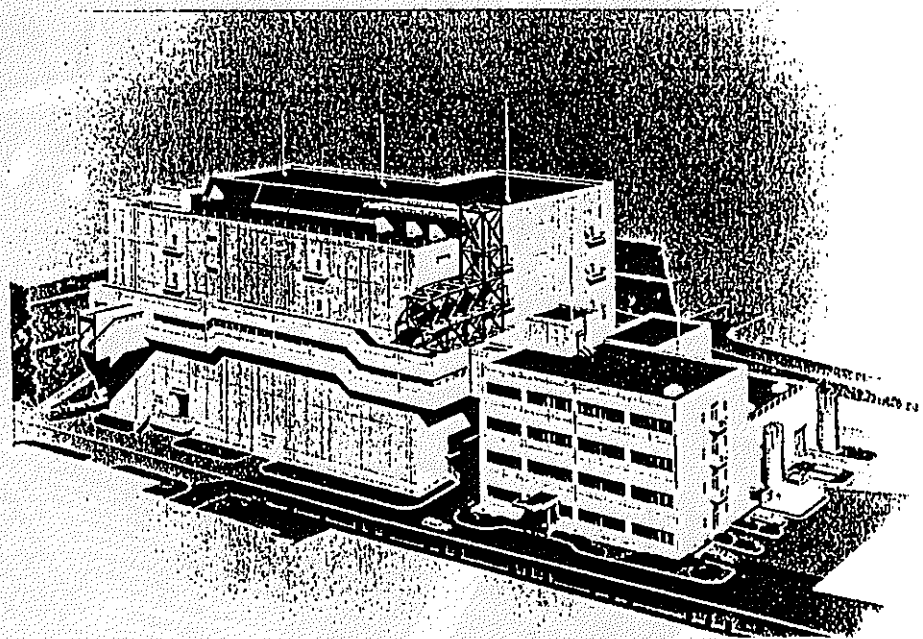


MOX Fuel Fabrication Facility



Fuel Pin Bundle being inserted to Wrapper Duct



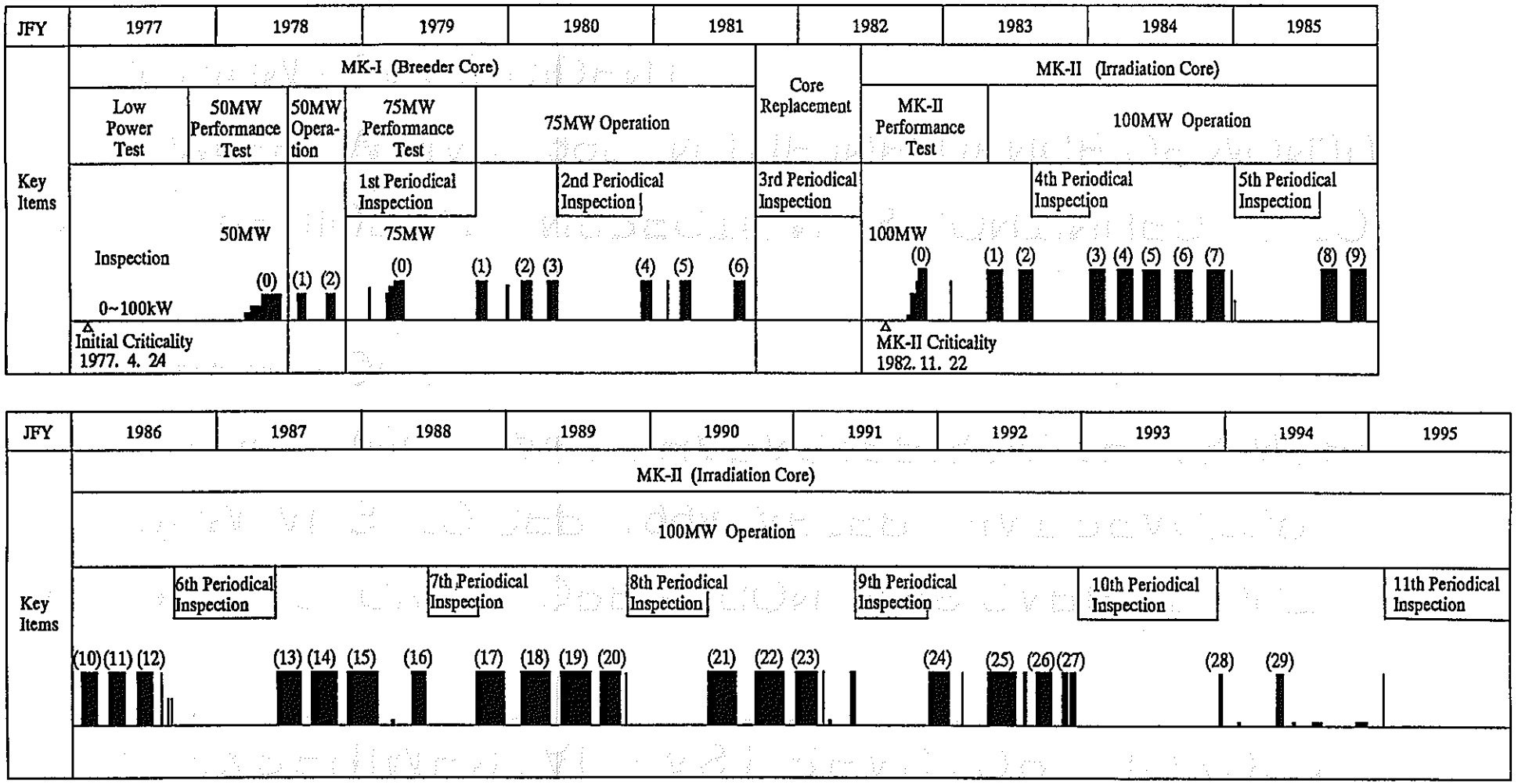


RETF
(Recycle Equipment Test Facility)

2. EXPERIMENTAL FAST REACTOR, "JOYO"(1)

- 29th DUTY CYCLE OPERATION WAS CARRIED OUT FROM AUG. TO SEP. 1994. AFTER THAT REACTOR PHYSICS AND PLANT CHARACTERISTIC TESTS WERE CARRIED OUT.

- 11th PERIODICAL INSPECTION IS CONTINUED UP TO NOW FROM MAY 1995 IN THE INFLUENCE OF MONJU SODIUM LEAK INCIDENT.



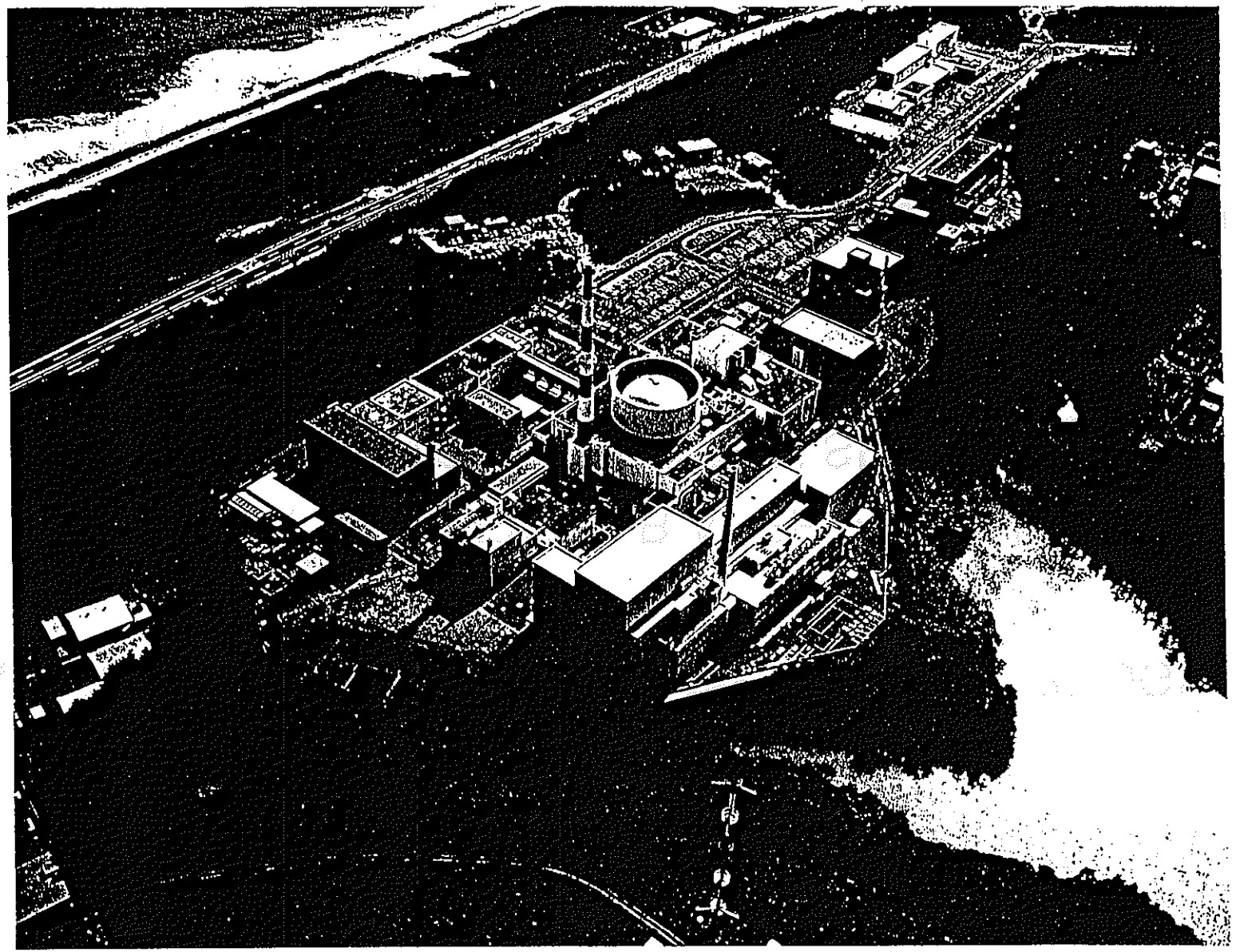
Operating History of JOYO (As of March, 1996)

2. EXPERIMENTAL FAST REACTOR, "JOYO"(2)

- JOYO UPGRADING PROGRAM, "MARK-III PROGRAM"

  - THE SAFETY REVIEW FOR APPROVAL OF INSTALLATION HAS BEEN COMPLETED IN SEP. 1995 .

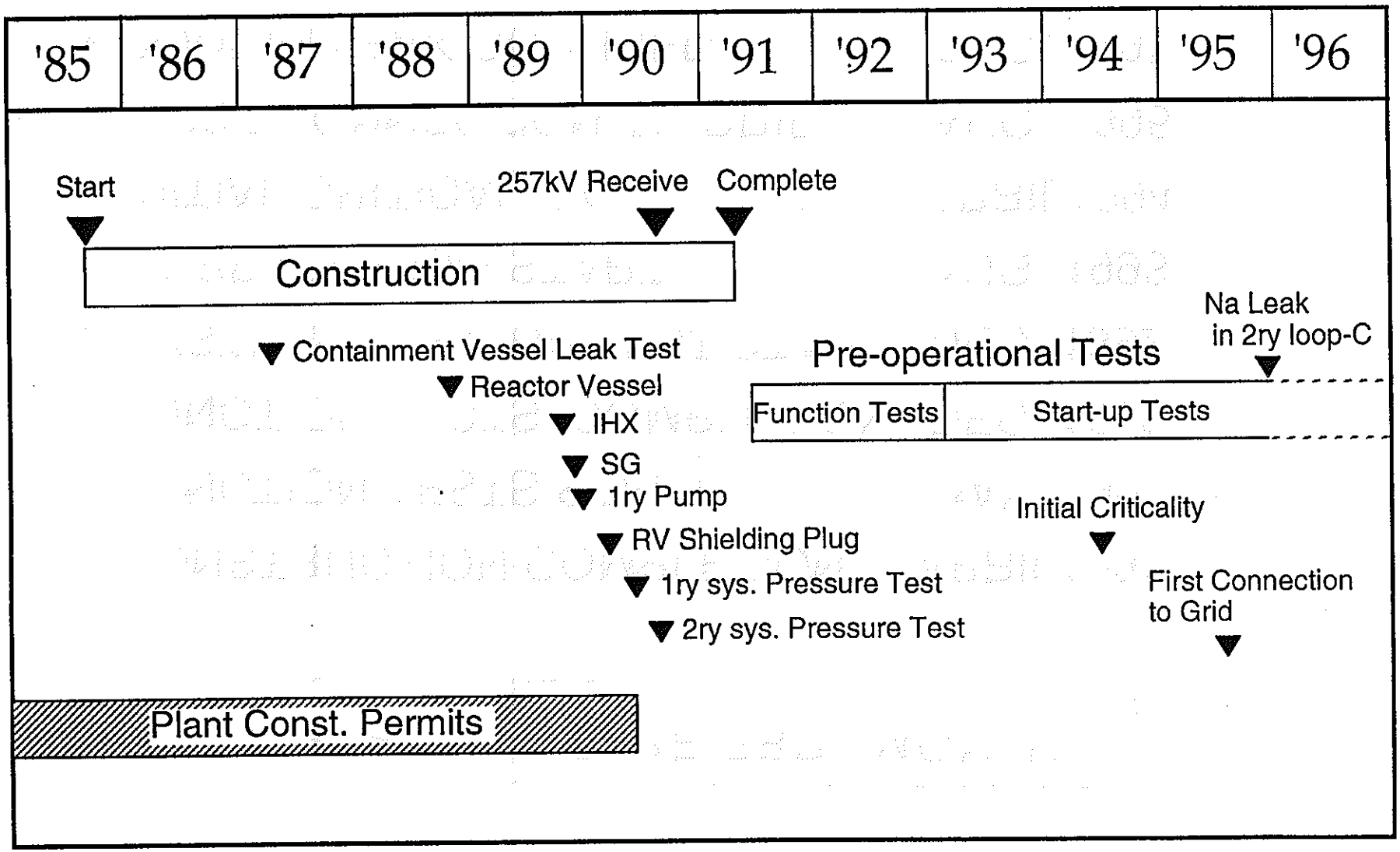
  - THE PLANT MODIFICATION SCHEDULE IS UNDER CONSIDERATION IN THE INFLUENCE OF MONJU SODIUM LEAK INCIDENT.



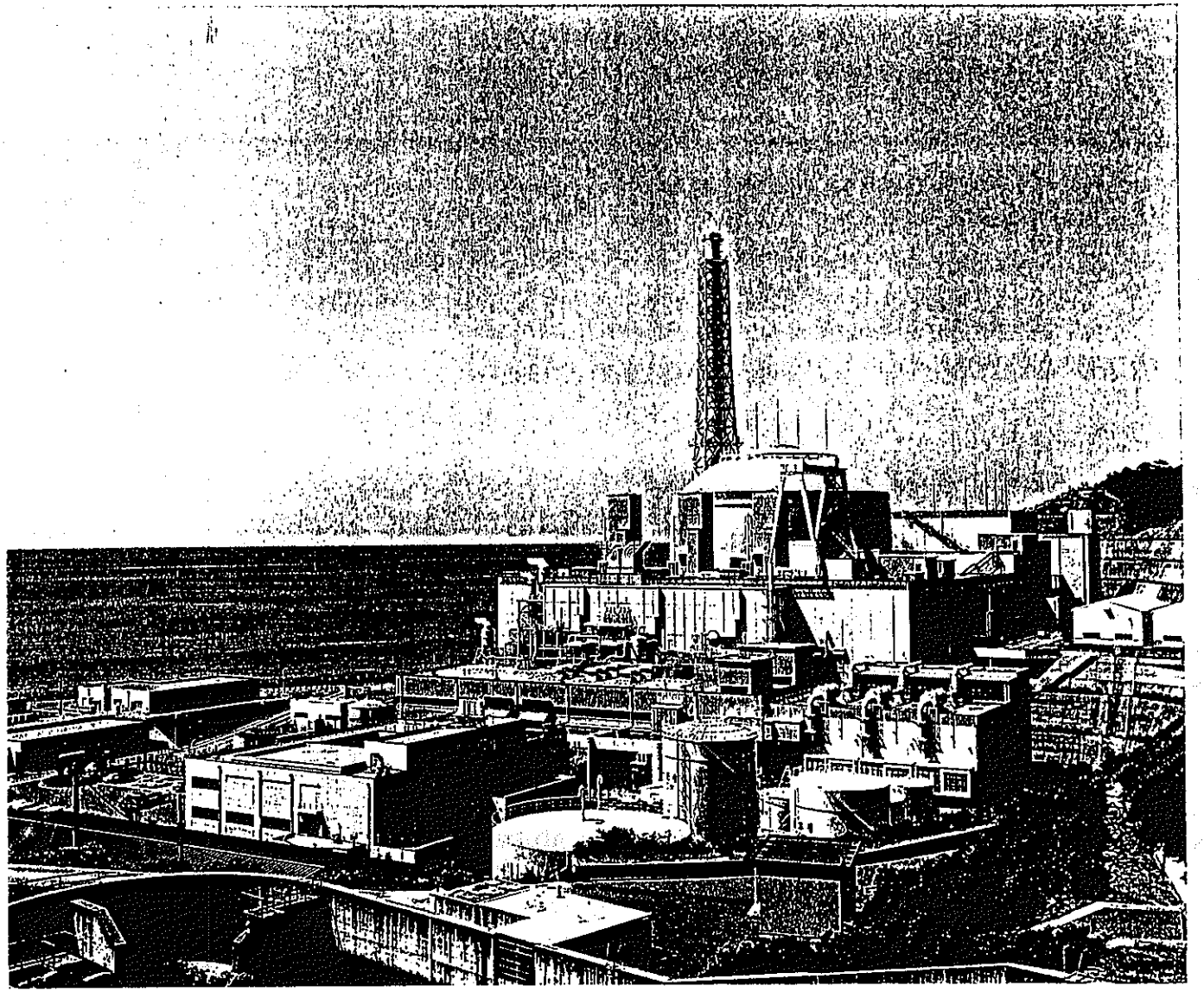
Experimental Fast Reactor JOYO

3. PROTOTYPE FBR "MONJU"

- CONSTRUCTION COMPLETION: APRIL1991
- FUNCTION TESTS START: MAY 1991
- FUNCTION TESTS COMPLETION: DEC. 1992
- SYSTEM START-UP TEST START: DEC. 1992
- CORE LOADING START: AUG. 1993
- INITIAL CRITICALITY : APRIL1994
- FIRST CONNECTION TO GRID: AUG. 1995
- SODIUM LEAK OCCURED: DEC. 1995

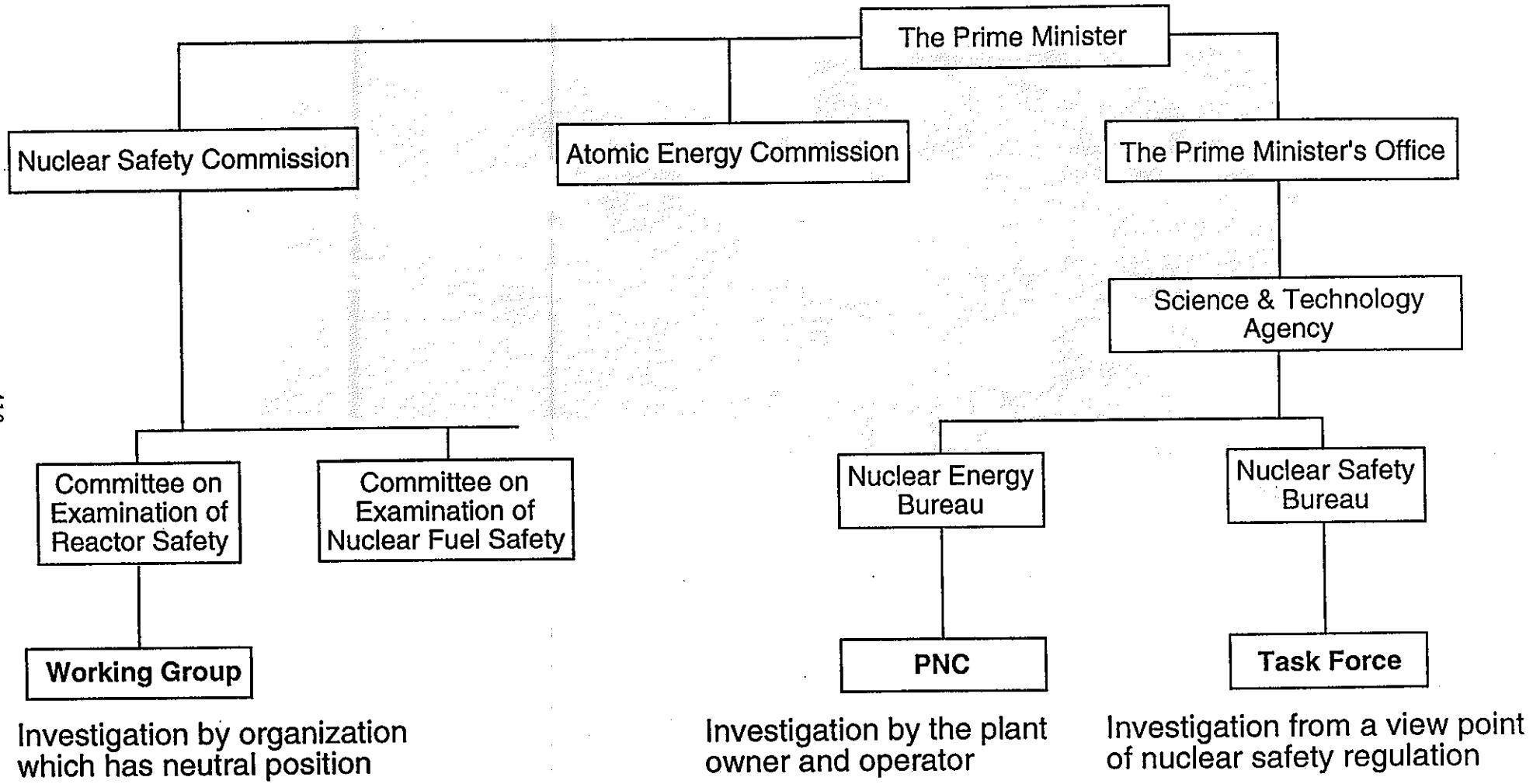


Monju Construction & Tests Schedule



Prototype FBR MONJU

Organization for Cause Investigation of Monju Sodium Leak



PNC: Power Reactor and Nuclear Fuel Development Corporation

Schedule of Cause Investigation

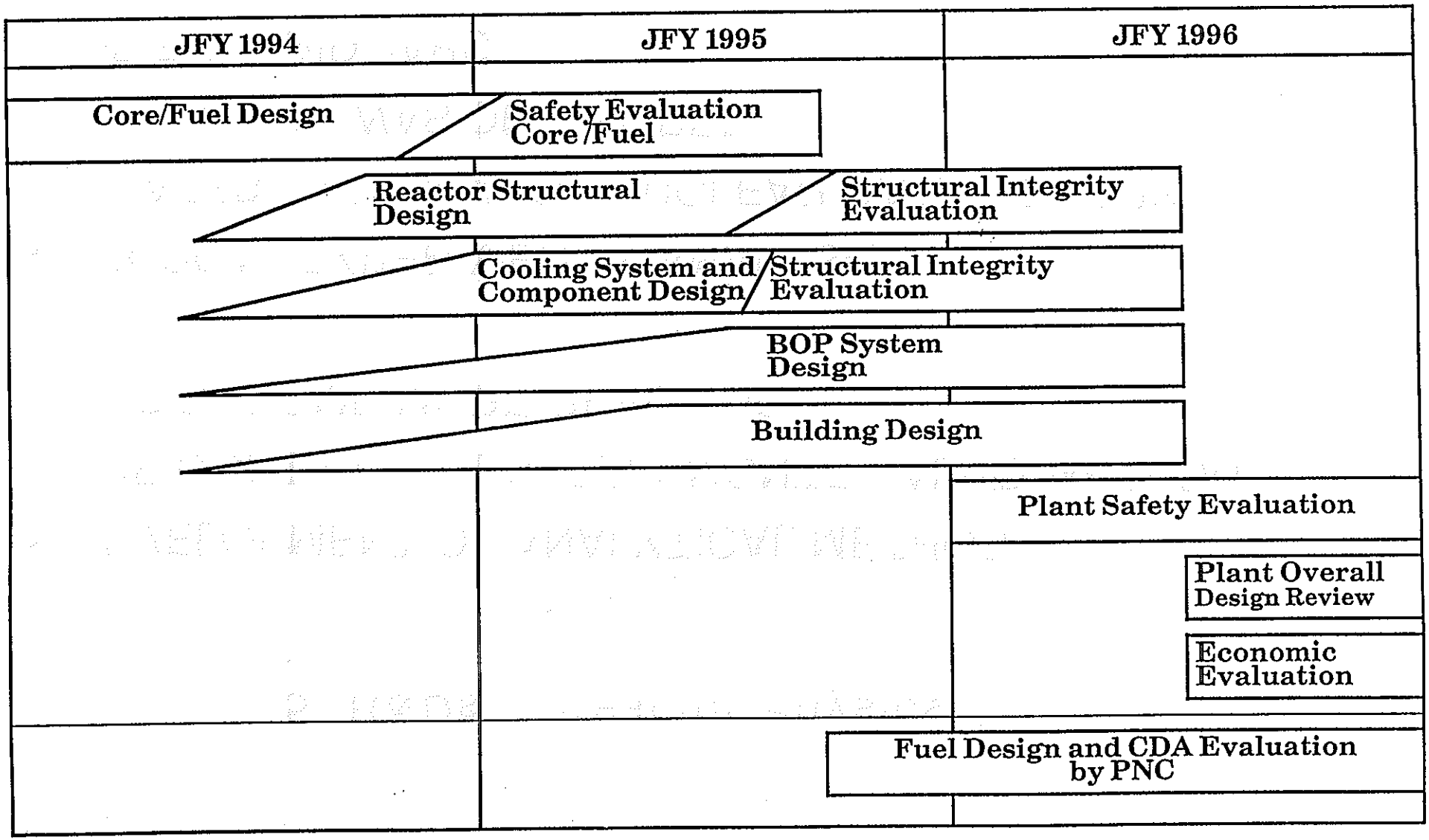
	Dec. 95	Jan. 96	Feb.	Mar.	Apr.
	Sodium leak occurrence ▼ 12/8	Confirmation that the well tube tip was lost. ▼ 1/8	Cut out the affected part ▼ 2/9	Discovery of the lost well tube tip ▼ 3/28	Recovery of the lost part ▼ 4/24
Investigation of the affected temperature sensor	Preparations	Cut out/ visual inspection		Metallographic examination	
Investigation of other temp. sensors in secondary loops			Integrity confirmation of A&B loop sensors	Integrity confirmation of C loop sensors Cut out/inspection of intact sensor	
Securing safety of intact loops			Seal reinforcement of temp. sensors in A&B loops		
Recovery of the lost well tube tip			1st/2nd stage	3rd stage	Recovery work
Survey of effects on facilities			Sampling and detailed examination of grating, ventilation duct, building concrete wall and floor liner		
Survey of effects on equipment Clean up	Inspection and clean up of equipment, control panels and areas contaminated by sodium composite/Recovery and analysis of sodium composite				
Estimation of total quantity of sodium leaked	Interim estimation/Evaluation by experiments and analysis				
Experiments and analysis for cause investigation	Analysis of flow induced vibration and hydraulic tests				
	Experiments and analysis of sodium burning behavior				

4. DFBR DESIGN STUDY

- DESIGN STUDY FOR PLANT OPTIMIZATION HAS BEEN CONDUCTED BY JAPC FOR 3 YEARS SINCE FY1994.

- FOCUSED ISSUES IN THIS STUDY ARE:
  - CORE SAFETY ENHANCEMENT INCLUDING THE APPLICATION OF GAS EXPANSION MODULES(GEM)
  - INCREASE OF DESIGN MARGINS FOR THE STRUCTURAL TOLERANCE OF THE CONTAINMENT FACILITIES
  - FEASIBILITY & LICENSABILITY OF SEISMIC ISOLATION PLANT FOR INTRODUCTION TO THE DFBR

Schedule of Design Study for DFBR Plant Optimization



5. R&Ds Reactor Physics(1)

- DEVELOPMENT OF ANALYTICAL METHOD:
  THE CONTINUOS-ENERGY"MONTE CALRO METHOD"
  HAS BECOME MORE REALISTIC.

- CRITICAL EXPERIMENT & ANALYSIS:
  A LATEST VERSION OF NUCLEAR DATA LIBRARY
  "JENDL-3.2" WAS DEVELOPED.
  OF PNC AND JAPC.

5. R&Ds Reactor Physics(2)

- CROSS-SECTION ADJUSTMENT:

  AN EFFORT HAS BEEN CONTINUED TO EXTEND THE APPLICABILITY OF CROSS-SECTION ADJUSTMENT & DESIGN ACCURACY EVALUATION SYSTEM TO THE "DOPPLER EFFECT".

- SHIELDING EXPERIMENT & ANALYSIS:

  POST-EXPERIMENT ANALYSIS OF JASPER WAS ALMOST COMPLETED.

  TYPICAL RADIAL SHIELDING EXPERIMENT OF JASPER IS NOW BEING RE-ANALYZED WITH "JENDL-3.2".

5. R&Ds SYSTEMS AND COMPONENTS

- R&D on SASS IS IN PROGRESS.
  A SYSTEM PERFORMANCE TEST USING AN ACTUAL REACTOR SUCH AS JOYO IS UNDER PLANNING.

- DEVELOPMENT OF ISI TEST EQUIPMENT FOR MONJU WAS COMPLETED.
  MOCK-UP TEST RIGS AND ISI TEST EQUIPMENT AT OEC WERE DISASSEMBLED AND TRANSFERRED TO MONJU.

- 1 MW DUPLEX TUBE SG MODEL TEST WAS COMPLETED AND POST TEST EXAMINATION IS UNDERWAY.

5. R&Ds FUELS & MATERIALS(1)

- FUEL PIN PERFORMANCE: 3-DIMENSIONAL FEM CODE FOR ANALYSIS ON DEFECTED FUEL PIN BEHAVIOR HAS BEEN MODIFIED WITH PIE RESULTS OF RBCB PIN IN EBR-II. DEVELOPMENT OF NITRIDE FUEL PERFORMANCE CODE IS IN PROGRESS.
- CORE MATERIAL:PNC 1520 ADVANCED AUSTENITIC STEEL WAS DEVELOPED FOR HIGH BURNUP FUELS. DESIGN BASE STANDARD OF PNC1520 WAS COMPLETED. EVALUATION STUDY IS IN PROGRESS TO APPLY MONJU AND DFBR CORES.

5. R&Ds FUELS & MATERIALS(2)

- THE FUEL SUBASSEMBLY WITH CEA CLADDING TUBES HAS BEEN IRRADIATED SINCE 1988 AND REACHED THE FINAL STAGE AT 125GWd/t.
- PHASE-II PROGRAM OF PNC/DOE COLLABORATIVE OPERATIONAL RELIABILITY TESTING IN EBR-II HAS BEEN SUCCESSFULLY COMPLETED BY THE END OF 1995.
- PNC /JAERI IRRADIATION TEST OF NITRIDE AND CARBIDE PINS HAS BEEN CONDUCTED IN JOYO SINCE 1994.
- CONSTRUCTION OF PIE FACILITY OF FUELS MONITORING(FMF-2) HAS COMPLETED.
  HOT OPERATION WILL START FROM LATE 1997.

5.R&Ds

STRUCTURAL DESIGN AND MATERIALS(1)

- EFFORT TO ENTIRE THE CAPAVILITY OF THE GENERAL PURPOSE NONLINEAR STRUCTURAL ANALYSIS PROGRAM "FINAS" IS CONTINUED. THE LATEST VERSION V12.0 WAS TRANSLATED INTO ENGLISH.
- CREEP - FATIGUE DESIGN GUIDE ,DESIGN RULES FOR WELDMENT AND STRAIN LIMIT CRITERIA ARE INVESTIGATED TO IMPROVE AND EXTEND THE CURRENT ELEVATED TEMPERETURE STRUCTURAL DESIGN GUIDE.
- STRUCTURAL TESTS AND EVALUATION ARE BEING PERFORMED TO IMPROVE STRENGTH PREDICTION METHODS.

5.R&Ds
STRUCTURAL DESIGN AND MATERIALS(2)

- SEISMIC ANALYSIS METHOD DEVELOPMENT AND VERIFICATION ON FBR CORE IN THE FRAMWORK OF "IAEA CRP" HAS COMPLETED IN 1995.
- DETERMINISTIC AND PROBABILISTIC FRACTURE MECHANICS METHODOLOGIES ARE BEING DEVELOPMENT.
- STRUCTURAL MATERIAL TESTS IN AIR ,IN SODIUM ,IN WATER/STEAM AND UNDER POST-IRRADIATION CONDITION HAVE BEEN CONDUCTED TO PREPARE A NEW VERSION FOR DFBR.

5. R&Ds SAFETY(1)

- SAFETY EVALUATION FOR NORMAL & ABNORMAL EVENTS

THERMAL HYDRAULICS EXPERIMENTS:

- THE TESTS FOR INTER SUBASSEMBLY HEAT TRANSFER & CORE PLENUM INTERACTION WERE COMPLETED.
- A SERIES OF TESTS IS UNDERWAY FOCUSING INTER WRAPPER FLOW UNDER CONDITIONS OF DHR BY NATURAL CIRCULATION.

DEVELOPMENT AND VALIDATION OF ANALYSIS CODE:

- SUBCHANNEL CODE "ASFRE" & "SABENA"
- 3-D THERMAL HYDRAULIC CODE "AQUA"
- FLUID STRUCTURE THERMAL INTERACTION "FLUSH"
- 3-D FLUID DYNAMICS CODE "SPLASH"

5.R&Ds SAFETY(2)

- DEGRADED CORE RESEARCH

FUEL FAILURE PROPAGATION DURING LOCAL FAULT:

- CABRI-FAST IN-PILE PROGRAM,JOINTLY CONDUCTED WITH CEA,WAS COMPLETED.
- A NEXTJOINT IN-PILE PROGRAM "CABRI-RAFT" IS INITIATED IN 1996.

PHYSICAL PHENOMENA DURING CDAs:

- FOR SAS4A,A REFERENCE CODE VERSION "REF-96" HAS BEEN DEVELOPED UNDER COLLABORATION WITH FZK & CEA.
- SIMMER-III VERSION 2 HAS ALMOST COMPLETED.

5. R&Ds SAFETY(3)

- PLANT ACCIDENT RESEARCH

  A STUDY ON THE NON-RADIOLOGICAL SODIUM FIRE:

  - A SERIES OF PROTOTYPICAL SODIUM FIRE EXPERIMENTS WERE SCHDULED TO CLARIFY THE SEQUENCE OF THE MONJU IHTS SODIUM LEAK.
  - TWO TESTS WERE CONDUCTED SO FAR TO EVALUATE THE SODIUM LEAK RATE IN THE INCIDENT.

  A STUDY ON THE RADIOLOGICAL SOUCE TERM:

  - A EXPERIMENTAL RIG HAS BEEN CONSTRUCTED FOR INVESTIGATION OF FP RELEAS EBEHAVIOR, PERFORMANCE TESTS ARE IN PROGRESS TO VALIDATE EXPERIMENTAL MEASUREMENTS.

5.R&Ds SAFETY(4)

- STEAM GENERATOR SAFETY RESEARCH

  IMPROVEMENT OF THE SODIUM -WATER REACTION EVALUATION METHOD:

  - A SERIES OF SIMULATION TEST FOR OVERHEATING FAILURE PROPAGATION PROCESS HAS BEEN CONDUCTED USING THE TUBE RUPTURE SIMULATION TEST RIG (TRUST).
  - A LARGE SCALE SODIUM-WATER REACTION EXPERIMENTS ARE PLANNED IN THE MODIFIED "SWAT-3" FACILITY TO SIMULATE SODIUM FLOW & TUBU COOLING EFFECTS.

5.R&Ds SAFETY(5)

- RESEARCH ON PROBABILISTIC SAFETY ASSESSMENT IMPROVING THE SYSTEMS ANALYSIS CODE NETWORK FOR LEVEL-1 PSA:
  - EVALUATION THE RISK DURING THE SHUTDOWN
  - DEVELOPMENT DYNAMIC RELIABILITY ANALYSIS PROGRAM "DYANA"
  - DEVELOPMENT A LIVING PSA SYSTEM "LIPSAS"

LEVEL-2 PSA TASKS FOR CONSEQUENCE ANALYSIS:

- CURRENT EFFORT INCLUDES A PLELIMINARY ANALYSIS OF AN EVENT/TREE OF ULOF ACCIDENT.

5.R&Ds PLANT DESIGN STUDY

- DESIGN STUDY ON A 1300MWe-SIZE PLANT FEATURED WITH PASSIVE SAFETY USING NITRIDE FUEL WAS COMPLETED AT PNC IN 1995.

- A CONCEPTUAL DESIGN STUDY FOCUSING CORE DESIGN IN ORDER TO STUDY FBR CORE VARIETY SUCH AS PLUTONIUM BURNING ,MINOR ACTINIDE BURNING AND USE OF NEW TYPE FUEL .

5. R&Ds FUTURE R&D PLAN IN MONJU

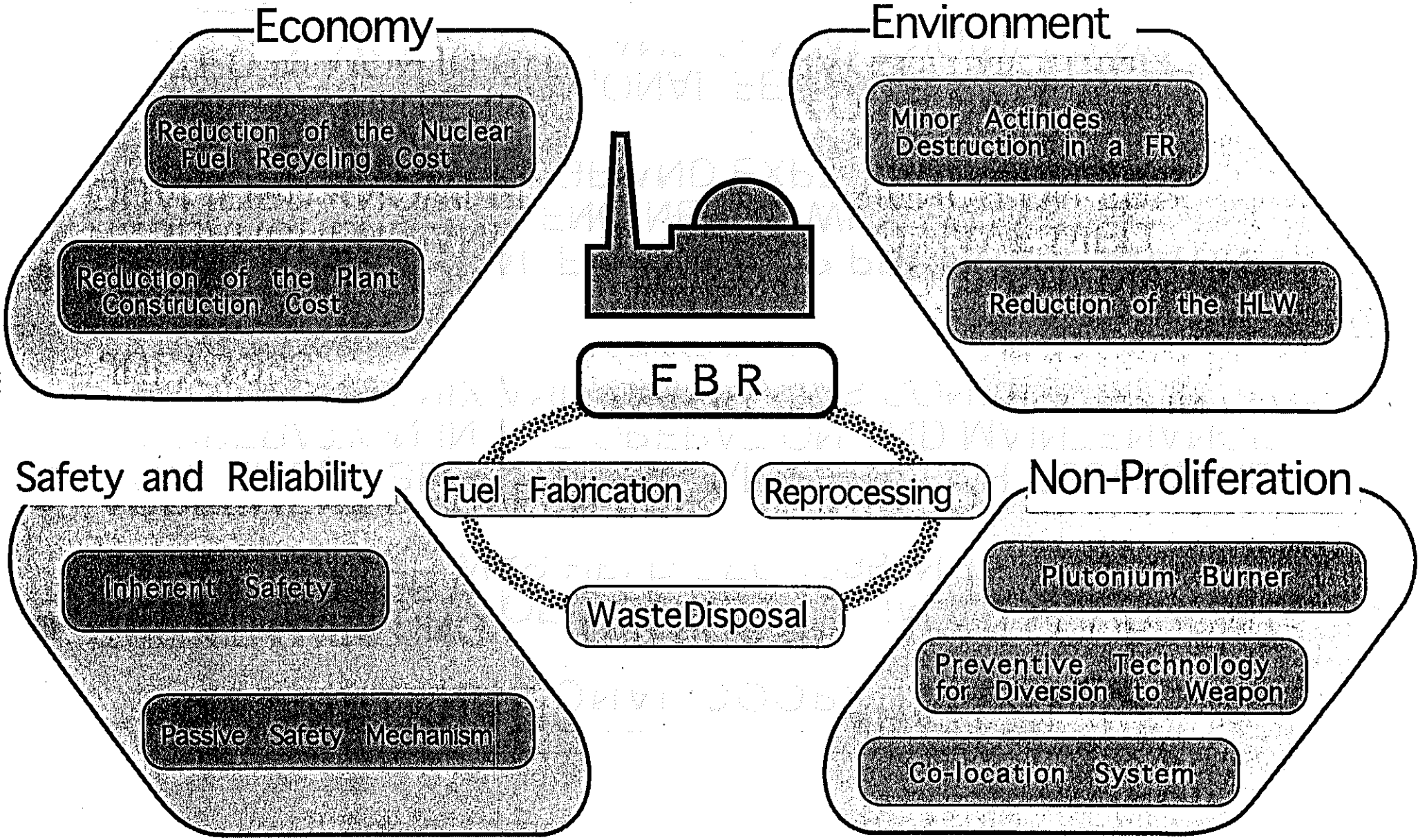
- OVERALL ESTIMATION ON FBR TECHNOLOGIES
  - MONJU DESIGN VERIFICATION
  - A DESIGN DATA BASE IMPROVEMENT etc.
- ADVANCED TECHNOLOGY DEVELOPMENT
  - APPLICATION OF VIRTUAL REALITY TECHNOLOGY FOR PLANT DESIGN & SIMULATION.
  - REMOTE MAINTENANCE & REPAIR TECHNOLOGY
  - ADVANCED FUEL HANDLING SYSTEM etc.
- ESTABLISHMENT OF THE FUEL CYCLE & Pu UTILIZATION
  - IMPROVEMENT FOR HIGH PERFORMED AND FLEXIBLE CORE
  - PRACTICAL USE PLANNING FOR IRRADIATION BED etc.

6. ADVANCED NUCLEAR FUEL CYCLE TECHNOLOGY

- LONG -TERM R&D ON ADVANCED NUCLEAR FUEL RECYCLING TECHNOLOGIES WILL BE CARRIED OUT FOR RECYCLING NITRIDE FUEL ,METALLIC FUEL ANDACTINIDE ELEMENTS.
  - TO PURSUE THE TECHNICAL OPTIONS FOR FUTURE NEEDS
  - AIMING REDUCTION OF IMPACT ON THE ENVIRONMENT AND ASSURANCE OF NUCLEAR NONPROLIFERATION, BESIDES IMPROVEMENT OF SAFETY,RELIABILITY AND ECONOMY.
  - RESEARCH FOR NEW RECYCLING SYSTEM USING NEW FUEL AND MINOR ACTINIDE.

- R&D PROGRAM ARE BEING DISCUSSED AT THE SUB-COMMITTEE ON ADVANCED RECYCLING PROGRAM OF JAEC. A TESTING REACTOR AND FACILITIES MAY ALSO BE CONSIDERED.

Objectives of Advanced Nuclear Fuel Recycle



7. INETRNATIONAL COOPARATION

- THE 4TH MEETING OF THE EUROPE-JAPAN COORDINATING COMMITTEE WAS HELD DEC.1995 IN TOKYO.

- AGREEMENT BETWEEN EDF AND PNC FOR TECHNICAL COOPERATION IN THE OPERATION AND MAINTENANCE OF SUPERPHENIX AND MONJU WAS CONCLUDED IN JULY 1995.

- "PNC INTERNATIONAL FELLOWSHIP PROGRAM" INVITING RESEARCHERS AND ENGINEERS WHO HAVE WELL QUALIFIED KNOWLEDGE AND EXPERIENCE ON FBR.

  IN 1995FY,13 INTERNATIONAL FELLOWS HAVE BEEN INVITED IN MONJU-SITE AND O-ARAI ENGINEERING CENTER.

REPORTING ON MONJU SODIUM LEAK

Presented at The 29th Annual Meeting of the International Working Group on Fast Reactors, Aktau, Kazakhstan

May 14-17, 1996

by Monju Construction Office/
Power Reactor and Nuclear Fuel Development Corporation

SCOPE OF PRESENTATION

BRIEFING OF MONJU PLANT

SUMMARY OF THE INCIDENT

LOCATION OF LEAKAGE

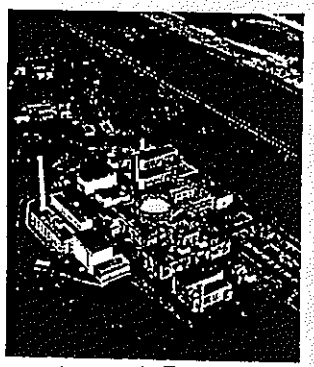
REACTION PRODUCT TRANSFER

AMOUNT OF SODIUM LEAKED

CURRENT SURVEY ON FAILURE MODE

CONCLUSION

FBR Development Program in Japan



Experimental Fast Reactor
JOYO

Experimental Reactor

"JOYO"
100MWt
Initial Criticality 1977

Prototype Reactor

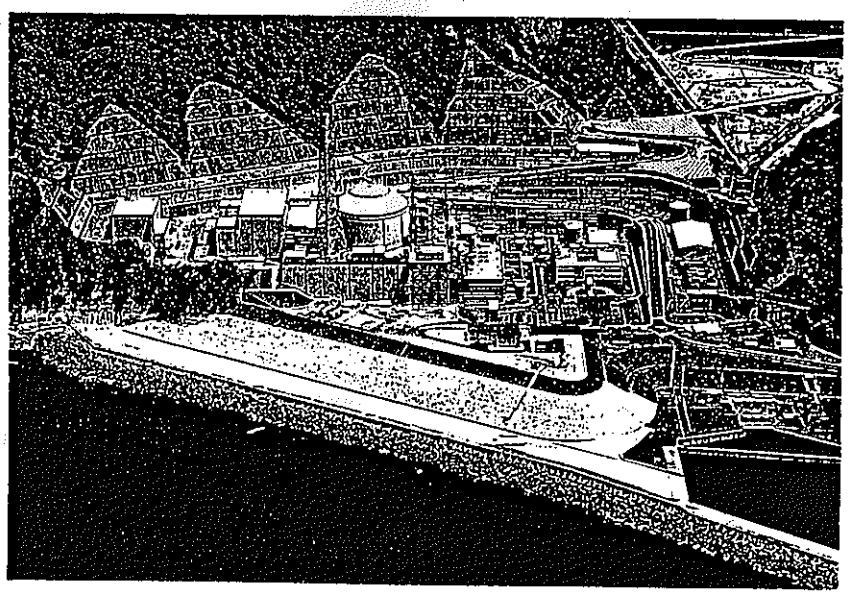
"MONJU"
280MWe
Initial Criticality 1994

Demonstration Reactor

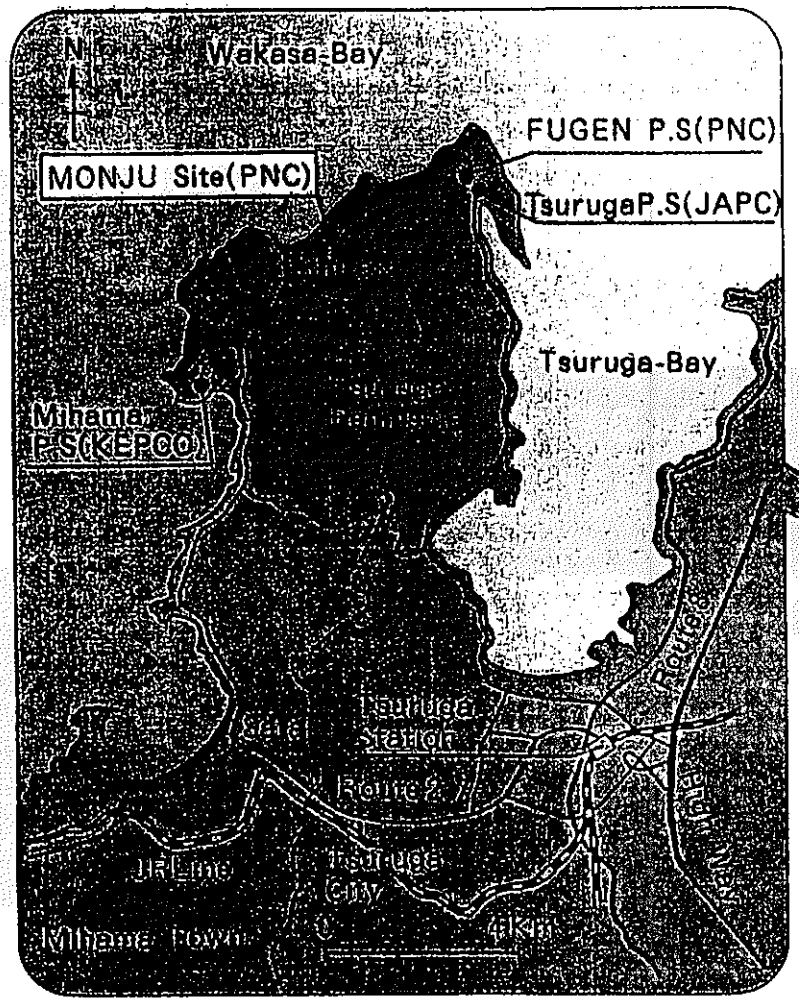
Construction
Beginning of 2000s

Commercialization

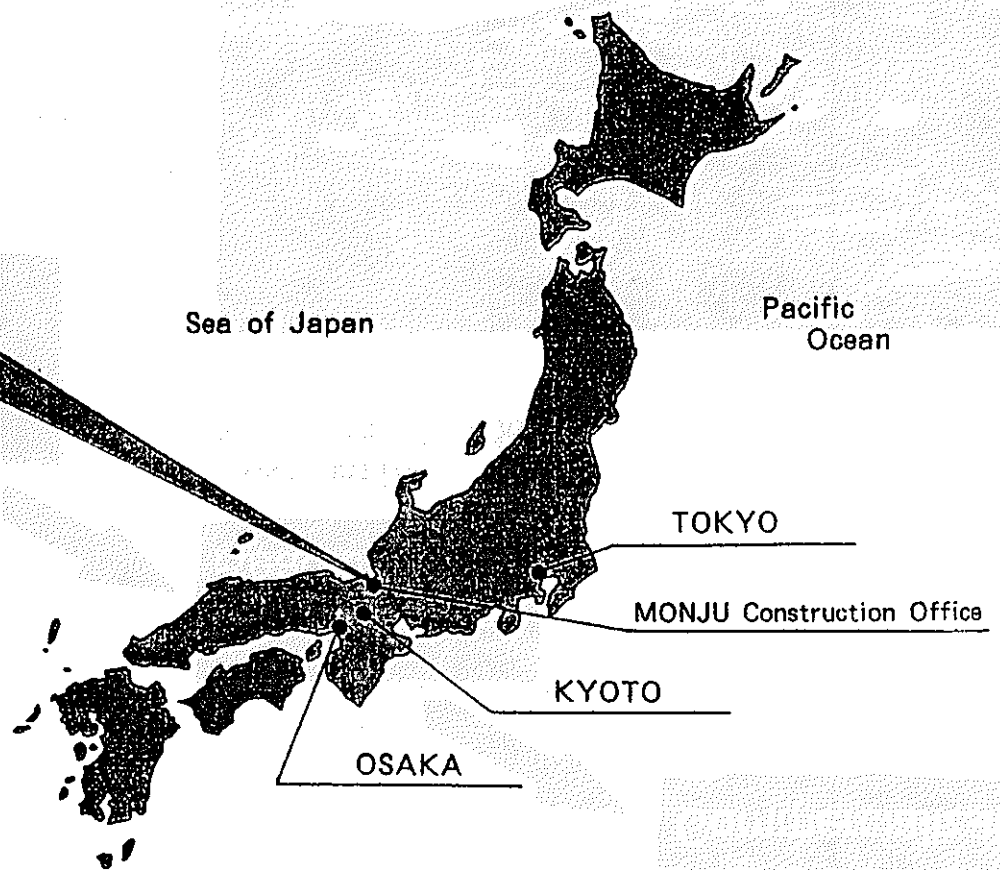
around 2030



LOCATION OF THE MONJU SITE

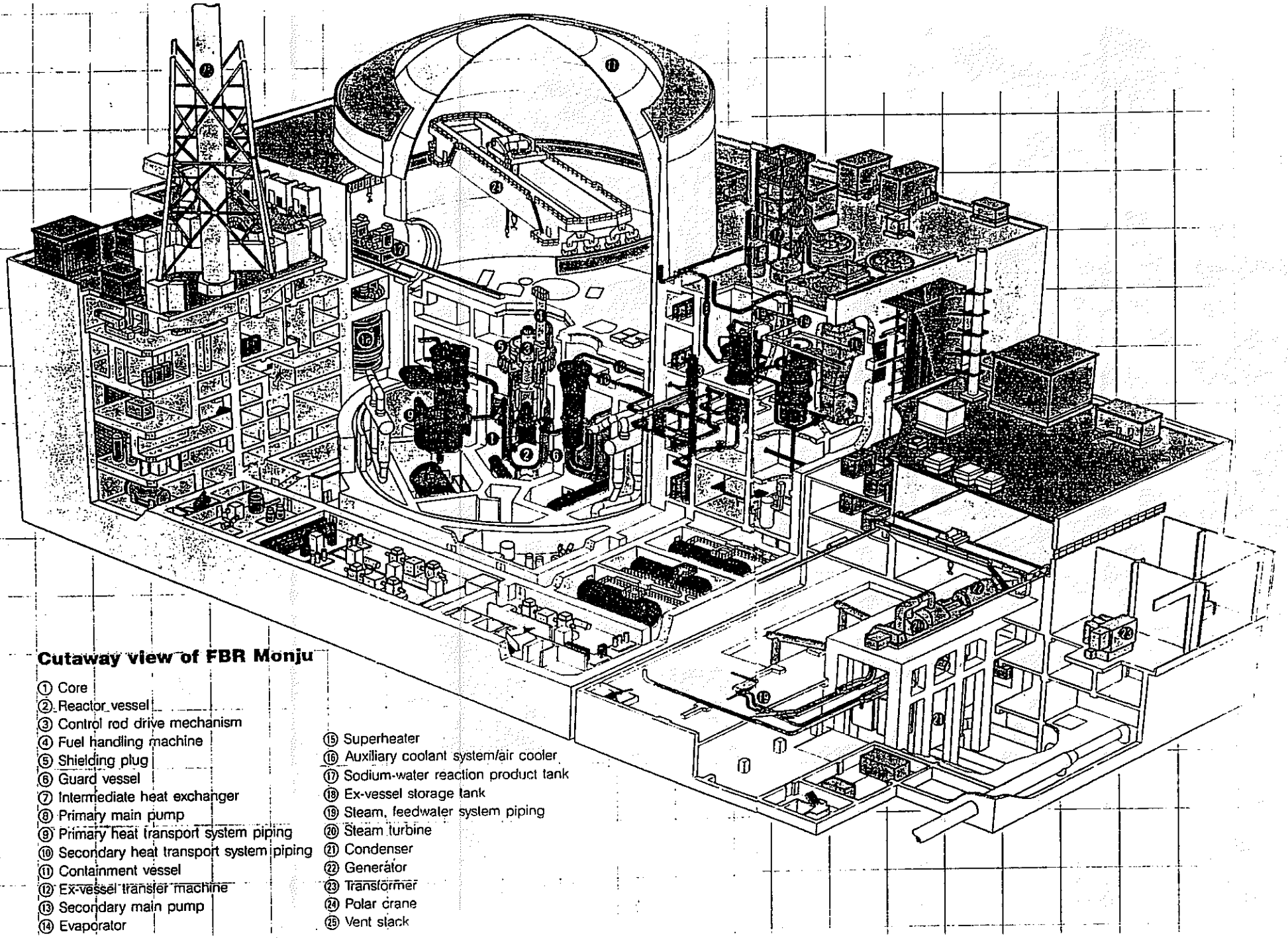


MONJU Site: Shiraki District, Tsuruga City
Fukui Prefecture, 4 0 0 km West of Tokyo



MONJU PLANT KEY PARAMETERS

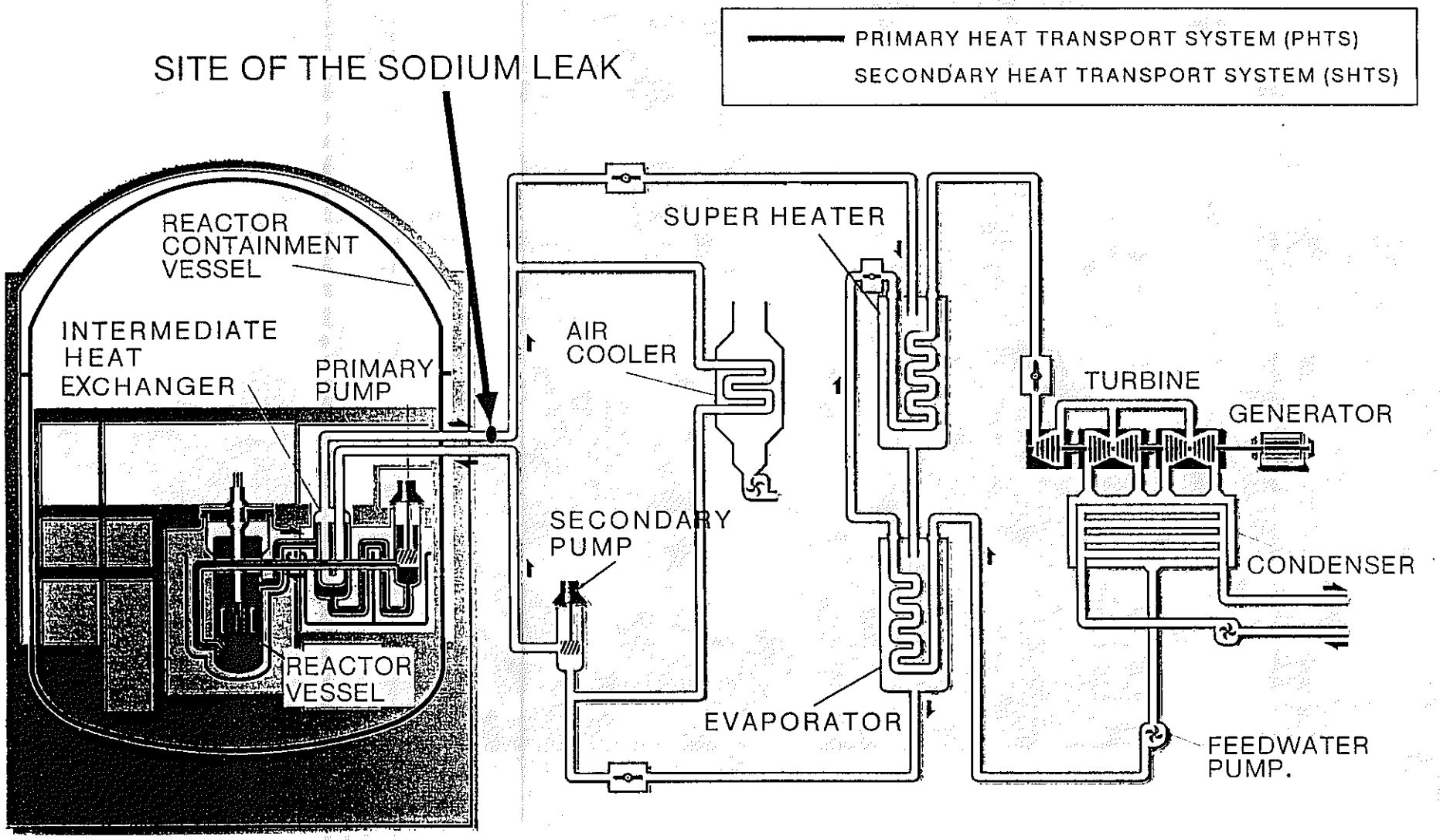
● TYPE	Na COOLED/3 LOOPs
● OUTPUT	714MWt/280MWe
● FUEL	PuO_2-UO_2
● FUEL INVENTRY	5.9 TON(METAL)
● Pu ENRICHMENT	16/21(EQUILIBRIUM)
● BURNUP	80,000MWD/TON
● BREEDING RATIO	1.2
● TEMP	397/529°C (in/outlet)
● REFUELING	every 6 months



Cutaway view of FBR Monju

① Core
② Reactor vessel
③ Control rod drive mechanism
④ Fuel handling machine
⑤ Shielding plug
⑥ Guard vessel
⑦ Intermediate heat exchanger
⑧ Primary main pump
⑨ Primary heat transport system piping
⑩ Secondary heat transport system piping
⑪ Containment vessel
⑫ Ex-vessel transfer machine
⑬ Secondary main pump
⑭ Evaporator
⑮ Superheater
⑯ Auxiliary coolant system/air cooler
⑰ Sodium-water reaction product tank
⑱ Ex-vessel storage tank
⑲ Steam, feedwater system piping
⑳ Steam turbine
㉑ Condenser
㉒ Generator
㉓ Transformer
㉔ Polar crane
㉕ Vent stack

MONJU NUCLEAR STEAM SUPPLY SYSTEM



SUMMARY OF THE INCIDENT

[DEC 8, 1995]

19 : 47 • ALARM OF SEC. HOT-LEG TEMP. HIGH

• FIRE ALARMs STARTED.

19 : 48 • SODIUM LEAK DETECTOR ALARMED

20 : 00 • REDUCED THE REACTOR POWER

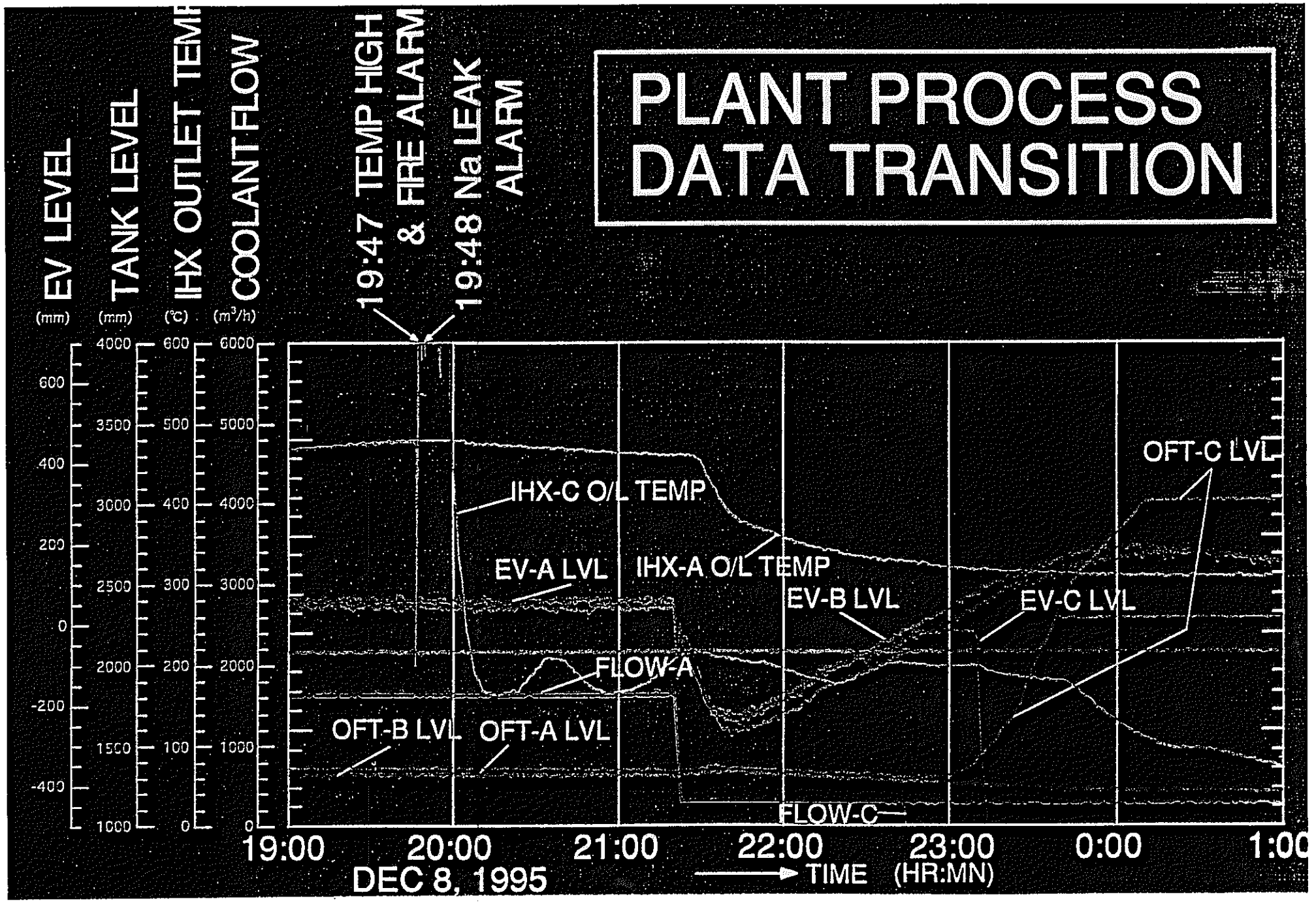
21 : 20 • TRIP THE REACTOR BY MANUAL

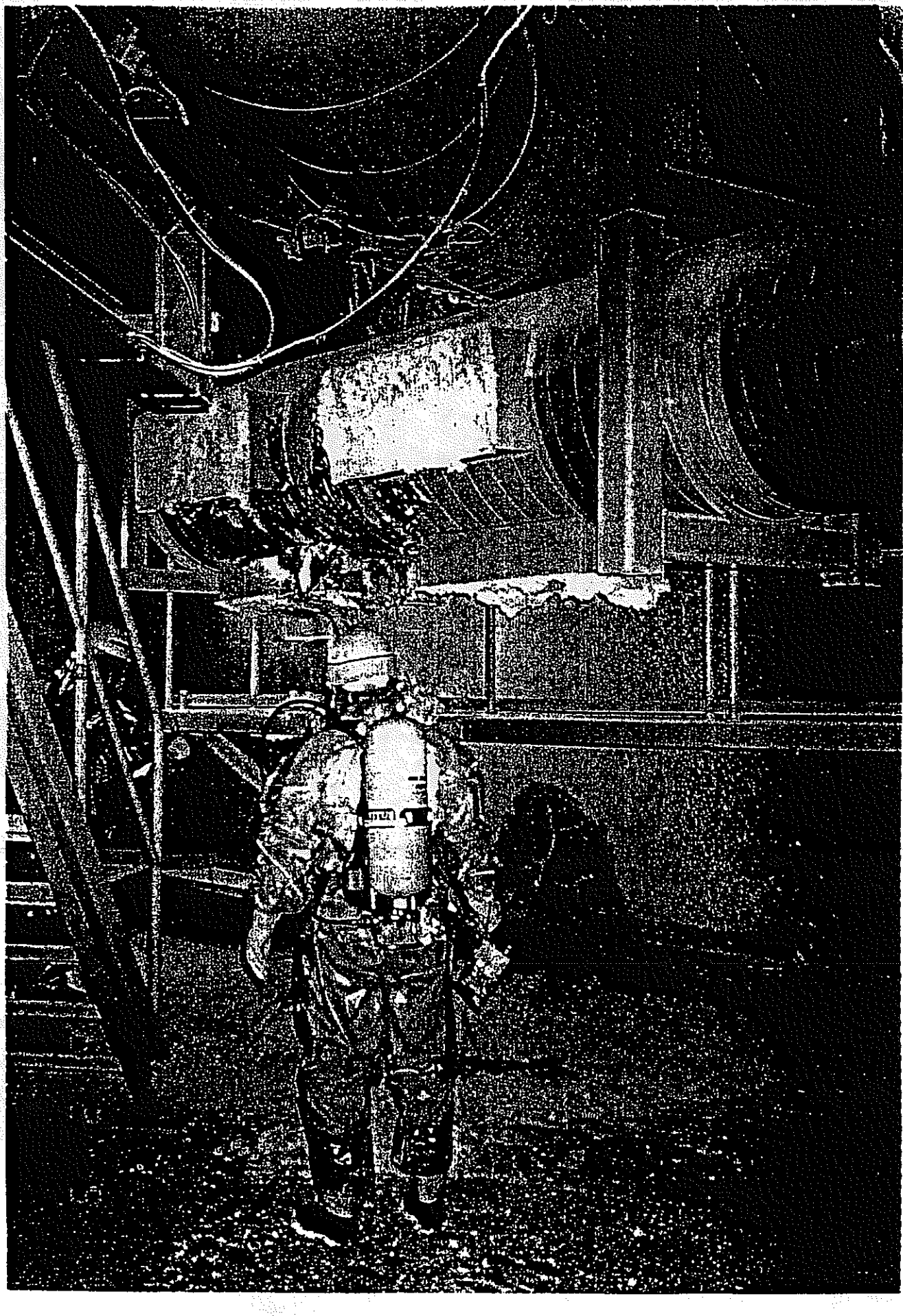
22 : 40 • SODIUM DRAIN OPERATION

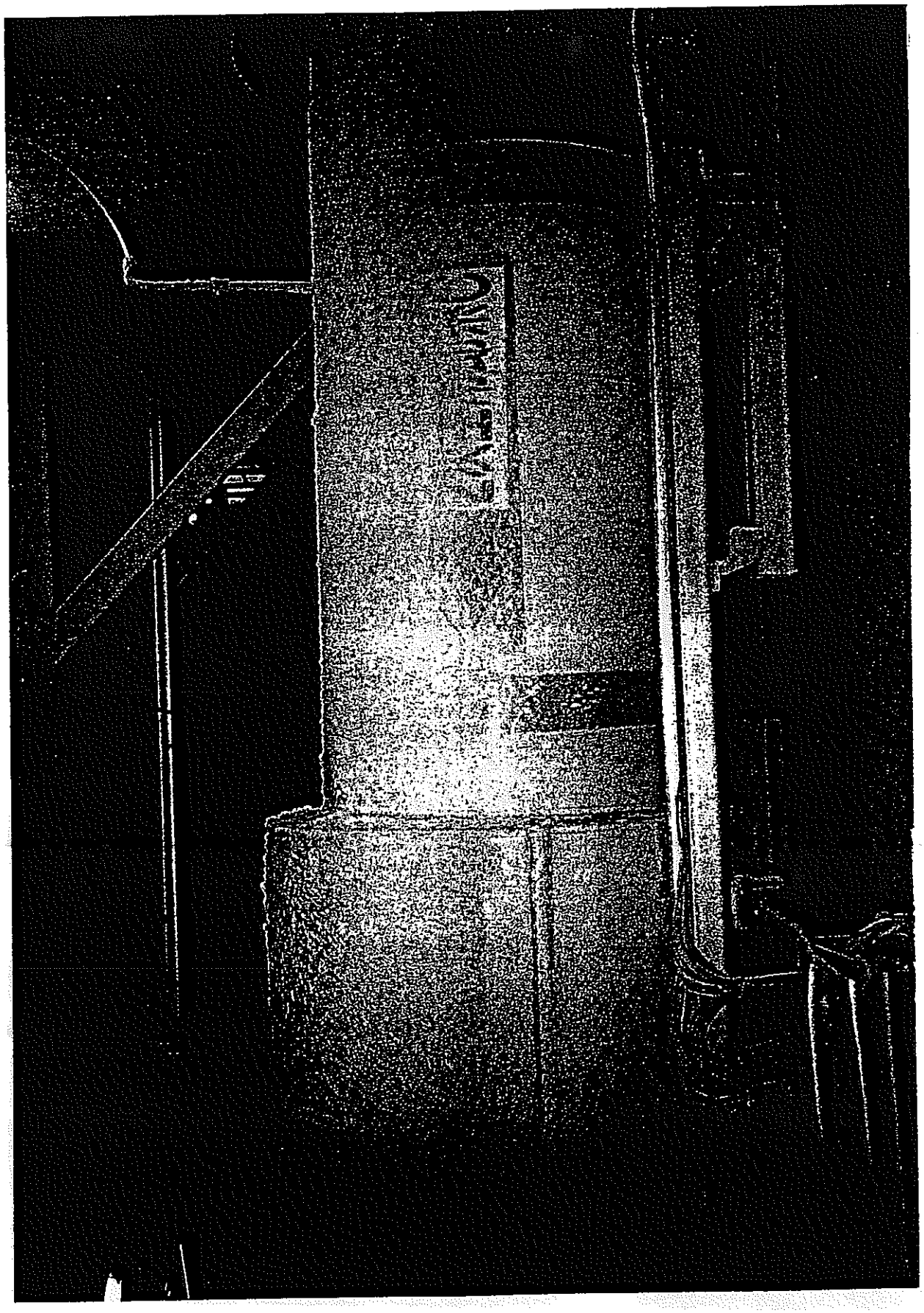
23 : 13 • HVAC BLOWER TURNED-OFF

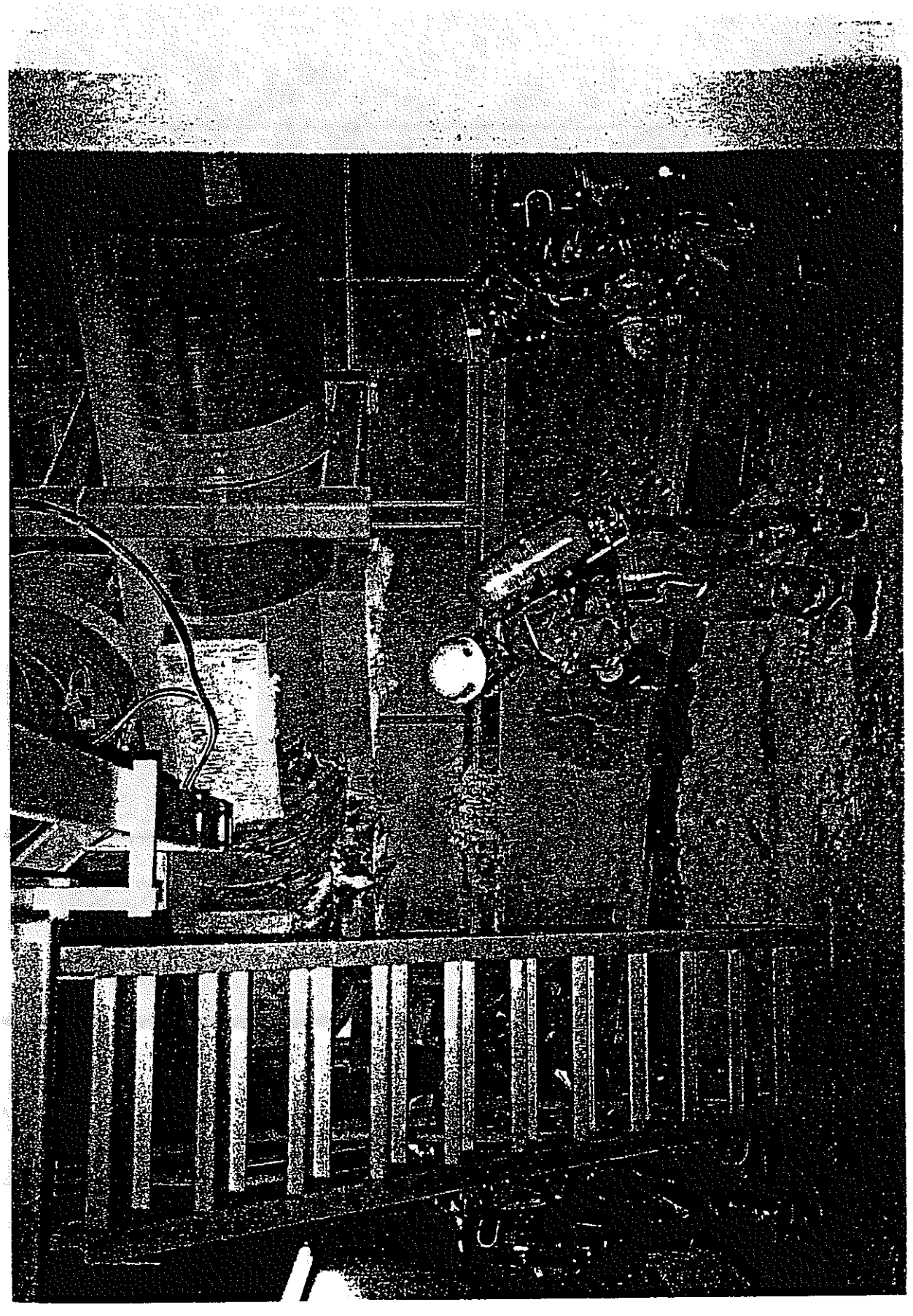
[DEC 9]

0 : 15 • COMPLETION OF DRAIN

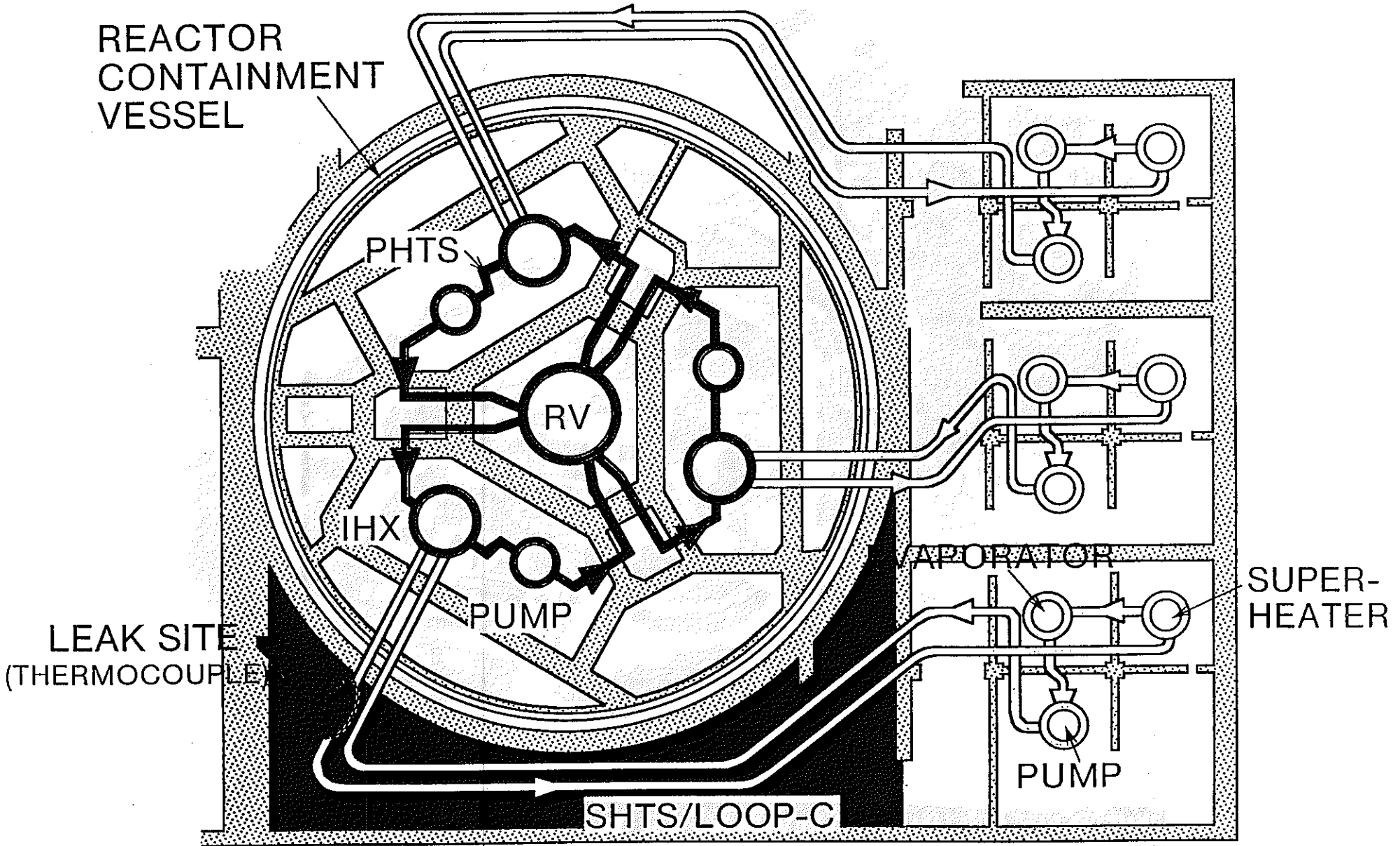

PLANT PROCESS DATA TRANSITION
EV LEVEL (mm)
TANK LEVEL (mm)
IHX OUTLET TEMP (℃)
COOLANT FLOW (m³/h)
19:47 TEMP HIGH & FIRE ALARM
19:48 Na LEAK ALARM
IHX-C O/L TEMP
EV-A LVL
IHX-A O/L TEMP
EV-B LVL
EV-C LVL
OFT-C LVL
FLOW-A
OFT-B LVL
OFT-A LVL
FLOW-C
19:00
20:00
21:00
22:00
23:00
0:00
1:00
DEC 8, 1995
TIME (HR:MN)



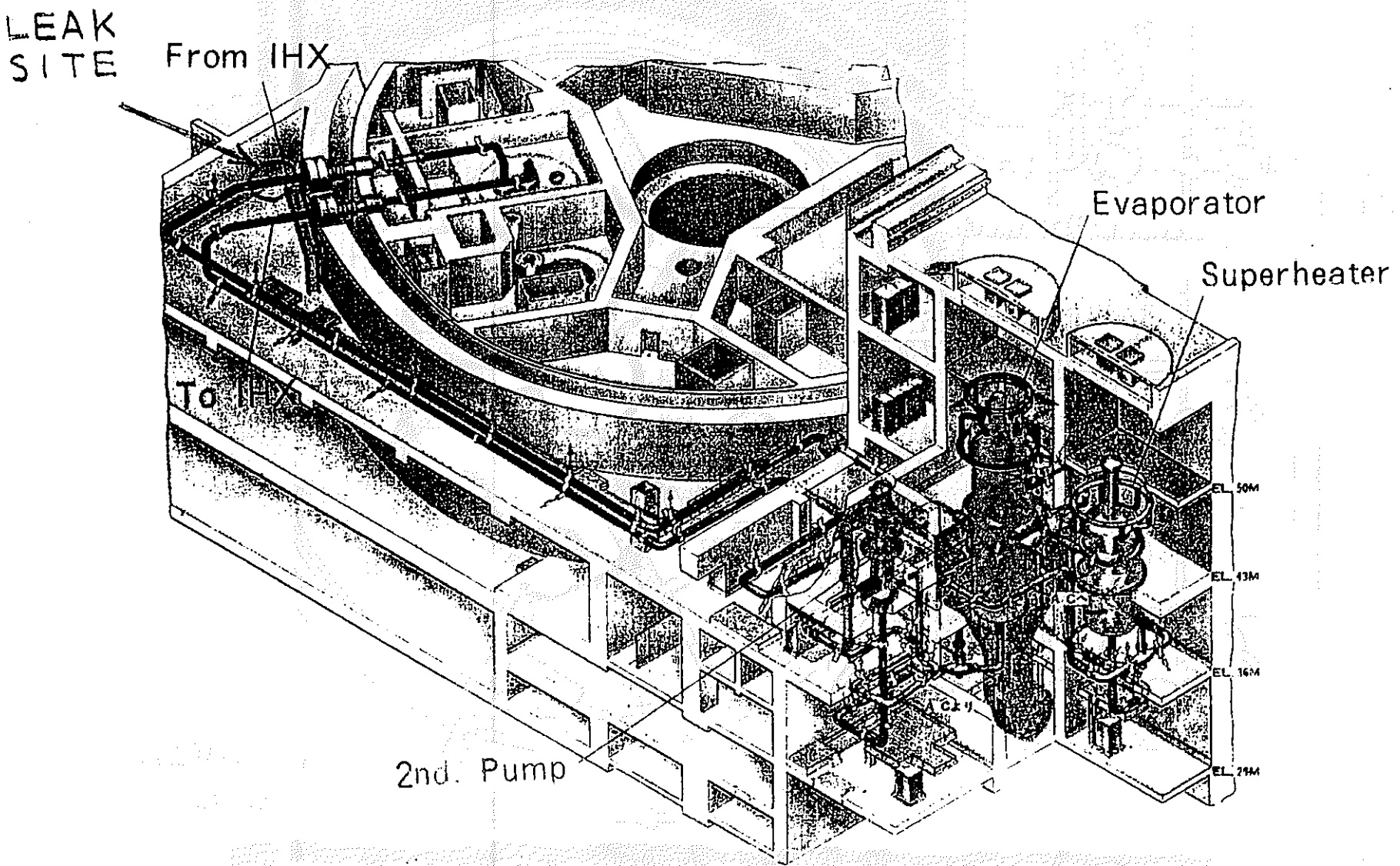


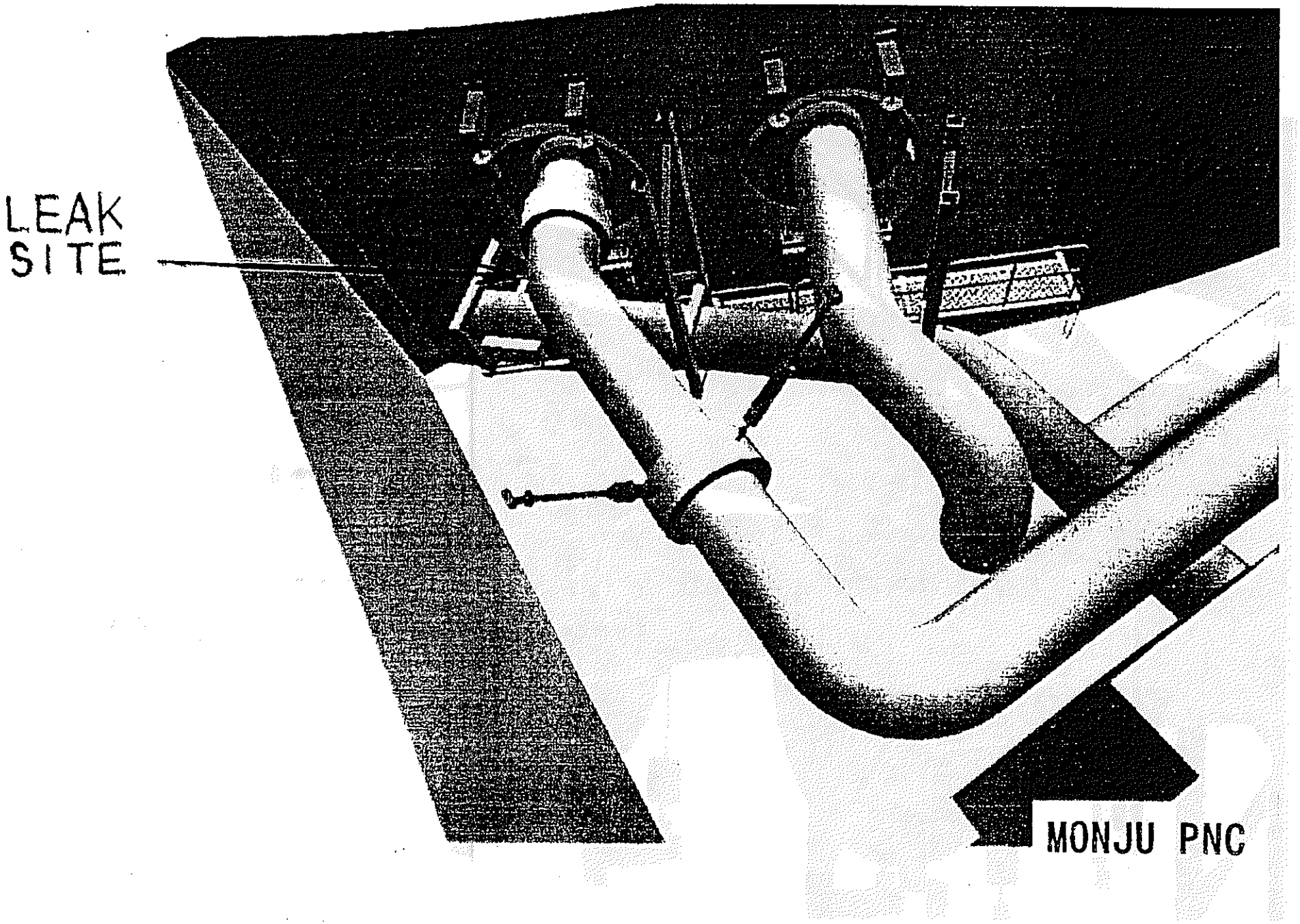


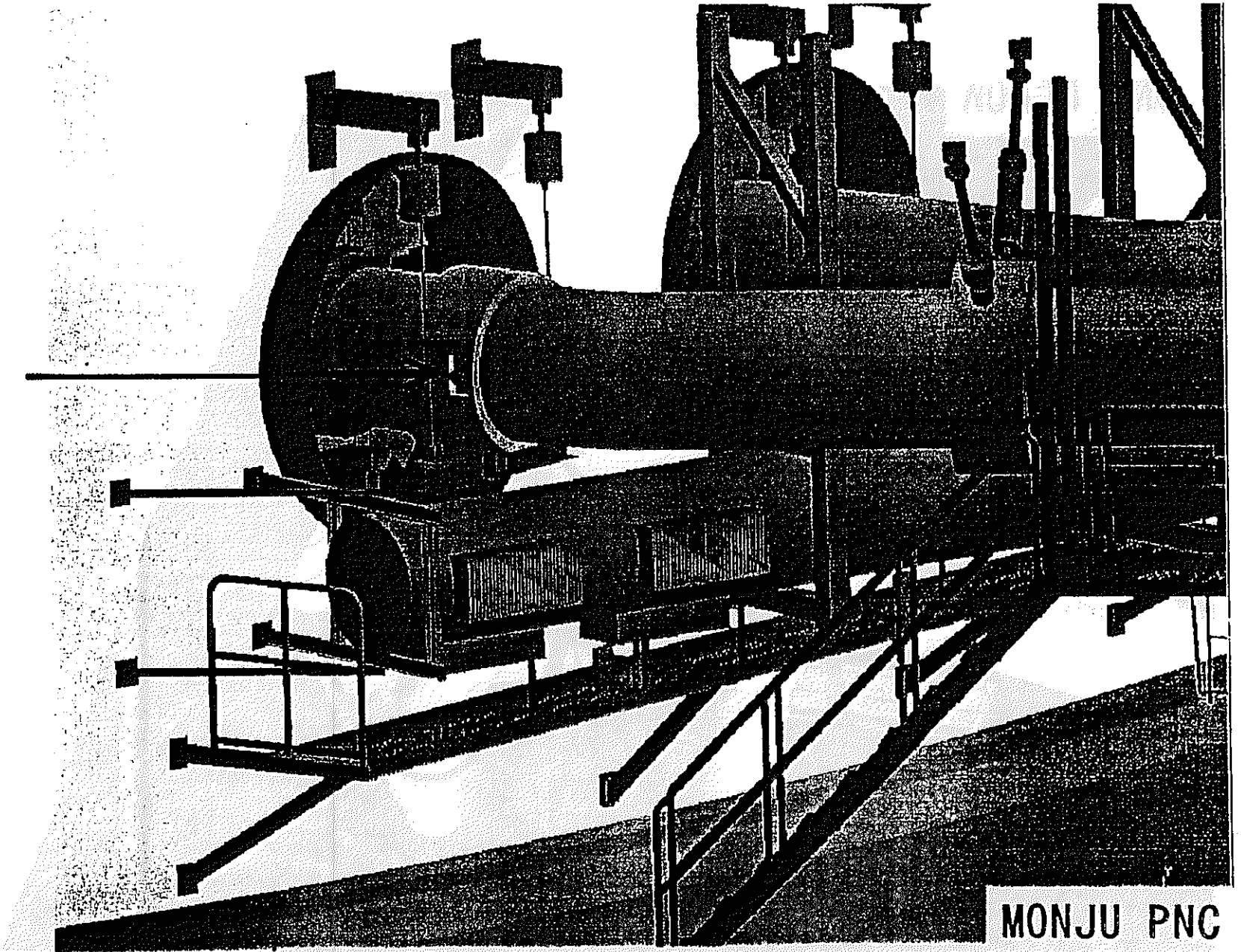
LOCATION OF LEAKAGE

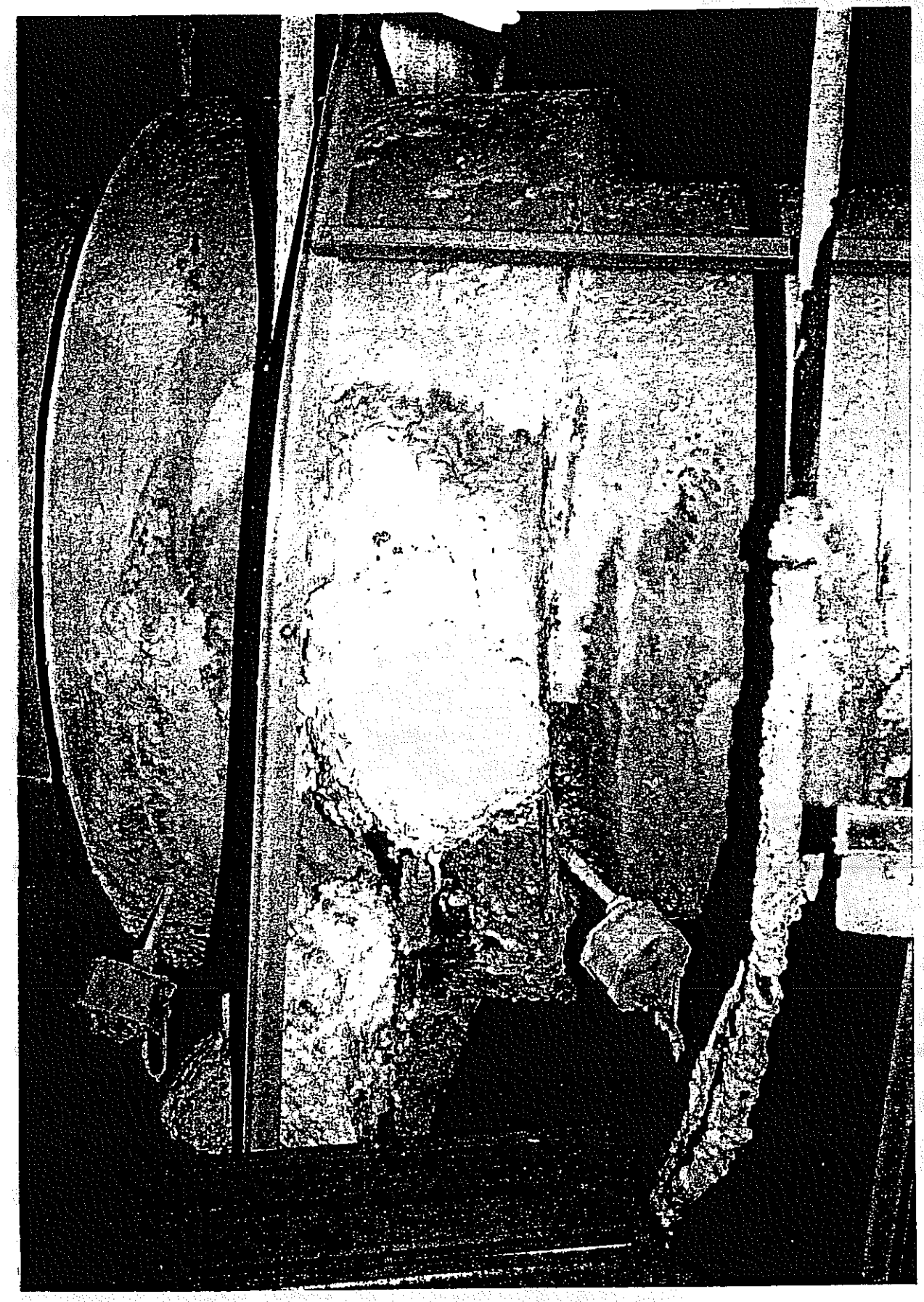


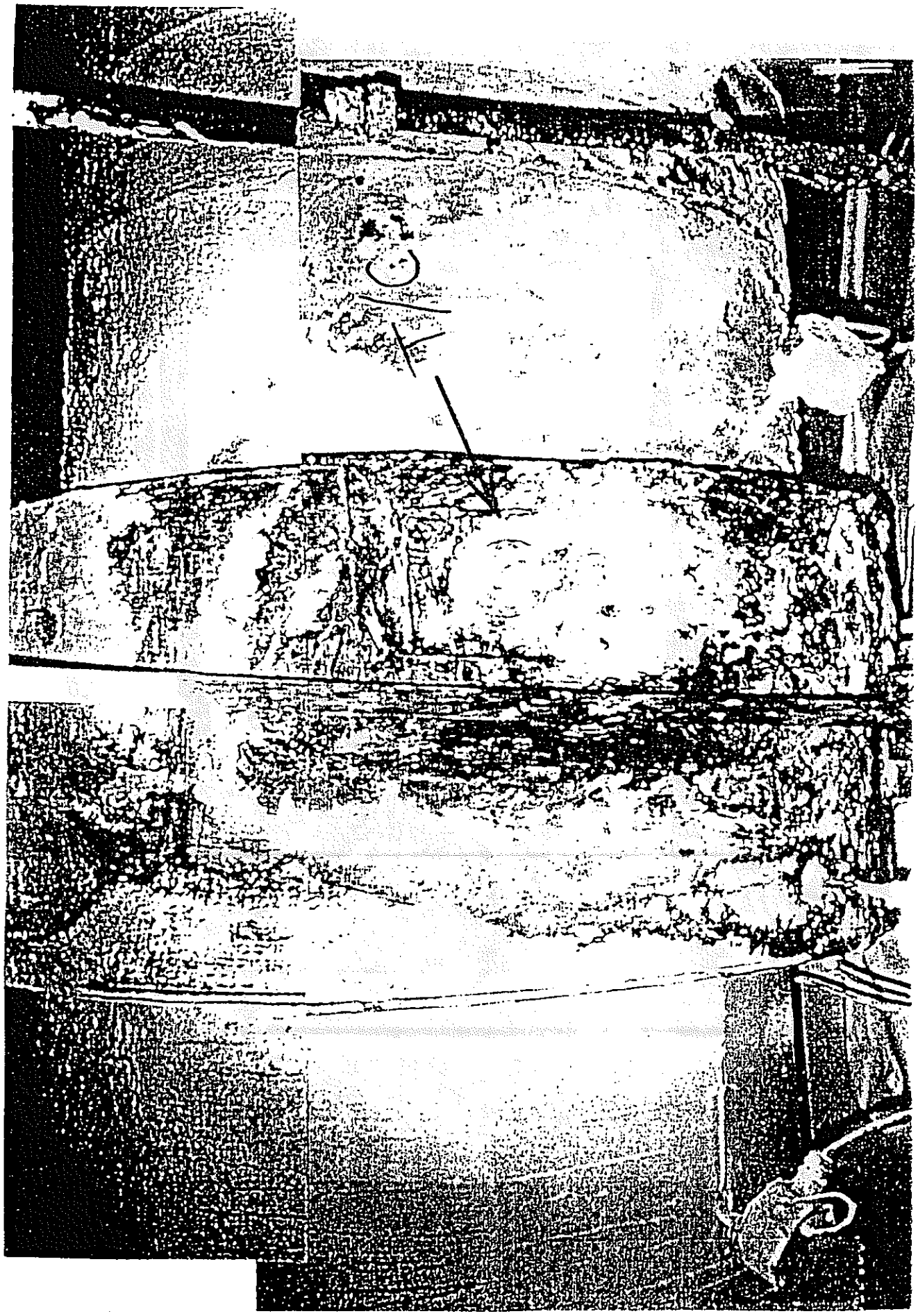
SECONDARY HEAT TRANSPORT SYSTEM




LEAK
SITE
MONJU PNC


LEAK
SITE
MONJU PNC



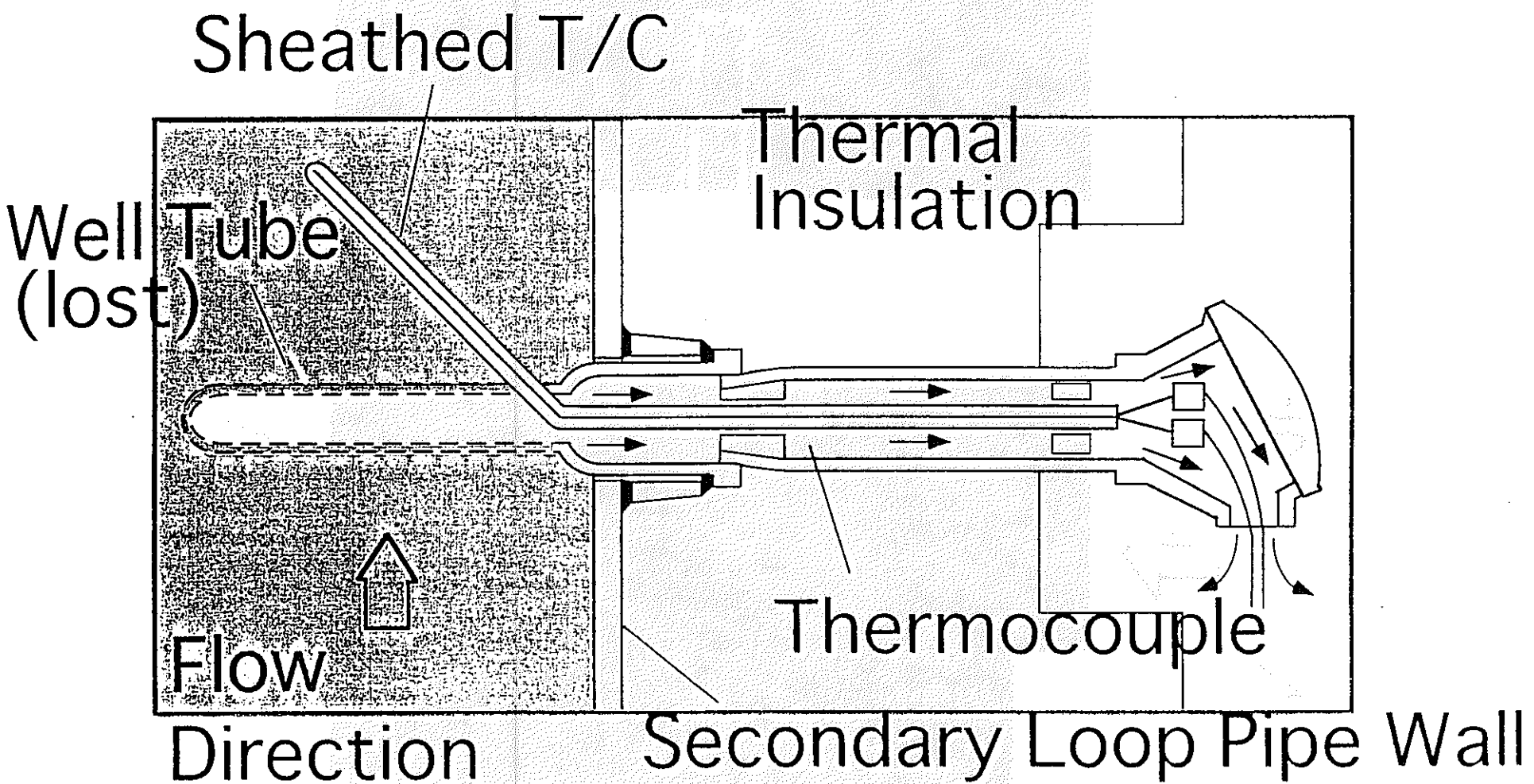


Na FLOW

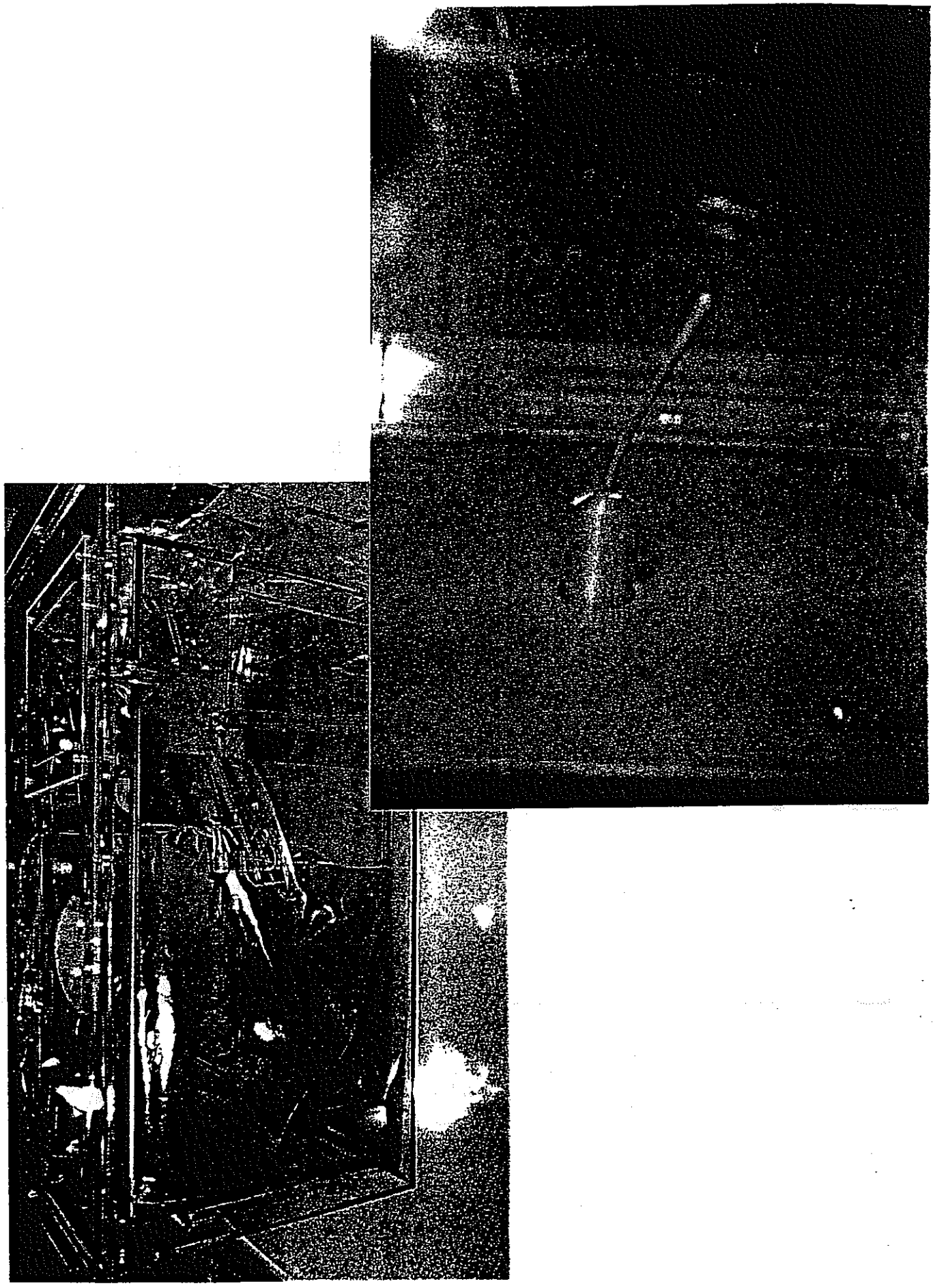
No.2

CH-1-A-G

BENT
T/C

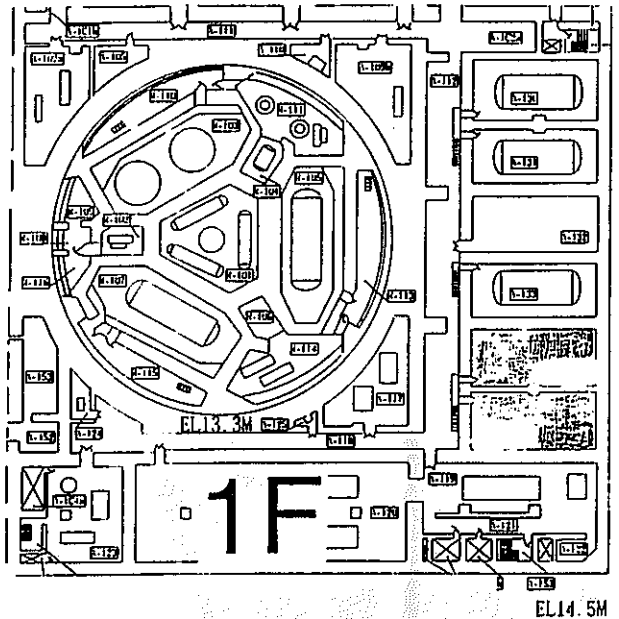


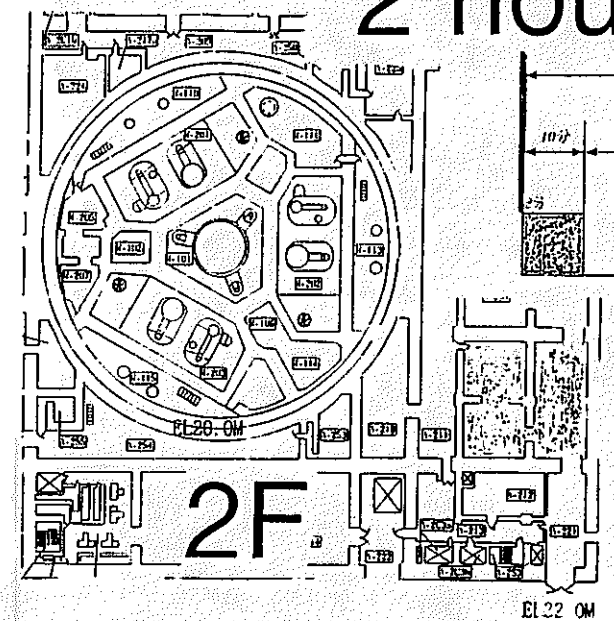
FLOW PATH OF
THE SODIUM LEAKED

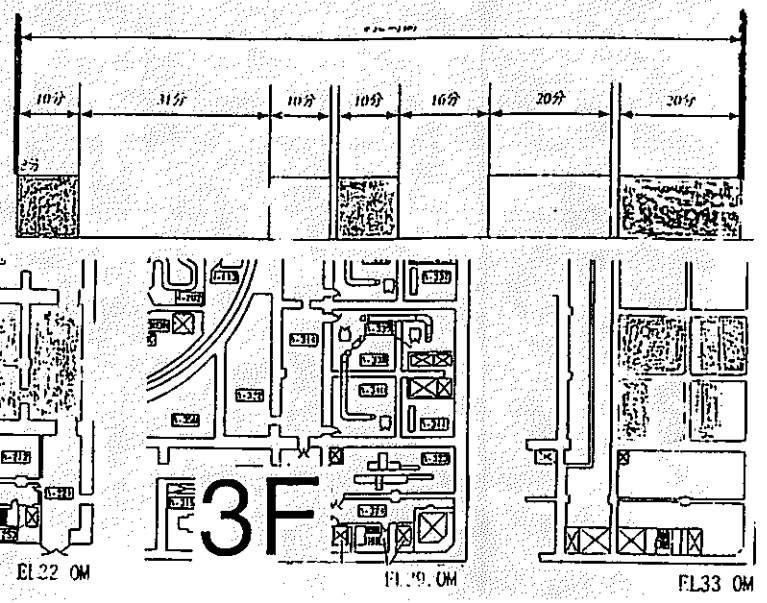


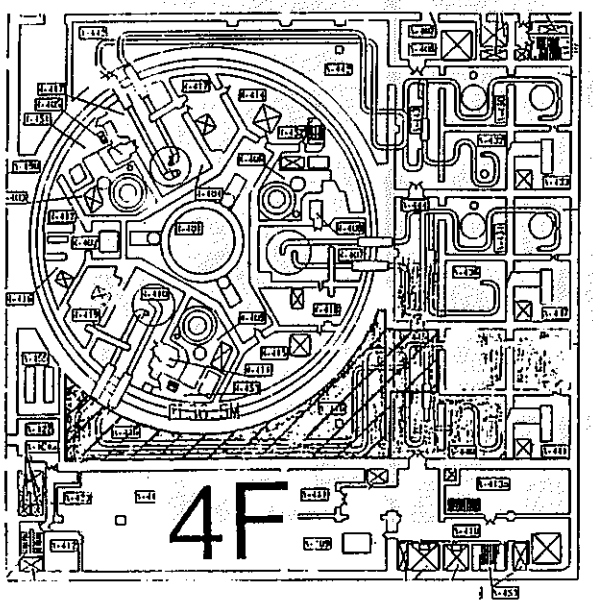
WORKING OF FIRE ALARMs

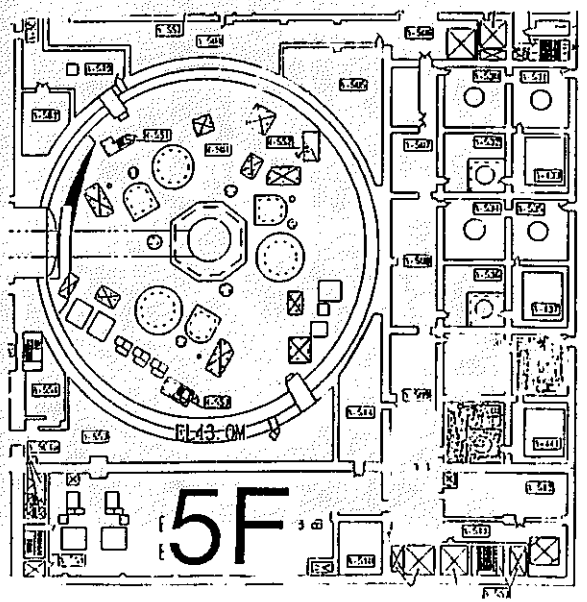
2 hours in duration

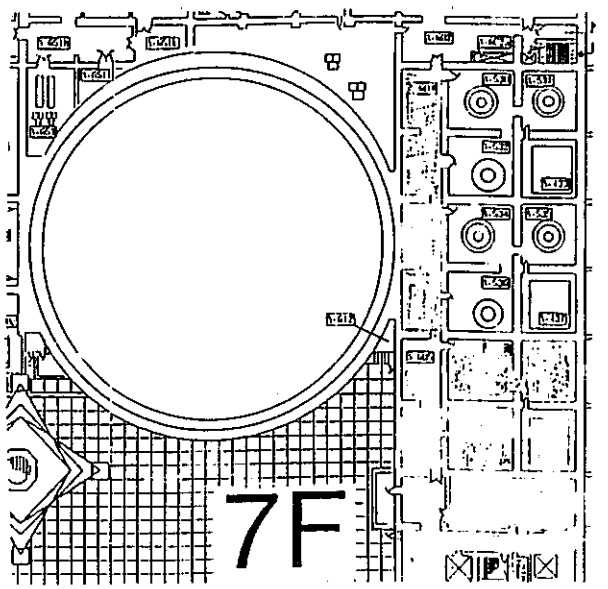


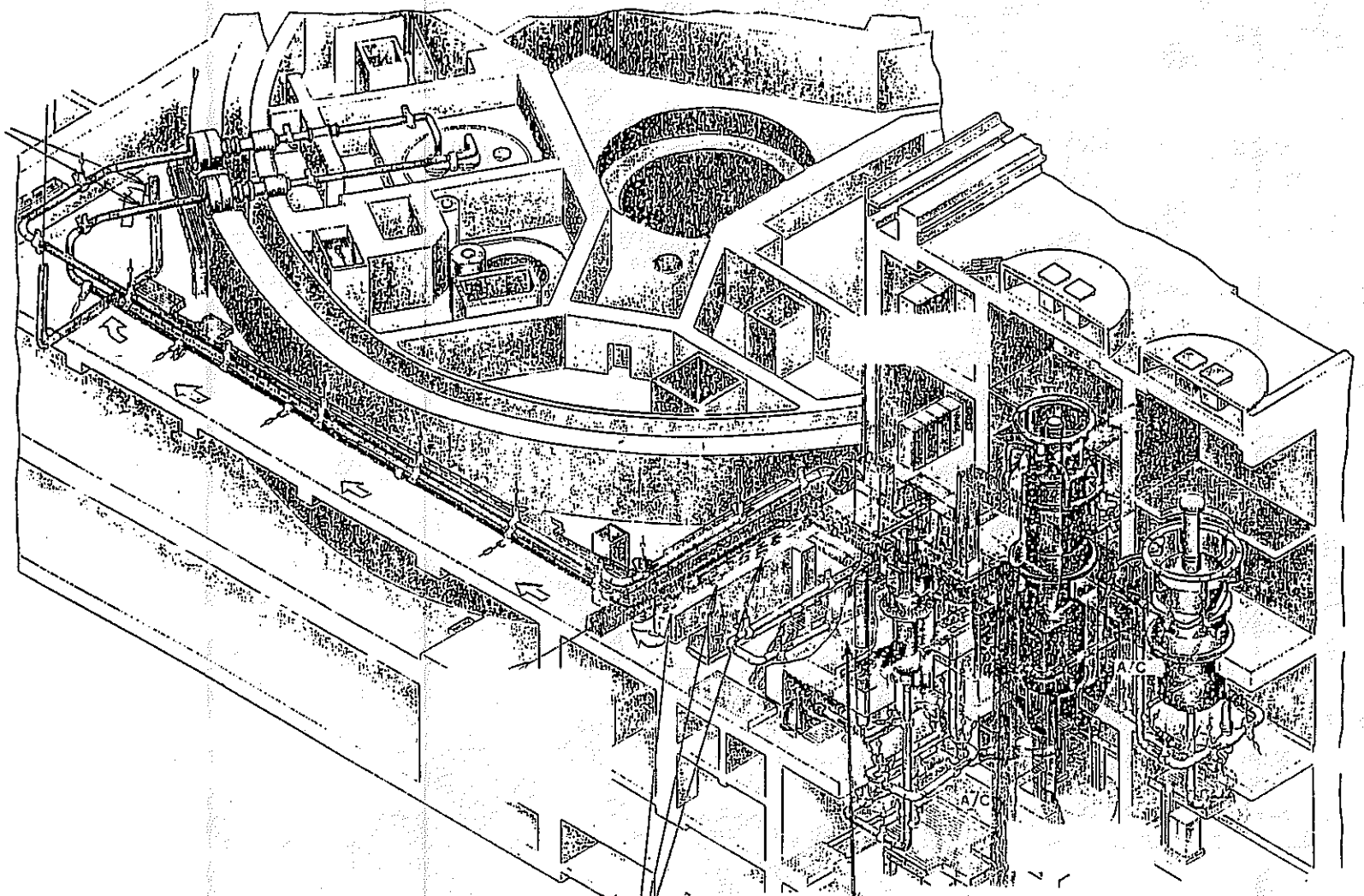












REACTION PRODUCT TRANSFER THROUGH HVAC DUCTs

AMOUNT OF LEAKED SODIUM ESTIMATED

AMOUNT: ≒ 700kg

LEAKRATE: ≒ 50g/sec.

TIME: ≒ 4hours

ESTIMATED FROM

(1) PROCESS DATA

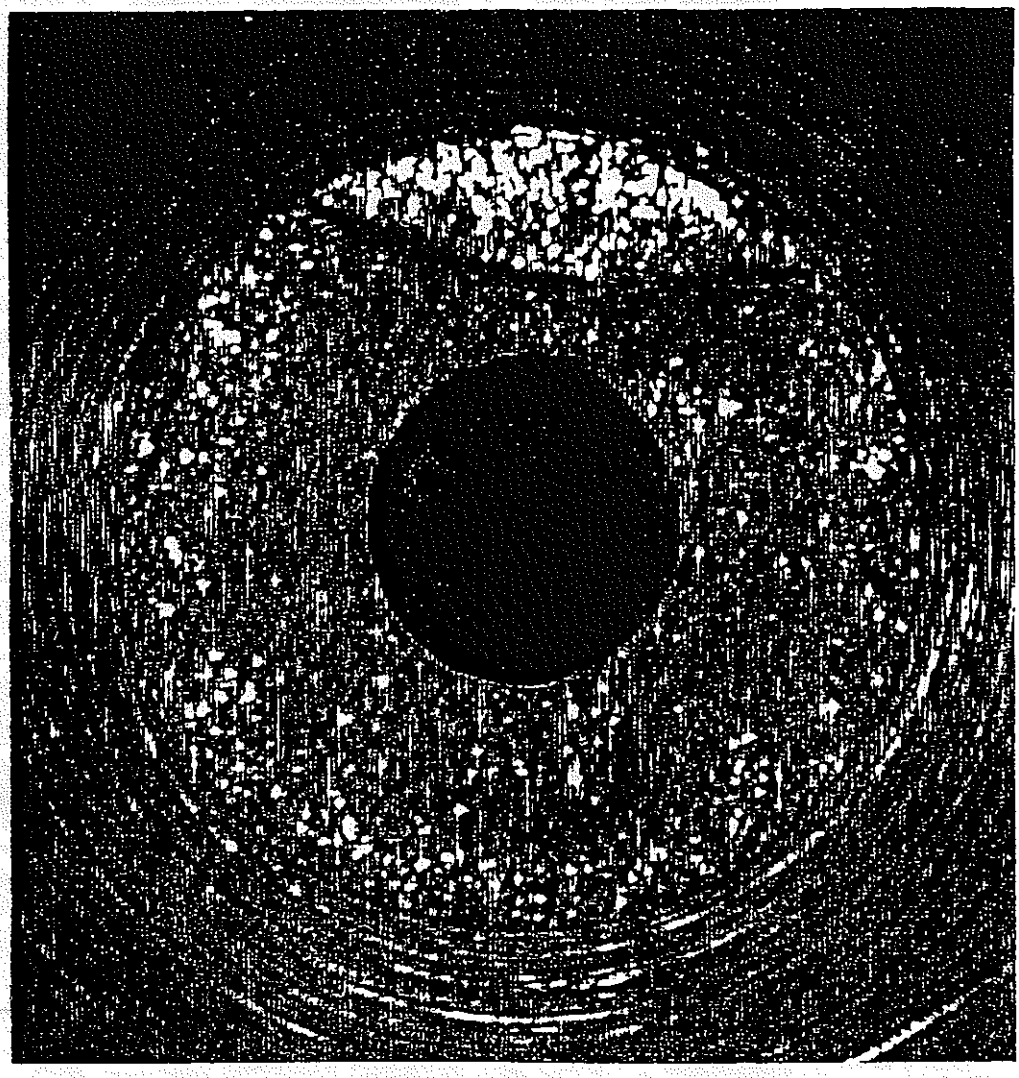
(2) CALCULATION

(3) HYDRAULIC SIMULATION TEST

DESIGN MAXIMUM OF MONJU SEC. SODIUM LEAK IN SAFETY ANALISYS

- SIZE OF OPENING— 1/4Dt/about 15 cm^2
- LEAKED SODIUM IN MAX.— 150 m^3
- ANALITICAL RESULTS—
  - INTERNAL PRESSURE IN BLDG. COMPARTMENT about 0.22 < 0.6 kg/cm^2
  - MAX. TEMP. OF CATCH-PAN about 460<530 ℃

FLOW DIRECTION



THE BREAKING SURFACE THROUGH DIGITAL MICROSCOPE (10x)

ASME SECTION III DIVISION 1–APPENDIX N

N-1324.1 Avoiding Lock-in Synchronization

(a) Avoiding both lift and drag direction lock-in :

$V/f_1D < 1$

(b) Lock-in being suppressed : $Cn > 64$

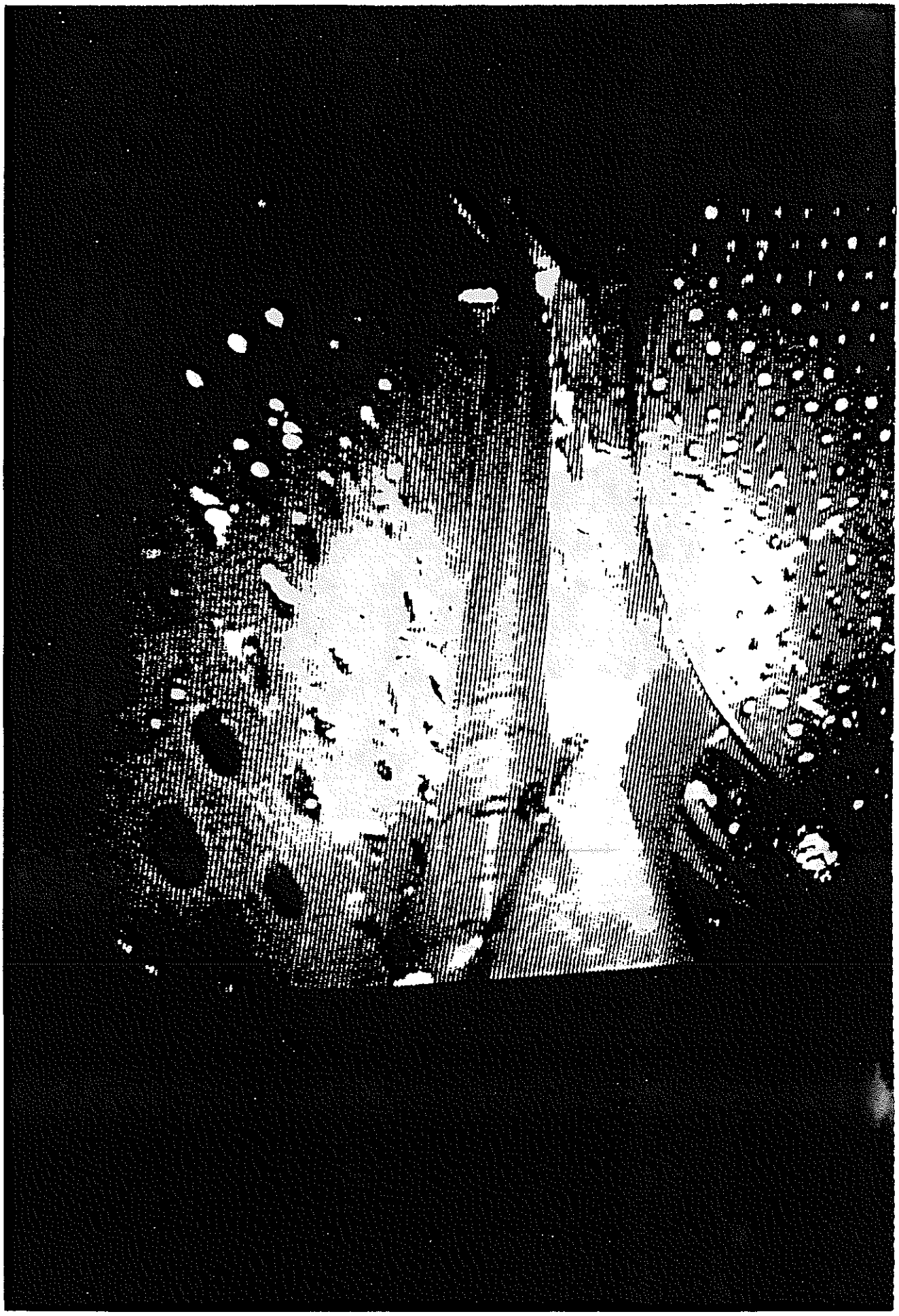
(c) Lift direction lock-in avoided & drag one suppressed : $V/fnD < 3.3$ & $Cn > 1.2$

(d) Lock-in in lift direction avoided :

$fn < 0.7fs$ or $fn > 1.3fs$

CONCLUSION

- THROUGH THE BROKEN THERMO-THIMBLE, AROUND 700 KG OF SHTS SODIUM WAS LEAKED AND REACTED WITH O_2 IN THE AIR, RESULTED IN THE SPREAD OF REACTION PRODUCT IN BLDG.
- THE CURRENT EXAMINATION SHOWED THE BREAKAGE WAS CAUSED BY HIGH CYCLE FATIGUE DUE TO THE FLOW-INDUCED VIBRATION OF THE THIMBLE.
- INTENSIVE INVESTIGATIONS ARE NOW IN PROGRESS IN ALL AREAS TO DETERMINE THE CAUSE AND COUNTERMEASURES.



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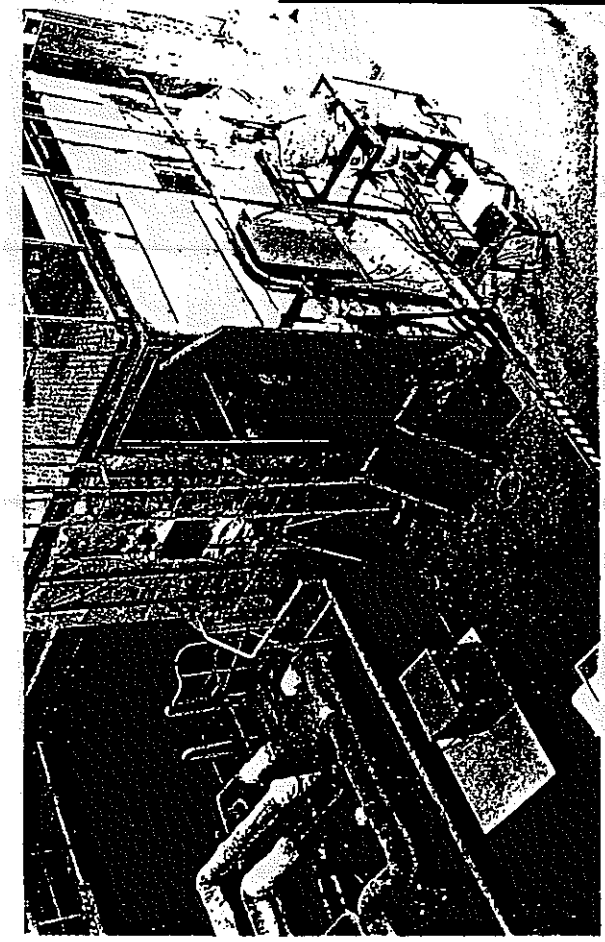
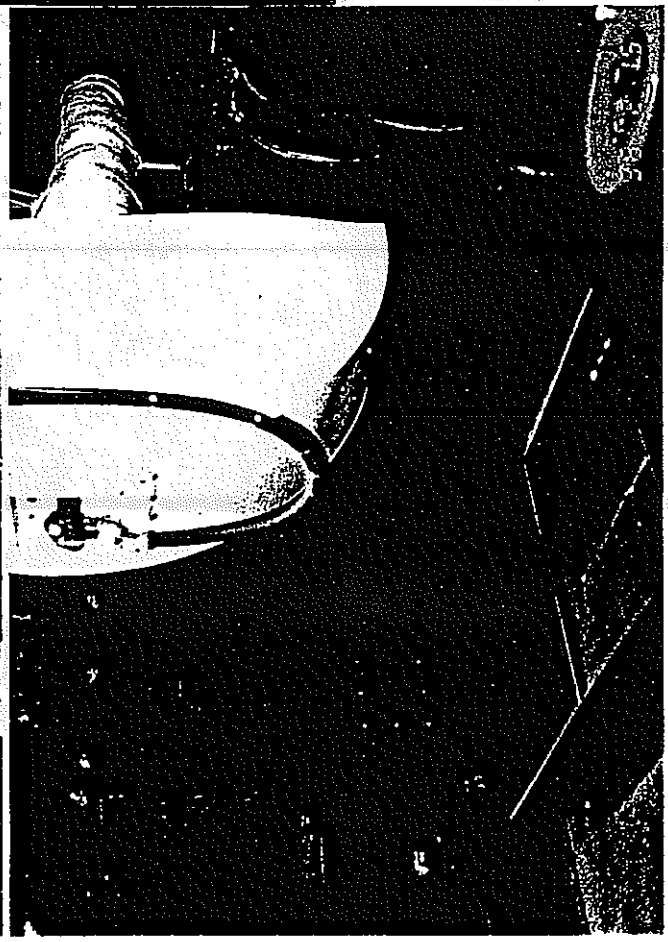
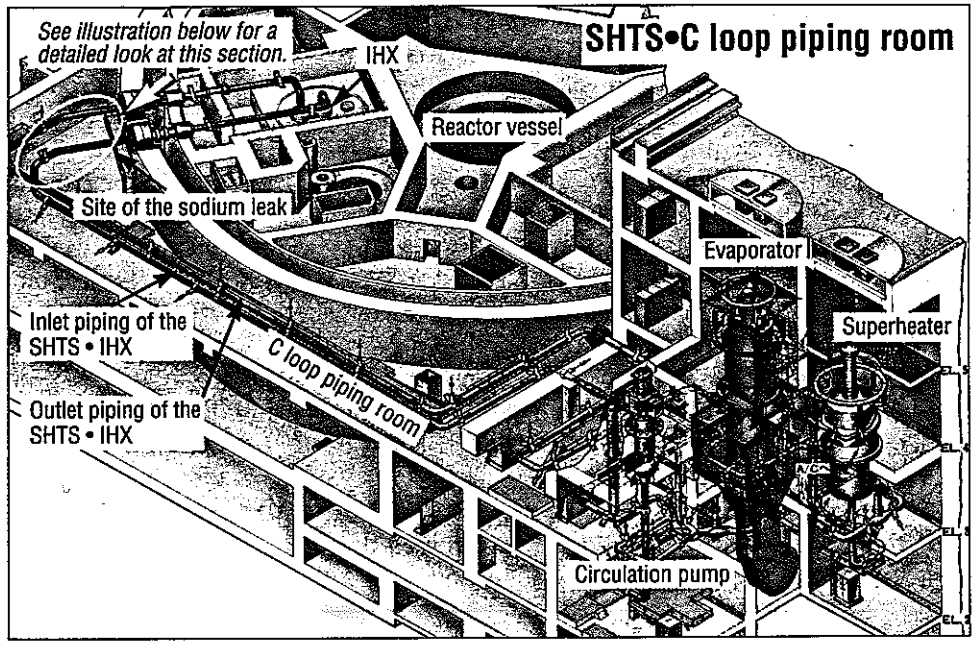




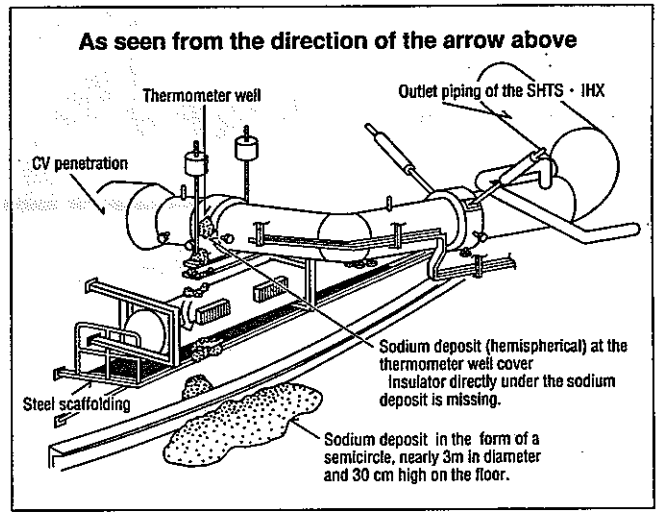
どうれん TOPICS No.3 FEBRUARY 1996

Leakage in the Secondary Heat Transport System (SHTS) of Prototype FBR Monju — 40% Startup Test (First Update)

The prototype fast breeder reactor Monju started operating on December 6, 1995. When power was being increased for the plant trip test, as part of 40% power test, an alarm sounded at 19:47 on December 8 indicating that sodium temperature was unusually high at the outlet of the intermediate heat exchanger (IHX) C loop in the SHTS. The fire alarm (smoke detector) sounded as well. Another alarm followed, indicating that there was a sodium leak in SHTS·C loop. Generation of smoke was confirmed when the door of the piping room of SHTS (C) was opened. The plant's·operators determined that a sodium leak had occurred and decided to begin manual shutdown operations. Reactor power-down operations began at 20:00.



The state inside the piping room (C) was checked again through the gap of its door and an increase in white fume was observed. Accordingly, electric power generation was stopped at 21:15, and the reactor was manually tripped at 21:20. After the trip, the reactor was cooled down by the auxiliary cooling system (ACS) and was shifted to low-temperature shutdown state. To minimize leakage, sodium was drained from the piping of SHTS·C loop beginning around 22:40 and operations were complete at 00:15, December 9.



Publisher: Public Relation Office, Power Reactor and Nuclear Fuel Development Corporation (PNC)

Multiple investigations of the piping room (C) on and after December 9 have confirmed solidified materials associated with the sodium leak near and around the thermometer well of IHX. Sodium was deposited on the floor in the form of a semicircle, nearly 3m in diameter and 30 cm high, approximately $1m^3$. In addition, sodium compounds resulting from the sodium leak was lightly spread across the room's floor. There were no further anomalies observed in the piping of SHTS. A sodium compound resulting from the leak was recorded on the entire floor of the steam generator (SG) room (C) connecting to the piping room and in the passage of the first and second floors.

The following works were performed to determine the cause of the incident from December 11, 1995 through January 31, 1996.

1. Sodium draining of a portion of the C loops in SHTS and Primary Heat Transport System (PHTS)
2. Removal and recovery of deposits from the floor inside the piping room (C) and sodium compounds around the leakage area
3. Cutting away and removal of sodium ventilation duct at IHX outlet
4. X-ray of area around thermometer well
5. Removal of piping heat insulator from around thermometer well

Results of works have showed the following.

1. Radiation effects

No effects of radiation to workers, residents and the surrounding environment have been observed as a result of the accident.

2. X-ray results

The small tip of the thermometer well well that is inserted into the piping could not be detected from the x-rays. The thermocouple was bent at a 45° angle in the piping in the direction of the flow of sodium.

3. Amount of sodium removed

Approximately 700 kg of sodium oxide compounds have been recovered as of present.

4. Heat insulator removal

The heat insulator from the the containment vessel to elbow #1 was removed. A small amount of the sodium compounds had adhered to the inside of the outer shroud and the surface of the insulator. There was no show of sodium compounds in the insulator.

The following tests and investigations will be implemented.

1. Investigation of cause of sodium leak

●After removing the thermometer well, detailed tests will be run to observe the fractured surface.

●Investigation of the design, fabrication, inspection and plant data.

2. Search for the missing thermometer well

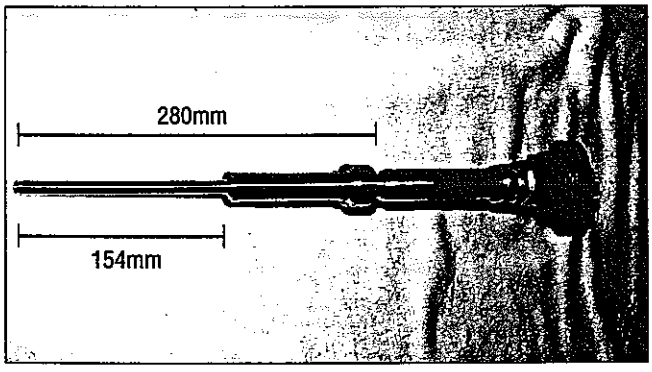
●Using x-ray and other methods, search for and retrieval of the lost part

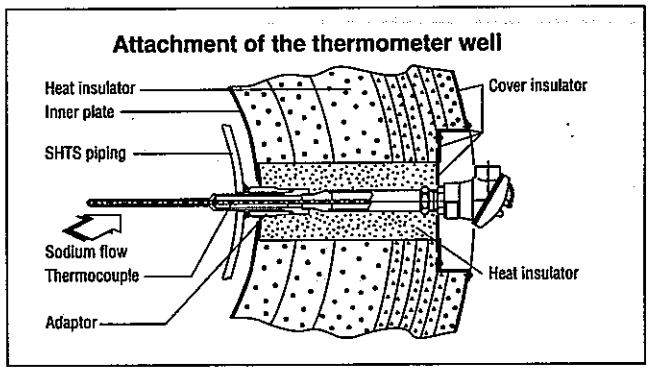
3. Study of effect on surrounding equipment

●In addition to investigating the leakage route, an investigation of the leaked sodium and resulting compounds, and the amount and chemical breakdown of the deposit

4. Behavior of the sodium compounds

●Study of the behavior of the sodium aerosol that spread through the building's rooms





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STATUS OF SODIUM COOLED FAST REACTOR

DEVELOPMENT

IN THE RUSSIAN FEDERATION

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ABSTRACT

This report describes the recent development and activities concerning fast ractors in Russia. The status of nuclear power in 1995 and operational experience of the BR-10, BOR-60 experimental reactors and of the BN-600 nuclear power plant are presented.

Main results of R&D program in fast reactor area are discussed.

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1. THE STATUS OF NUCLEAR POWER IN 1995.

As of January 1996, 29 commercial power units were operating in Russia in 9 nuclear power plants with a total installed capacity of 21242 MW. This figure includes 13 units with VVER (6 VVER-400 units and 7 VVER-1000 units), 15 units with uranium-graphite channel-type reactors (among them 11 RBMK units), and 1 unit with fast reactor (BN-600).

The NPPs generated $99.3 \cdot 10^6$ MW-h ($97.8 \cdot 10^6$ MW-h in 1994).

The mean load factor was 53.4 % (52.53 % in 1994).

During 1995 there were 95 events at the 29 units in operation. As to the international estimate of the events, only 3 of the above 95 incidents can be classified by the INES: 1 incident - level 2, 2 incidents - level 1.

From 28 unplanned outages 17 involved the triggering of emergency protection systems. As a whole, in 1995 safety indices of NPPs in operation were higher than in 1994.

2. FAST REACTOR OPERATIONAL EXPERIENCE.

2.1. THE BN-600 NUCLEAR POWER PLANT.

Operating histogram of the BN-600 reactor for 1995 is shown in Fig.1. During this year there have been two shutdowns of the reactor for planned maintenance work and core refueling and one unplanned shutdown due to two loops outages (see fig.1). During this period the load factor was 70.31 % (78.2 % in 1994). This reduction of the load factor compared to one in 1994 is caused by heat output for consumers (0.57 %), and by unplanned loop outages the reactor shutdown.

In Table 1 main BN-600 reactor characteristics for the last five years and from the start of operation are presented.

In 1995 there were no water-into-sodium leaks in steam generators.

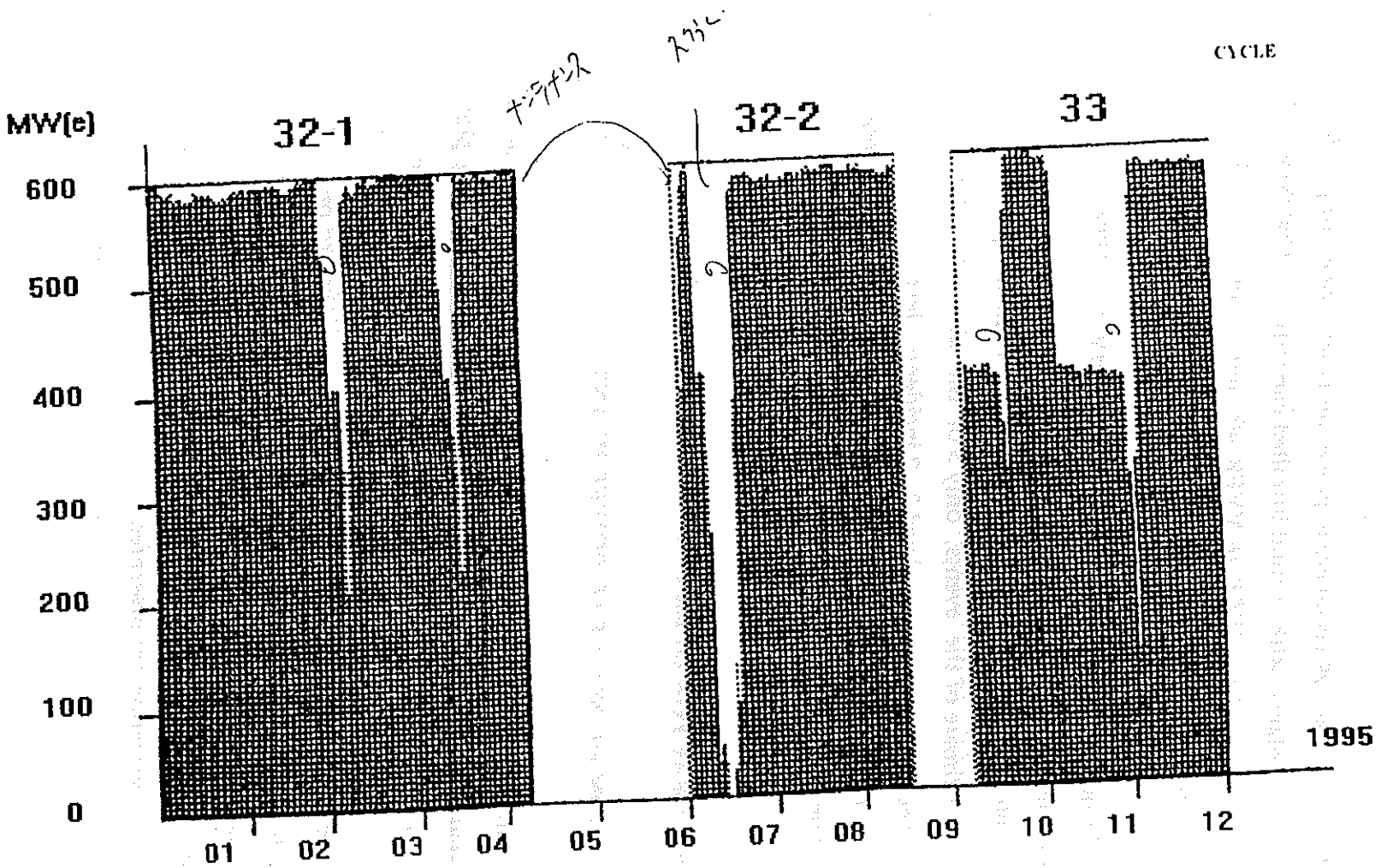

Fig.1. BN-600 Operating Histogram.

Unfortunately, in 1995 there were six unplanned loop outages. Main causes of these loop outages were as follows: turbogenerator, electric equipment, drain steam pipeline and small sodium leaks of steam generator valves after repair work. In Table 2 sodium leak statistics for a period of BN-600 operation and their classification are presented.

Operating experience with the core.

As known, after the 26-th run the BN-600 reactor core has been changed over to the third charge with a maximum fuel burn-up of 10 % h.a., the maximum dose being 75 dpa. The fuel subassembly duct material is ЭП-450 type ferritic - martensitic steel (13Cr-2Mo-Nb-P-B), the fuel pin cladding material is the cold worked austenitic steel (16Cr-15Ni-2Mo-2Mn-Ti-P-B).

In 1995 a decision was made to increase the time duration of the runs 32, 33 and 34 in order to confirm the performance of the core and the radial blanket up to burn-up of 11.3.% and 2 % h.a., respectively.

As a result the total lifetime of fuel subassemblies will be 540 eff.days.

In standard fuel subassemblies a maximum burn-up by the end of run 32 was 11.1.% h.a., a maximum dose was 80 dpa.

During the whole period of BN-600 operation in its core 12 S/As with pellet MOX fuel were irradiated. Till the end of run 32 one fuel subassembly with a maximum burn-up of 11.6 % h.a. was unloaded. At present 4 fuel subassemblies with pellet MOX fuel are under irradiation in the reactor.

Besides, in the BN-600 6 fuel subassemblies with vibratory compacted MOX fuel with a maximum burn-up of ~ 10 % h.a. and a dose of ~ 80 dpa were irradiated.

Table 1. Main BN-600 reactor characteristics for the last 5 years and from the start of operation.

Characteristic	Units	1991	1992	1993	1994	1995	From the start of operation up to Jan.1,1996
Electric power output	10^6 kWh	3670	4402	3915	3810	3695	56814
Load factor	%	69.83	84	80.3	78.2	70,31	69.3
The number of unit shut-downs		2	2	2	2	3	75
The number of loop outages		4	2	0	0	6	63

Table 2. Classification of sodium leak development causes.

Failure causes	Types of equipment			Total	Over circuit	
	equip.	valves	pipes	No/%	I circuit	II circuit
Manufactur-ing defects	2	1	3	6/20.7	2	4
Design faults	-	-	5	5/17.3	3	2
Repair faults	-	9	2	11/34.5	-	11
Operational causes	1	1	-	2/6.8	1	1
Ultimate lifetime	-	6	-	6/20.7	-	6
N0/%	3/10	17/57	10/33	30/100	6 /20	24 /80

2.2. THE BOR-60 EXPERIMENTAL NUCLEAR POWER PLANT.

In 1995 the BOR-60 plant was operated at 45 - 50 MW. The first loop of the reactor was provided with the reverse-type steam generator RSG1 and air exchanger, another one - with the reverse - type steam generator RSG - 2.

The results of plant operation in 1995 are presented in Table 3. In this table also are given the main results of BOR-60 operation from the commissioning up to 1996.

Table 3 . Results of BOR-60 plant operation .

Characteristic	1	2	3	4	1995	From commissioning up to 1996
Operation time, h	1960	1203	1747	1093	6003	
Availability factor	0. 9	0. 55	0. 79	0. 495	0. 685	
Maximum power level, MW (th)	50	50	45	45	50	
Power generation: thermal, Mwh electricity, Mwh	94505 16475	45988 8314	67119 13819	45794 7138	253406 45745	6129771 946901
Steam generators operation time, h RSG1 RSG2	 1890 1890	 1203 1203	 1665 1665	 1050 1050	 5808 5808	 75426 28540
Heat output for consumers, GKal	25830	5457	1952	9846	42835	189975

A large number of irradiation tests of fuel and structural materials of nuclear and thermonuclear reactors are carried out in the core and radial blanket and irradiation of electroceramics and electric insulation materials - in vertical channels.

The core charge mainly consisted of fuel subassemblies (S/A) containing ceramic (UO_2 , $UPuO_2$) fuel and metallic (UPu, UZr) fuel.

Most S / As with ceramic fuel had pins with vibratory compacted fuel.

One S / A with MOX vibratory compacted fuel reached a burn up of 29,2 % h.a. and fluence of $3.1 \cdot 10^{23}$ n/cm^2.

One S / A containing UZr sodium bonded fuel pins reached a burn-up of 9.7 % h.a.

2.3. THE BR-10 EXPERIMENTAL REACTOR EXPERIENCE.

In April 1995 40 years have passed since the day of the first in Russia experimental fast reactor (BR-2) commissioning. Its power was 100 kw and mercury was used as coolant. It was the predecessor of the present BR-10 reactor.

2.3.1. Main stages of reactor operation.

The BR-10 research reactor was commissioned in 1959. Until its updating in 1973, the reactor had been named as BR-5, it was the first in Europe sodium cooled fast reactor of 5000 kW power. In the secondary circuit of this reactor sodium-potassium alloy was used for coolant. Heat removal was performed by means of one air heat exchanger and one steam generator with double-walled tubes. The gap between the tube walls was filled with mercury. But soon after the steam generator was replaced by an air heat exchanger. Plutonium dioxide was used for the reactor first initial fuel charge. Reactor operation with plutonium oxide fuel was continued up to 1964.

In March 1965 new fuel-uranium monocarbide was loaded into the reactor. Tests of this fuel was continued until 1971.

From May 1971 to July 1973 redesign of the technological part of the reactor was carried out with the aim of increasing power up to 10 Mw. At reactor updating the centrifugal pumps the design of which was not quite successful were replaced by the electromagnetic ones. Besides, in the secondary circuit, two air heat exchangers were additionally introduced, and sodium-potassium coolant was replaced by sodium. Since then it was named BR-10. However, due to technological errors the reactor was operated at 7,5 Mw power.

During the third run from 1973 to 1979, plutonium oxide fuel was tested in the tractor.

reactor ?

From 1979 up to 1983 the second redesign, of the reactor was carried out. Its duration was determined by the complicacy of the problem - the replacement of the central part of the reactor including the safety element - a compensation cylinder. It was caused by that neutron fluence on the reactor vessel reached an ultimate value of $8*10^{22}$ n/cm^2. During this period vast experience on equipment decontamination and its disassembly was gained.

During the fourth run of the reactor (1983-1989) mass testing of core fuel elements with uranium mononitride fuel was carried out. The reactor was operated at a power of up to 8 Mw.

In 1989-90, updating of reactor systems was carried out aimed at increasing its safety and at improvement of the experimental base.

Since May 1990 up to now the reactor is operated with the fifth uranium mononitride fuel charge.

2.3.2. Experimental Capabilities of the Reactor.

The primary circuit has three experimental loops.

a) The central loop (CL)

The central loop channel is placed in the reactor centre and has a semiclosed cooling circuit with possible variation of sodium temperature at the channel inlet from 250°C to 350 °C. The direction of coolant movement in the central loop channel is from the bottom upwards. Sodium inlet into the central loop channel is made separately, sodium leaving the central loop channel is mixed with the main circuit sodium at he core outlet. There is also a possibility to arrange coolant movement in the central loop channel from the top downwards, in this case sodium heated up in the core is supplied into the experimental volume of the channel at a temperature of 470...500°C.

The procedure for a creep inner investigation developed at the Institute of Physics and Power Engineering allows to measure in the central loop channel the creep rate in the range of about 10^{-4} %/hr during no more than 100...200 hours to an accuracy of 15...20 %. During 3..6 months on one sample it is possible to obtain a data set sufficient for determining the character of the radiation creep rate dependence on stress, temperature, irradiation rate of material under investigation. Such tests are most attractive for a fast comparative estimate of radiation resistance of modified materials perspective for nuclear power.

b) Sodium spectrometric section

Sodium is sampled into it from the pump head of the first one primary loop. The pipe line of the sodium spectrometric section is cut in parallel to the sodium purification system.

c) Radioactive sodium loop and sampling system.

Sampling for them is made from the pump head of the second primary-circuit loop. Sodium is returned at the pump section of this loop.

Besides the central loop, instrumented devices can be placed in another two cells of the core.

In the nickel reflector of the reactor and in the graphite thermal column adjacent to it there are vertical and horizontal experimental channels and neutron beams at which various physical, biological and medical studies are carried out. Main characteristics of beams and channels are presented in Table 4.

Table 4 BR-10 reactor experimental channel characteristics.

№	Experimental channel name	Channel diameter, mm	Neutron flux, $n*cm^{-2}*s^{-1}$	Note
1.	2	3	4	5
		1. Vertical Channels		
1.1.	CLCh- central loop channel	25 20	$8.6*10^{14}$ $8.6*10^{14}$	Channel sodium "Dry" duct
1.2.	Core cells in the reactor centre	26	$8.4*10^{14}$	
1.3.	Core cells at the reactor periphery	26	$4.65*10^{14}$	
1.4.	M-O in nickel reflector	20	$0.40*10^{14}$	"Dry "duct Thermal neutron flux in the duct with water
1.5.	M-1,2 in nickel reflector	22	$1.6*10^{14}$	"Dry" duct
1.6.	OK-50 in nickel reflector	62 30	$6.0*10^{13}$ $0.5*10^{14}$	"Dry" duct Thermal neutron flux in duct with water
1.7.	OK-70 in nickel reflector	86 47	$6.9*10^{13}$ $0.8*10^{14}$	"Dry" duct Thermal neutron flux in duct with water
1.8.	K-1 in the thermal column	20	$3.0*10^{12}$	
1.9.	OK-3 in the thermal column	50	$1*10^{12}$	
1.10.	K-2 in the thermal column	20	$3.0*10^{11}$	
1.11.	AK in the thermal column	100	$8.0*10^{10}$	
1.12.	K-3 in the thermal column	20	$3.9*10^{10}$	
1.13.	K-4 in the thermal column	20	$1*10^{10}$	
1.14.	K-5 in the thermal column	20	$3.5*10^{9}$	
1.15.	K-6 in the thermal column	20	$1*10^{9}$	
1.16.	K-7 in the thermal column	20	$3.5*10^{8}$	

1.	2	3	4	5
	2. Horizontal channels			
2.1.	Channels (3) of the B-2 beam	70	$6.9*10^{13}$	
2.2.	Channels (3) of the B-2 Beam	50	$6.9*10^{13}$	
2.3.	The B-3 fast neutron beam	50	$3.0*10^{10}$	
2.4.	П-2 intermediate neutron beam	50	$1*10^{8}$	
2.5.	T-4 thermal neutron beam	50	$2.8*10^{8}$	
2.6	Channel in the thermal column	465	$1*10^{12}$	

The M-1, M-2 vertical channels are located in the movable nickel reflector. The OK-50, OK-70 and M-0 vertical channels are located in the fixed nickel reflector. Channels in the reflector have air-cooled steel jackets. The intra-channel test devices in the OK-50 and OK-70 channels developed at IPPE that are used for creep investigations ensure single-axial stressed state of material (extension or shear) permit to monitor and maintain the irradiation temperature at a preset level within 350...700°C, assure testing in the inert atmosphere. A version of remote recharging of samples is also possible.

In the thermal column 9 vertical and 4 horizontal channels are located. All vertical channels in the thermal column have no jackets.

In graphite of the thermal column a horizontal recess of 460 mm in diameter has been drilled in which an irradiation device with uranium target is expected to be positioned for the purpose of irradiation of lavsan film to obtain nuclear filters with a pore density of up to 10^9 pores/cm^2. At present start-up and adjustment activities at this unit are under way.

The Б-3, П-2 and T-4 neutron beams are covered with cast-iron gates 1000 mm thick. Into the Б-2 horizontal channel there is fitted a plug with six holes into which leak-tight jackets are placed. At the reactor neutron beams it is possible to carry out neutron-diffractometry of spent fuel rods, as well as to perform other work using various-energy neutrons.

To the Б-3 beam a medical centre is adjacent at which oncologic patients are treated and medical and biological investigations are carried out. Compex

equipment includes systems for neutron beam collimation, light guidance of the beam at a target, for TV observation and radiation monitoring.

The OK-70 and OK-50 vertical channels of the reactor are planned to be used for regular production of molybdenum-99, iodine-125, iodine-131, iodine-125, xenon-133 radionuclides used for diagnostic medical and treatment purposes.

It has been experimentally shown that at filling the OK-70, OK-50 channels with water there can be obtained neutron irradiation devices with the moderated neutron flux density of up to $8*10^{13}$ neutrons/cm^{2}sec.

An irradiation device with water mounted in the M10 channel allows to produce Au-198 and Hg-197 isotopes with a specific activity of 3300 Ci/g and 50 Ci/g, respectively.

In the reactor core regular irradiation of sulphur-32, sulphur-33 isotopes is carried out and phosphor-32, phosphor-33 isotopes with specific activity up to Ci/g are produced.

The core channels can be used for commercial production of phosphor-32,-33, niobium-93, tin-118,-119,-120, scandium-46, sulphur-35, manganese-54, iron-54, iron-55, cobalt-58 and other radionuclides.

2.3.3. Main lessons from reactor operation.

Sodium Leaks.

During the whole time of BR-10 reactor operation at the I and II circuits 19 incidents occurred related to losses of tightness of equipment, pipes, valves and level indicator sensors resulting in liquid metal coolant leaks of different volume. The statistics of coolant leak sites is presented in the Table 5.

Table 5. Statistics of coolant leaks.

N	Coolant leak location	The number of leak incidents	
		I circuit	II circuit
1.	Pipe lines	3	-
2.	Equipment	-	3
3.	Valves	4	3
4.	Level indicator sensors	3	3

Causes of the above elements loss of tightness are classified as follows:

- the development of hidden defects in material and welding - 12 cases;
- design faults - 3 cases;
- faults at assembling - 1 case;
-misoperation (violations of piping heating up sequence by operating personnel) -2 cases;
-burning-through of piping because of electric heater short circuit - 1 case.

The most serious by its consequences sodium leak occurred in 1959. At BR-10 reactor energy start-up, burning-through by short circuit current from the electric heater of the tube connecting the gas space of the central part [redacted] of the reactor with the level indicator tube [redacted]. Loss of tightness of tube resulted in gas (argon) release from the reactor central part space, in a rise of sodium level up to a hole formed in the tube and sodium pouring out outside the reactor vessel. The reactor was shut down by the scram system and sodium circulation through the primary circuit was stopped at once. To eliminate the consequences of this accident it was necessary to carry out disassembling of the reactor central part and shield, their washing from sodium oxides and reassembling. To prevent possible burning-through of heated pipes a circuit of electric supply for heaters from power network with isolated neutral conductor was mounted.

In 60-ties coolant microleaks were noted at places of coolant tubes-to-tubeplate attachment at air heat exchangers. Microleaks showed up in form of sodium-potassium alloy oxides build-up on tubeplate. However these leaks received no development, i.e., self-healing of microcracks took place. Since that time up to now no coolant leaks in air heat exchangers occurred.

BR-10 operation experience and special experiments have revealed an importance of complying with specified operations on heating up pipes and

equipment after sodium freezing, as sodium at its melting down increases its volume by ~4%.

During the years of reactor operation the following sodium leak monitoring systems were introduced and tested:

- short circuit of spark-plugs located within leak jackets gaps;
- radioactivity measurement;
- short circuiting of electric heaters to earth.

Radiometrical sensors are locateds:

a) near secondary popes for integrity monitoring of tubes and tubeplates of intermediate heat exchangers;

b) at air lines of I circuit rooms special ventilation.

Operating experience has revealed that the primary sodium leak monitoring system over Na-24 radioactivity should record sodium leakages of ~10 g. In this case sensitivity of the system is ~10^{-5} Ci/m^3 that allows to record an increase of sodium aerosol content in ventilation air lines of up to 2 mg/m^3.

As was indicated above, for the heaters of all pipe lines and equipment of the BR-10 reactor primary and secondary circuits a three-phase. 380 V isolated neutral electric supply system was created.

The isolated neutral electric supply system permits to implement continuous monitoring of heater insulation conditions that was used for creation of the system for monitoring coolant leak initiation at those sections of equipment and pipe lines where the heaters are made of nichrome (Ni-Cr alloy) and insulated from the pipes by porcelain beads. All the pipes and small-sized equipment of reactor sodium circuits are coated with such heaters, they are wound upon the popes sufficiently closely (with a pitch of ~20 mm), and at leak initiation the liquid metal damages insulation at once. The heater insulation monitoring system was supplemented by a failed-heaters automatic detection circuit . During 1 minute a group is detected in which a heater with a loss of insulation is located, and the detected group of heaters is cut off from the electric supply network. The above systems have been introduced now at all sodium rigs and reactor sodium circuits.

Operating experience of a steam generator with double-walled tubes.

A steam generator unit at one of secondary-circuit loops with a pilot once-through steam generator was in operation at the reactor from June 1959 up

to October 1960. The steam generator was made of two series-connected sections.

Each section is a shell-and-tube heat exchanger with 61 U-shaped steam-water tubes of 16*1,5 mm in diameter enclosed on straight sections into up upsetting tubes of 21*1,5 mm in diameter to prevent direct steam-water-to-sodium potassium contact at a loss of tube bundle tightness. The intertube space was filled with mercury. The steam generator, as well as all liquid-metal equipment of the reactor circuits were made of 1X18H9T stainless steel. This steam generator had been in operation at the plant for a total of 4200 hours. After 5-months operation at a full load the steam-water tube leaks in the steam generator evaporation section appeared. During August-September 1960 8 failed tubes were detected and plugged. An investigation of the steam generator tube bundles has revealed that the fault in the steam-water tubes represent longitudinal cracks 10 to 70 mm long located in the evaporation section (in the zone of evaporation termination and of superheating initiation) and are related to stainless steel intergranular corrosion under conditions of cyclic thermal changes and variable stresses in the downflow evaporator tubes.

In October 1960 a loss of tightness of the mercury-sodium-potassium tube bundle occurred. Mercury-into-sodium-potassium alloy leakage was related with a loss of tightness by one of casing tubes at place of its attachment to the tubeplate due to a fabrication defect at steam generator manufacture. After the above accident the steam generator was disassembled and replaced by an air heat exchanger. From operating experience of the steam generator with mercury interlayer between water and sodium-potassium alloy a conclusion can be drawn that a tendency to create of a universally safe steam generator (safe in all respects) results in a sophistication of its design and does not always increase its reliability.

Experience of Operation with different Fuel Types.

1959-1964. The First Reactor Charge.

The core contained plutonium oxide fuel pins. In early 1961 when plutonium burn-up exceeded 3% failed pins started to appear in the core. And by the 1-st of September 1961 when burn-up reached 5%, Cs^{137} activity in sodium increased by 200 times. In gas and sodium some other fission products

were also found. From September 1961 to March 1962 the reactor was shut down for total discharging of the core.

At the examination of discharged fuel subassemblies there was found out that 22% of fuel subassemblies had untaught fuel pins. Defects in form of cracks along the whole fuel pin length were revealed. One of the causes explaining it was the absence in the fuel pin design of a plenum for gaseous fission products collection. In 1962-1964 the reactor was operated at reduced power levels because of that some of the reloaded fuel subassemblies continued to lose their tightness. At this stage the release of solid and gaseous fission products from untight fuel pins into coolant and gas plenums was studied as a function of temperature and reactor power.

In November 1961 when a maximum plutonium burn-up reached 6.5% the reactor was shut down for passing over to a new fuel type.

1965-1971. The Second Reactor Charge.

Uranium monocarbide was chosen for fuel. Such a choice was caused by that due to its higher density and teat conductivity than those of oxides the monocarbide allows to improve physical characteristics of the reactor. New fuel pins were designed for higher burn-ups, and for gaseous fission products collection gas plenums were provided in them.

Maximum burn-up reached in uranium monocarbide fuel pons was 6.1%t.a. In this case in 0X18H9T steel cladding some insignificant defects in form of small cracks were observed, and in X16H15M3Б steel-clad fuel pins no defects were practically present. The presence of defects in fuel pins was mainly determined by a large circumferential temperature nonuniformity. It has been revealed that uranium monocarbide as compared to oxide fuel is less workable, it calls for total insulation from oxygen in the presence of which a severe uranium monocarbide-to-fuel pin cladding interaction goes on. Besides, significant pyrophority of this fuel was revealed.

1973-1979. The Third Reactor Charge.

The tested fuel was plutonium oxide with perspective material for pin cladding - X15H15M3Б stainless steel. In the third charge fuel pins all design drawbacks (disadvantages?) of the first two charges were taken into account. Maximum burn-up was 14.2% h.a.

1983-1989. The Fourth Reactor Charge.

After reactor updating (1979-1983) fuel pins of uranium mononitride were loaded into the reactor for testing. A maximum uranium burn-up of 9% h.a. was attained. This fuel type manifested its high workability, its that allowed to carry out the tests of single uranium mononitride pettets even in an open form without cladding. During this period in the core an extensive investigation of fuel pins containing various fuels and having different artificial defects in their cladding was carried out. Reactor operation with damaged fuel pins allowed to study kinetics of fission products release into coolant-gas plenum, their propagation in the circuit. Results of experimental investigations formed the basis for the development of fission products mass transfer mathematical models. At the reactor various methods for fuel pin condition were tested.

Experiments at the BR-10 reactor have shown that cold traps designed for sodium purification from oxides catch cesium and iodine efficiently enough.

1990-1995. The Fifths Reactor Charge.

Uranium mononitride is used for fuel. Burn-up has following problems:

- an investigation of fuel pin test samples;
- nuclear-physical, medical-biological and material research;
- the development of new technologies (manufacturing of nuclear
- a study of membranes, production of radioisotopes for medical and industrial uses);the mechanisms of fission products release from fuel pins and of their distribution in sodium circuits;
- the development of methods and procedures for coolant purification from fission fragments and corrosion products;
- carrying out of a broad range of physical studies using various energy neutron beams;
- treatment of oncological patients with the use of tumour irradiation by Б-3 beam fast neutrons.

3. R&D PROGRAM.

3.1. PHYSICAL RESEARCH AT CRITICAL FACILITIES.

3.1.1. The BFS critical facilities .

At the BN-800 type reactor model (the BFS-58 critical assembly) the core was modified (BFS-58-4). Reactivity-compensating boron rods were inserted into the core to a position close to the state at the beginning of the run, the reactivity loss being compensated by additional loading of 261 fuel rods at the core periphery.

The first cycle of the control rods efficiency measurement has been carried out, processing and analysis of the experimental data obtained are under way.

In accordance with the Program of Investigations, in 1996 it is envisaged to complete the studies at the BFS-58-4 critical assembly which include:

- the completion of control rod efficiency measurements;
- the measurement of radial and height distributions of plutonium-239, uranium-235, 238 fission reaction rates determining the heat release field in the core;
- measurements of the sodium void effect of reactivity (SVER) in the sodium volume and the core, for a sector comprising all three radial fuel zones.

There has been made processing of experimental data obtained at the BFS-58-1U1, critical assembly with inert-diluent and weapon-grade plutonium-based fuel in the central zone of 40 cm radius sufficient to preclude the influence of the driver zone upon the central characteristics. Additional calculations were made for making corrections to the parameters depending on the low energy neutron spectrum part the contribution of which for such type fuel was significant. Further work on evaluation of the accuracy of experimental data corrections determination is required.

A model of a critical assembly with the central zone having a composition of the plutonium utilizer-reactor with 39% of weapon-grade plutonium in oxide fuel has been developed. The outside driver zone contains uranium oxide fuel. The core is assembled at the BFS-1 critical facility and the following measurements in the core center have been performed:

- the sodium void effect of reactivity;

- of boron control rods efficiency;

- of average fission and capture cross-sections ratios by several detector types;

- of central reactivity coefficients of reactor material samples;

- of Doppler effect for fuel samples.

The measurements of fission distributions with height for main fissionable isotopes have been carried out as well.

A preliminary analysis of the results obtained has confirmed the expected changes in the measured characteristics related to a neutron spectrum variation. Quantitative estimates of the observed changes will be made after a careful calculation analysis.

Processing and initial calculation analysis of the experimental data obtained on a set of the BFS-67 critical assemblies have been completed.

The experiments and their analysis were carried out in cooperation with specialists from the Nuclear Center in Cadarache (France). The central part, about 300 l in volume, of a model simple in composition and configuration contained mixed fuel similar to the composition of the Superphenix reactor and surrounded by the driver zone with uranium oxide fuel (BFS-67-1).

In the core center there were measured:

- SVER (in a volume of about 30 l) including the distribution with height in a single fuel channel;

- natural and enriched boron carbide based control rod efficiency;

- the ration of average fission and capture cross-sections including minor actinides;

- reactivity coefficients of a set of samples including neptunium and americium-241.

The same functionals were measured in the BFS-67-2, 3B critical assemblies in which 13% and 6.5 % of uranium-238 nuclei in the central part of the plutonium subzone were substituted by neptunium nuclei, respectively. A maximum amount of the neptunium dioxide introduced was 10 kg.

The changes in fuel neutron properties at neptunium introduction (neutron spectrum hardening) result in a marked (by about 3 times) increase of the SVER positive component in the center of the core, in a decrease by 5 - 10 %

of the boron rods efficiency, in changing of the hydrogen reactivity sign from a positive to a negative one.

The results of measurements of spectral indices and reactivity coefficients confirm a marked change (hardening) of the neutron spectrum.

Preliminary calculations fulfilled in IPPE and in Cadarache have mainly revealed a satisfactory agreement of the calculated values with the corresponding experimental ones.

The difference of 5 - 10 % is observed for plutonium-238 and americium-241, 243 spectral indices and of about 20 % - for curium-244. More detailed analysis is supposed to be performed in the first half of 1996 and to present the results at the International Conference on Reactor Physics (Physor 96) in Japan.

In connection with the continuation of investigations at the BN-800 reactor model, work on studying of characteristics of the BN-Type reactor model with non-breeding blankets will be started in the 4-th quarter of 1996.

3.1.2 .The COBRA critical facility.

The CBR- 22 critical assembly is arranged of the uranium - thorium core and depleted uranium oxide blanket, the core contains 90 % - enrichment metallic uranium pellets 5,6 mm thick and thorium pellets 10 and 40 mm thick. The core diameter is 70 cm, the height is 60,5 cm. High enrichment of fuel in the core (~25%) results in a very hard neutron spectrum, the fraction of neutrons with energies below 10 kev not exceeding 0,5 %.

In the first quarter of 1996 all scheduled experiments on this assembly were completed, summarising main results of the measurements let us note as follows:

1. The critical core charge making up 250 fuel rods was determined. Calculations carried out by the FFCP, CRAB and MMK codes with the BNAB - 90 constants overestimate keff by about 1 %.

2. U - 238 and Th-232 relative cross-sections in the core centre were measured. Calculated U-238 cross-section values agrees with the measured ones within the experimental error, whereas for Th-232 the calculation was about 12% less than the experiment.

3. By means of the activation method the relative Np-237 capture cross-section was measured which proved to be ~ 8% greater than the corresponding calculated value.

4. A large set of track detectors was used for transuranium nuclides fission cross-section measurements.

U-233, U-236, U-238, Th-232, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-242, Cm-244 fission cross- sections were measured. Now calculation studies of cross-section values for these nuclides using group constants from various libraries (BNAB-90, ENDF/B-VI, etc.) are under way.

3.2. PLUTONIUM AND MINOR ACTINIDES UTILIZATION.

In 1995 studies of various plutonium utilization scenarios were detailed and developed.

A nuclear center of the " Mayak" PO Type with several BN-800 reactors and required fuel-cycle plants would allow to utilize the accumulated and replenished plutonium inventory of both grades and to convert plutonium into a so-called "spent fuel standard". Reprocessing of this fuel will be postponed until the appearance of the necessity in uranium economy .

An increase of the spent fuel protection can be achieved by means of addition of neptunium at fuel pin fabrication.

In 1995 it proved to be possible to join the efforts of design institutes to provide economical estimates of the plutonium utilization schemes and moreover, to develop the program of joint activities on this problem with an American group of specialists.

Within the frames of work on plutonium utilization some features of using thorium as a diluent of plutonium and of plutonium-free cycles as well have been considered:

- uranium oxide in the matrix of thorium metal in the VVER-1000 reactor;
- enriched uranium oxide-thorium oxide in RBMK;
- uranium-233 oxide-thorium oxide in the NP-500 reactor;
- plutonium oxide in thorium matrix for BN-800-plutonium burner.

Reactor physics at the presence of thorium in the cores has some advantages due to formation of uranium-233 favorably affecting the safety

characteristics. The results of measurements on the thorium-bearing assembly have been obtained.

The actinide balance in the reactor and fuel-cycle systems was studied simultaneously with the plutonium balance. Fuel recycle in the fast reactor systems allows to reduce substantially the amount and radiotoxicity of actinides under condition of achieving low non-recoverable losses at each fuel reprocessing cycle.

Recycling of actinides and plutonium in fast reactors can be carried out using technologies of incomplete separation from fission products. As a separation criterion an activity or radiotoxity equal to those of actinides can serve. So, e.g., plutonium should not be obligatory carefully cleaned from neptunium and americium-from lanthanide's.

The latter problem has been considered as applied to the BN-800 reactor under the assumption that lanthanides and americium-curium do not separate at all. Even under such conditions, according to reactor characteristics a uniform addition in fuel in amount of a few percent of the plutonium amount is possible. The real possibility will be determined by radiation conditions at manufacturing.

3.3. RADIOLOGY.

A very important factor of carrying out investigations aimed at further increase of BN-type reactors radiological and ecological safety is taking into account of long-lived radionuclides formation not only upon the main structural material elements but upon insignificant impurities as well. Within the frames of the subject for the first time there was carried out a wide-scale analysis of possible contribution to the activation from all stable nuclei-target which could be present in the composition of BN-reactors materials and of promising materials: EP-450, EI-847, 18Cr10NiTi steels and others; in low-activated chromium- manganese steel; in boron carbide; in titanium, tantalum, tungsten and their compounds, etc. The calculations have been performed on the basis of the activation constants section of the BNAB-93 libraries and of the well-known foreign FISPACT code.

As an example, the table 6 presents the values of specific long-lived activity for sodium coolant. From the table it follows that the long-lived activity of sodium coolant will be determined not by activation of the main element but

by activation of its impurities, with the main contribution to activity being given by various impurities at various storage limes. The consideration performed is extremely important at the development of BN-reactor decommissioning projects and at choosing the strategy of waste management.

Table6. The value of specific activity of long-lived radionuclides formed from impurities in primary sodium during 50 years of operation as a function of storage time, Bq/kg

№	In-sodium impuri-tes	Content, mln.$^{-1}$	Formed radio-nuclide, ($T_{1/2}$, years)	Storage time, years					
				0	5	50	150	10000	100000
1	Na	$1,0 \cdot 10^6$	Na-22	$4,6 \cdot 10^7$	$1,2 \cdot 10^7$	$7,1 \cdot 10^1$			
2	C	2-13,8 (14)*	C-14 (5730)	$2,8 \cdot 10^4$	$2,8 \cdot 10^4$	$2,8 \cdot 10^4$	$2,8 \cdot 10^4$	$5,0 \cdot 10^3$	
			Be-10 ($1,6 \cdot 10^6$)	$2,8 \cdot 10^2$	$2,8 \cdot 10^2$	$2,8 \cdot 10^2$	$2,8 \cdot 10^2$	$2,8 \cdot 10^2$	
3	N	1.-8.1	C-14	$1,5 \cdot 10^6$	$1,5 \cdot 10^6$	$1,5 \cdot 10^6$	$1,5 \cdot 10^6$	$4,3 \cdot 10^5$	
4	O	7-57 (50)*	C-14 (5730)	$6,2 \cdot 10^3$	$6,2 \cdot 10^3$	$6,2 \cdot 10^3$	$6,2 \cdot 10^3$	$1,9 \cdot 10^3$	
5	Cl	1-130 (100)*	Cl-36 ($3,0 \cdot 10^5$)	$1,8 \cdot 10^5$	$1,8 \cdot 10^5$	$1,8 \cdot 10^5$	$1,8 \cdot 10^5$	$1,8 \cdot 10^5$	
6	K	600-870 (800)*	Ar-39 (269)	$1,0 \cdot 10^7$	$1,0 \cdot 10^7$	$9,0 \cdot 10^6$	$7,0 \cdot 10^6$		
			Cl-36 ($3,0 \cdot 10^5$)	$1,0 \cdot 10^3$	$1,0 \cdot 10^3$	$1,0 \cdot 10^3$	$1,0 \cdot 10^3$	$1,0 \cdot 10^3$	$8,2 \cdot 10^2$
7	Ca	1-12 (10)*	Ca-41 ($1.0 \cdot 10^5$)	$2.3 \cdot 10^3$	$2.3 \cdot 10^3$	$2.3 \cdot 10^3$	$2.3 \cdot 10^3$	$2.3 \cdot 10^3$	$1.2 \cdot 10^2$

* - The value used in calculations.

3.4. RADIOACTIVE ISOTOPES PRODUCTION IN FAST REACTORS.

The production of radioactive isotopes can be considered as one of promising tasks for fast power reactors.

Depending on the radionuclide kind and requirements for the product its production can done by various methods:

- by means of complex reprocessing of irradiated fuel and transactinides irradiation;

- due to the use of threshold reactions followed by chemical reprocessing of the target that allows to obtain isotopes with high specific activity without carrier;

- by means of targets irradiation in special individual irradiation devices with moderator;

- by means of creation of special zones for isotope production , and in the limit - of the moderating screen.

The value of the achievable specific activity of nuclides is rather high: e.g., for cobalt -60 it is over 100 Ci/g, for iridium - 192-700 Ci/g. By this parameter the fast reactors as a whole surpass the possibilities of thermal research reactors with a flux of several units by 10^{14} n/cm^{2}s and, all the more, of VVER-and RBMK - type power reactors.

Estimates of total production are also very considerable: during a year at a large power reactor there can be produced up to 10 MCi of cobalt - 60 or up to-1-2 kCi of carbon - 14.

The restraining factors are reactor neutron field disturbances and the reduction of reactivity margin at using moderator.

The experimental results as a whole confirm the parameters of isotope production in individual irradiation devices and point to safety of their irradiation in the radial blanket as well.

In the experiment at the BN-350 reactor cobalt-60 with a specific activity of about 100 Ci/g was obtained. Experiments at the COBRA critical assembly with irradiation device mock-ups have shown that at using special absorbing elements the disturbances in the reactor neutron field will be small. However, it results in a significant reactivity loss. At present one of the stages of work has been completed on the verification of characteristics of absorber-free irradiation devices surrounded by buffer subassemblies without fuel pins and strong absorbers. The use of such irradiation devices results in a considerably lower loss of reactivity at an acceptable effect upon neutron field.

3.5. MOLTEN FUEI- COOLANT INTERACTION INVESTIGATIONS.

7 experimental runs with pins filled with Fe_2O_3 +Zr thermite charges have been carried out: 2 "dry" (sodium- free) ones, 3 single under-sodium ones and 2 with 7 fuel pin bundles under sodium. The main results are as follows:

- At a Na temperature of 550°C only 1 pin (in a single version) with a charge of 38 g has failed.

- All pins were damaged on two sides. Averaged specific area of pin wall failures was 14 and 19% (total charges of 165 and 488g), a maximum failure of the central pin was 61 and 64%.

- Sodium heating up after the interaction corresponds to the energy stored in the charge.

- The values of acoustic energy emission in the pin bundle experiments registered by the acoustic sensor are proportional to the total charge of the sides.

- Pressure pulses registered by the sensors positioned in gas and sodium do not exceed some atmosphere units.

- The total masses of fragments collected after pin bundle experiment (107 and 298g) are proportional to the charges.

- The screen and microscopic analysis of fragments 0-0,16 mm to >1,0 mm

has shown that normal-logarithmic size distribution of particles takes place, with the following parameters:

median size- 0,24 mm,

standard deviation -3,5,

maximum fragment sizes -6,5-8 mm.

3.6. SODIUM FIRE INVESTIGATIONS.

The experiments simulating instantaneous rupture of sodium pipeline were carried out at a rig of 8 m^3 volume, under the following conditions:

- sodium mass -12 kg;
- temperature of sodium - 500°C;
- sodium jet velocity -17 m/c;
- pipe diameter - 15 mm.

Pressure within the vessel at outflow reached 0,16 MPa, a maximum temperature was ~ 700°C in the upper part of the vessel (outflow was made from below upwards), visual inspection of the inner part of the vessel gave an estimate of the burnt sodium of ~5-10%.

3.7. EXPERIMENTAL STUDIES OF COOLANT BOILING.

A possibility of durable (for more than 1 hour) stable heat removal from the pin bundle due to coolant boiling under natural convection conditions at a heat flow density up to 200 kW/m^2 has been experimentally demonstrated.

Recent experiments on boiling-up of sodium-potassium coolant in a seven-pin bundle characterize the following main points.

Two types of boiling conditions were revealed. In the first type of boiling a stable flow rate of coolant at the working section inlet was observed, some other parameters of the process had also a stable character. In the experiments conducted these conditions were available at a heat flux $q < 150$ kW/m^2.

At a higher heat flux ($q > 150$ kW/m^2) pulsed conditions of liquid metal coolant boiling arose characterized by significant oscillations of flow rate and pin cladding temperatures.

Depending on conditions determining the presence of the coolant film upon the pins surface in the process of formation of large vapor volumes in the pin bundle (among these conditions there should be especially noted the conditions of the pins surface, its roughness) the pulsed boiling conditions could develop in two directions.

At the presence of such film on the pin surface the pin cladding temperature did not increase higher than the saturation temperature; in other case, for the pulsed boiling conditions, the jumps of the pin temperature relative to the saturation temperature are characteristic that increase with the thermal load.

At a heat flux $q \sim 270$ kW/m^2 and a gradual decrease of coolant flow rate a pressure relief in the boiling area from $p1 = 0.06$ MPa to $p2 = 0.05$ MPa was accomplished. In did not result in fast boiling up of coolant within the whole volume of the working section. However, 200s later the first pressure jump with an amplitude $\Delta p = 0.065$ Mpa followed, and some part of coolant was ejected from the working section. Therewith circulation ceased , evaporation of coolant film from pin surface was noted. In 140 s, after the first pressure jump a sharp jump of pin temperature was observed accompanied by melting down of claddings of the three most looded of them on a section of 50 mm long at the end of the heated section. The first pressure jump was followed by the second

one with an amplitude $\Delta p = 0.132$ Mpa. After switching off, the heaters power supply the working section was again filled with coolant.

At analysis of experimental data for boiling has revealed a number of important factors which should be taken into account when carrying out rig investigations at a pin bundle:

- complexity and ambiguity of the boiling process under coolant natural circulation conditions in the liquid-metal circuit;
- a possibility of stabilization, suppression or accidental development of pulsed boiling conditions;
- a danger of pressure reduction in the circulation circuit;
- the shortage of information on features and parameters of some conditions, on the influence of different factors upon the boiling process.

3.8. FAST REACTOR ECOMONIC.

Work on evaluation of comparative technical and economic characteristics of BN-800 as compared to VVER-640 (NP - 500) is carried out jointly by IPPE-S-Pt. AEP. The main aim at the work is to reveal the effect of the technological process features in fast and thermal reactors on NPP economic characteristics.

To exclude the effect of the site region factor upon the NPP capital investments the costs for Kola NPP-2 have been converted to Southural NPP site conditions (Chelyabinsk region). For this purpose a basis-index method of costs scaling was used . Besides, the expenses specific for both sites were excluded wherever possible. In Table 7 the comparable capital investments into NPPs construction with BN-800 and VVER-640 for Chelyabinsk region are presented.

In addition to the cost characteristics there has been analyzed structure metal demand for main components of the reactors compared. Construction sheets of the above NPPs were taken as the basis. Structural metal demand of the BN-800 and VVER-640 reactor and turbine sections are presented in Table 8.

Table 7. Comparable capital investments into NPP construction with the exception of cost investments specific for the sites.

Nuclear Power Plant	S-U.NPP: 3 x BN-800	K NPP - 2 : 3 x VVER-640
Capital investments into industrial construction, Mln, roubles 1991	4154	3357
Specific capital investments, rouble 1991/kW	1728	1734
including:		
construction and assembly work	635	484
equipment	684	967
other expenses	409	283

Table 8. Specific characteristics of the reactor plants and turbine halls for the BN-800 and VVER-640 NPPs.

Reactor section, t/MW	BN-800	VVER-640
Heat-mechanical and technological equipment	12.1	12.2
Pipe lines	2.2	2.2
Electric engineering equipment and instrumentation	4.1	0.5
Turbine section (machine hall), t/MW	BN-800	VVER-640
Heat-mechanical and technological equipment	8.9	7.3
Pipe lines	6.5	1.6
Electrotechnical equipment and instrumentation	2.7	1.6
Specific labour consumption for main construction objects, man-day/kW	5.8	2.9

Research Activities on Fast Reactors in Switzerland

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1 Future Electricity Supply Mix Options

The current domestic Swiss electricity supply is primarily based on hydro power (approximately 61%) and nuclear power (about 37%). The contribution of fossil systems is, consequently, minimal (the remaining 2%). In addition, long-term (but limited in time) contracts exist, securing imports of electricity of nuclear origin from France. The total yearly net electricity generation by domestic power plants in the hydrological year 1993/1994 was 62.9 TWh; the domestic demand was 51.0 TWh (including 7% losses). During the last two years, the electricity consumption has been almost stagnant, although the 80s recorded an average annual increase rate of 2.7%.

The future development of the electricity demand is a complex function of several factors with possibly competing effects, like increased efficiency of applications, changes in the industrial structure of the country, increase of population, further automation of industrial processes and services. Under the basic assumption of economic growth, two possible electricity demand level cases were postulated by VSE (Association of the Swiss Utilities): a high-growth demand case corresponding to a yearly increase of 2% from year 1995 to year 2010 and 1% from year 2010 to year 2030, and a low-growth demand case corresponding to a yearly increase of 1% from year 1995 to year 2010 and 0.5% from year 2010 to year 2030.

The electricity generating systems whose capacity, according to VSE, is largely secured are herewith referred to as the "base" supply and is predominantly hydropower.

Due to decommissioning of the currently operating nuclear power plants and expiration of long-term electricity import contracts there will eventually open a gap between the postulated electricity demand and the base supply. The assumed projected demand cases, high and low, as well as the secured yearly electric energy supply are shown in Figure 1 for the period of interest.

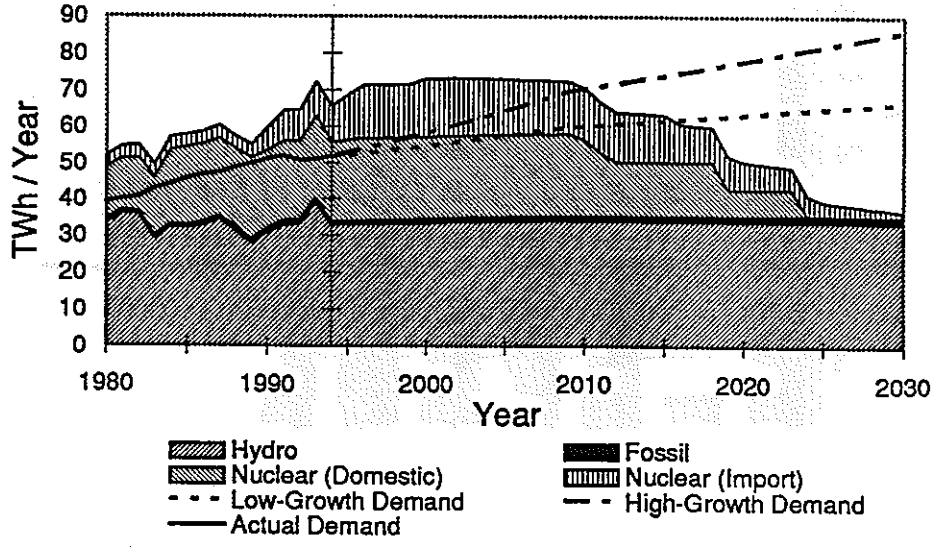

Fig. 1: Electric energy demand cases considered by VSE and the secured yearly electricity supply for the period 1995-2030.

Supply mix options		System shares in the cold season to cover the gap of electric energy
1	Conventional thermal	50% CC(gas); 25% CC(oil); 25% Coal
2	Nuclear	100% Nuclear domestic
3	Mix Nuclear/Gas	Today's installed nuclear capacity; rest of supply CC(gas)
4	Import 100%	10% CC(gas); 30% Coal; 60% Nuclear
5	Import 50%	Import: 10% Coal; 40% Nuclear. Domestic: 25% CC(gas); 25% Nuclear.
6	Conventional thermal + PV	as Option 1 with PV substituting 5%
7	Mix Nuclear/Gas + PV	as Option 3 with PV substituting 5%

Table 1: Definition of the supply mix options assumed by VSE to cover the future electricity gap.

For these 7 supply mix options and for the base supply life cycle analysis (LCA) to evaluate the total amounts of greenhouse gases (GHG) were performed (Fig. 2)

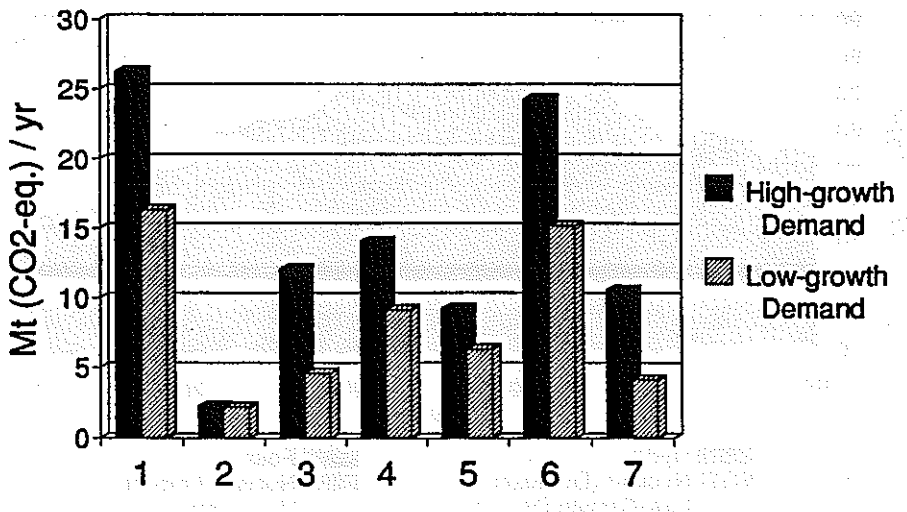

Fig. 2: Greenhouse gas emissions for the 7 supply mix options of the high- and low-growth demand cases in year 2030.

In Figure 3 the total GHG-emissions in Switzerland are compared with those due to the electricity production. The GHG emissions are naturally only one criteria among several others to decide on the future supply mix in Switzerland. A complete documentation of this subject is given in Ref. [1].

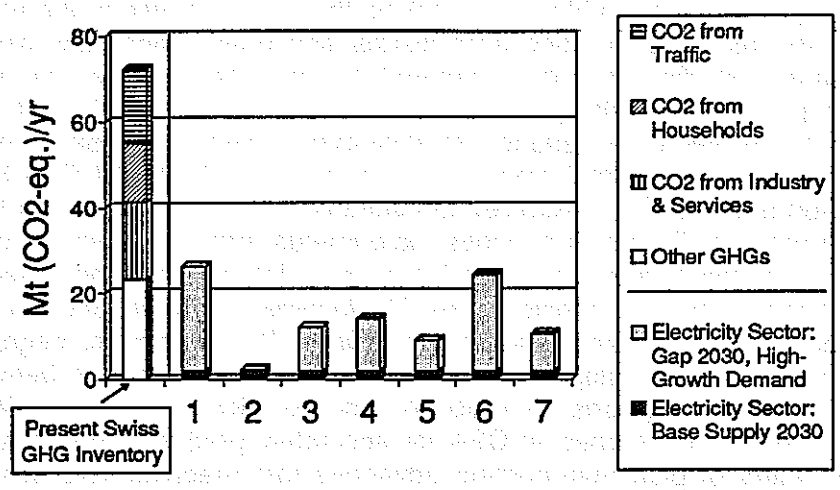

Fig. 3: Comparison of the present Swiss GHG emission inventory with LCA results for high-growth demand case options for year 2030.

2 Physics Aspects of Pu-Burning Fast Reactor Configurations

Introduction

Several countries worldwide have decided to take effective steps for managing the long term growth of their plutonium stockpiles, thus making the maximum use of the energy source present in plutonium and reducing the long term toxicity and proliferation risks associated with stored stockpiles. Various studies have demonstrated that fast reactor cores can be designed to burn the degraded plutonium resulting from LWR multi-recycle; nevertheless, the required modifications of conventional fast cores tend to alter their overall physics and safety behaviour. Within this framework, a major experimental program is undertaken at the MASURCA facility at CEA Cadarache in support of the CAPRA (Accrued Consumption of Plutonium in Fast Reactors) project [2]. The CIRANO program [3] which provides experimental data on mock-up configurations of Pu-burning cores, is primarily aimed at validating nuclear data and methods used to model the novel features of plutonium burning reactor cores: Replacement of radial and axial blankets by reflectors consisting of stainless steel and sodium, higher Pu-enrichment, degraded Pu-vector, and more heterogeneous core design. The ongoing collaborative program between CEA in France and PSI in Switzerland supports the validation and qualification of the ECCO/ERANOS deterministic code system [4] developed at CEA for accurately predicting operating and safety-related physics parameters of plutonium-burning advanced fast reactors. This scheme of calculations, which is based on the JEF-2 evaluation, is compared with other deterministic code systems, such as the PSI system MICROX/TWODANT, and with the continuous energy Monte Carlo reference method, using additional data evaluations. The ECCO/ERANOS predictions are also compared with measured values, provided by the CIRANO program, of reactivity, central spectral indices, sodium void worths, reaction rate traverses, as well as neutron and photon heat depositions along the radial and axial directions.

System Description and Results

Neutronic analyses, including coupled neutron/photon calculations and sensitivity studies to methods and data, are performed for the first two cores in the CIRANO experimental program, namely ZONA2-A and ZONA2-A3 [5]. The ZONA2-A core is the first ever totally plutonium-fueled core built in MASURCA. It contains UO_2PuO_2 rodlets with about 25 % Pu-enrichment, and Na rodlets. The core is axially and radially surrounded by a breeding region (blanket) consisting of natural UO_2 and Na rodlets, itself surrounded on all sides by a stainless steel shield region. The ZONA2-A core as such constitutes the starting reference point for the changes outlined above. These changes have begun with the replacement of the radial and axial fertile blankets surrounding the core by sodium/steel reflector assemblies.

The second configuration analyzed, ZONA2-A3, forms an intermediate step in this process, as the radial fertile blanket, but not the axial one, has been replaced by sodium/steel elements. This replacement results in reduction of the critical radius (reflector gain), significantly higher flux and heat deposition rate in these sodium/steel elements than in the original radial blanket, and therefore in larger complexity of the physics phenomena involved. The correct collapsing of cross sections in the reflector, modelling of resonance shielding and back scattering into the core, as well as determination of gamma sources, are particularly difficult because of the enhanced role played by structural material resolved and unresolved resonances in the whole fast energy range, and by anisotropic cross sections.

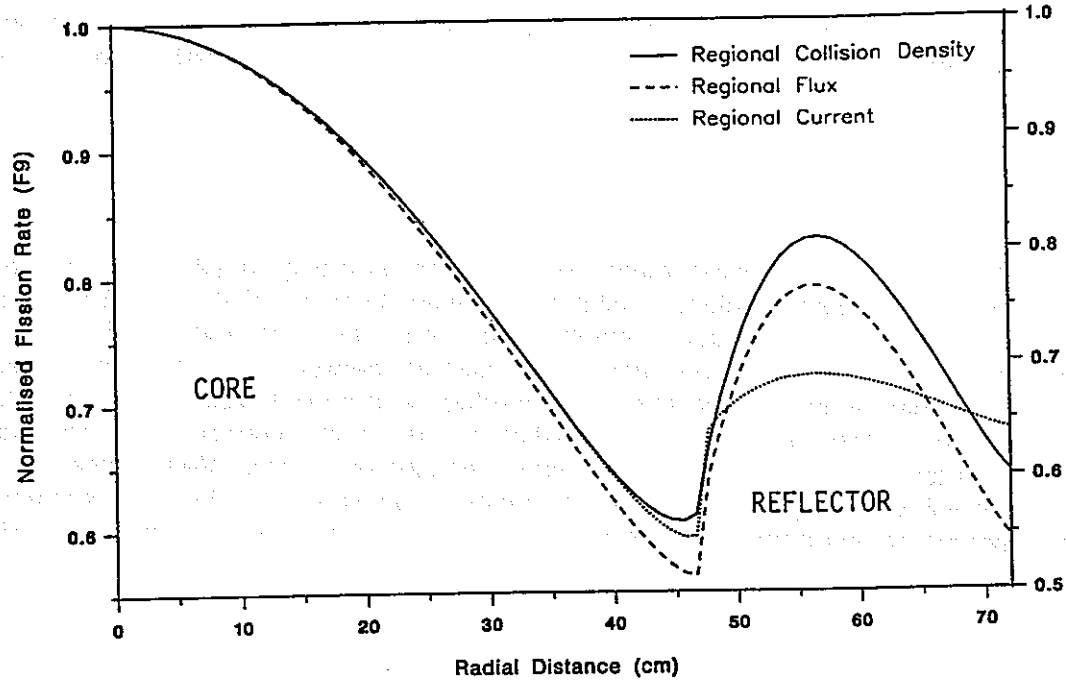

Fig. 4: ZONA2-A3 axially centered Pu239 fission rate radial profiles using different homogenisation procedures in the cell calculation

Globally, consistency between deterministic, stochastic, and experimental results is achieved for both configurations. The MICROX/TWODANT predictions of experimental values, using a small number of broad groups, became more satisfactory after the original procedures of homogenisation of anisotropic cell cross sections in space (with either regional fluxes or currents), as well as the generation of fission spectra for individual actinides, were improved. As the previous figure shows, the use of the new homogenisation method (with regional collision densities) leads to higher computed fission rates of ^{239}Pu in the radial reflector region of ZONA2-A3. These rates are in much better agreement with the experimental values [3].

Despite these improvements, some unexplained discrepancies between deterministic, stochastic, and experimental results are still noticed. In particular, because the computed reflector gain appears to be mispredicted by all methods, the effect of using different sets of iron data in the various regions is evaluated. In the following Table 2, relative variations of the reactivity of ZONA2-A3 are given in Δk/kk' in %, with respect to the reference k, the JEF-2 value.

^{56}Fe Data from	Radial Reflector	Axial Blanket	Core	All Regions
JENDL-3	– 0.22	– 0.03	– 0.11	– 0.39
ENDF/B-VI	– 0.20	– 0.01	+ 0.02	– 0.23
BROND-2	– 0.41	+ 0.07	– 0.07	– 0.59

Table 2: Reactivity Variations for ZONA2-A3 in Δk/kk' in %, k is the JEF-2 Value

This table and additional detailed analysis indicate strong sensitivity to both the level and fine shape of the iron cross section: The computed reactivity and radial power distribution are particular sensitive to the elastic scattering cross section between 100 keV and 1 MeV, with the maximum at 300 keV.

Conclusion

The first results of the CIRANO experimental program are providing valuable information about Pu-burning fast reactor configurations. Collaborative work between PSI and CEA on current experiments is continuing with particular emphasis on code validation and qualification. Efforts are concentrating on unexplained discrepancies observed between calculated and measured reaction rate traverses across the core/reflector interface and sodium void worth. The CIRANO program will provide further experimental information on fast reactor configurations characterised by a variable plutonium isotopic vector and higher enrichment levels. Measurements of heat deposition rates will be analyzed to broaden the range of application of the code systems. This additional experimantal information will be used to assess the quality of methods and data for calculating Pu-burning fast reactor configurations.

3 Swiss research relating to residual heat removal in Fast Breeder Reactors

Introduction

The LMFBR-related thermal-hydraulics research programme in Switzerland consists of experimental and analytical investigations on the mixing between two horizontal fluid layers of different velocities and temperatures.

The main purpose of the experimental part of this programme is to investigate the thermal-hydraulic phenomena in horizontal mixing-layers with particular attention paid to the effects of Richardson number (experiments in a wide range of velocity and temperature differences between the two streams) and Prandtl number (use of two fluids water and liquid sodium). For mixing layers with different temperatures, buoyancy forces influence considerably the intensity of mixing. Stratification is therefore an important phenomenon for heat exchange between sodium streams of different temperature in the upper plenum of FBRs under decay-heat-removal conditions.

Further development and adaptation of suitable computer codes for mixing layer calculation represent the analytical part of the programme. To accurately predict the flow fields and temperature distribution in the pools a satisfactory validation of codes, based on a reliable experimental data base, must be performed.

Short review of achieved results

In the first phase of the programme experiments with water were performed in a test-section made of acrylic glass (WAMIX I experiment). Visual observations of flow primarily lead to a better qualitative understanding of the phenomena in the developing mixing layer.

From the different experimental techniques tested, the laser sheet technique, using luminescent dye diluted in water was found to be a very efficient method to investigate generation, formation and interaction of vortices.

The mixing-layer thickness as a function of important parameters (distance from the splitter plate, Richardson number and velocity ratio) was determined and semi-empirical correlations for their calculation were given.

In order to investigate the formation of streamwise vortices (visualisation by a moving laser sheet) a conveyer system was used. It was found, that with increase of Richardson number (e.g. increase of the temperature difference between the layers) the formation of vortices is suppressed and the growth of mixing layer reduced.

The experimental results were used for comparison with codes, particularly with direct numerical simulations of the code FLOW-SB [6]. Extensive work of the WAMIX experiment was also related to the design of sodium experiment NAMIX, leading to decision to build a test-section with homogeneously-turbulent inlet flows.

A survey of the WAMIX tests together with main evaluated results was given in [7] and [8]. Complete results of the first investigation phase as well as their analysis are summarised in [9].

The main objective of the analytical work is to check the ability of general-purpose fluid dynamics codes (like FLOW3D [10] or ASTEC [11, 12]) to reproduce (with reasonable accuracy) the overall development of stratified mixing layers at various Prandtl numbers and to provide a code for a direct numerica1 simulation (modification of code FLOW-SB).

Up to now, most of the work has been carried out with the code ASTEC (calculations of velocity profiles in the flow conditioning part of the test-section and investigation of the effect of channel wall inclination on mixing-layer development). Particularly important was the participation of ASTEC as well as FLOW3D in different benchmark calculation tasks.

A survey of this work was presented at the international "Computational Fluid Dynamic Services (CFDS)" conference [13]. Some results were also included in a contribution to NURETH-6 [7].

Very good progress has already been achieved adapting the pseudo-spectral code FLOW-SB for the direct numerical simulation (DNS) of stratified mixing layers. Many modifications have been successfully implemented, particularly important to mention the possibility to visualise flow structures and temperature fields. A number of interesting calculation results on mixing-layer development with 2D and 3D models were compared with experimental as well as with the results of other numerical calculations [8, 14].

Progress report 1995

In 1995 the experimental work has been concentrated on preparation of second phase of investigation in water (WAMIX II) and providing a comparable testing arrangement in sodium (NAMIX experiment).

To carry out systematic measurements of local velocity and temperature as well as their fluctuation in WAMIX test-section, a new instrumentation (laser Doppler and hot wire anemometry) was chosen and ordered. Some modifications on the water loop and test-section were also made in order to allow the application of the new measuring techniques.

After determination of some details of NAMIX test-section design (particular tests with water i.e. WAMIX experiment, were made to clarify these details), the construction was started parallel to work needed to provide other components of experiment. Particular attention was put to define the process control system and its interaction to the data acquisition system.

Participation on Benchmark exercises was continued also in 1995. Velocity and temperature distribution for vertical buoyant jets were calculated with FLOW3D. This work started already in 1994, was performed for a benchmark meeting, organised by the IAHR Working Group on Advanced Nuclear Reactors Thermal Hydraulics [15]. After the meeting additional comparisons were made at PSI. It was found that the FLOW3D predictions are in closest agreement with the experimental results of all the codes using $k - \varepsilon$ model [16].

The analysis of stratified mixing layers at different Ri and Pr numbers was continued with the code FLOW-SB. Particular attention was paid to entrainment and mixing in the layer. The calculation for air and sodium has been finished, that for water is under way. The results were reported at the EUROMECH 339 conference [17].

As examples of these results Figs. 5 to 7 are added here. The non-dimensionalized times at which the isosurfaces are shown correspond to characteristic times, when the flow has reached a large degree of three-dimensionality and therefore ribs are visible. The important parameters which influence the results are initial conditions (deterministic and random distribution of velocity perturbations), temperature difference between the flows (Ri - effect) and the choice of fluid (Pr - effect).

The combination of deterministic and random initial conditions simulation corresponds well to the observations in experiments. The effect of stratification can be seen by comparing Fig 5 and 6. The baroclinic torque (i.e. torque produced by density gradients normal to the gravity direction) plays an important role in stratified cases because it influences the vorticity level. Cross-section cuts show that in stratified cases the vorticity is weakened .

Based on the comparison between the results for air and liquid sodium (of particular interest) at Ri = 0.05 (compare Figs. 6 and 7), it can be concluded that besides stratification also the high thermal conductivity of sodium suppresses the growth of the mixing layer.

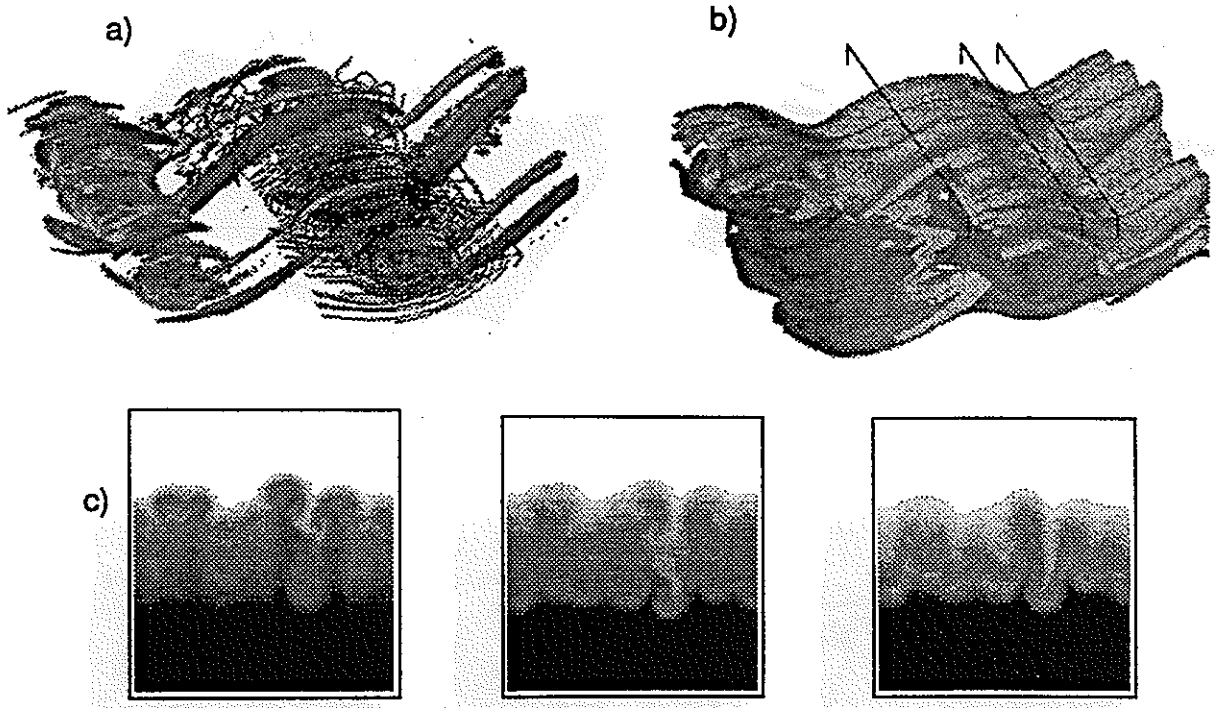

Fig. 5: Unstratified mixing layer calculation with combined deterministic and random initial conditions for air simulation at Richardson number Ri = 0.00.
Vorticity isosurfaces (see a), passive scalar isosurfaces (i.e. surfaces with equal marker substance concentration; see b), and cross-sectional cuts of the passive scalar (see c), at different streamwise locations (as marked in b) are given.

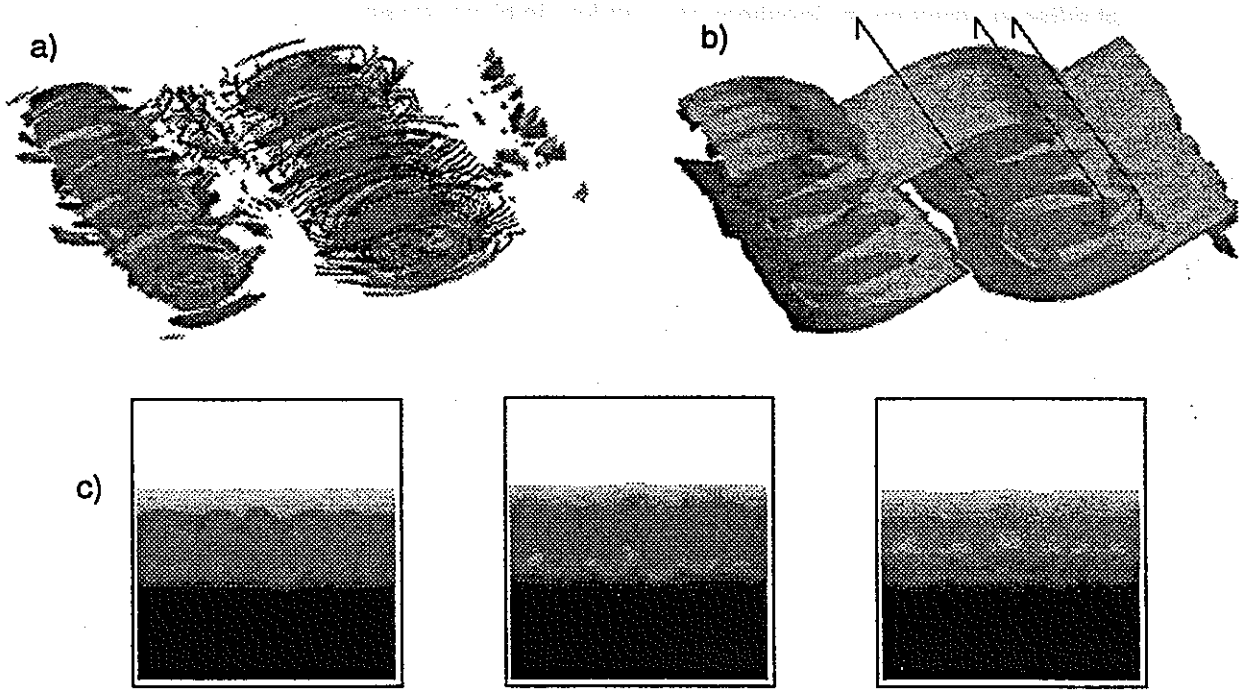

Fig. 6: Stably-stratified mixing layer calculation with combined deterministic and random initial conditions for air simulation at Richardson number Ri = 0.05.
Vorticity isosurfaces (see a), passive scalar isosurfaces (i.e. surfaces with equal marker substance concentration; see b), and cross-sectional cuts of the passive scalar (see c), at different streamwise locations (as marked in b) are given.

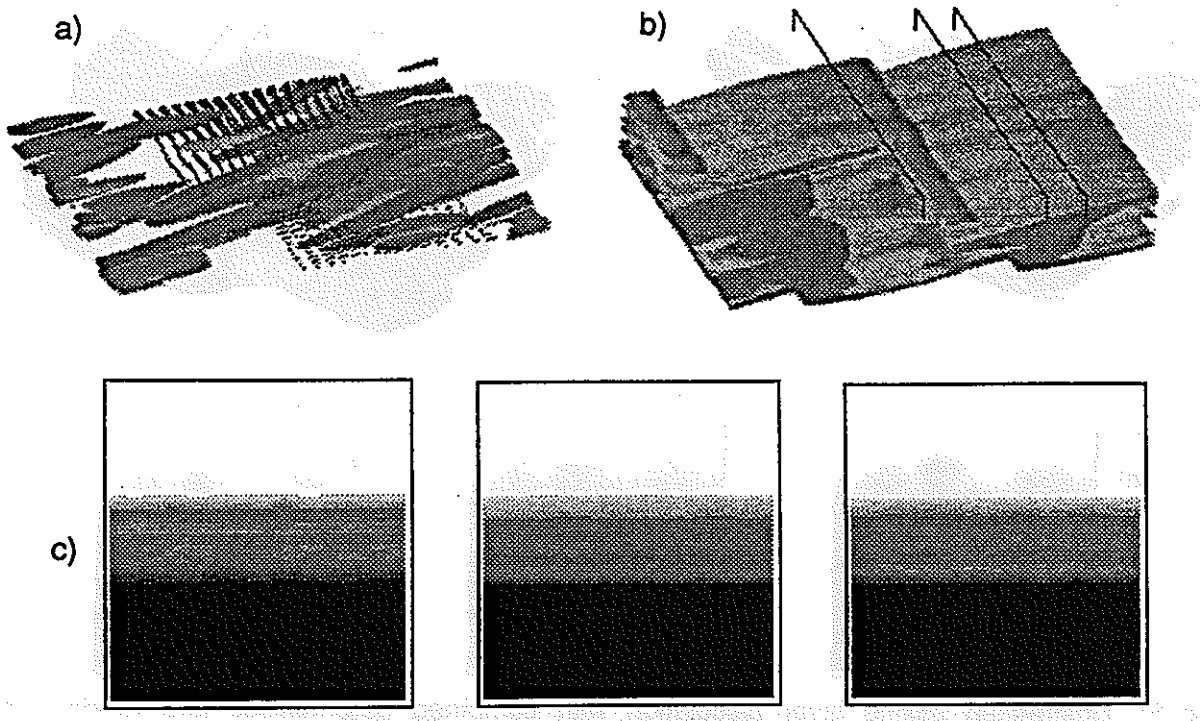

Fig. 7: Stably-stratified mixing layer calculation with combined deterministic and random initial conditions for liquid sodium simulation at Richardson number Ri = 0.05.
Vorticity isosurfaces (see a), passive scalar isosurfaces (i.e. surfaces with equal marker substance concentration; see b), and cross-sectional cuts of the passive scalar (see c), at different streamwise locations (as marked in b) are given.

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A REVIEW OF THE UK FAST REACTOR PROGRAMME

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Abstract

The general position with regard to nuclear power and fast reactors in UK during 1995 is described. The status of fast reactor studies made in UK is outlined and a description and statement regarding the conclusions of the programme of studies associated with the closure of the Prototype Fast Reactor is included.

1. The UK Nuclear Industry

The latest available statistics on electricity supply relate to 1994 and show that the total electricity generated was 325 TWh, an increase of 0.7% on 1993. Of this 49% was from coal, 28% nuclear, 13% gas, 5% oil and 5% others. Nuclear generation was 88.3 TWh, down 1.2 % on 1993. This was due in part to the closure of PFR, and also to significant increases in generation from combined-cycle gas turbine plant and from renewable sources.

The Sizewell B Pressurised Water Reactor (PWR) station, the newest in the UK, had a very successful initial year in 1995. Criticality was achieved in January, the plant was first connected to the grid in February, and full power operation was reached in June. Since then it has operated reliably with a very high load factor.

In May 1995 the Government's review of the nuclear industry was published. The main points were as follows.

- There is believed to be a significant role for nuclear power in the future.
- The perceived environmental risk of nuclear power can be counteracted by emphasis on safety, a rigorous regulatory regime, and a good safety record.
- Long-term economic interests in the European Union are best served by a free market in energy.
- In the short-term nuclear power is not essential to meet environmental targets.
- In the current UK market the combined-cycle gas turbine system is the most competitive for generation.
- The "nuclear levy" on electricity sales is to be stopped.
- The Advanced Gas-Cooled Reactors (AGR) and PWR stations can be privatised in 1996.
- It is not practical to privatise the Magnox stations.
- The financing of long-term nuclear liabilities is to be separated from power generation.
- Fast reactors were not mentioned specifically.

The nuclear generation industry is being reorganised in preparation for privatisation. A new holding company, "British Energy plc", has been set up to take over the AGR and PWR stations of Nuclear Electric and Scottish Nuclear. It is intended to privatise British Energy when appropriate. The older Magnox stations are to be retained in national ownership in a new company "Magnox Electric plc". Eventually Magnox Electric may become part of BNFL (who already own the two original Magnox stations at Calder Hall and Chapelcross).

As was noted in the UK position paper in 1995, the United Kingdom Atomic Energy Authority has been divided into two organisations. These are AEA Technology plc and UKAEA Government Division. By Act of Parliament, AEA Technology plc. has been vested as a private limited company, wholly owned by the UK Government, with effect from 1 April 1996. It is the UK Government's intent to privatise this company, when appropriate, during 1996. It is the aim of AEA Technology to become a leading scientific and technical services company.

UKAEA, which is to remain in the public sector, is responsible for discharging the decommissioning of nuclear facilities (including the Prototype Fast Reactor (PFR)) and Radwaste management arising out of the past activities of the United Kingdom Atomic Energy Authority. The Intellectual Property arising from the processes of decommissioning PFR will be held by UKAEA.

2. PFR Decommissioning

Since the closure of PFR at end March 1994, UKAEA has been responsible for the decommissioning of the reactor. This is to be divided into three active stages separated by periods of care and maintenance. The first stage comprises initial decommissioning and preparation for care and maintenance and will last until the early 2000s. An initial sub-stage has proceeded to schedule and the removal of core components and their replacement by dummy components was completed in 1995. In addition, the secondary sodium circuits and the NaK-filled decay heat rejection circuits have been drained down, non-essential services have been isolated and redundant plant items have been removed.

During the second sub-stage, a sodium disposal plant is to be constructed and operated on site to dispose of the liquid metal inventory of the reactor. The second sub-stage of decommissioning will also include the conditioning of the primary circuit for long term storage, the draining and cleaning-out of the buffer store and the dismantling non-essential plant and buildings. On the present schedule, the plant will then be placed in long term care and maintenance, including surveillance of the integrity of the secondary containment building, until 2070.

The Stage 2 decommissioning, scheduled to begin in 2070, will comprise the dismantling of the secondary sodium and steam circuits, the construction of a new containment around the primary circuit and irradiated fuel cave and the dismantling of the original secondary containment building. This Stage 2 decommissioning is scheduled to take 6 years. The Stage 3 decommissioning, comprising dismantling of the reactor internals and the concrete shielding, will follow a further period of care and maintenance and will take 20 years.

3. PFR Closure Experiments

A programme of experiments associated with the closure of PFR was carried out. The aim, with only a very modest budget, was to carry out work in some areas for which the European partners were wholly dependent upon the UK facilities and to learn as much as possible of value to future fast reactor development. The programme, as carried out, was in three parts:

i) Steam generator leak detection studies in PFR
ii) Sodium-water reaction tests in the Super Noah rig
iii) Destructive examination of materials from PFR

The first two studies were completed on 31 March 1995 and were described in the 1995 paper. The programme to examine materials began in 1994 and the examinations were completed at end March 1996.

These examinations have been restricted to the secondary sodium circuit, since the funding available precluded the study of primary circuit components such as the Above Core Structure or Intermediate Heat Exchangers. The secondary circuit studies were in three parts:

i) Examination of carbon steel components exposed to sodium environments.
ii) Study of delayed reheat cracking in austenitic steel weldments.
iii) Study of secondary pipework transition welds.

3.1 Examination of carbon steel components exposed to sodium environments

Following cracking of 15Mo3 steel in the Superphénix storage drum and Kalkar sodium storage vessels, questions remained over the use of as-welded carbon steels for structures in contact with sodium or sodium vapour. Examination of as-welded joints in the PFR carbon steel sodium storage tanks that have been exposed to sodium vapour were carried out to provide some assurance that similar steels, likely to be used for EFR roof structures, would not be susceptible to the type of cracking observed in the molybdenum-containing 15Mo3 steel.

Carbon steel could then be used in the as-welded state for roof constructions, avoiding the need for costly stress relief heat treatment.

A suitable vessel that had operated at up to 150°C for a period of several months, prior to the sodium filling of PFR, was selected. The section thicknesses of the vessel were generally 12.7 mm. Following removal of lagging, the weld seams and a number of nozzle welds were ultrasonically inspected from the exterior of the vessel. No defect indications of significance were detected. The man-way nozzle, a smaller temperature connection nozzle and a section containing the junction of a circumferential weld seam and a longitudinal weld seam were removed for examination. These were bagged-off to prevent air ingress and the sodium was removed using an ethanol wash. A bolted patch arrangement was used to effect a repair at each sample removal site. The sections removed were examined with dye penetrant. No indications of significance were found. The metallurgical examination of these samples has yet to be reported but preliminary studies suggest that there is no cracking of the type observed in the 15Mo3 steel, implying that carbon steels in the as-welded condition can operate successfully in contact with sodium without cracking.

3.2 Study of Delayed Reheat Cracking in Austenitic Steel Weldments

Delayed Reheat Cracking (DRC) was the main mechanism attributed to the cracking occurring at 500°C to 520°C in the weldments of the PFR reheater and superheater shells, between 1987 and 1993. It was also a major contributory factor in two earlier cracking incidents in PFR secondary pipework. All of these incidents have involved the Ti-stabilised Type 321 stainless steel.

DRC is expected to be exacerbated by large section thicknesses, high levels of constraint and high heat input welding. The published literature indicates that Type 316 steel is less susceptible to DRC than the Ti-stabilised Type 321 steel of PFR, however, the data base of experience of DRC on Type 321 steel is itself meagre. In particular there was little information on section thickness effects and relatively low temperatures (≤600°C) of operation. Suitable welds chosen for examination in PFR were those of the secondary pipework and a superheater vessel shell. The aim was to look for cracking, if any, in both good geometry and poor geometry welds of the thin section pipework and to characterise known cracks detected by ultrasonic inspection in the shell of Superheater 3.

A section of pipework on the line from the IHX to the Superheater 3 inlet was removed and bagged off. Repair of the remaining pipework was effected using welded end caps. Three sections, containing known defects, were removed from the Superheater 3 vessel shell and cleaned in ethanol. Repairs to the vessel were made using bolted patches.

The metallurgical examination of the pipework and superheater samples has not yet been reported formally. Some very limited cracking has been found at one of the pipework welds, suggesting that more extensive cracking, possibly leading to leakage, might have occurred over a much longer period than the 56,000 hours of high temperature operation experienced.

A large crack in a circumferential weld of Superheater 3, detected during the first ultrasonic inspection in 1988, and showing negligible growth in the intervening period, was found to be a hot tear in the weld formed during the vessel fabrication. A smaller crack at a longitudinal seam weldment at the junction with a circumferential weld was also examined. This crack had shown progressive growth over a three year period. It had an intergranular characteristic consistent with either DRC or conventional creep crack growth.

3.3 Study of a Secondary Circuit Transition Weld

At present, design rules for transition welds are not well validated. Such welds will be required between the roof and the primary vessel and between the secondary sodium pipework and the steam generators of future fast reactors. There is a history of problems with transition welds in conventional power stations, but this is usually associated with thick section welds

operating at typically 600°C, where creep-related failures have occurred in a decarburised zone in the ferritic steel.

A transition welds in PFR which operated at high temperature was selected for examination for microstructural changes. This weld was one between the 2.25Cr1Mo steel evaporator inlet nozzle and the Type 321 secondary pipework steel. These welds were made using Inconel filler and operated at or below 490°C.

Following removal of the lagging from the lower region of Evaporator 1, which was the unit with the longest operational exposure, ultrasonic inspection was carried out on two of the three inlet nozzles on lines from Reheater 1 and two of the three on lines from Superheater 1. One of the two inspected nozzles from the Superheater line was removed, and a repair effected, in the same way as the sodium pipework. The nozzle weld removed was subjected to destructive metallurgical examination. This examination has not yet been reported formally, but no characteristics were found that suggest the onset of any changes which might ultimately lead to failure. This weld had operated for about 67,000 hours.

4. Fast Reactor Development

Participation in the European collaboration continued under the leadership of BNFL with support from Nuclear Electric, NNC, and AEA Technology. This involved

- BNFL and NE in the activities of the European Fast Reactor Utilities Group (EFRUG),
- NNC in EFR Associates' reactor design studies, mainly in the areas of rector physics, safety, in-service inspection and repair, containment, decay heat rejection, and feedback of experience from PFR,
- BNFL, NNC and AEA Technology in R&D studies, mainly in connection with the CAPRA project and associated fuel cycle strategies (reported in more detail below), and
- Secondees with EFR Associates at Lyon and CEA at Cadarache (MASURCA).

Collaboration with Japan takes the form of a continued attachment at the MONJU site, and participation in studies of the safety of advanced reactor systems.

In addition the activities of the IWGFR have been supported.

5. UK Activities in Support of the CAPRA Project

The CAPRA project is described in detail by our French colleagues. This report concentrates on the UK contribution.

At the end of 1994 the feasibility of a mixed oxide core designed to consume around 70 kg of plutonium per TWh of electrical generation was established. In 1995 the work was extended to refine the design of the oxide core, to examine the feasibility of a high-consumption core containing no uranium, and to investigate the effect of plutonium-consuming on nuclear fuel cycle logistics in Western Europe.

5.1 Reactor Physics

The oxide core has considerable flexibility which can be exploited for various purposes. One attractive possibility is to increase the fuel residence time. This may be possible because, compared with a conventional fast reactor core such as EFR, the ratio of clad dose to burnup is low. If the limit on irradiation is set by dose rather than the fraction of heavy atoms fissioned the fuel residence time could in principle be increased by about 30%. The sodium void coefficient would be increased because of the higher concentration of higher plutonium

isotopes, but this can be compensated for by increasing the amount of moderator in the core. Studies by NNC have shown that, provided the fuel performance can be demonstrated experimentally, the high burnup version of the mixed-oxide CAPRA core appears to be feasible and has attractive features.

An obvious disadvantage of a core without uranium is the loss of the negative Doppler coefficient associated with U 238. This is offset partly by the negative Doppler of the higher Pu isotopes, and partly by the fact that positive feedback from the sodium void reactivity is lower. Studies by AEA Technology have shown, however, that there are difficulties in calculating the Doppler effect in non-uranium cores, and have indicated how standard calculation routes may need to be improved. There is a world-wide lack of experimental data on the Doppler effect of Pu 240.

5.2 Reactor Safety

Because of the heterogeneity there are difficulties in predicting the performance of the mixed-oxide core in a core-disruptive accident. The lead in tackling this problem has been taken by our German colleagues, and UK organisations have participated in the discussions to set the context of these aspects of the safety assessment.

NNC have made a detailed study of the transient response of a preliminary non-uranium core design to a range of design-basis and beyond-design-basis accidents. As expected the response of the intact core is strongly affected by the small Doppler coefficient, especially in the case of "high-quality" plutonium which has lower Pu 240 content. Nevertheless for a range of initiating conditions core damage and fuel melting can be avoided provided there is adequate negative feedback from a source such as the expansion of the control rod supports . The reliability and acceptability in a safety case of this form of negative feedback would of course require further investigation.

5.3 Fuel Performance

It is an objective of the CAPRA project that the fuel should be compatible with the PUREX reprocessing route. For the mixed-oxide core this has limited the plutonium fraction of the fuel to 45 % in order to preserve solubility in nitric acid. For the non-uranium core it has ruled out oxide as a fuel material, so that interest has centred on plutonium nitride fuel.

Compared with oxide, nitride has high thermal conductivity and high density. The high conductivity results in lower fuel temperatures. This is advantageous in that it tends to guard against dissociation of the nitride which may be a problem at high temperature, but may also have the drawback that more fission-product gas is retained within the fuel matrix.

The high density is a disadvantage to a CAPRA core. Compared with a conventional fast reactor core the power per unit mass of fuel is high because the diluting effect of the uranium has been lost. Fuel pin and subassembly designs have to be adapted to accommodate the very high power density, and high fuel density exacerbates the problem.

AEA Technology have been involved in determining the properties of nitride fuel materials and its behaviour in the reactor, both in normal operation and in off-normal transients.

5.4 Fuel Cycle Scenarios

It is necessary to support the studies of plutonium-burning reactors of the CAPRA type by a clear understanding of the effect the deployment of such rectors would have on inventories of plutonium and other actinides and waste products. Studies of this type have been undertaken in the UK and other countries for some time. During 1995 a combined West-European assessment was started in connection with the CAPRA project.

NNC have set up a model of the flows of fuel cycle materials within the nuclear power industry. Initially this was applied to the UK, but it has been applied to Western Europe to show the effect of plutonium burners on plutonium stocks and waste arisings, in comparison with other options such as recycle of plutonium in LWR reactors in the form of MOX fuel. Plutonium stocks can be reduced by CAPRA reactors but it is clear that, if nuclear power generation is assumed to continue indefinitely into the future, there is a minimum plutonium stock level which has to be maintained.

5.5 The Third International CAPRA Seminar

In November 1995 BNFL hosted a seminar at Lancaster at which the results of the work done in many countries in connection with the CAPRA project were exchanged and discussed. About 60 delegates from 8 countries attended. 29 papers were presented in the course of two days of discussion. The meeting concluded with a visit to the BNFL THORP reprocessing plant at Sellafield.

INTRODUCTION TO THE INTERNATIONAL NUCLEAR SAFETY CENTER

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INTERNATIONAL NUCLEAR SAFETY CENTER

MISSION AND GOALS

The mission of the International Nuclear Safety Center is to promote continuing improvement in nuclear safety and enhance core competencies in nuclear technology.

The International Nuclear Safety Center will:

- **Work to maintain technical excellence in design of safe nuclear reactors.**
- **Undertake cooperative programs in development of safety technology.**
- **Provide a focus for open exchange of safety information.**

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INTERNATIONAL NUCLEAR SAFETY CENTER

TECHNICAL OBJECTIVES

- Improved computer models and codes for system simulation, design analysis, and safety evaluation, including applications of advanced computing techniques.
- Improved experimental support for accident prevention features, consequence mitigation, and accident management.
- Improved technology for monitoring of equipment performance and diagnosis of failures.
- Information exchange using modern computer and communications technology.

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TYPES OF COLLABORATIVE PROJECTS

- **Joint assessments of analysis methods.**
  - Code-to-code comparisons.
  - Comparisons of codes to experiments.
- Joint development and extension of analysis methods.
- Joint design, performance, and analysis of experiments.
- **Joint evaluation of advanced safety design concepts.**
- **Technology transfer and training in analysis methods.**

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AREAS OF TECHNICAL INTEREST

- Reactor physics, neutronics, and thermal hydraulics; system simulation.
- Structural and seismic analysis.
- Fuel performance; extended burnup; failure detection.
- Materials; aging phenomena; materials properties.
- Artificial intelligence applications in diagnostics and monitoring.
- Severe accident phenomenology; accident management.
- Safety of non-reactor facilities.
- Technology of decontamination and decommissioning.

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CURRENT RESEARCH AND DEVELOPMENT PROJECTS

Accident Management Technology

- **Applications of Accident Management Technology**
  - Summarize current US accident management technology and assess applications to other designs.
  - Investigate technology transfer.
  - Develop collaborative projects -- experiments, analysis, interpretation.

- **Severe Accident Materials Properties** (current collaboration with Nuclear Safety Institute of Russian Academy of Sciences - IBRAE)
  - Database of existing information.
  - Measurements to extend the available data.
  - Interpolation and extrapolation methods.

- **Analysis of Severe Accident Experiments** (current collaboration with IBRAE)
  - Application of advanced computational techniques to codes used for analysis and interpretation of severe accident experiments.

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CURRENT RESEARCH AND DEVELOPMENT PROJECTS

System Diagnostics and Monitoring

- **System and Component Monitoring**
  - Develop a generic expert system for sensor and component operability surveillance.

- **On-line Plant Transient Diagnostics and Management**
  - Develop a generic expert system for diagnosis of abnormal thermal-hydraulic system conditions.

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CURRENT RESEARCH AND DEVELOPMENT PROJECTS

Advanced Simulation

- **Integrated Plant Simulation**
  - Use advanced computational science techniques with existing thermal-hydraulic and neutronics codes to improve plant simulation.
  - Provide high fidelity, fast running computer codes for use in design, training, and operations support.

- **Advanced Structural Modeling and Simulation**
  - Simulate behavior of metal and concrete structures under accident loads.
  - Fluid structure interaction simulations.
  - Analysis of seismic response.
  - Introduction of advanced computational science techniques.

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INTERNATIONAL NUCLEAR SAFETY CENTER

INSC DATABASE

Purpose

- Provide for free and open exchange of safety information.
- Facilitate access to detailed information on reactors worldwide promoting improvement in safety standards.
- Provide a resource for the data, tools, and information necessary to perform safety analyses to accepted international standards.
- Rapid access to the database through the Internet.

Status

- Database operational at Argonne.
- Counterpart database being developed at Nuclear Safety Institute of Russian Academy of Sciences (IBRAE).

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INTERNATIONAL NUCLEAR SAFETY CENTER

INSC DATABASE

<u>Implementation</u>

- **Interactively accessible resource database.**
- **Distributed database architecture.**
- **Graphical and forms-based interface to obtain database information.**
- **Provision for controlled access to specific database elements based on user-domain and access level authorization:**
  - Server side control.
  - Oracle security mechanisms.
  - Combination of both.
- **Procedures for database verification and maintenance.**

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INTERNATIONAL NUCLEAR SAFETY CENTER

INSC DATABASE

Database Contents

- Plant-specific general, operating, and design information.
- Material properties for safety analysis.
- Bibliography of available safety evaluations and risk assessments.
- Analysis methods and computational tools.
- Network links to other information sources.

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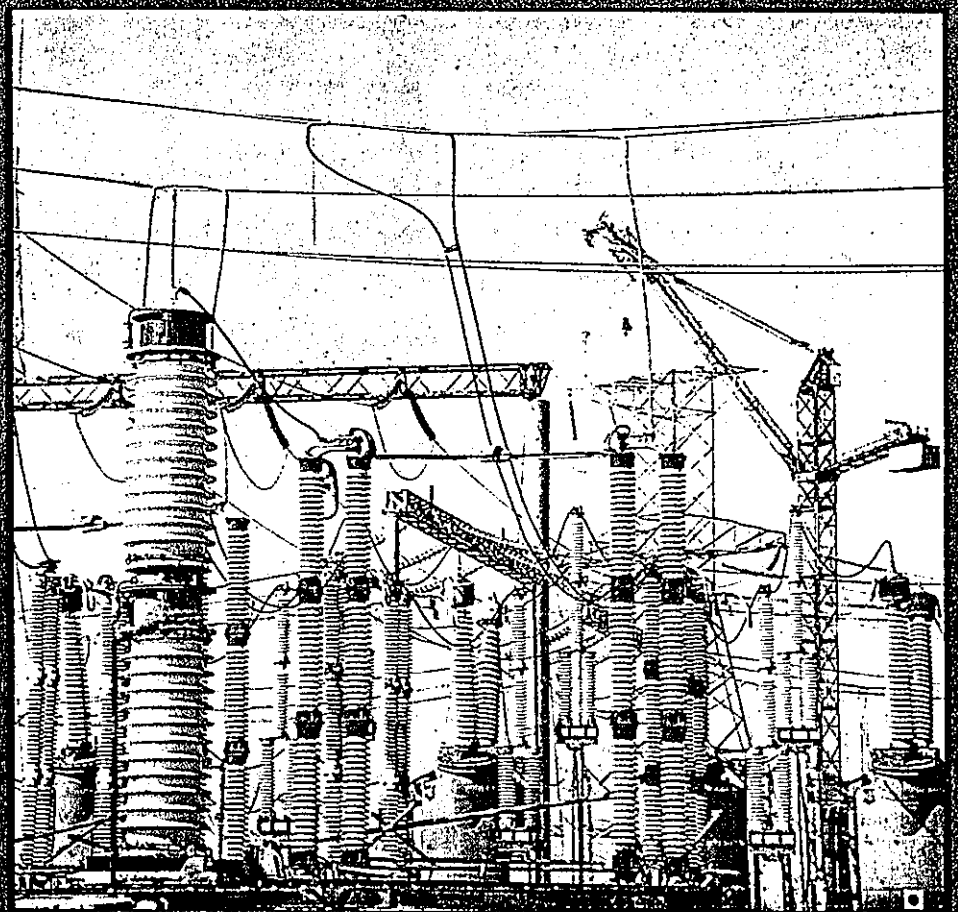
МАНГЫШЛАКСКИЙ ЭНЕРГОКОМБИНАТ

им. 60-летия СССР



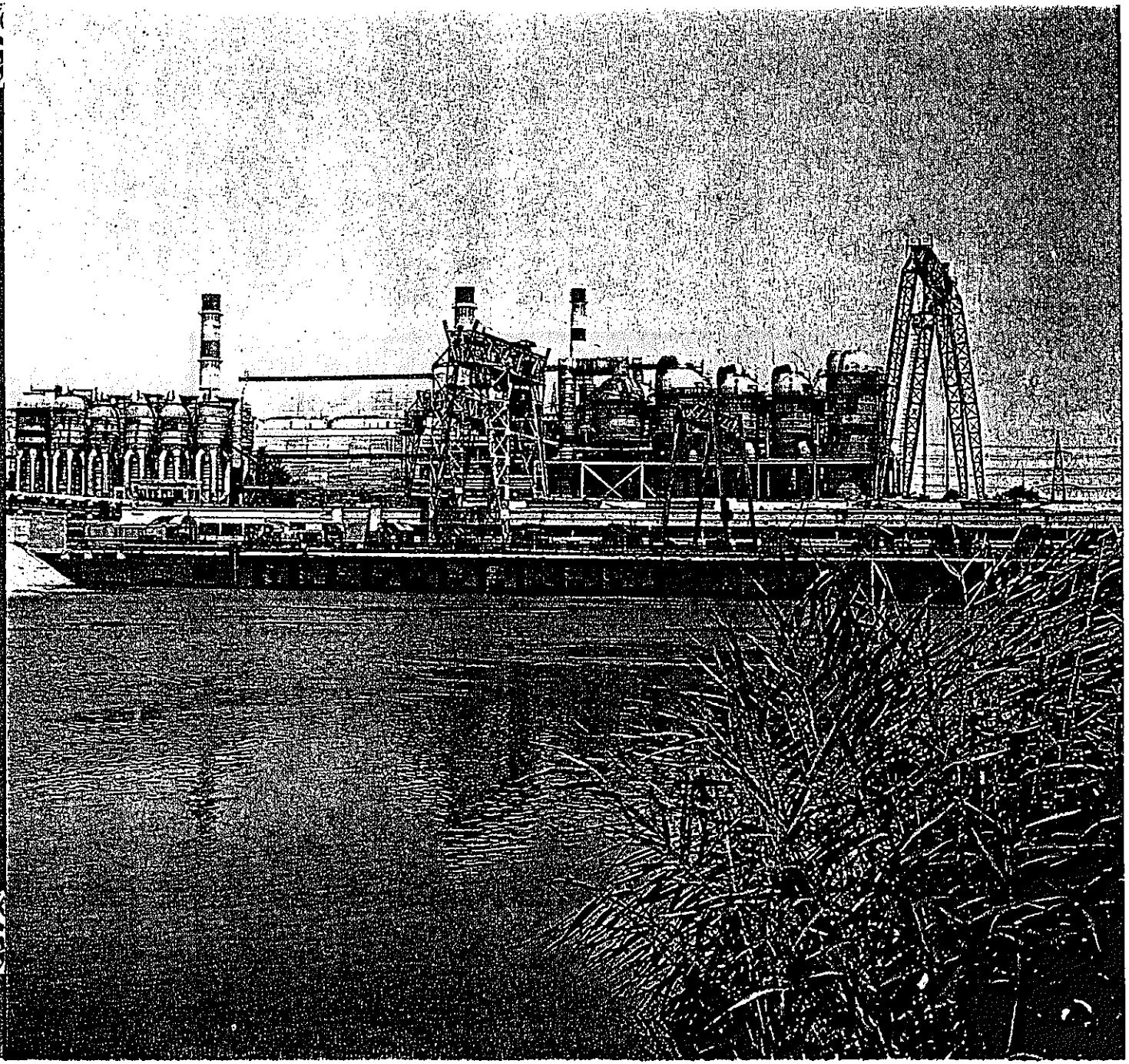
МАНГЫШЛАКСКИЙ ЭНЕРГОКОМБИНАТ

им. 60-летия СССР

МИНСК «ПОЛЫМЯ» 1989

"ГЛАВНОЙ ЗАДАЧЕЙ ПРЕДПРИЯТИЯ ЯВЛЯЕТСЯ ВСЕМЕРНОЕ УДОВЛЕТВОРЕНИЕ ОБЩЕСТВЕННЫХ ПОТРЕБНОСТЕЙ НАРОДНОГО ХОЗЯЙСТВА И ГРАЖДАН В ЕГО ПРОДУКЦИИ"

Из закона СССР о государственном предприятии



Мангышлакский энергокомбинат имени 60-летия СССР как самостоятельное предприятие образован 1 июля 1968 года.

Полуостров Мангышлак находится на западе Казахстана. Это район многовекового обитания казахских племен, которые кочевали по суровой пустыне, занимающей большую часть пространства полуострова в поисках пастбищ.

Суров климат Мангышлака. Жара африканской пустыни летом и холод Арктики зимой. Достигающая 30 градусов разница температур между днем и ночью. Сильные ветры и пыльные бури. Сухая, засоленная почва, поверхность которой нагревается до 60—70 градусов. Лишь редкий кустарник да верблюжья колючка приспособились к жизни в этих условиях.

Северо-восточное побережье Каспия впервые нанесено на географические карты в сороковые годы прошлого века. Тогда же были обнаружены нефть и другие полезные ископаемые. Однако первая геологическая карта полуострова была выполнена лишь в 1928 году. После этого начались планомерные поисковые работы. Искали не только полезные ископаемые, но и пресную воду, источники которой на полуострове практически отсутствовали.

В 1956 году в северной части самой глубокой в стране впадины Карагие, расположенной на 132 метра ниже уровня мирового океана, было открыто месторождение слабоминерализованных вод. Для питья вода не годилась. Но полученный к тому времени опыт опреснения морской воды на термических опреснителях в Баку и Красноводске вселял надежду на успешное решение проблемы опресненной питьевой воды. Задачу эту в промышленных масштабах еще предстояло решить. Пока же воду доставляли через море танкерами.

Поиск нефти и газа увенчался успехом в 1961 году, когда в январе на месторождении Узень ударил первый фонтан газа, а в июле на площадке Жетыбай — первый фонтан нефти.

В октябре 1961 года XXII съезд КПСС принял решение о создании в Западном Казахстане новой нефтедобывающей базы страны. Началось бурное развитие промышленного потенциала Мангышлакского края.

Энергетика Мангышлака начала развиваться в 1960 году с прибытием первого энергопоезда. В 1962 году вводится в эксплуатацию дизельная электростанция, а в 1963 году — первые турбоагрегаты ТЭЦ-1. И тогда же первую опресненную воду дала опытно-промышленная испарительная установка (ОПИУ) производительностью 150 тонн дистиллята в час.

В 1964 году начинается строительство атомного реактора на быстрых нейтронах, энергия которого должна была обеспечить опреснение воды и выработку электроэнергии. Начался новый этап в развитии атомной энергетики.

В целях безопасности была принята трехконтурная тепловая схема реактора, исключающая распространение продуктов деления во внешнюю среду.

В 1966 году приступили к монтажу металлоконструкций реактора и строительству ТЭЦ-2, которая была предназначена для работы в едином комплексе с реактором и установками по производству дистиллята. Выбор оборудования станции определялся условиями надежной работы всего комплекса таким образом, что на периоды отключения реактора пар на турбоагрегаты должен был поступать с котлоагрегатов ТЭЦ с обеспечением его устойчивой подачи на опреснительные установки. В декабре 1968 года было закончено строительство главного корпуса ТЭЦ-2 и введен в эксплуатацию ее первый турбоагрегат мощностью 60 МВт. Поэтапно вводятся в эксплуатацию не имеющие аналогов в стране уникальные гидротехнические сооружения: водозаборный морской канал протяженностью три километра и морская насосная с пропускной способностью, равной стоку реки Урал.

29 ноября 1972 года был произведен физический, а 16 июля 1973 года энергетический пуск реактора, ставший началом его промышленной эксплуатации. Более чем 15 лет надежно, безопасно и стабильно работает промышленный реактор на быстрых нейтронах с натриевым теплоносителем. И все эти годы проводятся исследования и эксперименты, которые дают богатый материал для совершенствования реакторов на быстрых нейтронах большой мощности, создание которых является генеральной линией перспективного развития ядерной энергетики.

Сегодня Мангышлак стал крупным центром нефтедобывающей и химической промышленности, строительной индустрии. Здесь добывается 85 % республиканской нефти и 90 % природного газа. Промышленный комплекс полуострова является крупнейшим в стране поставщиком полистирола и азотных удобрений. Здесь производятся сжиженный газ и газовый бензин, камень-ракушечник и другие строительные материалы.

Растут энергетические мощности комбината. В марте 1983 года и в октябре 1984 года введены в строй два блока тепловой электростанции мощностью по 210 МВт каждый. Увеличилось число опреснительных установок, суммарная производительность которых сегодня превышает 150 тысяч кубометров дистиллята в сутки.

В каждой вновь вводимой установке реализуются результаты

последних исследований и накопленный опыт в области опреснения морской воды. Особенностью действующих опреснительных установок является их способность производить глубокообессоленный дистиллят, что позволяет использовать его для подпитки водой парогенераторов реактора, а также и то, что они не требуют реагентов для борьбы с накипеобразованием — рециркулирующие затравочные кристаллы (мелкоразмолотый природный карбонат кальция) надежно защищают теплопередающие поверхности от карбонатной накипи.

Питьевая вода гидрокарбонатного класса, производимая комбинатом, благодаря наличию в ней необходимых человеку солей и микроэлементов, соответствует самым высоким требованиям. Но работа над повышением ее качества продолжается.

Научно-исследовательскую работу на предприятии возглавляет научно-технический совет и его секции.

Все более широкое применение в системе управления экономической, хозяйственной и научной деятельностью находит современная вычислительная техника.

Неоднократно посещали комбинат делегации советских и зарубежных ученых.

— «Я удовлетворен знакомством с единственной в среднеазиатской зоне СССР атомной электростанцией»,— сказал, обобщая свои впечатления от посещения комбината, Х. Бликс — директор Международного агентства по атомной энергии (МАГАТЭ).— «Удачной представляется идея с помощью реактора на быстрых нейтронах сочетать производство электричества и питьевой воды в пустыне. Уровень надежности и безопасности БН-350 не уступает реакторам-размножителям других стран».

Комбинат надежно и экономично обеспечивает города, поселки и промышленные предприятия полуострова питьевой и хозяйственной водой, электрической энергией, осуществляет постоянный контроль за состоянием окружающей среды и влиянием на нее его производственного комплекса.

Многолетние наблюдения подтверждают безвредность для окружающей природы не только атомного реактора, но и опреснительных установок, что убеждает в возможности и целесообразности использования атомной энергии для освоения безводных районов.

В озере Караколь, в которое комбинат сбрасывает техническую воду, гнездятся птицы и водится рыба. На зиму прилетают лебеди и фламинго.

Территория комбината поражает непривычной для здешних мест зеленью деревьев, живописными клумбами цветов, создающими прохладу фонтанами.

Вокруг промышленных площадок развиваются огороды рабочих и специалистов.

Для полива деревьев, цветов и огородов успешно используется вода, уже отработавшая в технологических процессах и прошедшая необходимую очистку.

База отдыха трудящихся комбината «Ивушка» раскинулась на берегу моря. С мая по сентябрь привлекают сюда на семейный отдых ласковые волны и песчаные пляжи, спортивные площадки и водолечебница, кафе и дискотека. Для детворы построены бассейн и детский городок.

Цехом здоровья стал спортивный комплекс комбината «Орбита», располагающий крытым бассейном с морской водой, двумя спортивными залами, открытым теннисным кортом.

Культурно-массовую работу в коллективе комбината проводит клуб «Энергетик».

Шевченко — город энергетиков и химиков, геологов и нефтяников, строителей и педагогов, многих других специалистов, необходимых для освоения природных богатств края. В 1988 году городу исполнилось 25 лет.

Построенный между морем и пустыней, город строг и величавен, когда смотришь на него из иллюминатора самолета, удобен и уютен, когда прогуливаешься по микрорайонам или заходишь в квартиры его жителей.

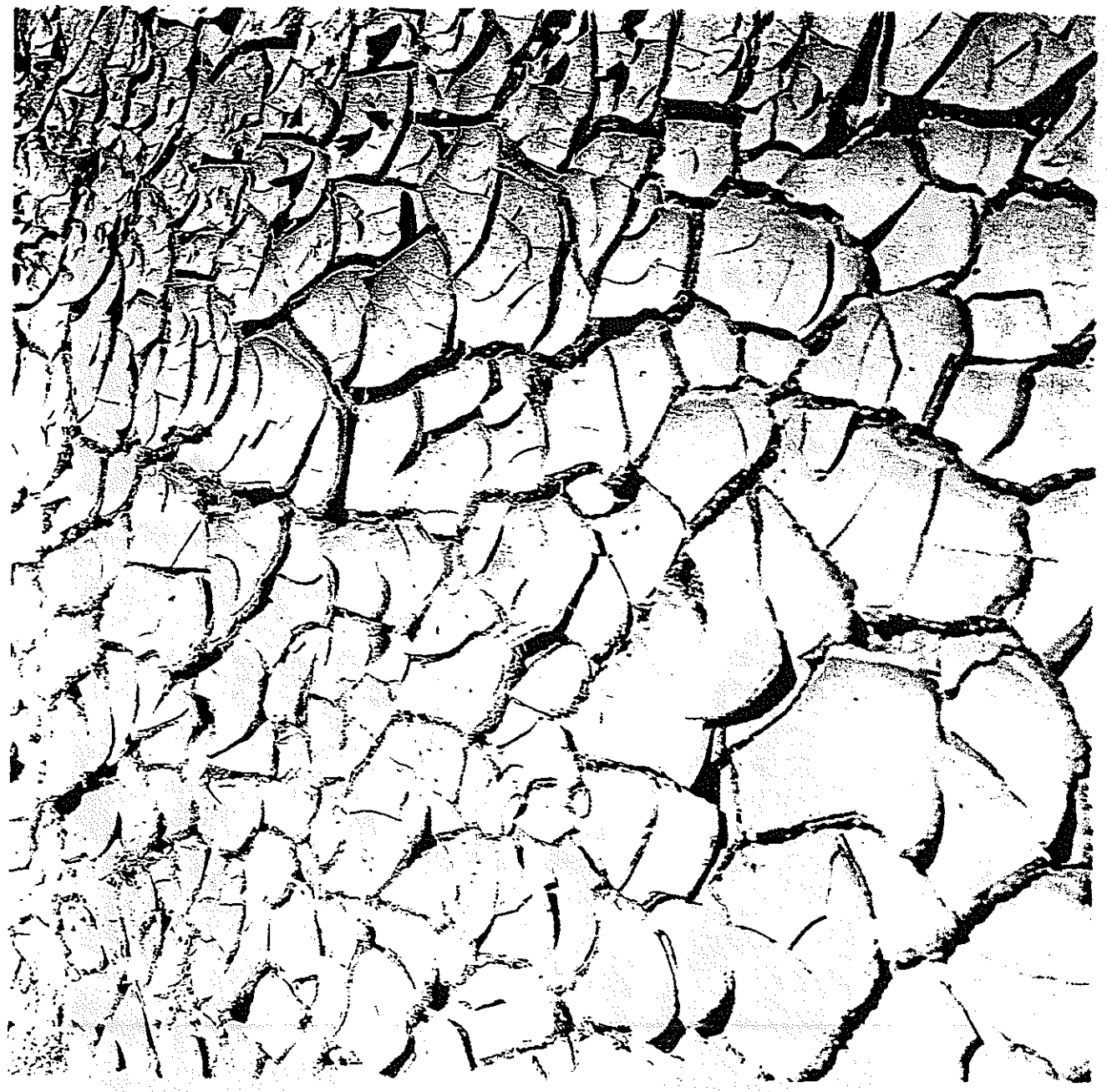
Кукольный театр и камерный оркестр филармонии, дворцы культуры и кинотеатры предлагают интересные встречи с искусством.

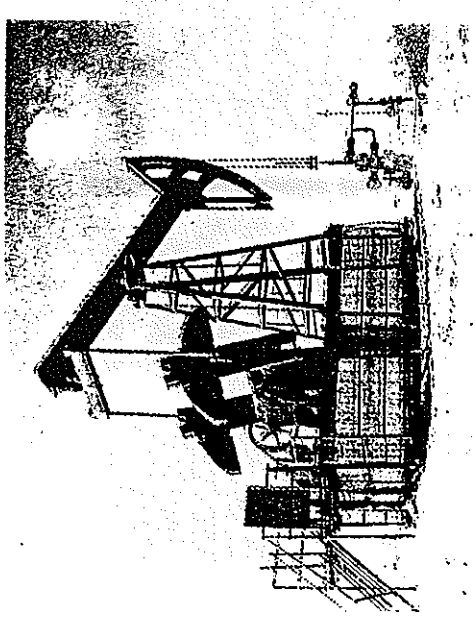
Стадион и спортивные площадки приносят всем желающим радость физического совершенствования.

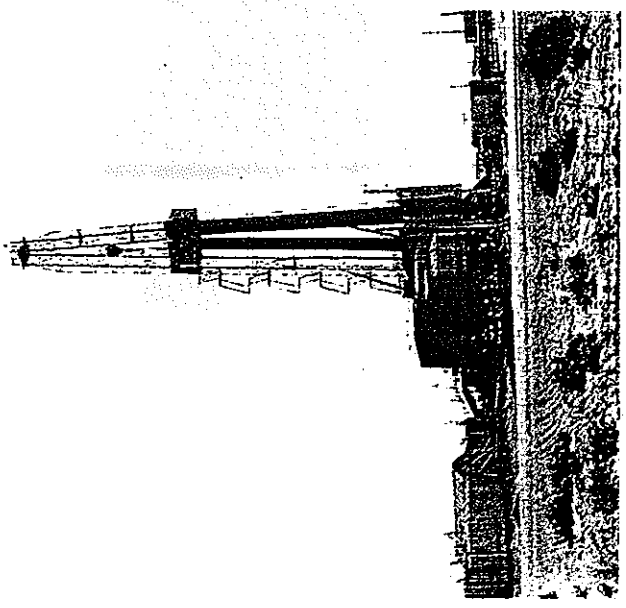
За создание гуманистической среды в пустыне городу вручена золотая медаль им. Патрика Амберкромби.

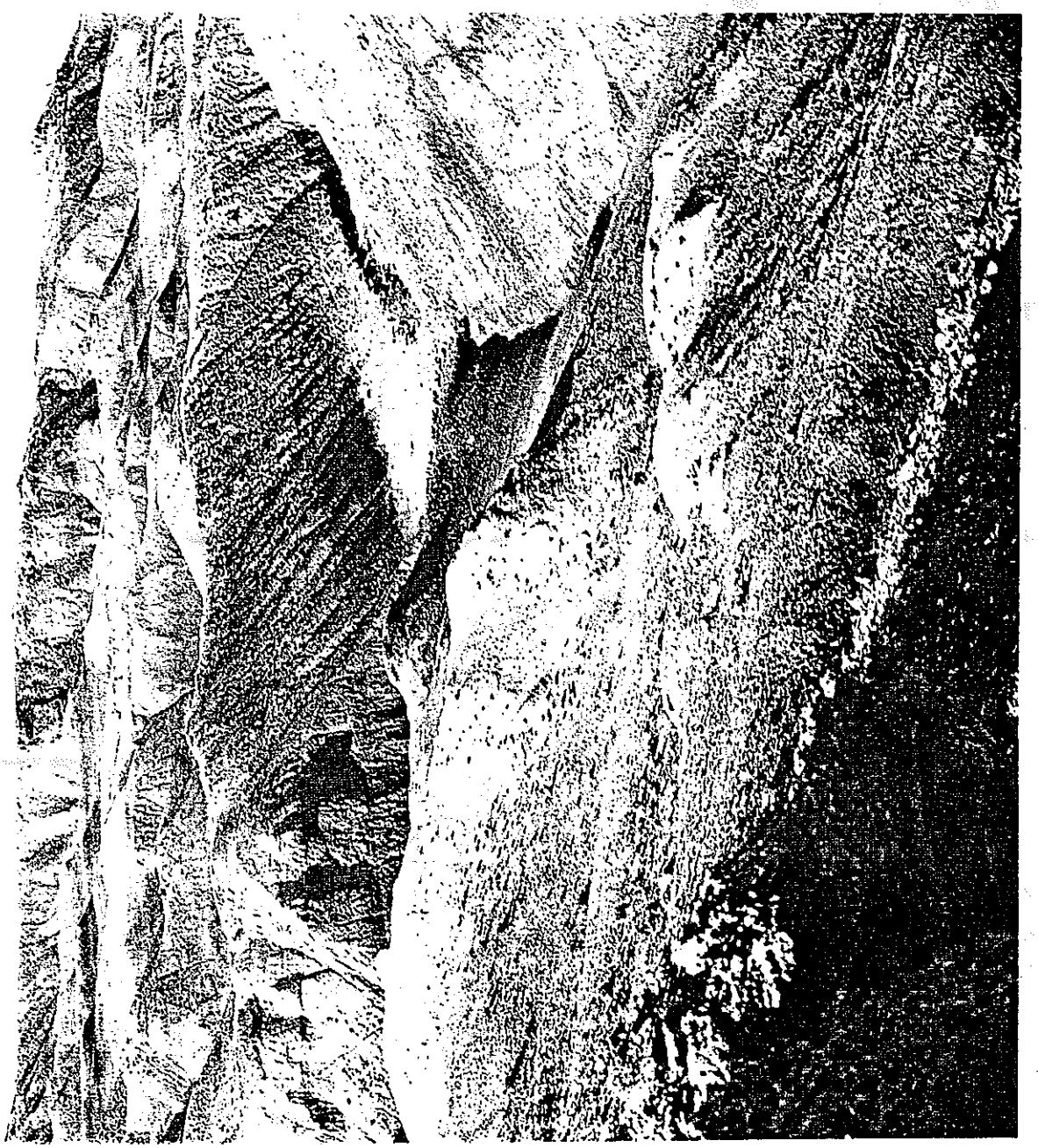
Город развивается вместе с расширением промышленного потенциала Западного Казахстана. Это стимулирует непрерывный рост энергетических и опреснительных мощностей Мангышлакского энергокомбината им. 60-летия СССР.

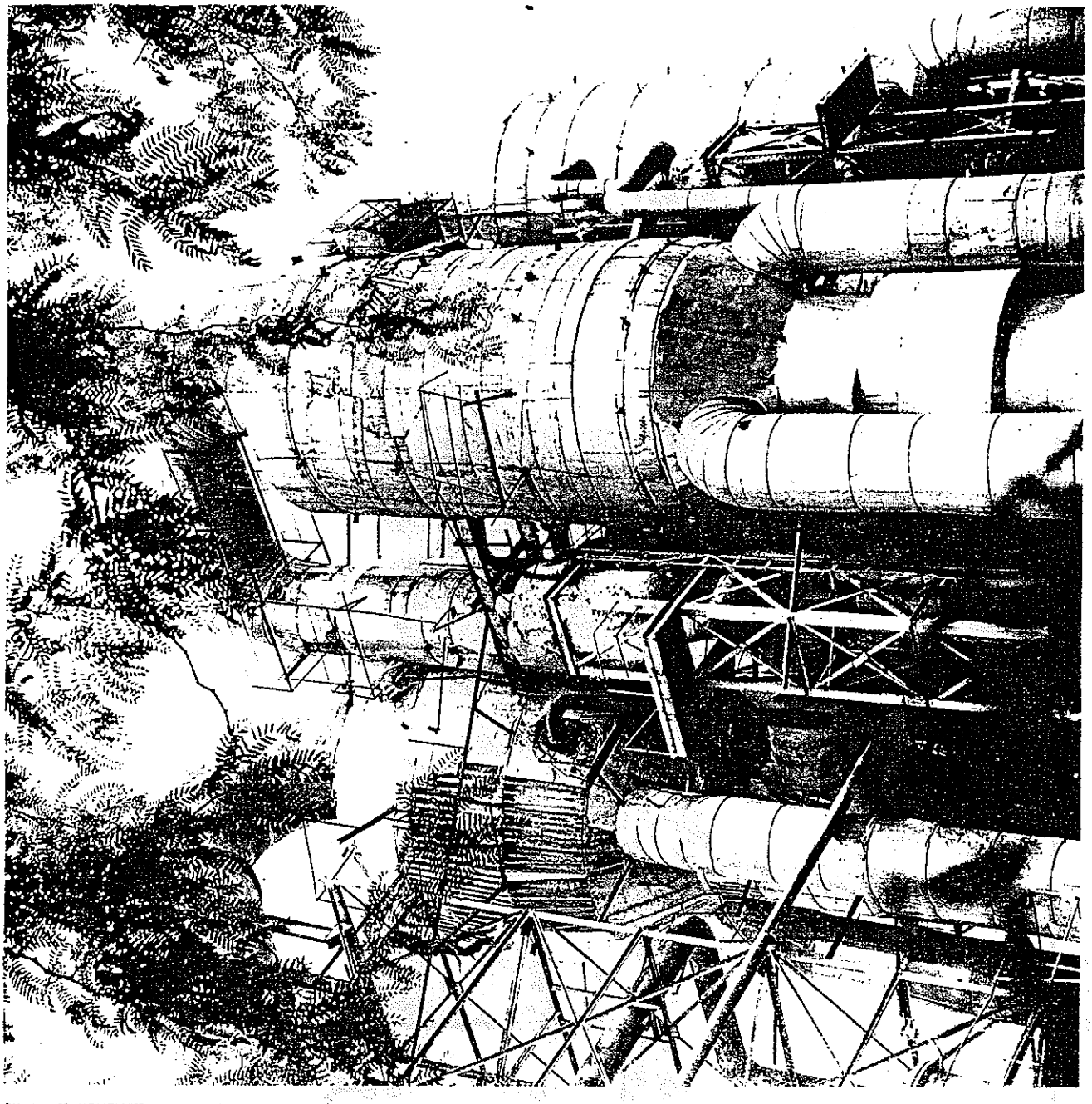
Опыт жизнедеятельности 180-тысячного города на опресненной воде моря несомненно будет использован будущими поколениями людей.



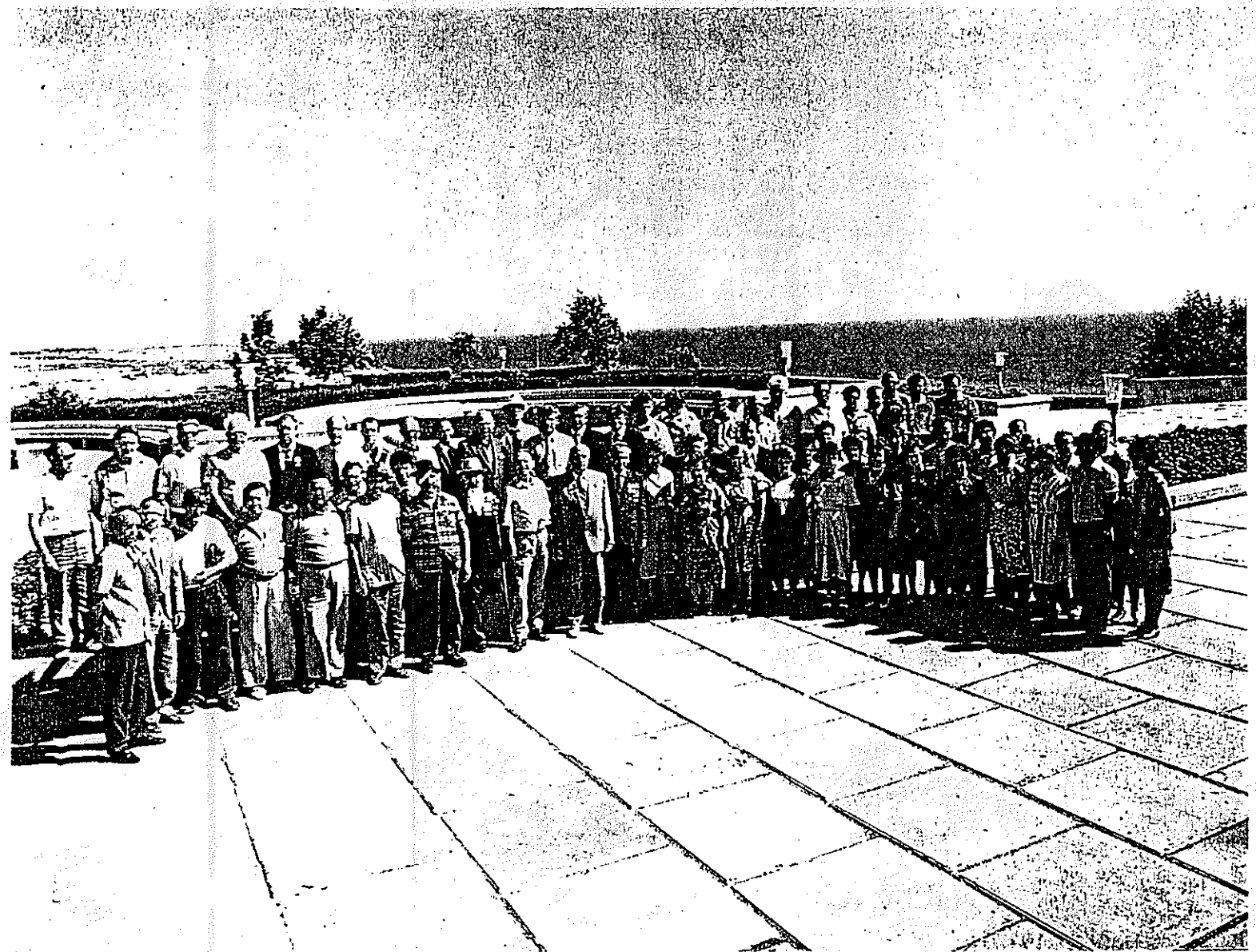




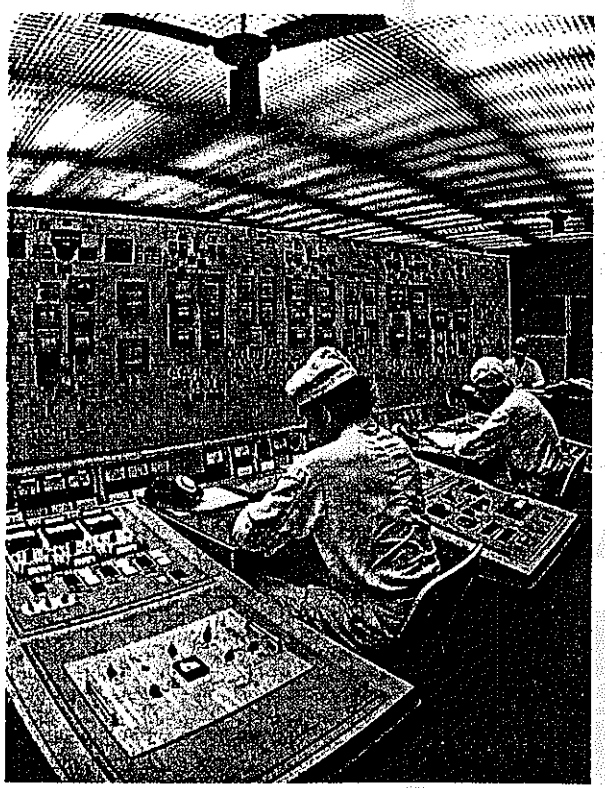


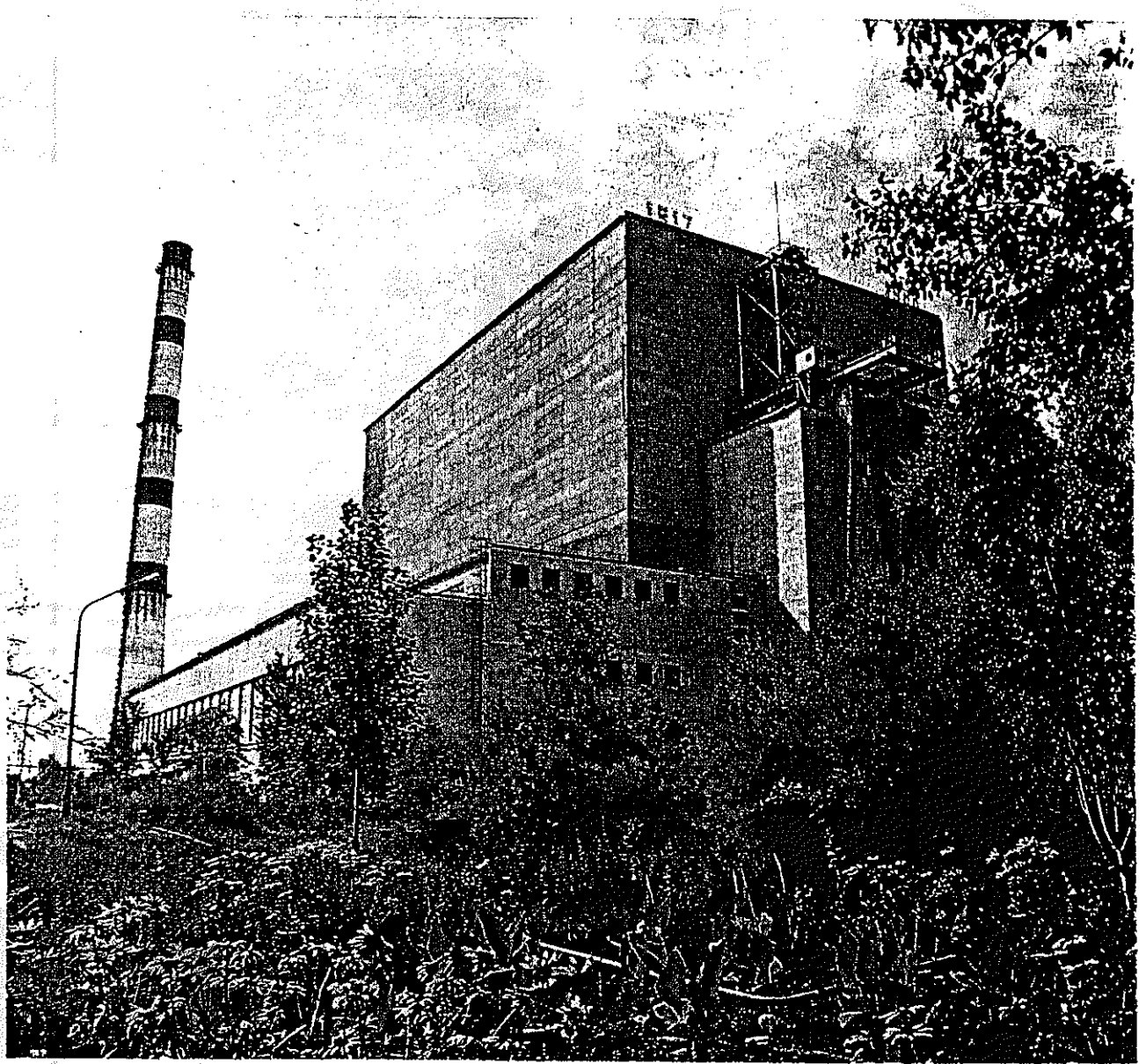


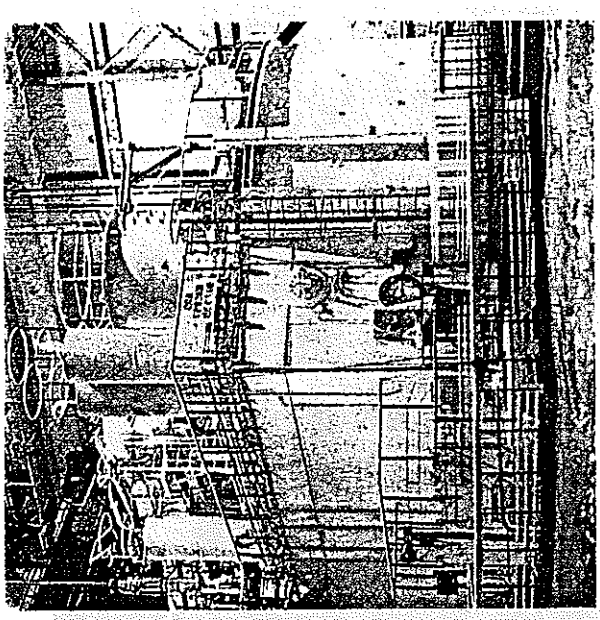
ОНИ БЫЛИ ПЕРВЫМИ

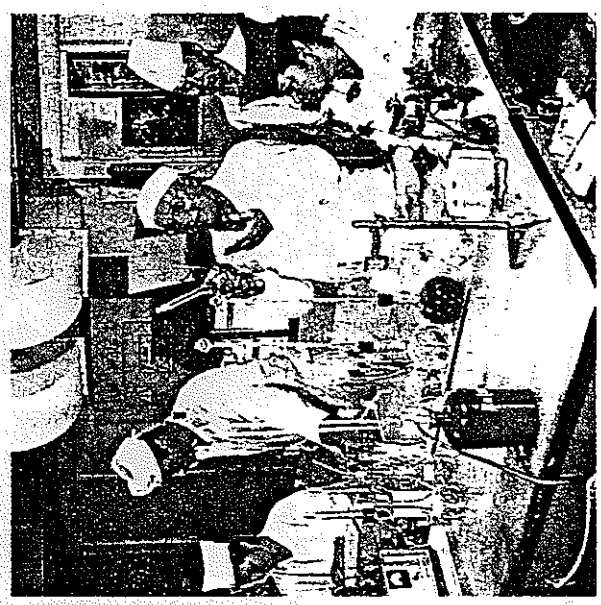


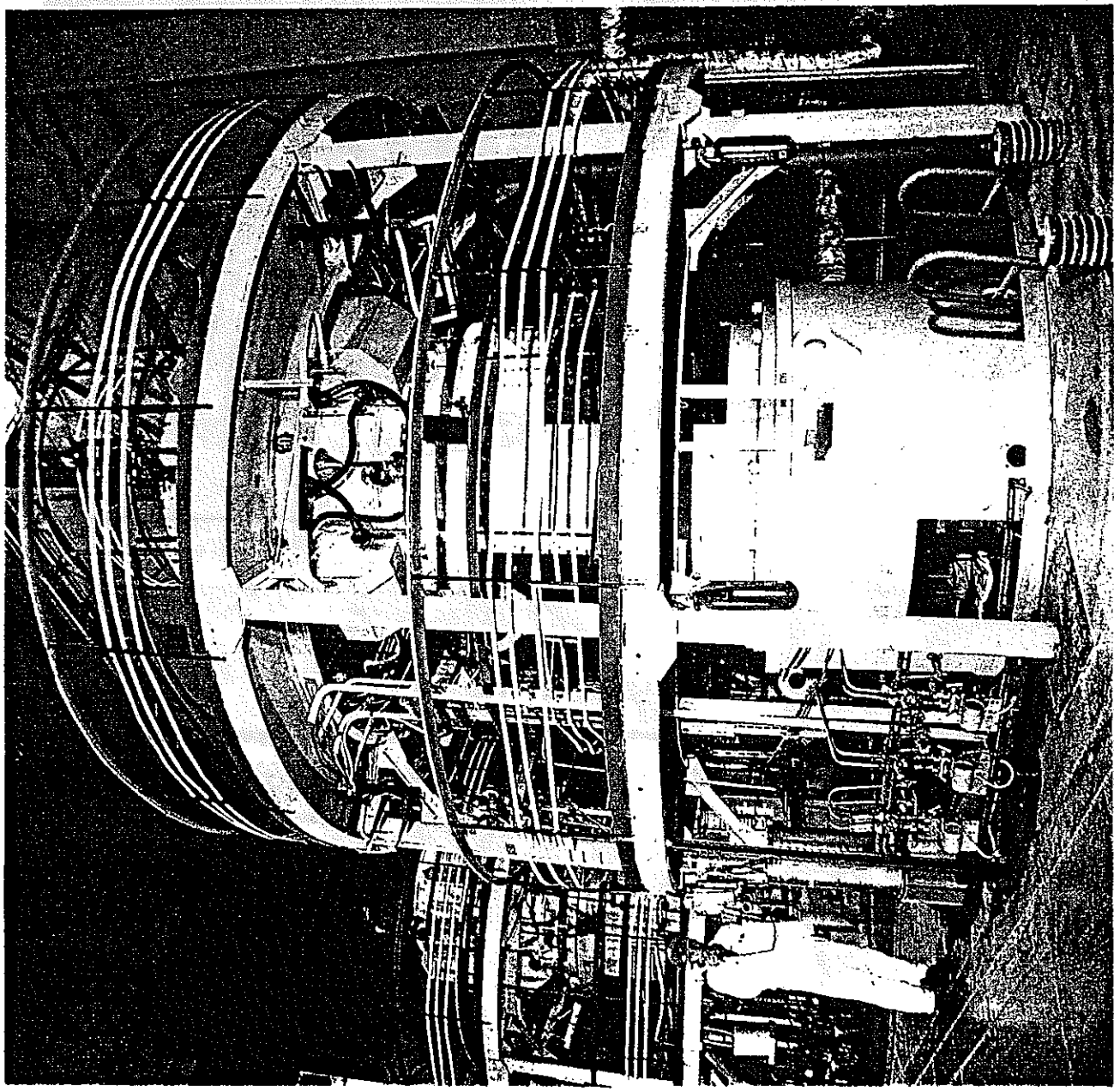
НА РЕАКТОРНОМ ЗАВОДЕ БН-350

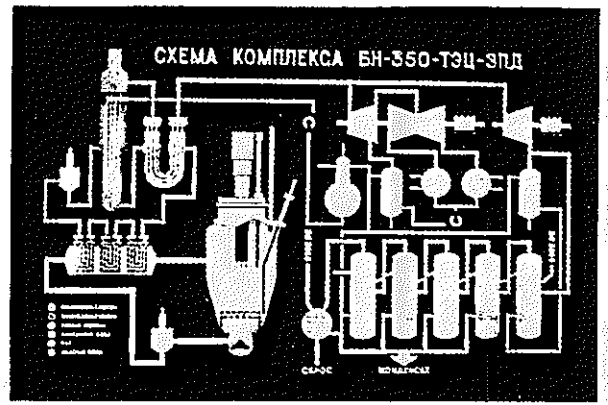








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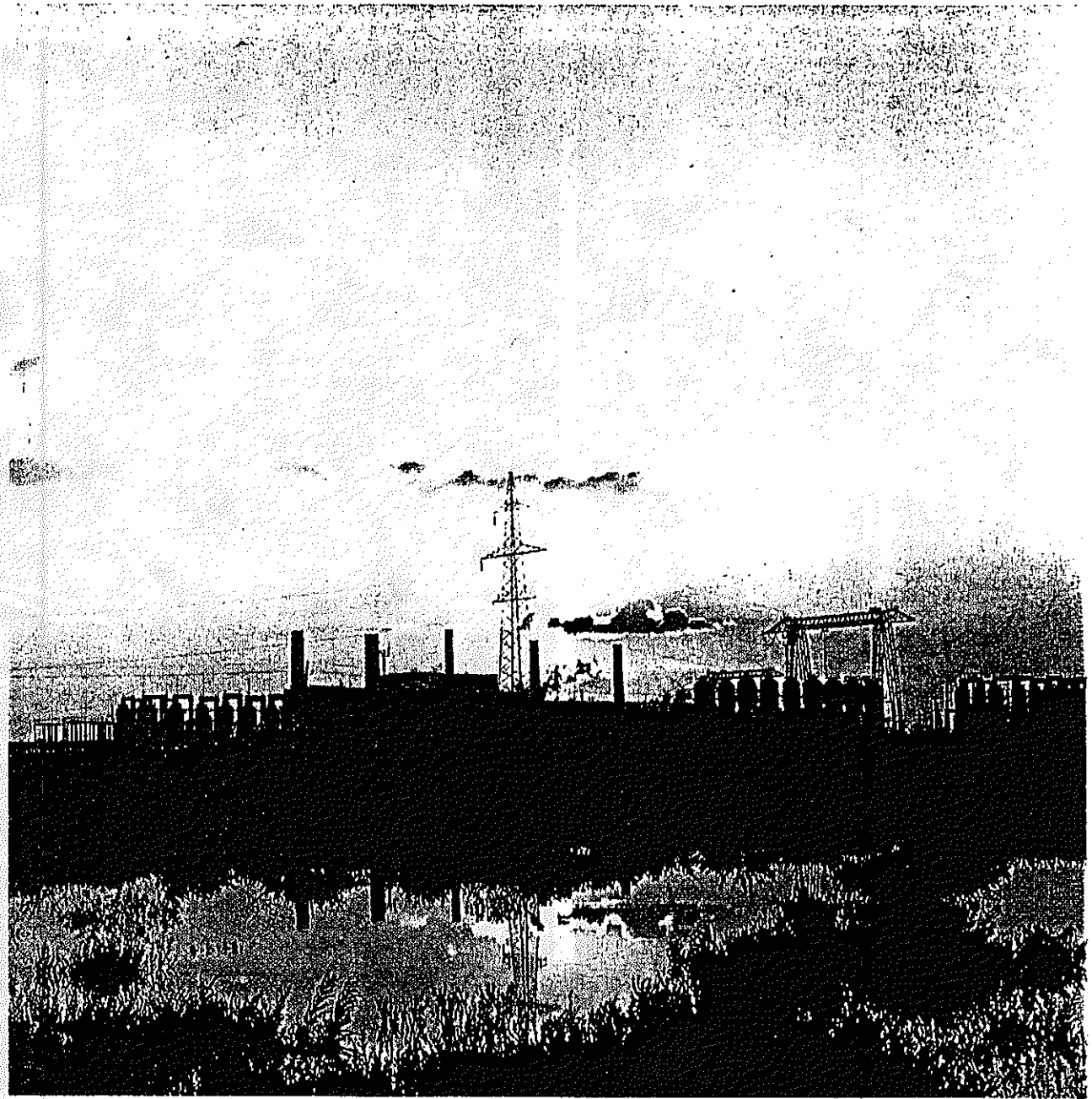

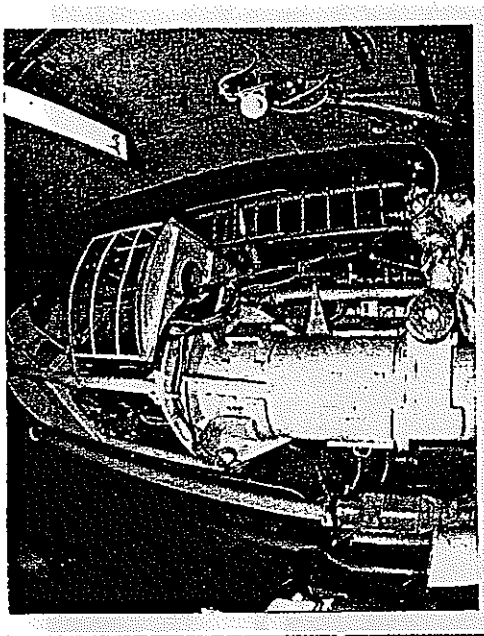

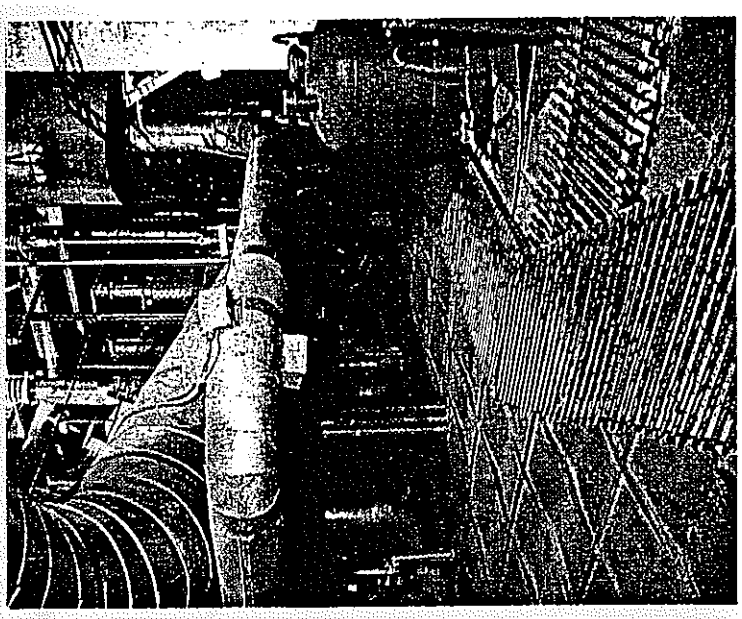

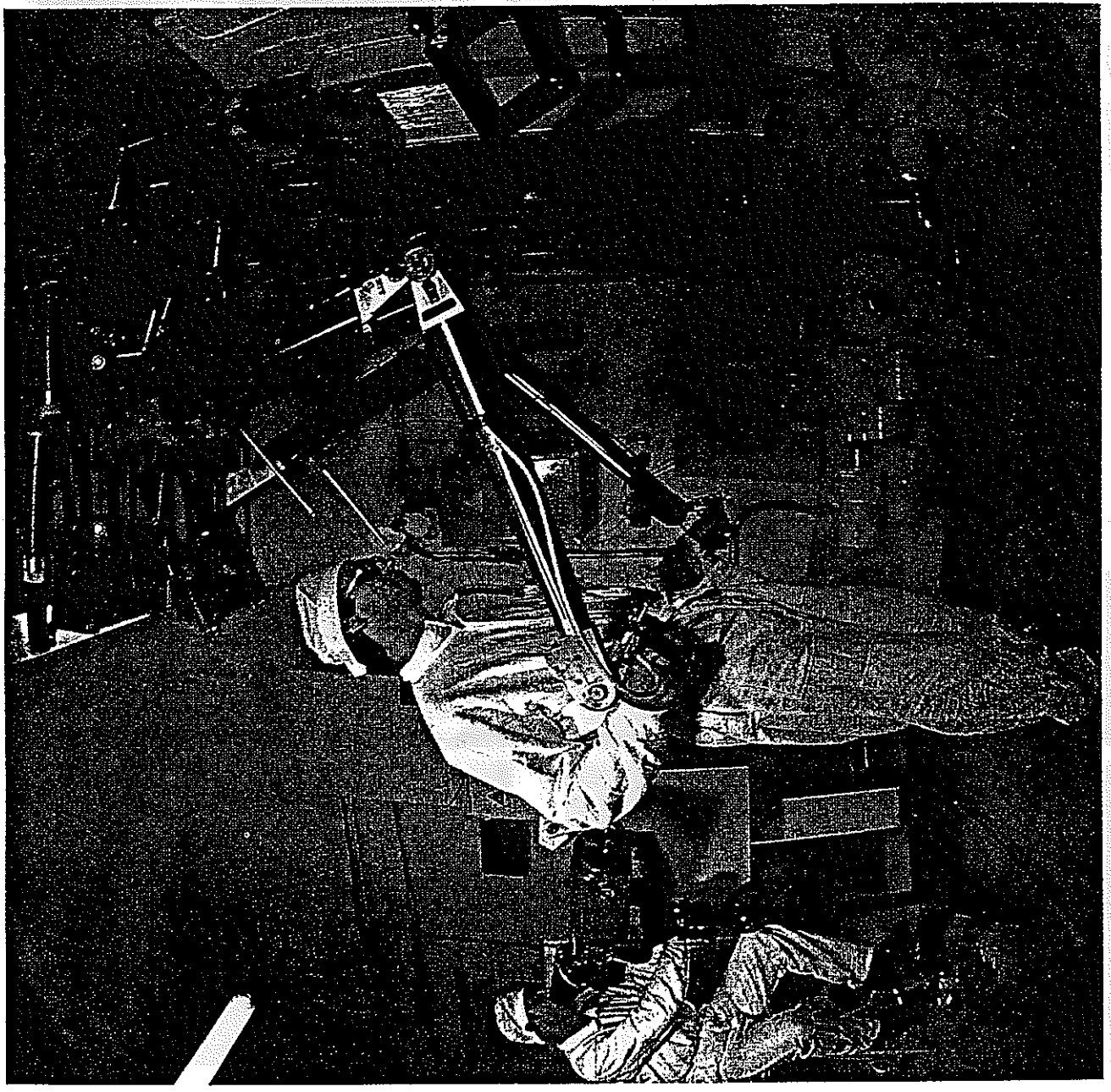

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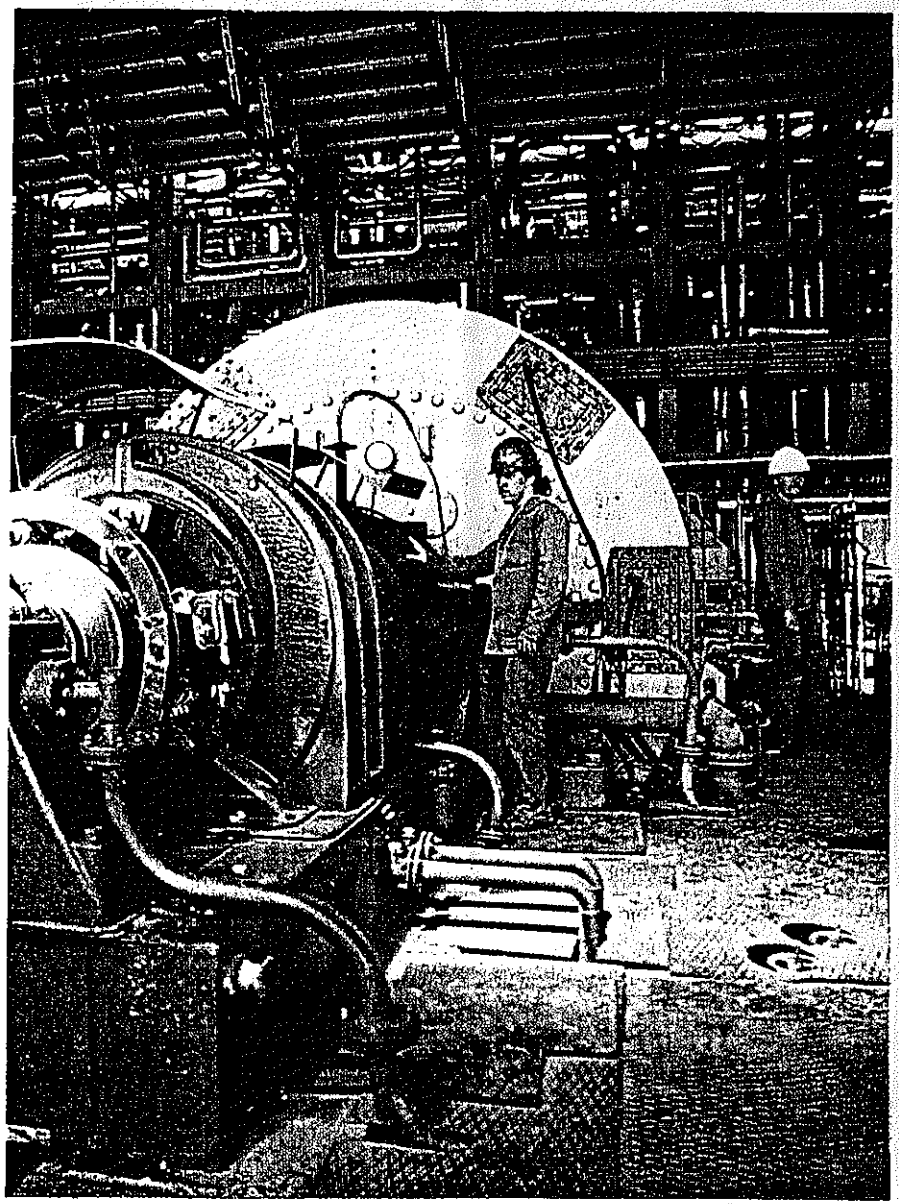

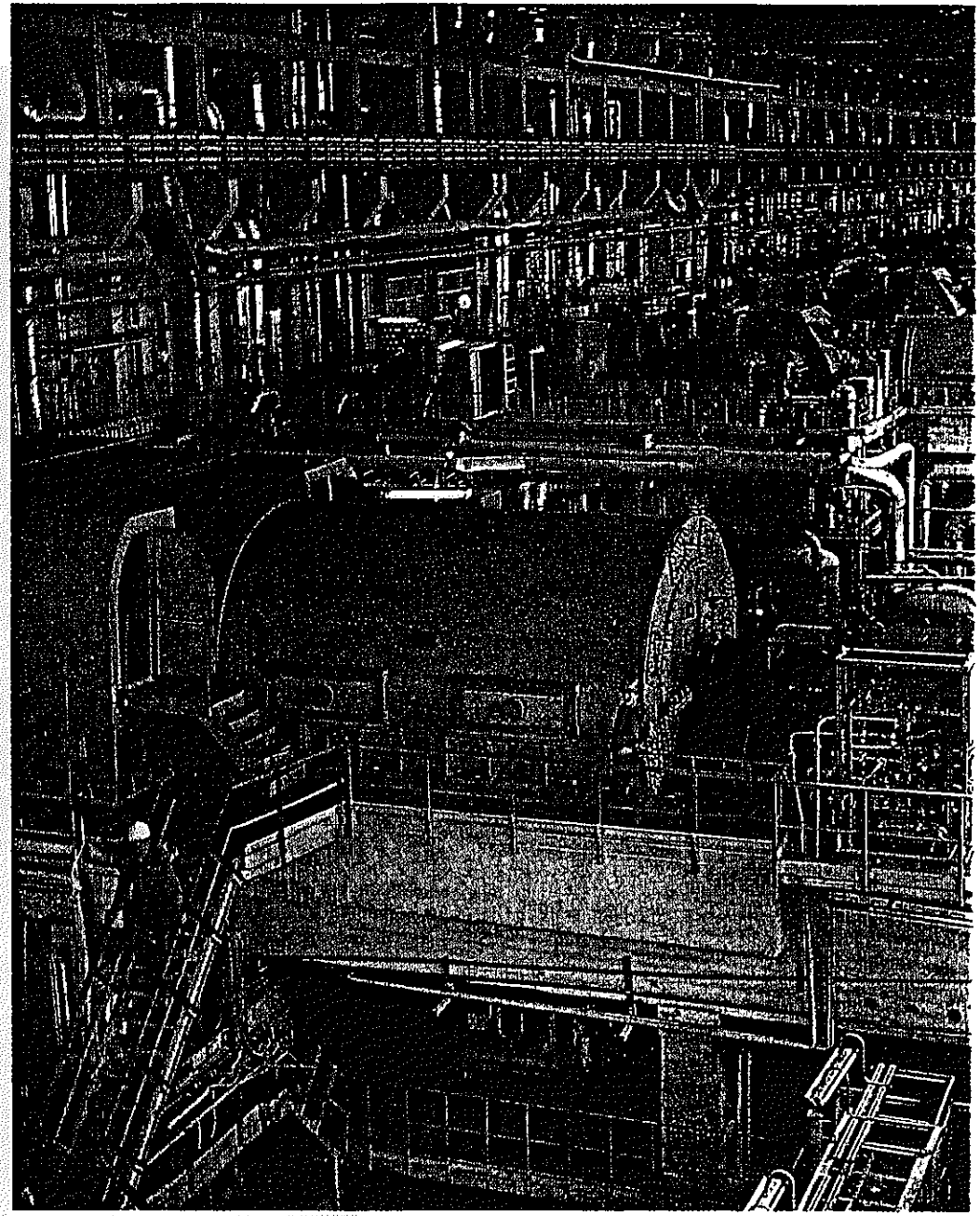

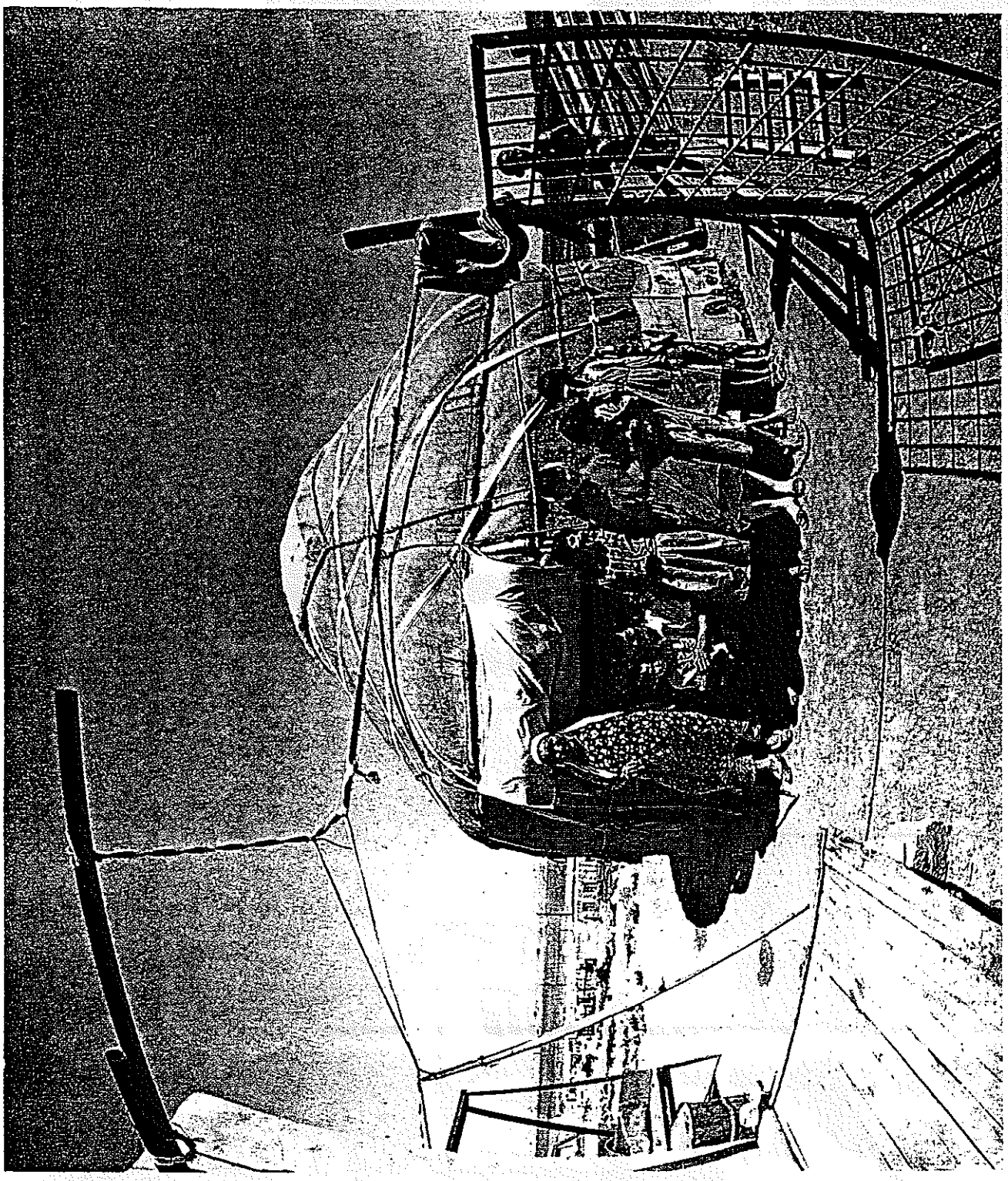

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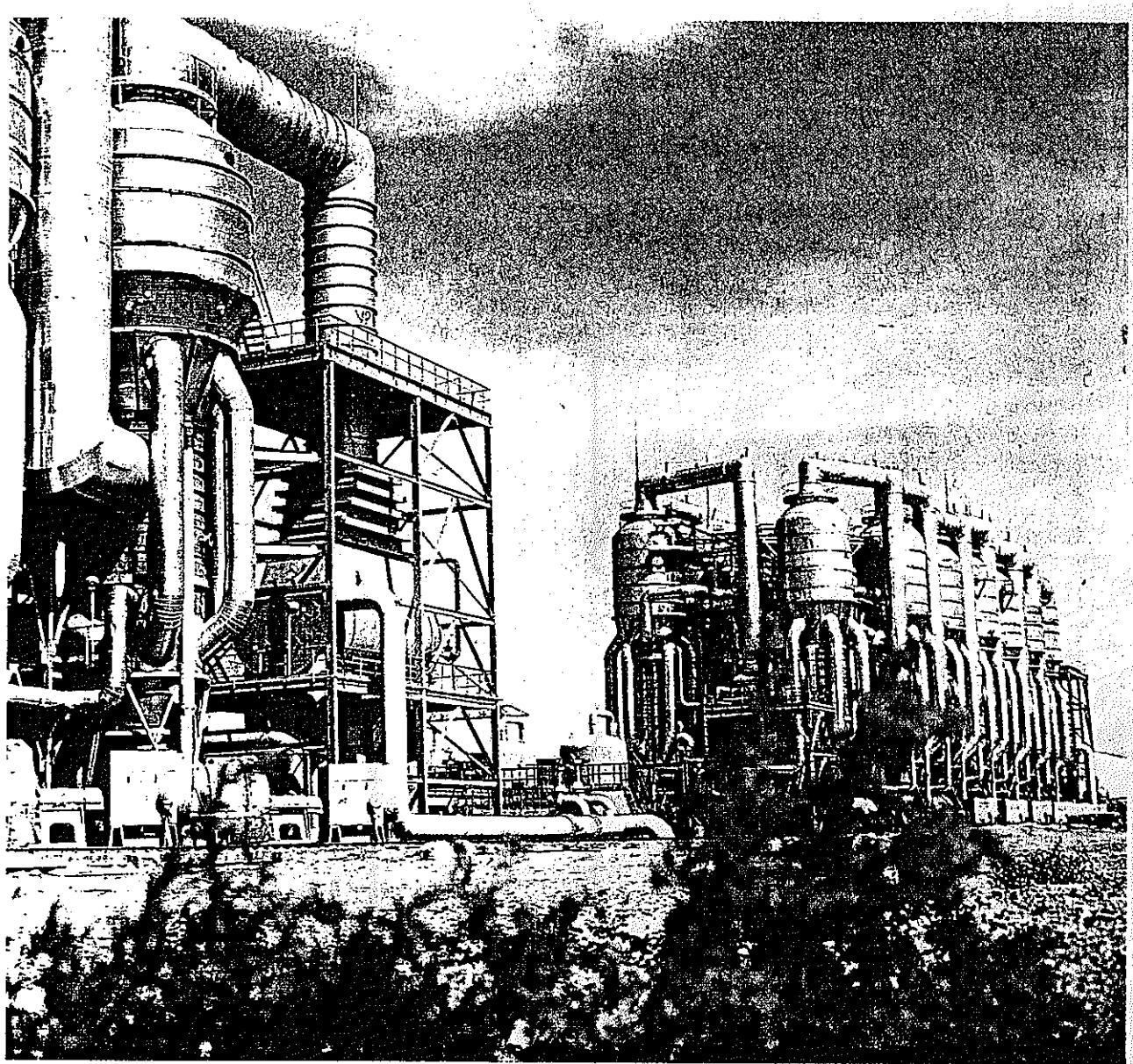

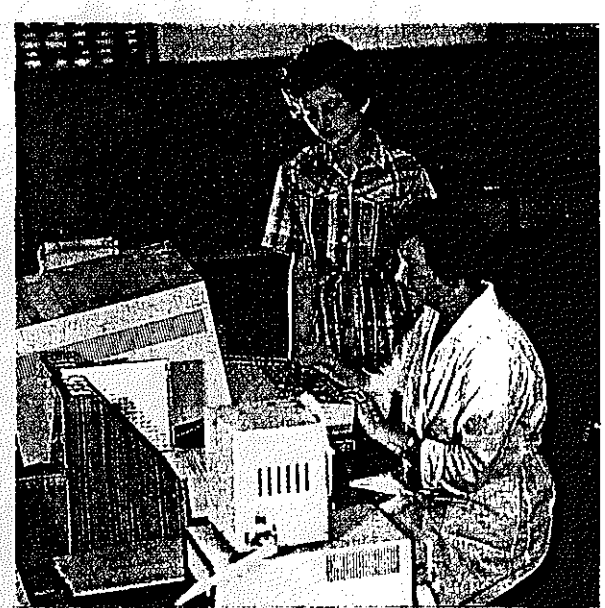

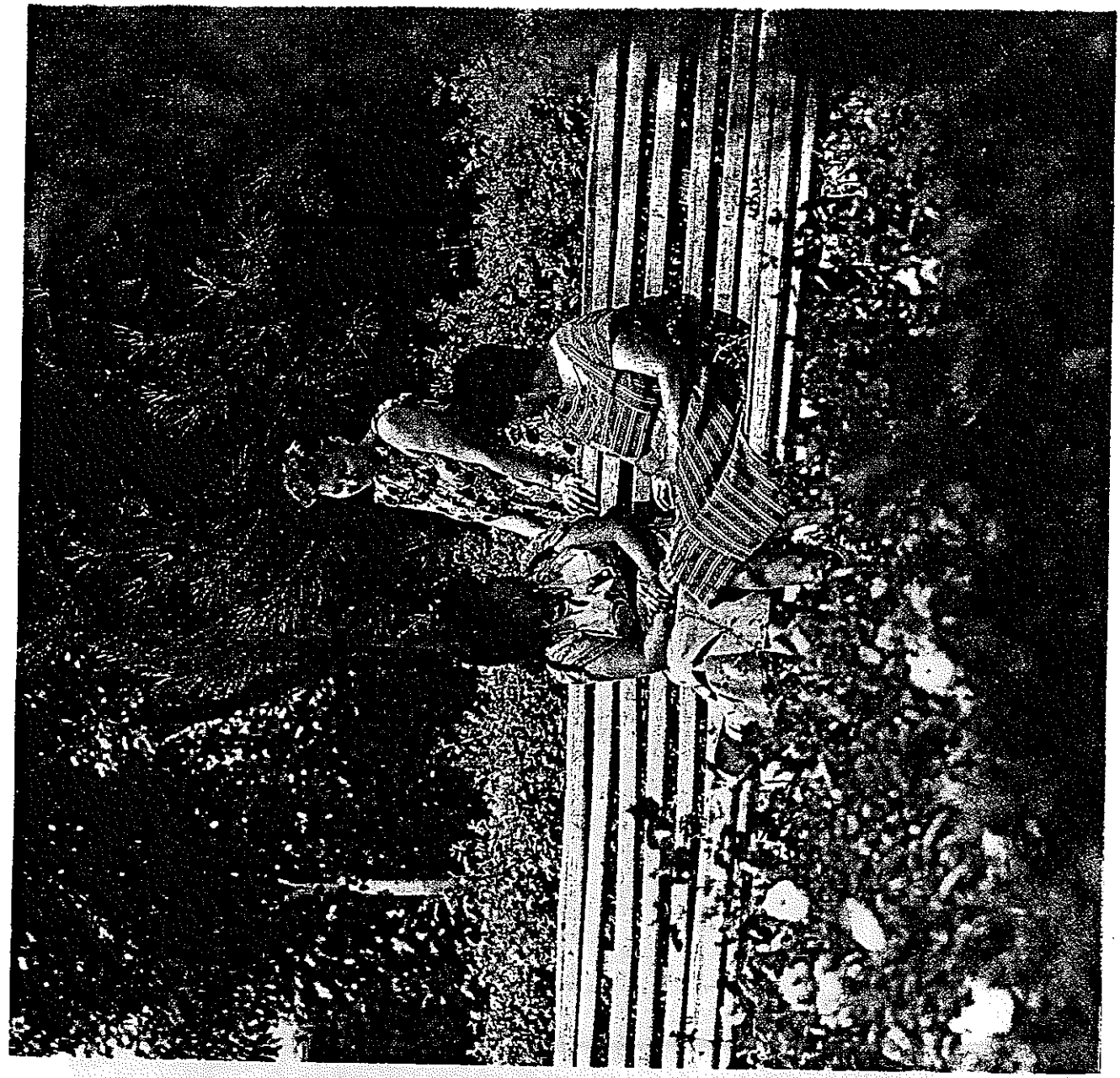

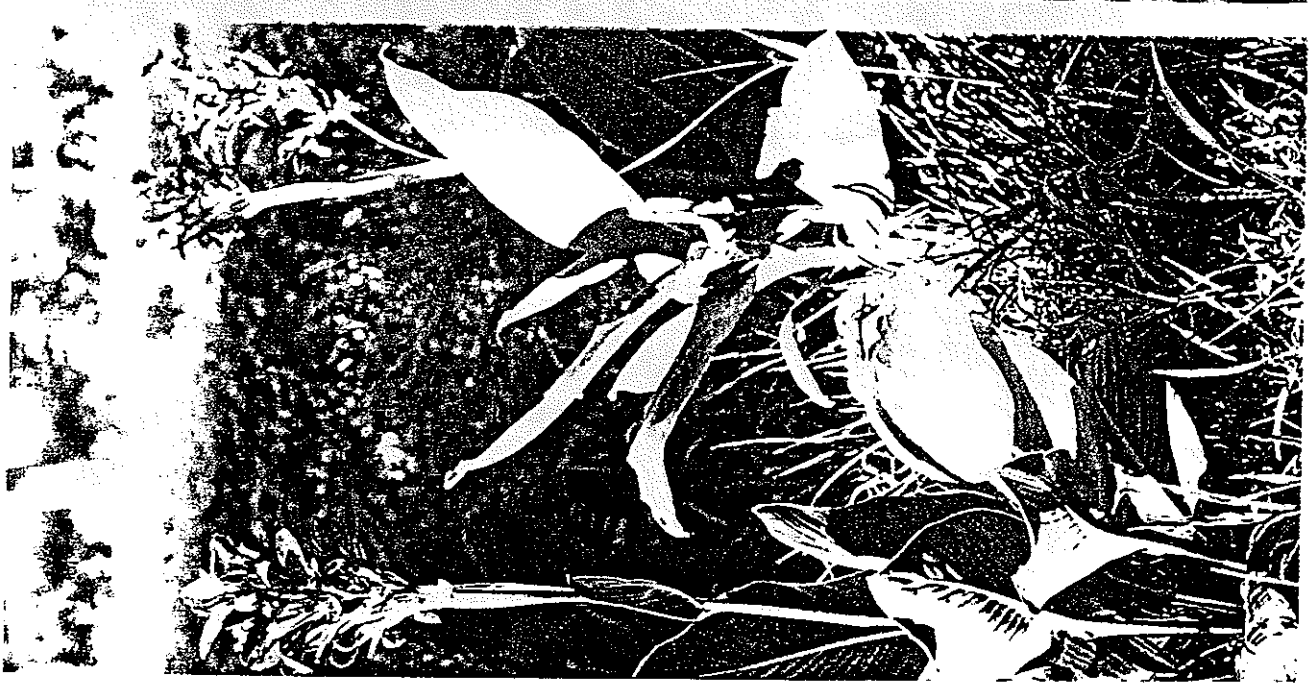

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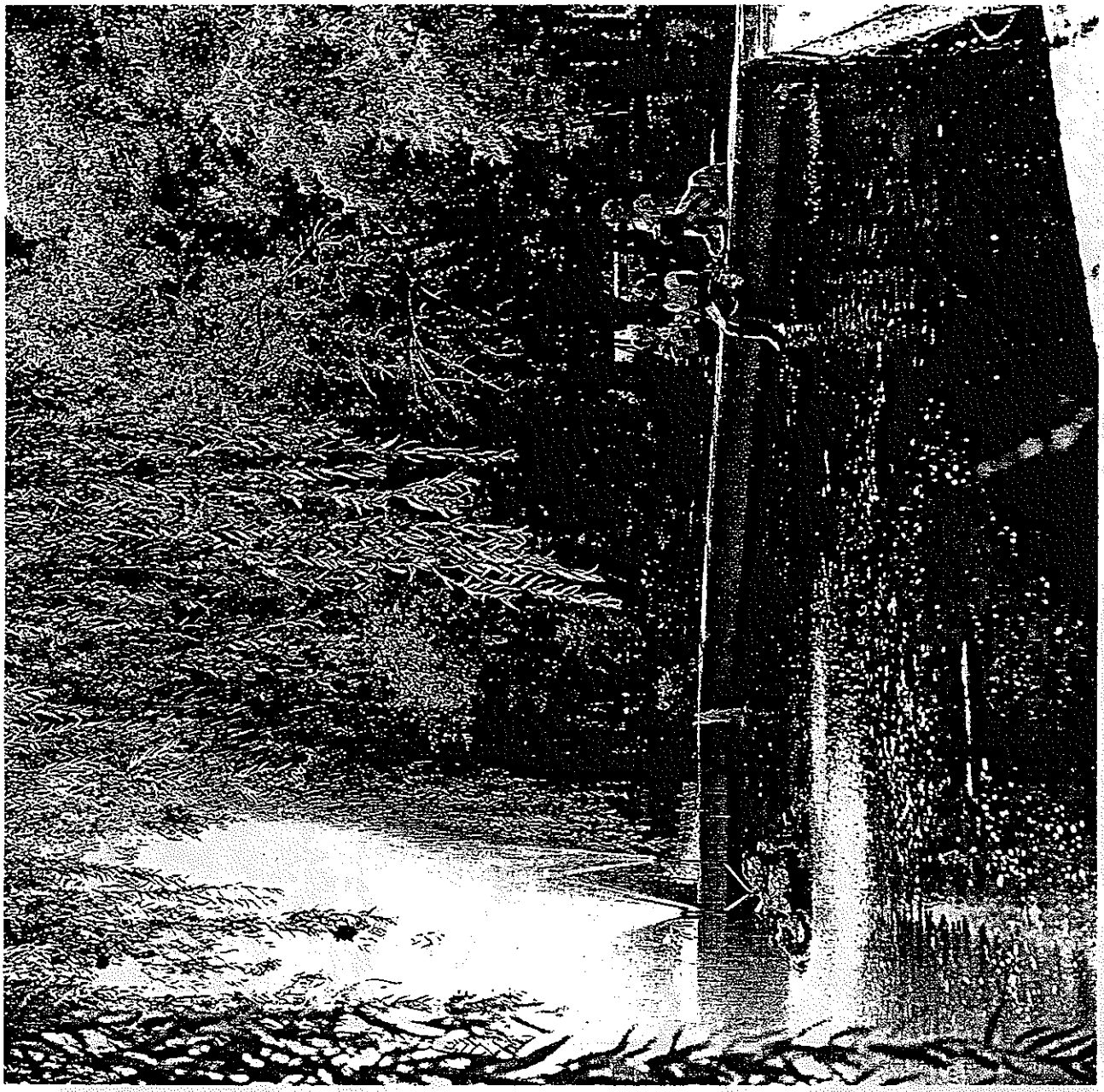

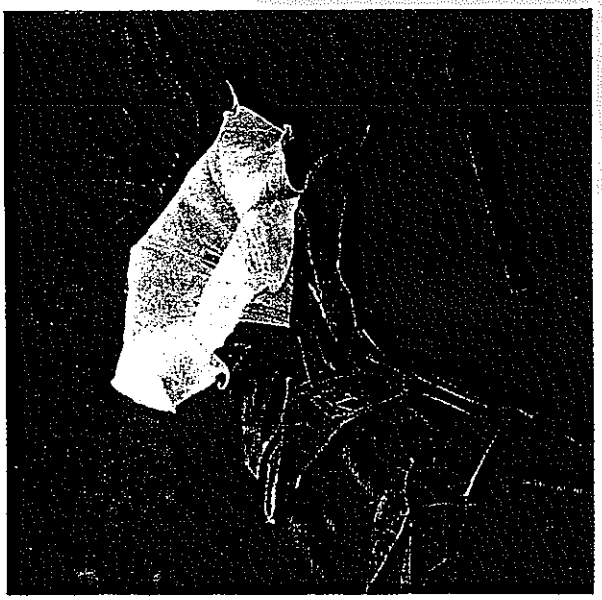

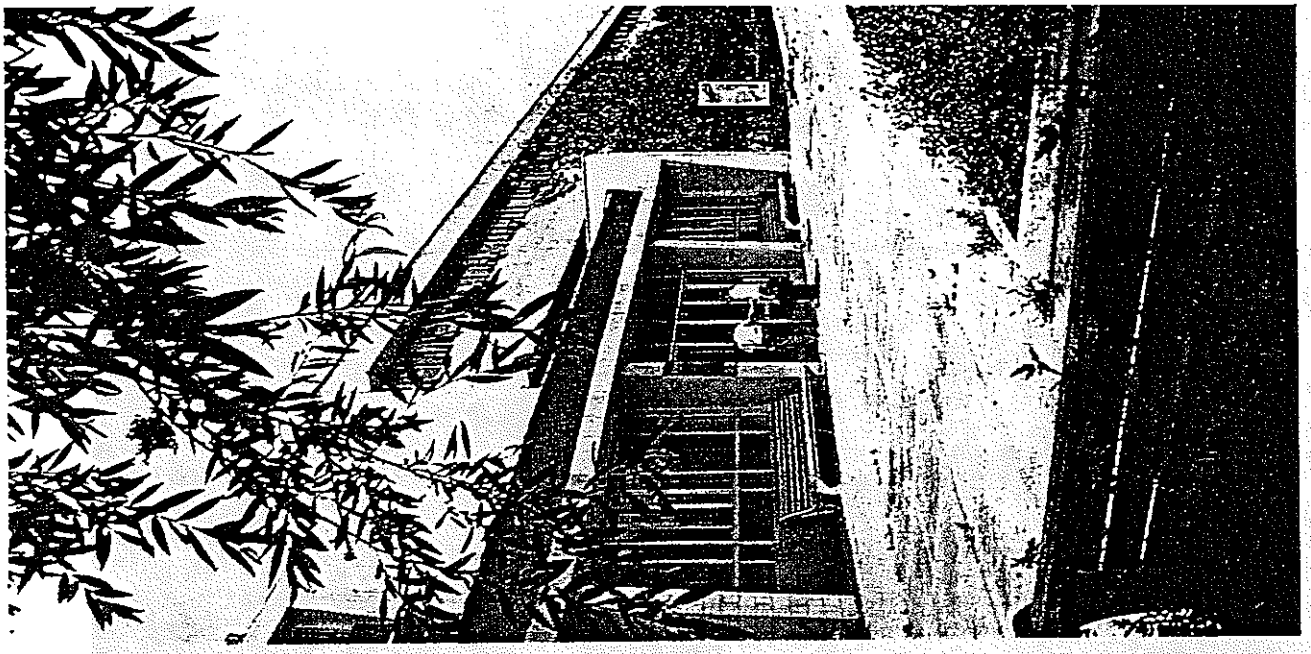

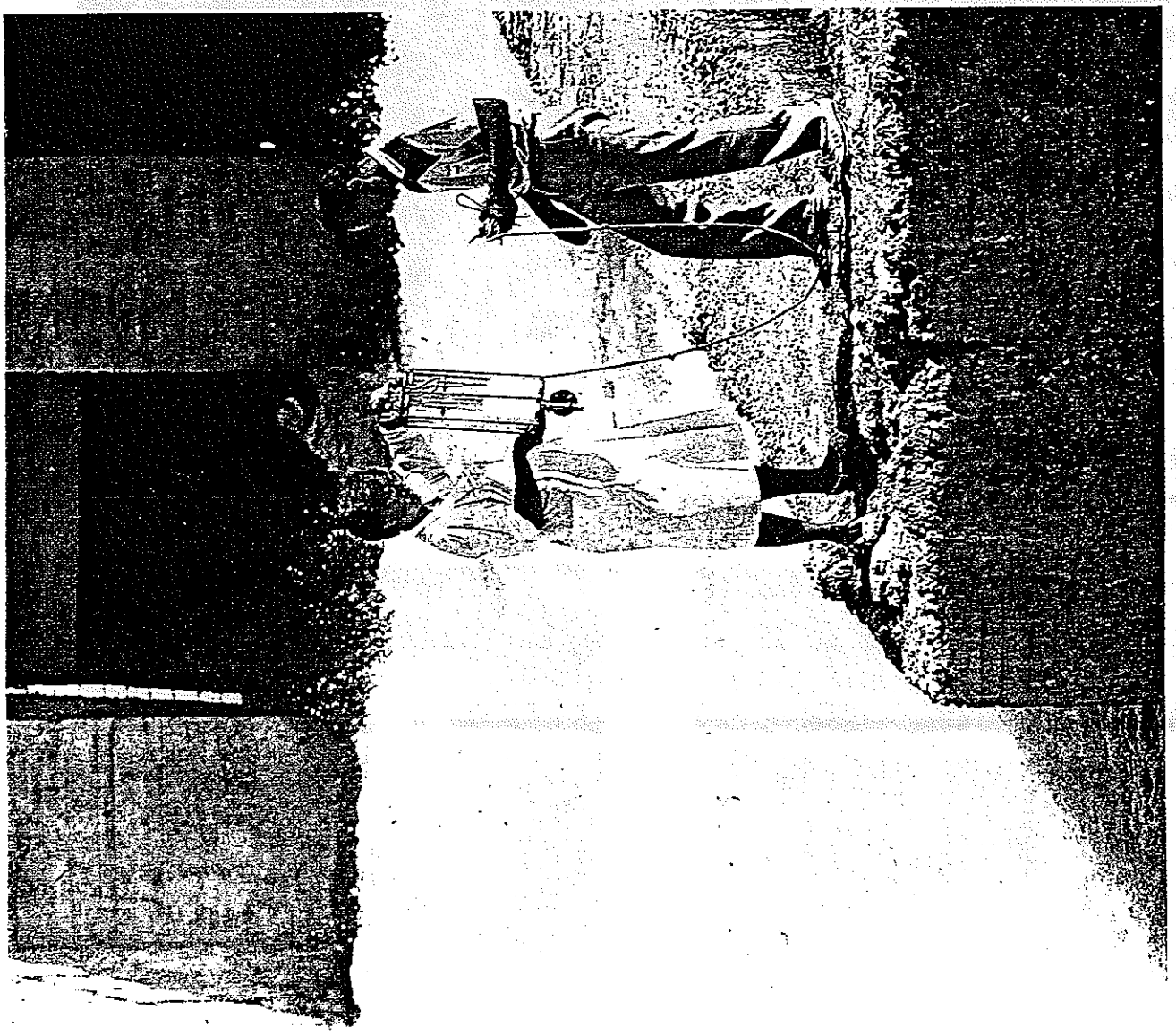

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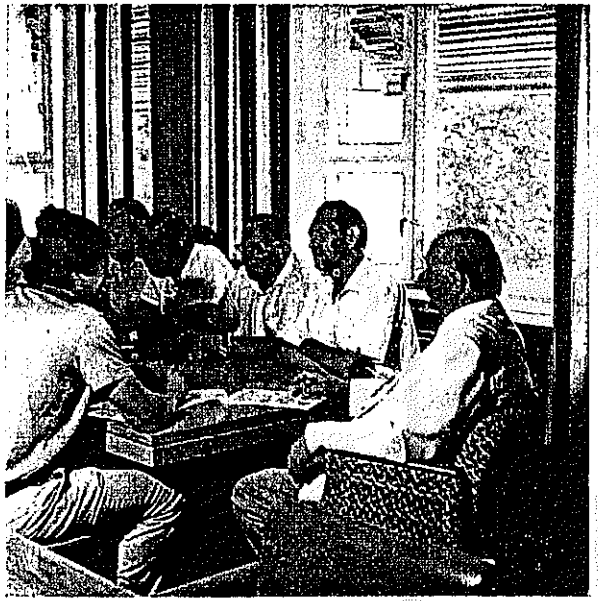

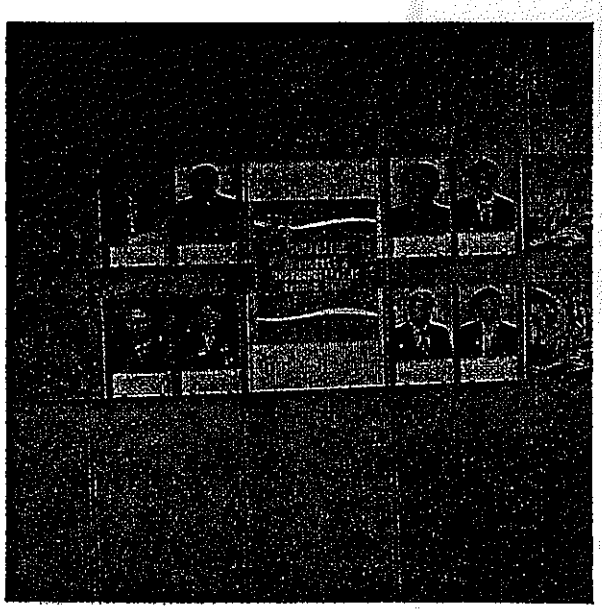

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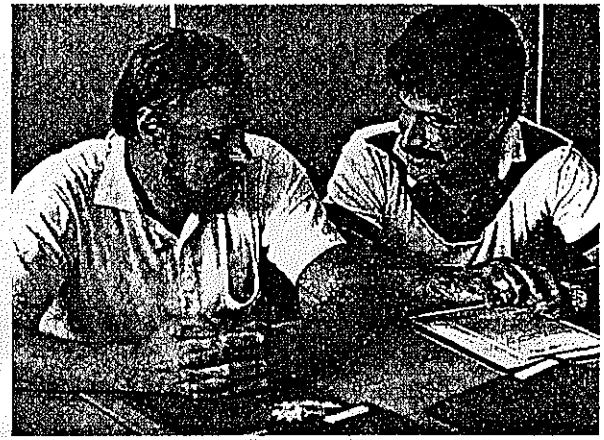

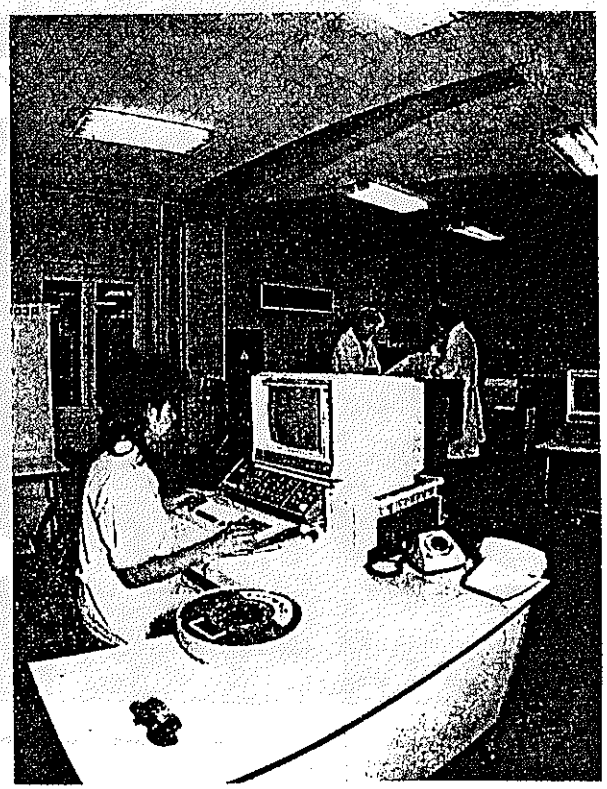

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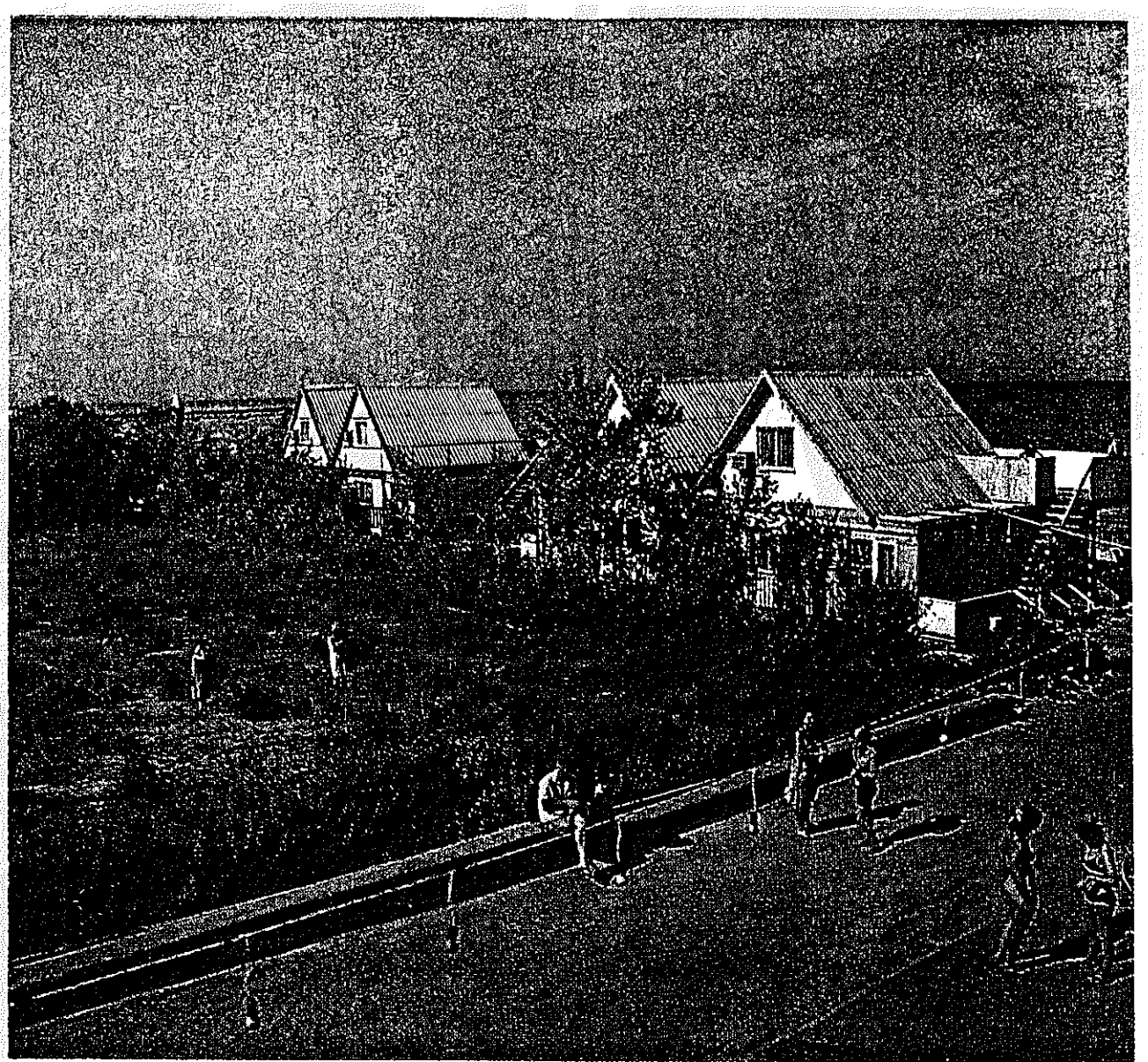

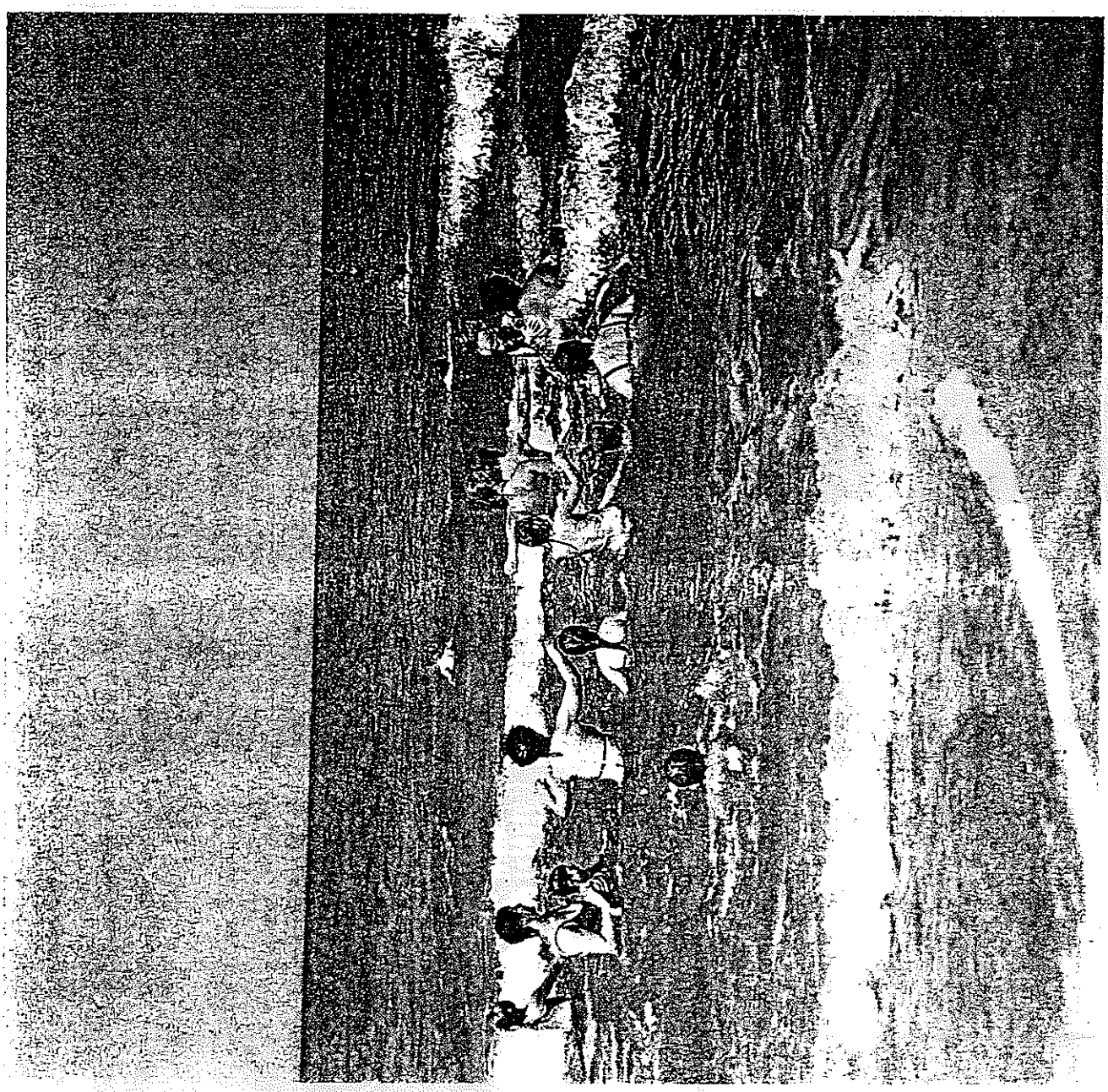

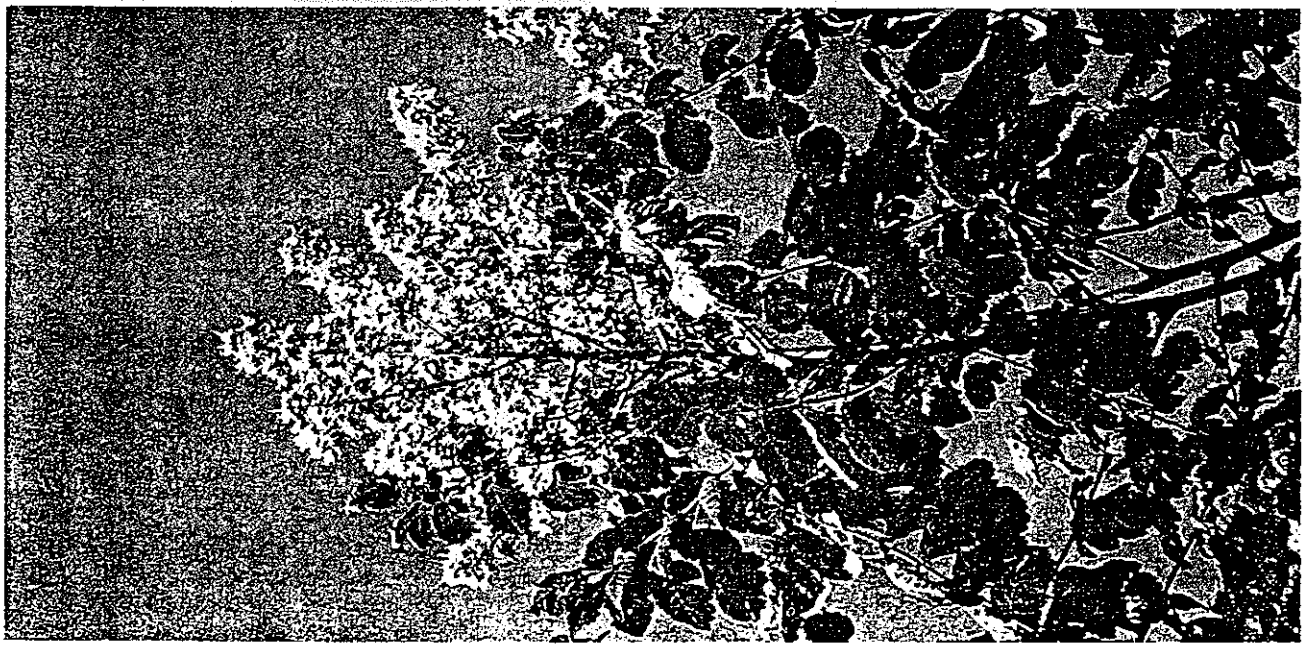

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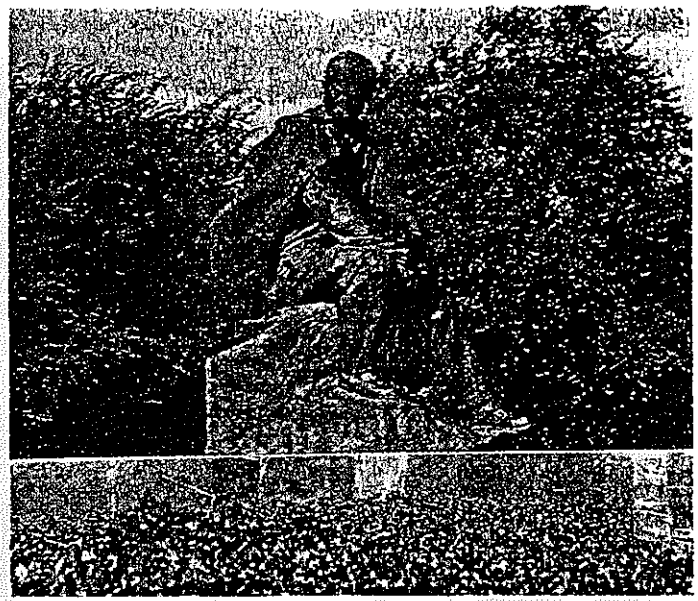

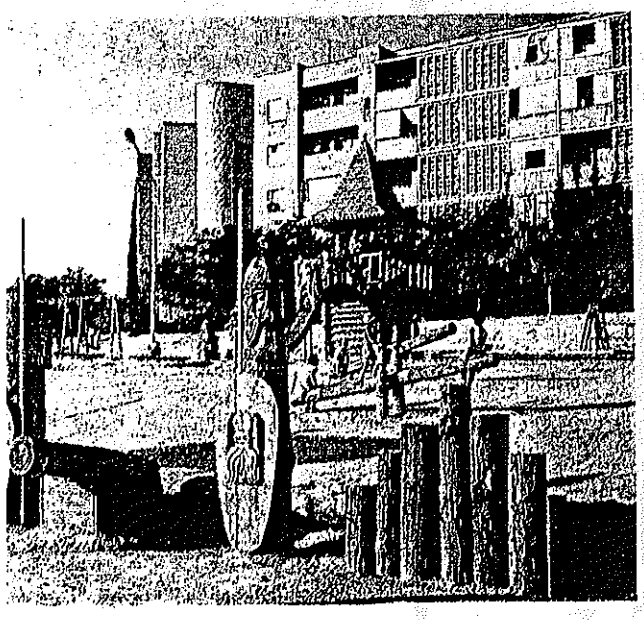

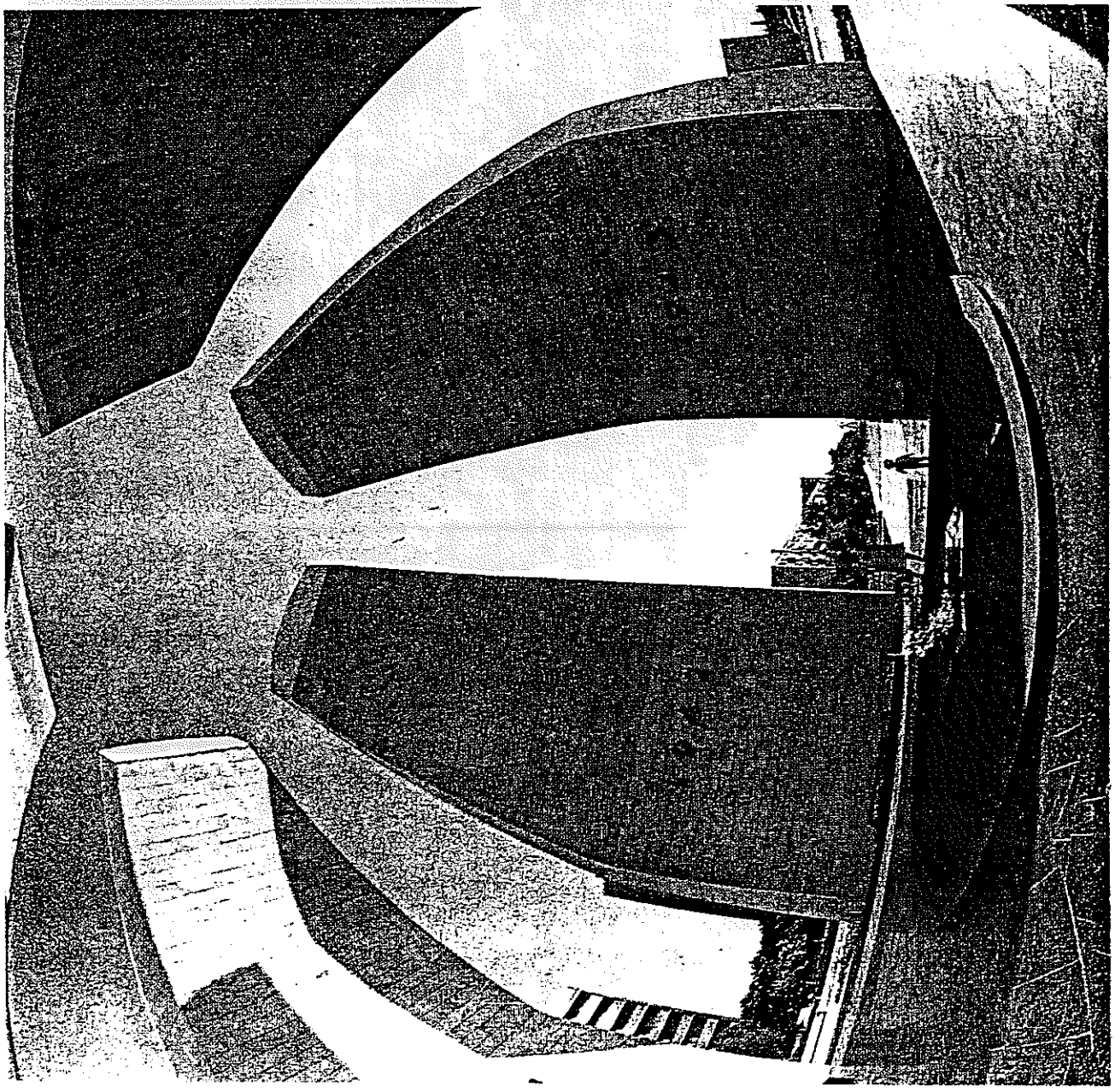

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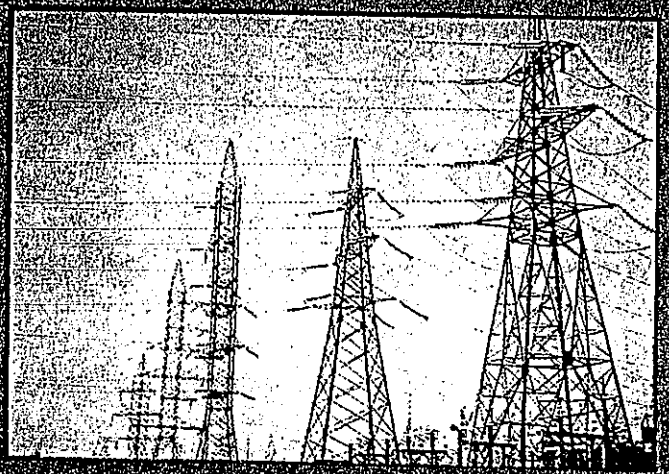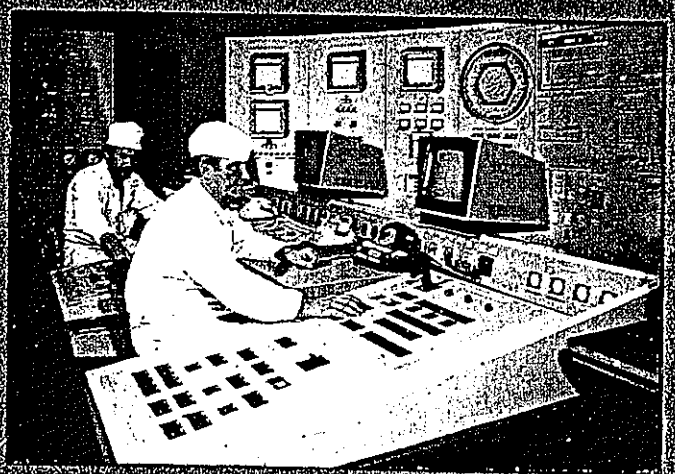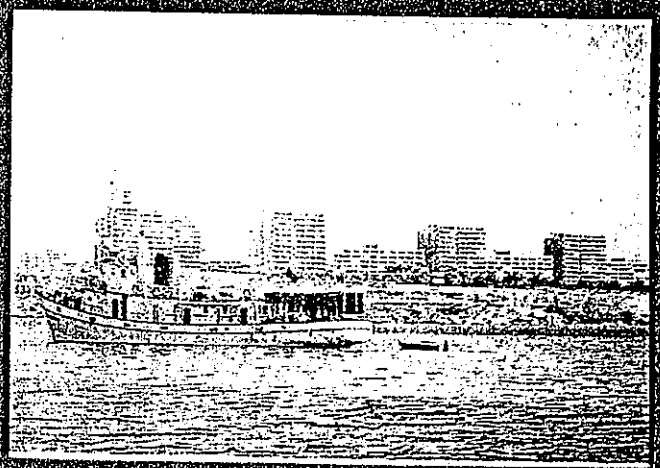
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