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第31回IAEA/IWGFR定例年会報告

1998年5月

動力炉・核燃料開発事業団

第31回IAEA / IWGFR定例年会報告要旨

平成10年5月20日
動力炉開発推進本部　技術開発室　山下　英俊

平成10年5月12日～14日、ウィーンのIAEA本部で開催された第31回IAEA / IWGFR定例年会に、日本委員として出席した。出席国（人数）は、中国（1）、フランス（2）、ドイツ（1）、インド（1）、日本（1）、カザフスタン（1）、韓国（2）、ロシア（2）、イギリス（3）の9ヶ国及びIAEA（5）であった。今回、ブラジル、イタリア、スイス、アメリカは欠席した。

本年会では、IAEAの高速炉に関する1997年の活動のレビュー、1998～1999年の活動計画の審議・調整を行うとともに、メンバー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。日本からは、高速増殖炉開発の在り方に関する議論の結果、動燃改革の状況について報告するとともに、1997年4月から1998年3月の期間における「常陽」、「もんじゅ」、実証炉開発の状況及び関連する研究開発概況をレビューした。

今後の活動計画では、「高速炉における炉物理評価、及び炉内・炉外中性子束の測定」を候補テーマとするIAEA主催の技術会議を、1999年秋頃、日本で開催する予定となった。

海外の主な状況は以下のとおり。

○イギリス、ドイツは、CAPRA計画に参加し基礎的な研究を進めている。また、原型炉PFRやSNR-300の廃止措置を進めている。

○フランスは、フェニックスの第50サイクル運転の準備を行うとともに、CAPRA計画の見直しやスーパーフェニックスの廃止計画の検討を進めている。放棄決定の影響を受け、EFRの設計成果とりまとめを1998年末までに行う。

○ロシアは、実験炉BOR-60で振動充填燃料を装荷し照射を進めている。特にBN-600は、ロシア国内の原子力プラントの中で最高の稼働実績をあげている。

○インドは、1997年7月に実験炉FBTRの発電運転に初めて成功し、今後出力を上昇させていく予定。原型炉PFBRは2001年の建設開始を目標に詳細設計を実施中。

○中国は、1997年8月に実験炉CEFRの基本設計を終了し、現在詳細設計を実施中。1999年4月のコンクリート打設開始、2003年7月の初臨界達成を計画している。

○韓国は、原型炉KALIMERの概念設計研究を行っているが、1997年7月に計画を見直し、建設完了予定を2010年代中頃に修正した。

次回、第32回定例年会は、1999年5月18日～19日、ウィーンのIAEA本部で開催されることとなった。

第31回IAEA/IWGFR定例年会報告

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第31回IAEA / IWGFR定例年会報告

1. 開催期間：1998年5月12日（火）～14日（木）

2. 開催場所：オーストリア、ウィーン　IAEA本部　C-07VI会議室

3. 出席国　：中国（1）、フランス（2）、ドイツ（1）、インド（1）、
（人数）　日本（1）、カザフスタン（1）、韓国（2）、ロシア（2）、
イギリス（3）及びIAEA（5）
今回、ブラジル、イタリア、スイス、アメリカは欠席した。
なお、フランス委員、ロシア委員、イギリス委員に交替があった。
（添付資料－2　参照）

4. 議事概要

イギリス委員のAinsworth(BNFL：今回で退任)が議長を務めた。議事の概要を以下に示す。

4.1　1997年におけるIWGFR活動成果のレビュー（添付資料－3　参照）

（1）1997年に開催された技術会議

①「新型炉の炉心・燃料構造材料挙動の照射影響」

1997年6月16日～21日、ロシア、オブニンスクにおいて開催。フランス、イギリス、ドイツ、中国、インド、ロシア、IAEAより29名が参加。日本からの参加はなかった。

今後MOX燃料のスウェリング、ガス放出、過渡時・事故時の挙動等をテーマとした専門家会合を準備することが提案された。また、核融合炉や軽水炉から要求される高照射場としても高速炉の役割は大きく、高速炉を使って継続的な照射を行えるよう努力していく重要性が指摘された。

②「通常運転時及び廃止時におけるナトリウムの処理・処分」

1997年11月3日～7日、フランス　Aix-enプロバンスにおいて開催。ブラジル、中国、フランス、ドイツ、インド、イタリア、日本、韓国、ロシア、イギリス、アメリカが参加し、ナトリウムの除染、処分、廃止措置等に関する29件のペイパーが提出された。

1999年に「高速炉の廃止措置に関する技術会議」を設定すること（アメリカは、非公式に1999年春にANLでの開催を提案した）、1998年に「BN-350プラントの廃止措置に係る技術オプション」に関するコンサルタント作業会を計画する提案がなされた。

(2) コンサルタント作業会

①「BN-800ゼロボイド炉心のULOFに関するIAEA/CEC共同ベンチマーク」

ベンチマーク第2フェーズの進捗確認を目的として1997年6月30日～7月3日、ブラッセルにおいて、1997年12月11日～12日、ウィーンにおいて開かれた。ロシア、フランス、ドイツ、インド、日本及びEUが検討に参加している。日本はSAS-4Aを用いてULOF事象の過渡解析等に貢献している。

②「高速炉の運転経験と将来プロジェクトへの反映」

1997年12月17日～19日、高速炉の運転経験をレビューし今後解決していくべき技術課題を摘出すること等を目的としてウィーンで開催された。カザフスタン（BN-350)、ロシア（BN-600）、イギリス（PFR)及びIAEAが参加した。日本からの参加はなかった。

③「BN-350の余寿命期間中の安定運転と廃止措置プログラム開発に関する支援」

1997年10月24日～25日、ウィーンで開催され、フランス、ドイツ、カザフスタン、ロシア、IAEAが参加。廃止措置や安全運転継続に係る技術的支援、資金的支援の項目がIAEAに提案された。

(3) 共同研究

①「免震構造に関する解析手法の相互比較研究」

1996年から1999年までの4カ年計画でスタートした。

第3回会合は、「免震技術に関するPost-SMiRT Seminor」と合わせ、1997年8月25日～29日、イタリア、Taorminaで開かれ、インド、イタリア、日本、韓国、ロシア、イギリス、アメリカ、EC及びIAEAが参加した。今回は参加国が開発している免震解析コード、解析手法に関する議論が行われた。種々の免震タイプの実験データの提供分担、解析分担及びスケジュールが確認された。

②「実験データを用いた高速炉用構造・熱流動コードの検証と調和」

フランス（FRAMATOM）提供のデータ（フェニックスで90000時間運転後に欠陥が発見されたT継ぎ手部）を基に、各国が有するコードで「２次系におけるT継ぎ手」のベンチマーク問題について計算と評価を行うことを目的とし、1996年に3年計画として活動が開始された。

第2回会合は1997年10月6日～10日、ロシア、オブニンスクで開かれ、フランス、インド、日本、韓国、ロシア、イギリスが参加し、各国の解析手法の差違を定量化するとともに、フェニックスで測定された実現象と比較するための検討・議論が行われた。

最終会合は、1998年4月20日～22日、フランス、リヨンFramatome-Novatomeで開催される。

4.2　1998年のIWGFR活動計画

(1) 1998年に開催予定の技術会議準備状況

①「実炉条件における燃料・吸収ピンと集合体、熱流動パラメータの解析のための手法と解析コード」（添付資料ー4　参照）

1998年7月27日～31日、ロシア、オブニンスクで開催する。ロシア政府の受け入れが確認され、公式案内が出された。参加申し込みの期限は6月20日まで。

②「高速炉運転中における異常な事象：原子炉システム経験のレビュー」

1998年11月9日～13日、ウィーン　IAEA本部の開催を予定する。議論のテーマは、コンサルタント作業会「高速炉の運転経験と将来プロジェクトへの反映」（1997年12月17日～19日）で摘出された事項を中心に行う計画。現在会議開催案内を事務局で準備中。

(2) コンサルタント作業会

①「BN-800ゼロボイド炉心のCDAに関するIAEA/CEC共同ベンチマーク」

本作業のレポートドラフトの検討を行う最終作業会を1998年6月29日～7月3日、ロシア、オブニンスクで開く予定。

(3) 共同研究

①「免震構造に関する解析手法の相互比較研究」

解析結果と実験結果の報告、議論を行う第4回会合を1998年5月の最終の週にに、イギリス、HertfordのTARRCで開催する予定。

②「実験データを用いた高速炉用構造・熱流動コードの検証と調和」

最終会合を1998年4月20日～22日、フランス、リヨンFramatome-Novatomeで開催。

(4) アドバイザリーグループ会合

「舶用原子炉技術の民生適用」（添付資料ー4　参照）

舶用炉の経験と反映、系統・機器設計、燃料交換なしの中小型炉概念、新型舶用炉の概念・性能目標・コスト・開発、舶用炉の民生適用の見通し等をトピックスに1998年7月20日～24日、ロシア、オブニンスクで開催。

参加申し込みの期限は6月20日まで。

4.2　1998年のIWGFR活動計画

(1) 1998年に開催予定の技術会議準備状況

①「実炉条件における燃料・吸収ピンと集合体、熱流動パラメータの解析のための手法と解析コード」（添付資料－4　参照）

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①「免震構造に関する解析手法の相互比較研究」

解析結果と実験結果の報告、議論を行う第4回会合を1998年5月の最終の週にに、イギリス、HertfordのTARRCで開催する予定。

②「実験データを用いた高速炉用構造・熱流動コードの検証と調和」

最終会合を1998年4月20日～22日、フランス、リヨンFramatome-Novatomeで開催。

(4) アドバイザリーグループ会合

「舶用原子炉技術の民生適用」（添付資料－4　参照）

舶用炉の経験と反映、系統・機器設計、燃料交換なしの中小型炉概念、新型舶用炉の概念・性能目標・コスト・開発、舶用炉の民生適用の見通し等をトピックスに1998年7月20日～24日、ロシア、オブニンスクで開催。

参加申し込みの期限は6月20日まで。

4.3　1999年以降のIWGFR活動計画

(1) 1999年に開催する技術会議テーマの選定（添付資料－5　参照）

日本からの提案として「破損燃料ピンの運転性」（1985年に提案）及び「高速炉の信頼性評価」（1991年に提案）が候補テーマとして残されていたが、これらの提案を取り下げるとともに、新たに「実プラントにおける炉物理試験の評価」を提案した。また、インドから新たに「高速炉における炉外（周辺）中性子フラックスの計測と予測」が提案された。

議論の結果、IWGFRとしては、日本の提案にインドの提案も含めた「高速炉における炉物理評価及び炉内または炉外におけるフラックスの測定」をテーマとする技術会議の開催を予定することとなった。

提案国である日本の受け入れについて打診があり、開催時期としては1999年10月～11月頃を考えたい旨を非公式に回答した。

(2) 1999年以降のIWGFR活動計画（添付資料－6　参照）

IAEA事務局より、新たな共同研究計画として下記2つのテーマの提案があった。各委員とも重要な研究テーマであるとの共通認識のもと、IWGFR活動計画案を原則的に了承し、サポートすることとなった。

① 「高速炉における反応度効果の計算不確かさを減じるためのコード、解析手法の改良」　1999年～2003年

高速炉の安全解析に必要なドップラー係数、ナトリウムボイド係数等を解析と実験から検証、改良を図り不確実さを減じることを目的とする。

② 「放射性廃棄物の効率的燃焼に向けた原子炉のオプション」2000年～2003年

2010年までに、3000トンのPu、100トンのNp・Am、250トンのTc、90トンのCs、60トンのIを含む使用済み燃料が蓄積される見通しがある。これらのMA・FPを効率的に燃焼又は核変換し、放射性廃棄物を減らすために、原子炉、加速器、溶融塩炉等様々な選択肢の技術的な検討を行う。

4.4　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し、IAEA技術レポートとしてメンバー各国に既に配布されている。前回会合では、IWGFRよりIAEA事務局に、これを有効に活用するためにコンピュータデータバンク化を図るよう要請していたが、事務局の作業を終了した旨報告があった。

下記により高速炉データベースにアクセスできる。

Address	: http://www.ippe.rssi.ru/IAEA/IWGFR/FRDBNew/index.html 又はhttp://frdb.ippe.rssi.ru/
User name	: frdb
Password	: breeders

5. 各国における高速炉開発状況

各委員より、報告があった開発概況は以下のとおり。一カ国あたりの報告と質疑持ち時間を30分に制限された。詳細は各国提出の資料（添付資料－７）を参照されたい。

【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年末に国家計画委員会（SPC）の承認が得られた。1995年初めに開始された基本設計は1997年8月に終了し、現在は詳細設計を実施している。

現在のマイルストーン

・1998年4月：コンクリート打設開始、
・2000年末　：原子炉建物建設完了
・2003年7月：初臨界

建設用地として、中国原子能科学研究院（北京の西方50kmの敷地に隣接する土地10haを地方自治体より新たに追加取得した。

なお、機器の製作発注を既に開始しているが、中国においてはナトリウム中における動的機器の経験が乏しいため、ポンプ、制御棒駆動機構、燃料取扱い機は輸入するとのことであった。

【フランス】

原型炉フェニックスについては、3回の常設顧問会が開催され、安全性に関するレビューの結果、1998年の1サイクル運転、その後1年かけて点検等の作業を行い、2000年～2004年までに6サイクルの運転を認める結論を出した。第50サイクル運転の開始は計画より遅延している。

スーパーフェニックスは、1997年6月に政府が正式に放棄を決定し、さらに2月の時点でこれまでの運転許可のデクレが取り消されたため、1997年は運転されなかった。このため、知識獲得プログラム（PAC）の進捗に影響している。

スーパーフェニックスの廃止については、IAEA分類のレベル1の段階まで措置するため、ナトリウムドレンと炉心燃料の取り出しを行う条件に関するデクレを1998年内に取得する予定である。現在の計画では炉心燃料の取り出しは1999年の中頃に開始し、その作業には18カ月を要する予定である。

EFR開発に関しては、スーパーフェニックス放棄決定の影響を多いに受け、プロジェクトの停止を余儀なくされている。設計研究のとりまとめを1998年末までに行うこととしている。

CAPRA計画は、1991年の廃棄物管理法に関する応えを準備すべく、従来のプルトニウム燃焼に加え、MAや超寿命FPの燃焼・変換に重点をおく計画に見直された。

【ドイツ】

高速炉に係る研究は、FZKに限定され、欧州諸国および日本とのCDA解析コード（SIMMER-III、SAS4A）の共同開発、CAPRA計画、BN-800過渡事象解析作業等に参加している。また、TRABANTといった照射試験を進めている。

今回は原型炉SNR-300の廃止状況についての報告はなかった。

【インド】

高速実験炉FBTRは、11.5MWtに出力が上げられ、1997年7月に初めて発電を行われ、連続22日間運転に成功した。燃焼度は目標の5万に対し、3.9万に到達した。今後さらに15MWtまで出力を上げ、運転を行う予定である。

原型炉PFBRについては、詳細設計を実施中であり、現在のところ2001年の建設開始を目指している。

【日本】

原子力委員会　高速増殖炉懇談会における高速増殖炉開発の在り方に関する議論、動燃改革検討委員会における改革の基本的方向について、概況を紹介するとともに、「常陽」、「もんじゅ」、実証炉の状況を報告した。

MK-IIIに関する炉心改造の仕様、「もんじゅ」運転再開の見通し、新法人発足の計画などに質問が集中した。

【カザフスタン】

BN-350は、1973年の発電運転開始以来1998年7月に25年目を迎える。設計寿命は1000MWt運転で20年であったが、20年間平均では600MWtで運転されてきたため原子炉容器、炉内構造物等の健全性は問題ないと評価している。1997年の運転は、崩壊熱除去のために5ループを確保し原子炉出力を最高420MWtに制限している。

1998年1月に、原子力庁（KAEA）、国立原子力センター（NNC）、大学、原子力学会等の機関から独立させて原子力技術安全センター（NTSC）を設立した。プロジェクトや安全性等に関して専門的意見を下す機関として機能する。

アメリカとの合意のもと、ANLの技術的支援を受けている。第一ステップは、BN-350プラントの使用済み燃料の長期保管と廃止措置計画の安全性に関する技術的支援を受けることになっている。

【ロシア】

実験炉BOR-60では、振動充填MOX燃料の標準集合体を装荷している。最高燃焼度は、標準集合体で15.5%h.a.、試験用集合体で26%h.a.に達している。10%Zrナトリウムボンド燃料要素を入れた集合体（1体）では8%h.a.に至っている。

原型炉BN-600の1997年中の稼働率は73%であり、国内の原子力プラントのなかで最高の稼働率を達成している。

BN-800については、ベロヤルスクと南ウラルへ建設する許可が1997年12月に下りた。南ウラルの建設は予算獲得の方向に向かっており、ベロヤルスクの建設は合同企業体を設立して継続する計画である。1997年11月、原子力省大臣とスベルロフスカヤ地方の首長との間で「ロシアにおける原子力開発計画」を改訂することに合意し、ベロヤルスク発電所の4号機としてのBN-800の建設を2005年までに完了すべきとした。

【韓国】

韓国原型炉KALIMER（Korea Advannced LIquid Metal Experimental Reactor)は、15万MWe、プール型、ウランージルコニウム合金燃料、ガス膨張機構付炉心、1次・2次とも電磁ポンプ、免震構造採用を基本的概念としており、アメリカPRISM型炉の概念が基本になっている。

1997年には概念設計フェイズを終了し、1997年8月から5カ年にわたる基本設計を経て、その後詳細設計を進め2011年には初臨界に至らしめるという計画であったが、1997年7月に計画の見直しが図られ、現在も概念設計を継続しており、基本設計フェイズも約4～5年遅れている状況にある。

【イギリス】

欧州統合高速炉EFRの開発への関与は、EFRUGへBNFLが、設計グループへ民間のNNCが、CAPRA計画及び新型炉・サイクル開発へAEA Technology、NNC、BNFLが参加している。

原型炉PFRの解体は、1995年に炉心燃料の撤去と模擬炉心の構築を完了している。ナトリウムの処理プラントを1998年9月完成の予定で建設を継続するとともに、ナトリウム処理後、低レベル液体廃棄系に移送する前のプロセスとしてセシウムを除去するプラントを建設する契約を行った。処理ナトリウム（NaK含む）量は、PFRの1次・2次ナトリウム1150トンとその他の研究施設分350トンを合わせて約1500トンである。

6. 次回の予定

次回第32回年会は、1999年5月18日～19日、ウィーンIAEA本部で開催することとなった。

31st Annual Meeting
of the International Working Group on Fast Reactors
(IWGFR)
VIC, 12-14 May 1998
C-07VI

Preliminary Programme

Topic	Organization Presenter	Time
Tuesday, 12 May 1998		
1. Opening and introductory remarks	IAEA	09:30 - 09:45
2.a Chairman's remarks		09:45 - 09:50
2.b Introduction of the participants by themselves	All	09:50 - 10:00
2.c Adoption of the Programme	Chairman\All	10:00 - 10:10
3. Progress reports on National Programmes and brief comments on areas of interest for future cooperation within the frame of IAEA IWGFR (Time for presentation should be maximum 25-30 minutes including discussions)		10:10 - 11:10
China		
Coffee break		11:10 - 11:30
Continuation item 3		11:30 - 12:30
France		
Germany		
Lunch break		12:30 - 14:00

Topic	Presenter	Time
Continuation item 3		14:00 - 15:30
India		
Japan		
Kazakhstan		
Coffee break		
Continuation item 3		15:30 - 15:50
Korea, Rep. of		15:50 - 17:30
Russia		
Switzerland		
United Kingdom		
Adjourn 1st day		17:30
Wednesday, 13 May 1998		
4. Summary of activities within the frame of the IWGFR (since the 30th IWGFR)	A. Rinejski, IAEA	09:00 - 10:00
4.a Technical Commettee Meeting (TCM) on "Influence of high dose irradiation on advanced reactor core and structural materials" (Obninsk, Russia, June 1997)		
4.b Coordinated Research Programme (CRP) on "Intercomaprison of Analysis Methods for Seismically Isolated Nuclear Structures" (Taormina, Italy, August 1997)		
4.c CRP on "Harmonization and Validation of FR Thermomecanical and Thermohydraulic Codes ....." (Obninsk, Russia, October 1997 and Lyon, France, April 1998)		

Topic	Presenter	Time
4.d TCM on "Sodium Removal and Disposal from LMFR in Normal Operational and in the Framework of Decommissioning" (Aix-en-Provence, France, November 1997)		
4.e Consultancy on "IAEA/EC Comparative Calculation of the ULOF and UTOP in the BN-800 Reactor" (EC, Brussels, Belgium June 1997 and Vienna, VIC, Austria, December 1997)		
4.f Consultancy on "LMFR Operating Experience and Feedback for Future Project" (December 1997)		
4.g Consultancy on "Technical Options for the Decommissioning fo the BN-350 Reactor in Kazakhstan" (Obninsk, Russia, February 1998)		
5. Summary report by the IAEA on activities of interest to IWGFR		10:00 -11:20
5.a Activities in the Division of Nuclear Power - Development of reliability assurance programme guidelines	B. Cho, IAEA	
5.b Activities in the Division of Nuclear Fuel Cycle and Waste Technology - Symposium on MOX fuel technology (1999)	K. Fukuda, IAEA	
5.c Activities in the Department of Nuclear Safety - Enginering Safety	A. Guerpinar, IAEA	
6. TCM arrangments for 1998	A. Rinejski, IAEA	

Topic	Presenter	Time
6.a Methods and Codes for Calculation of Thermohydraulic Parameters for Fuel, Absorber Pins and Subassemblies of LMFR with Traditionals and Burner Cores (Obninsk, Russia, 27-31 July, 1998)		
6.b Unusual Occurrences During LMFR Operation:Review of Experience and Consequences for Reactor Systems (VIC, Vienna, Austria, 1998)		
7. Programme of activities for 1999-2000 within the frame of IWGFR	Chairman\All	11:40 -12:30
7.a Selection of topic for 1999 meetings		
7.b Discussion of programme for 1999\2000		
Lunch break		12:30 -14:00
7.c Review and discussion of the scope of future CRPs and TCMs and other activitiesby dividing into two Working Groups:		14:00 -15:30
Working Group 1 - Room C07VI, Chairman.....		
Working Group 2 - Room....., Chairman.....		
Coffee break		15:30 - 15:15
7.d Reports by the Working Group (WG) Chairman of draft scopes for future CRP, TCM and other activities and discussion	WG Chairman/All	15:50 - 16:50
8. Preparation of the IWGFR Summary Report, conclusions and recommendations of the meeting	Chairman/All	16:50 - 17:30
Adjorn 2nd day		17:30 -
Thursday, 14 May 1998		
Continuation item 8		09:00 - 10:30
Coffee break		10:30 - 10:50

Topic	Presenter	Time
9. Remaining items including review of the Terms of References of the IWGFR, proposed data and location of the next IWGFR		10:50 -12:30
Lunch break		12:30 -14:00
Continuation item 9.		14:00 -16:00
Adjourn 31st IWGFR		16:00 -

Department of Nuclear Energy
Division of Nuclear Power

Issue No. 1
4 May 1998

NOTIFICATION OF AN AGENCY SPONSORED MEETING

Title of meeting: Technical Committee Meeting of the International Working Group on Liquid Metal Cooled Fast Reactors (IWGLMFRs) (622-13-TC-385.68)

Dates, inclusive: 12-14 May 1998

Place: IAEA Headquarters
Room C-07VI, Ext.: 21361

Opening Meeting: 09:30 a.m.
Scientific Secretary: A. Rinejski, Room A-2575, Ext: 22812

Secretary: M. Bramante, Room A-2571, Ext: 22865

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KAZAKHSTAN			
Mr. Sergey KRECHETOV Atomic Energy Agency of the Republic of Kazakhstan (KAEA)	Deputy Director Nuclear Safety Division Republic Sq. 13, Almaty 480013 Tel: Fax: 007 3272 633844/637613 (KAEA) E-mail: sergey@kaznet.kz		**9-16 May 1998**

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Mr. Soo Dong SUK Korean Atomic Energy Research Institute (KAERI)	P.O. Box 105, Yusong, Taejon Tel: 0082 42 868 8929 Fax: 0082 42 868 8256 E-mail: sdsuk@nanum.kaeri.re.kr		**12-14 May 1998**
RUSSIAN FEDERATION			
Mr. Yu. ASHURKO Institute of Physics and Power Engineering (IPEE)	Bondarenko Square - 1 249020 Obninsk, Kaluzhskaja Region Tel: 007 084 39 98938 Fax: 007 095 230 2326 or 095 88 33 112		**9-16 May 1998**
Mr. K.V. ZVEREV MINATOM	Ministry of the Russian Federation for Atomic Energy 109180 Moscow Fax: 007 095 2392588 or 9533053		**9-16 May 1998**

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Mr. K.F. AINSWORTH BNFL British Nuclear Fuels plc	Nuclear Fuel Expert Risley Warrington Cheshire WA3 6AS Tel: 001925 834 832 Fax: 001925 833 704		**12-14 May 1998**
Mr. Charles PICKER AEA Technology	Risley Risley, Warrington WA3 6AT Cheshire WA3 6AS Tel: 001925 834 832 Fax: 001925 833 704		**12-14 May 1998**

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IAEA			
Mr. H.H. ROGNER			
Mr. A. RINEJSKI			
Mr. J. CLEVELAND			
Mr. B. CHO			
Mr. K. FUKUDA			
Mr. A. GUERPINAR			
Ms. H. TEZUKA			

INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors (IWGFR)

Reports of the IWGFR Meetings for the period since 30th Annual Meeting

April 1998

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CONTENTS

1. 30th Annual Meeting of the IWGFR (Beijing, China, 13-16 May, 1997)

2. Technical Committee Meeting (TCM) on "Influence of High Dose Irradiation on Advanced Reactor Core and Structure Materials (Obninsk, Russia, 16-21 June, 1997)

3. Consultancy on "IAEA/EC Comparative Calculation for the ULOF (Unprotected Loss of Flow) in the BN-800 Reactor" (Brussels, EC, Belgium,, 30 June - 3 July, 1997)

4. Research Coordination Meeting (RCM) on "Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures" (Taormina, Italy, 25-29 August, 1997)

5. RCM on "Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes and Relations Using Experimental Data" (Obninsk, Russia, 6-10 October 1997)

6. TCM on "Sodium Removal and Disposal from LMFRs in Normal Operation and in the Framework of Decommissioning" (Aix-en-Provence, France, 3-7 November, 1997)

7. Consultancy (CS) on "IAEA/EC Comparative Calculation for the ULOF (Unprotected Loss of Flow) in the BN-800 Reactor" (Vienna, Austria, 11-12 November, 1997)

8. Consultancy on "LMFR Operating Experience and Fead-back for Future Project" (Vienna, Austria, 17-19 December 1997)

9. CS on "Technical Options for the Decommissioning of the BN-350 LMFR in Kazakhstan" (Obninsk, Russia, 23-27 February, 1998

10. IAEA Documents of the IWGFR

Summary Report

of the International Working Group on Fast Reactors
30th Annual Meeting, Beijing, China, 13-16 May 1997

1. INTRODUCTION

The 30th Annual Meeting of the IAEA International Working Group on Fast Reactors (IWGFRs) was held in Beijing, China, from 13-16 May 1997. The meeting was attended by the Members and some additional participants from IWGFR Member States - Brazil, China, France, Germany, India, Japan, the Republic of Kazakstan, the Republic of Korea, the Russian Federation, the United Kingdom and by an observer from Switzerland. (See list of participants - Attachment 1).

The meeting received the congratulation letter from Mr. Zhang Huazhu, Vice General-Manager of China National Nuclear Corporation (CNNC), which was read out by Mr. Zhu Xuhui, Chief Engineer of Science and Technology Bureau of CNNC.

The welcome addresses were made by:

(1) Mr. Xu Naicheng, Director of International Organization of China Atomic Energy Authority;
(2) Prof. Yang Tianlu, Vice President of China Institute of Atomic Energy (CIAE) and Director of Fast Reactor Engineering Department of CIAE;
(3) Academician Ruan Keqiang, Chief Scientist of the Specialist Committee in Energy Field of the National High Technology Programme led by State Science and Technology Commission.

The meeting was opened by Mr. A. Rinejski, Scientific Secretary of the IWGFR, International Atomic Energy Agency and was chaired by Professor Xu Mi of China.

During the meeting, a technical tour to the China Institute of Atomic Energy took place.

2. INTRODUCTION OF IWGFR PARTICIPANTS

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

3. APPROVAL OF THE AGENDA

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

4. REPORT OF THE SCIENTIFIC SECRETARY

Reports of the IWGFR meetings for the period since the last, 29th Annual Meeting of the

Working Group had been distributed among the participants in advance of the meeting. The Scientific Secretary summarized the main points of the IWGFR activities.

Present trends in LMFR design are aimed at providing a very high degree of safety, economically competitive with other reactor types. Because of the relatively high capital cost of LMFRs, this implies the need for both high availability and plant lifetimes in excess of 40 years. Reduction of over-conservative design, while avoiding failure over lifetimes of at least 300,000 hours is therefore a motivator for the development of improved design standards. To this end, two TCMs (on *Creep Fatigue Damage Rules To Be Used in Fast Reactor Design*; and on *Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors)* were held in the United Kingdom and Japan respectively, providing an important opportunity to review developments in design rules for creep-fatigue conditions, codes and method for evaluation of radioactivity release and discussion of safety design principles against sodium fires.

The CRP on *Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures* started in 1996. The purpose of this CRP was to validate reliable numerical methods used for both detailed evaluation of dynamic behavior of isolation devices and isolated nuclear structures of various NPP types. The first Research Coordination Meeting (RCM) held in May in St. Petersburg, Russia discussed in detail the experimental data on seismic isolators and performed a comparison of the results of calculations based on such data.

Another CRP, aimed at *harmonizing and improvement of FR thermomechanical and thermohydraulic codes and relations using experimental data* was started in 1996. The first RCM, which was held in November in Lyon, France, discussed the status and scope of contributions to the CRP from participating organizations, reviewed the case setup data provided by FRAMATOME and discussed first results of thermohydraulic and thermal calculations provided by the participants.

A joint IAEA/EC benchmark exercise for hypothetical severe accidents (unprotected loss of coolant flow) in a BN-800 reactor with near zero void core has been continued during 1996. Two consultancies were held: one in June in Brussels and another in December in Vienna, in order to review and agree the results of preboiling transient calculations provided by organizations from France, Germany, India, Italy, Japan and Russia. The IWGFR supports the continuation of this activity.

In addition, a consultancy was held in October 1996 in Vienna to harmonize the international assistance to ensure stable operation during the remaining lifetime and the development of an effective decommissioning programme of the BN-350 fast reactor in Kazakstan.

5. CONSIDERATION OF IAEA'S SYMPOSIUM ON FAST REACTORS

The background and scope of this Symposium, to be held in Obninsk, Russia in Spring 1999 were explained by the Scientific Secretary. It was pointed out that the securement of long-term fuel supply for electricity generation through breeding remains the ultimate aim of fast reactor development. There is an emerging interest in using fast reactor for transmutation of minor actinides and disposal of plutonium in several countries. During the last decade, a high level of understanding regarding fast reactors and MOX fuel technologies was attained in countries operating fast reactors. In this context, the decision to hold a topical Symposium on LMFR was considered to be timely.

The objectives of the Symposium are to provide a forum for the review of and discussions on LMFRs:

- MOX core physics aspects;
- MOX fuel cycle technology;
- lessons learned from the construction and operation of a fast reactor;
- transmutation of minor actinides and plutonium disposal;
- alternative potential for fuel cycles, including thorium utilization;
- alternative coolants (e.g. lead, steam, gas, molten salt)
- fast reactor deployment strategies.

Mr. Rinejski pointed out that for the preparation of the Symposium, a Steering Committee consisting of representatives from Member States and international organizations should be appointed.

All the representatives expressed their support to this symposium, and they suggested slightly modified the title to: Fast Reactors and their Related MOX and other Fuel Technologies for Medium and Long Term Utilization: Experience, Advances and Trends.

6. CONSIDERATION OF MEETING ARRANGEMENTS FOR 1997

TCM on *Influence of High Dose Irradiation on Advanced Reactor Core Structural and Fuel Materials*, to be held from 16-21 June 1997 in Obninsk, Russia. An official agreement by the Russian Government to host this TCM has been received by the IAEA in February 1997. The official invitation letters to Member States have been sent in March. The Scientific Secretary asked the IWGFR members to assist in timely nomination of participants.

TCM on *Sodium Removal and Disposal from LMFRs During Normal Operation and in Framework of Decommissioning*, to be held from 3-7 November 1997 in Aix-en-Provance, France. The Host Government letter was sent to the French Mission in January 1997. After the official agreement is obtained, invitation letters to Member States will be dispatched. IWGFR members were asked to assist in timely nomination of participants.

7. SELECTION OF TOPICS FOR MEETING FOR 1998

As a result of voting, it was recommended to the IAEA that the following TCMs should be held within the framework of the IWGFR:

(1) Unusual occurrences during LMFR operation: review of experience and consequences for reactor systems. This TCM will be held in Vienna in October, 1998. and

(2) Methods and codes for calculations of fuel and absorber pins and subassemblies, thermohydraulic parameters in the real reactor conditions

Russian delegate confirmed his willingness to host this meeting and proposed to be held in Obninsk in June, 1998.

8. ACTIVITIES

Fast Reactor Database: The Secretary informed the IWGFR that the reference document has been published and distributed to the Member States.

It was suggested by Dr. G. Heusener and agreed by all representatives to put the Fast Reactor Database on computer data bank to facilitate its utilization.

Fast Reactor Technology - Status Report: The Secretary informed the participants that the document is under preparation for publication.

Mr. A. Rinejski informed the meeting about new publications by the IAEA. Dr. C. Le Rigoleur and other representatives asked Mr. A. Rinejski to send the complete list of the publications issued by the IAEA since its beginning.

9. PRESENTATIONS AND DISCUSSION OF NATIONAL FAST REACTOR PROGRAMMES

Presentations on fast reactor development were made by Members of the Group from Brazil, China, France, Germany, India, Japan, the Republic of Kazakstan, the Republic of Korea, the Russian Federation, the United Kingdom and Switzerland. Mr. A. Rinejski informed on fast reactor activities of EC. A discussion followed. Summaries of presentations are given below:

Brazil Fast reactor activities in Brazil are mostly research-oriented and have the objective of establishing a consistent knowledge base which can serve as a support for a future transition to the activities more directly related to design, construction and operation of an experimental fast reactor. Due to the present economic difficulties and uncertainties, the programme is modest, with a potential to grow; however, it is affected by the present "eye on the market" philosophy which is permeating everything which concerns nuclear energy in Brazil. Under a cooperation agreement with the Instituto de Engenharia Nuclear (IEN, Nuclear Engineering Institute), neutronics calculations are being continued with different methodologies in order to chose the best route for more refined calculations if necessary. The objective here is to understand the influence of various nuclear data sets, etc. A natural circulation circuit using water as the circulating fluid has been recently installed in the Department of Mechanical Engineering of the Instituto Tecnolólogico de Aeronáutica (ITA, Aeronautic Technological Institute), as part of a formal agreement between the Advanced

Studies Institute and ITA. Using a 8 KWt heat source, the circuit will be used for studying basic phenomena and also for computer code (passive system analysis) validation. In the area of heat transfer/removal, an exchange between the researchers at the Indira Ghandi Centre for Atomic Research (IGCAR, India) has been initiated in order to identify topics of common interest for a future collaboration. As in the ALMR (USA), HT-9 is being considered as the reference for cladding material. In 1996, a sample was experimentally obtained and first analyses have indicated the need for minor adjustments in its composition, which are presently being done. Under a formal cooperation agreement with the Instituto de Pesquisas Energéticas e Nucleares (IPEN, Nuclear Research Institute, Sao Paulo), metallic uranium recovery by electrorefining is being investigated, in order to understand the basic processes involved. A first experiment performed in 1996 in which a small amount of uranium was electrodeposited, has been successful in the sense that it allowed the identification of problems which deserve further attention. Among those can be included the uncertainties in the elecrodeposition current to be applied, whose values in the literature are conflicting. A small electromagnetic pump has been recently built and successfully tested at the Instituto de Estudos Avançados (IEAv, Advanced Studies Institute). The pump uses an Iron C-type magnet with a continuous-current coil to obtain the required magnetic induction. Present research involves extensive testing and also the utilization of rare-earth permanent magnets ($SmCo_5$) in place of the C-magnets in order to reduce the pump's geometric dimensions. R&D work on alternating current EMPs is planned to be initiated in 1997. After many years "on the shelf", the first of the three sodium loops which were part of the contract with ANSALDO/NIRA (Italy) will finally be assembled at the Institute de Engenharia Nuclear (IEN, Nuclear Engineering Institute, Rio de Janeiro), with completion expected by mid-1997. The loop (named SS-10) is to be used as a bed for experiments related to sodium purification and transferring.

China In 1996, with the increase of about 15 GWe electrical capacity, the national electricity generation has reached 225 GWe. Two nuclear power plants, the Qinshan Phase 1 and the Daya Bay have been in good operation. The load factor of Qinshan Phase 1 was 84.7% and for Daya Bay 76.1% for Plant 1 and 64.1% for Plant 2. During the ninth 5-year plan (1996-2000), four nuclear power stations consisting of eight reactors with 6620 MWe installed will be constructed. Within the framework of the High Technology Programme, the Chinese Experimental Fast Reactor (CEFR) with a capacity of 65 MWth/25MWe is under progress. The experimental reactor CEFR-25 may use UO_2 fuel for first core; subsequent core will use MOX fuel. It is a sodium-cooled pool type reactor. There are two primary sodium pumps and four intermediate heat exchangers (IHX) in the main vessel, two secondary loops, single steam/water circuit and one turbine-generator set. The decay heat removal is provided by two independent passive residual heat removal systems.

France In 1996, the total amount of electricity produced in France was 488.9 TWh, out of which 378.2 TWh (77%) were produced by nuclear power plants, 4.8 TWh (8.5%) by conventional thermal plants, and 68.9 TWH (14%) by hydraulic plants. 72.6 TWH was exported, 3.5 TWH was imported, 5.9 TWH was used for pumping, while line losses came to 31 TWH. Consequently, the net electrical power consumption in France was 383 TWH, corresponding to an increase of 3.8% in comparison to 1995 consumption. At the end of 1996, "Electricité de France" had 55 PWR units in operation. Total capacity was 59,795 MWe. The availability factor for these units was maintained at 82.7%. 1996 was marked by a

decrease of unexpected shutdowns and a new reduction in programmed shutdown periods. In 1996, a good safety level was maintained. In the field of Fast Reactors, the main events of 1996 were the following. At the end of February 1996, the PHENIX reactor was ready for operation. After review meetings, in February 1996, the French Safety Authority did not rule on the possibility of extending the lifetime of PHENIX for another 10 years, and did not want to authorize performance of the 50th cycle. The Safety Authority has requested safety improvements and technical demonstrations, before is examines the possibility of authorizing a new start-up of PHENIX. 1996 was devoted to this work. Concerning the CREYS-MALVILLE plant (SUPERPHENIX), the reactor was restarted at the end of 1995. The reactor's power was increased in successive steps: 30% Pn up to February 6, followed by 50% Pn up to May, then 60% Pn up to October and 90% Pn during the last months of 1996. A programmed shutdown period occurred during May, June and mid-July 1996. The reactor has been shut down at the end of 1996 for the 10 year inspection of the steam generators. A special panel, the CASTAING Commission, was appointed by the French government to evaluate the relevance of the Knowledge Acquisition Programme (PAC in French), to be performed at SUPERPHENIX. The Commission delivered its report in June 1996, recommending the pursuit and extension of this programme. The CAPRA project, initiated in February 1993, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high as possible rate. The first two-year phase of the CAPRA project studies was completed. Complementary studies, now in the framework of the CAPRA programme, were performed in 1996. 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 EFR programme launched by the European Fast Reactor Utilities Group (EFRUG) aims to keep the fast reactor option open (common technology for breeders and burners).

EDF objectives and development of the FR studies (status first months of 1997) A new contract between EDF and the French Government was signed on 8th April 1997. It set the main objectives for the company for the 1997-2000 period. EDF is confirmed as a Public Service, able to compete in the European electricity open market with a commitment to reduce the price per Kwh by 14% in next 4 years. The reduction is possible thanks to good performance of NPPs, control of O&M costs and debt reduction. EDF objectives are to preserve or to improve the current high level of safety and availability of NPPs but also to prepare to resume a successful NPP construction programme when need for new plants reappears in the future (advanced LWRs). Most experts consider that the need of FRs in the French nuclear facility, to face hardening of the uranium market, will not appear before the second half of the next century. However EDF maintains interest and support for FR development, as a contribution to the studies on the different possible options for the management of the back end of the fuel cycle for getting best value from SPX operating experience and to preserve expertise and previous investments. In that contract EFR programme studies aims notably to provide a "paper work house" (Reference EFR) for assessment of the economic or safety interest of R&D results. The EFR programme studies have been recently adjusted to take account of the limitation of resources. In the present situation, Creys Malville needs a new license to resume operation, the previous one having been cancelled by the French Supreme Administrative Court. Nevertheless this did not prevent the planned test and maintenance operations being carried out as Originally intended during the present situation. Activities cover decimal hydraulic tests of 3 of the 4 SG examination (US) of 37 tubes, several maintenance operations and modifications and ISI of the reactor block internals.

Germany The general situation in Germany remained unchanged in 1996 and can be characterized as follows: (1) nuclear electricity production of about 160 TWh/a, i.e. about 1/3 of total production and (2) extremely good availability of running power stations despite hostile political environment. Fast reactor activities at Forschungszentrum Karlsruhe (FZK) include neutron physical and safety calculations of selected core configurations for burner fast reactor (CAPRA type) and post-irradiated examination of structural and cladding materials. They are embedded in the European fast reactor development programme. First calculations of the initiation phase of an oxide core melt-down behavior were performed with the newest version of the SAS4A code. To determine swelling and in-pile creep of FR cladding material, a pressurized tube experiment was carried out in the PFR reactor. Samples of various alloys (model plain Fe-15Cr-15Ni, commercial German 1.4970, etc.) were tested in the temperature range between 420℃ and 600℃ and doses reached between 61 and 106 dpa. After shut-down of the PFR, the samples were transported to FZK for PIE and the non-destructive measurements of diameter and length were completed. In collaboration with CEA, a ferritic-martensitic wrapper material (W.Nr. 1.4914) was irradiated in the PHENIX reactor for 105 dpa and impacts on service life and integrity should be examined by testing the mechanical properties and microstructure. Particular attention was paid to the welds connecting the martensitic wrapper material with the austenitic (AISI 316) top and bottom sleeves.

Under decommissioning are the two German breeder power plants, SNR 300 and KNK II. SNR 300 at Kalkar site was stopped in 1991 because the operating license could not be gained for political reasons. During the following years, the circuits were drained of the sodium and the main components like pumps, intermediate heat exchanges an steam generators were removed and dismantled. The 20 MWe experimental fast breeder power plant KNK II was in operation for approximately 20 years at the nuclear research center at Karlsruhe. Its operation was discontinued in the wake of the SNR 300 termination. Afterwards, a decision was taken to scrap the plant to green field. During the recent years, the sodium from the secondary and primary systems has been taken out and shipped off to Dounreay. In addition, the tertiary system was completely dismantled; the secondary system is being cut out in the course of 1997. It is anticipated that the complete decommissioning work will be completed around 2000.

India About 395 TWh of electricity was generated during the period April 1996 - March 1997. The growth of installed capacity in the VIII Five-Year Plan (1992-97) is not satisfactory due to cut in the public spending in the power sector. Privatization in the power sector has not picked up as was expected and this has caused power famine in the country. Industry, agriculture and residential power supply were affected very badly during the summer months. Oil import bill during the years was $9.3 billion. The 13 party coalition government is discussing the IX plan proposal in the power sector. The operating performance of the thermal power reactors has considerably improved during the year and has enhanced the confidence level in nuclear energy in the government and the public. Remote repair of the clandria over pressure relief device and replacement of coolant channels has been an important repair

activity at RAPS (PHWR). Construction of 4 PHWR, 2 VVER (Russian collaboration) and 500 MWe PFBR have been proposed in the IX Plan (1997-2002). Operation of fast breeder test reactor (FBTR) at 10.5 MWt is continuing. The maximum fuel burnup reaches is 32,000 MWd/t without any failure. One fuel subassembly was taken out at 25,000 MWd/t burnup and post irradiation examination has been completed. The fuel performance is very good and the targeted fuel burnup is 50,000 MWd/t. Turbine was rolled up to synchronous speed of 3000 rpm several times during the year and the operation was found to be smooth. Synchronization with the grid could not be done because of problems with oil leak in the bearing, failure of auxiliary oil pump motor and problems in getting the required vacuum. Regulatory authority has approved the proposal to raise the power from 10.5 to 12.5 MWt with the addition of fuel subassemblies in the core. Design and R&D of 500 MWe Prototype Fast Breeder Reactor is continuing. The main options for the reactor are sodium coolant, pool type primary sodium circuit, MOX fuel, two primary sodium pumps, two secondary sodium loops with 4 SG in each loop. The important design activities carried out during the year are plant dynamic studies, decay heat removal analysis, design of pump to gird plate pipe, scram and LOR parameters, location of secondary sodium pump in the secondary circuit and design of the fuel handling machines. The important experimental R&D work carried out during the year are testing of prototype primary sodium pump in water, operation of a large sodium test rig to study the heat and mass transfer in the cover gas, fabrication of six annular linear induction pumps, testing of dummy fuel subassembly in water test rig for pressure drop and vibration measurements, testing of the full length sector model of SG for flow distribution, pressure drop and vibration aspects and testing of 1/3 scale grid plate model in air for pressure drop and flow distribution measurements. 30 KWt U-233 fueled KAMINI reactor was made critical in October 1996 and will be used for activation analysis and neutron radiography. Post irradiation examination of experimental fuel pins and one full fuel subassembly has been completed and the results are very encouraging. The non-destructive examinations carried out include eddy current inspection of cladding tubes and ultrasonic examination of secondary sodium pipe welds of FBTR. In the field of materials development, tensile, creep and creep-fatigue interaction effects on SS 316 LN and 9 Cr-1Mo, microstructural characterization and environmental effects on SS 316 LN and 9 Cr-1Mo, weldability of SS 316 LN and inconel 718, aluminizing of inconel 718 thermal ageing of trimetallic joints (SS 316 LN/Alloy 800/9 Cr-1Mo) and fracture properties of 9 Cr-1Mo welds have been studied.

Japan During the period April 1996 - March 1997, the 30th duty cycle operation has commenced in the experimental fast reactor Joyo. The cause investigation on sodium leak incident has been completed and total review of the plant are being performed on the prototype fast reactor Monju. The design study for demonstration FBR plant optimization has been completed by JAPC in the three years since fiscal year 1994. Related research and development are underway, being discussed and coordinated by the Japanese FBR R&D Steering Committee, which was established by PNC, JAPC, JAERI and CRIEPI. In November 1996, the Japan Atomic Energy Commission organized a meeting to discuss the significance of future FBR development.

Republic of Kazakstan has a nuclear scientific-industrial complex which was set up as part of the nuclear infrastructure of the former USSR. More than 50% of the uranium resources of the former Soviet Union are in Kazakstan, with seven uranium mines. Two UO_2

plants produced up to 35% of the total uranium in the USSR in 1990. There are extensive facilities for producing UO_2 pellets for VVER fuel elements from Russian-enriched uranium. Kazakstan has several research reactors and one operating nuclear power plant, the BN-350 fast reactor, which started operation in 1973 with a design life of 20 years. Work on its lifetime extension has the intention of bringing it into compliance with current safety standards. 1995 and 1996 were devoted to this work. In October 1996, experimental investigation on "accident-proof" decay heat removal by natural circulation was carried out. The reactor BN-350 was restarted in February 4, 1997 at a power level of 420 MWth.

Republic of Korea There is a significant dependence of electricity generation on nuclear energy in the ROK and this dependence raises the issue of the need of efficient utilization of uranium, which Korea imports and of spent fuel storage. A fast reactor is thought to be a practical solution to these concerns, and the Korea Atomic Energy Commission decided to develop and construct a liquid metal reactor of 150 MWe by 2011. KAERI is developing KALIMER (Korea Advanced liquid Metal Reactor), which will be the first of its kind in the country. To develop KALIMER, design methodology research is underway. The main features of the KALIMER concept are to be formally determined in July 1997. To develop this concept, a comparative study is being conducted of various possible design alternatives. The study is focused on making KALIMER more safe, economic, with lesser impact on the environment and more resistant to nuclear proliferation. The safety is to be enhanced by implementing an additional passive shutdown system which works on the Curie point temperature of a magnet, a primary system based on a pool, a passive reactor vessel cooling system whose structure is very simple and which is reliable. To improve the KALIMER's economics, the system is using seismic isolation device, electro-magnetic pumps, simplified safety grade reactor heat removal system (RHRS) and metal fuel. The use of metal fuel will also make KALIMER more resistant to nuclear proliferation. The high thermal efficiency of KALIMER also improves the economics and makes for less discharge to the environment. After determining the main features of KALIMER, the basic design work will be carried out over the next five years, until 2002. During the first three years, the KALIMER's system design will be framed out quantitatively, mainly for NSSS and the BOP systems physically connected to the NSSS. In the remaining two years, the finished design will be evaluated for its overall performance, safety and economics. Based on the overall evaluation results, the design will be refined further and the remaining systems which are not covered during the first three years will also be designed. There is a lot of equipment and technology which require R&D for implementation of KALIMER. To name a few, the submerged electro-magnetic pump, a seismic isolator, high temperature structure design, metal fuel, passive Curie point shutdown system and the steam generator. R&D of these items require a massive investment to the facility and it would therefore be desirable to carry out this work through an international collaboration. The collaboration on the systems engineering and methodology development is also needed to make up for the lack of Korea's experience in liquid metal reactor systems engineering.

Russian Federation In 1996, the amount of energy generated by 29 NPPs in Russia, their total installed power being equal to 21.2 GWe, was $108.8.10^6$ MW-hours. Average value of load factor was 58.32% (as compared to 53.4% in 1995). During 1996, only two incidents occurred, which were classified at first level events on the INES scale. The BN-600 reactor

operation was successful, its load factor being equal to 76.32% (as compared to 70.31% in 1995). Since its first start-up, the power unit has operated as electricity generating plant during 76.5% of time, 21.1% of time having been spent on refueling and planned repair work, which 2.3% of time was taken by unscheduled reactor shutdowns. Currently, the BN-800 reactor design undergoes examination according to the regulations now in force in Russia, in order to obtain the license to continue its construction on both Beloyarsk and South-Ural sites. In order to fulfill the construction of the 4th power unit of the Beloyarsk NPP, a public joint-stock company was established, incorporating several large enterprises and institutions whose financial potential is sufficient for accomplishing the 7-year construction. The economic analysis of the BN-800 reactor design, including the revision of some expense items is carried out, aimed at a cost reduction. Much work on the fuel and structural materials irradiation was carried out on the BOR-60 reactor. Fuel burnup of 30.5% has been achieved in some of the pilor vibro-packed MOX fuel elements. Experimental studies are underway on the BFS critical facilities, modeling various fast reactor core designs for plutonium utilization and minor actinides burning. Work on uranium-thorium fuel cycle is being carried out on the COBR test facility. Analytical studies were conducted on the former weapons plutonium utilization in the BN-600 and the BN-800 reactors. Experimental studies have been focused on the issues of sprayed sodium burning and sodium leaks under thermal insulation of the pipelines. Also, sodium boiling and several aspects of sodium technology were studied experimentally.

Switzerland The Fast Reactor activities reported here are the results of two small research program, one on liquid metal thermohydraulics and the other fast reactor physics, and a general evaluation of the development of advanced reactors including fast reactors. The mentioned two research programs which had been supported by the utilities and Swiss Department of Energy were terminated at the end of 1996 due to budgetary reason. However some of the scientific knowhow from the abandoned programmes are now incorporated in the growing research activities for actinide transmutation and waste incineration. The **liquid metal thermohydraulics programme** centered around the investigation of mixing layers of two overlaying horizontal streams of liquid metal of different temperatures and velocities. First a series of analoge experiments were performed with water. The experiments with sodium are starting now. The **reactor physics work** was predominately in the computational field to validate methods and data in cooperation with CEA using the European Code System ECCO/ERANOS. PSI performed an evaluation of various **advanced reactor systems** for the Swiss Department of Energy. The aim was to assess the development potential of the various nuclear systems taking all the relevant aspects into account.

United Kingdom Statistics for 1995 show that nuclear power comprised 28% of electricity supplied. There were 35 nuclear plants in operation with a total installed capacity of 14,168 Mwe. The nuclear industry has been restructured as a consequence of the UK Government Reiew, published in 1995. A new utility company 'British Energy plc', which is in private ownership, owns and operates the 14 Advanced Gas Cooled Reactors and the Sizewell B PWR. The 20 Magnox reactors are operated by the newly formed 'Magnox Electric' which remains in public ownership. The former United Kingdom Atomic Energy Authority has been divided into two organizations 'UKAEA' and 'AEA Technology plc'. UKAEA, which remains in public ownership, is responsible for decommissioning of nuclear facilities and radwaste

management arising from the UK Atomic Energy Authority. AEA Technology is now a privatised company offering technical services on a commercial basis and has been vested with the Intellectual Property from the UK fast reactor system development programme. British Nuclear Fuels plc remains 100% government owned and retains the responsibility in the UK for nuclear fuel cycle development including fast reactor. It is likely that Magnox Electric and the 20 UK Magnox reactors will become part of BNFL under the last phase of the industry restructuring. The UK continues to participate in the European Collaboration, mainly funded by BNFL. Activities focus on the further development of the EFR design, on the CAPRA programme led by CEA and on broader fuel cycle studies. Support to international events, including IWGFR meetings has been maintained where appropriate. A number of international secondees to FR programmes have also been maintained. The experiments associated with the closure of PFR have now been completed by the destructive examination of materials from the secondary circuit and sodium storage tanks. The results have been reported to the European partners. The experiments showed no cause for concern with respect to failure during longer term operation. In the first stage of decommissioning, the PFR core has been removed and replaced by a dummy core. Decommission activity is currently concentrating on the provision of a sodium disposal plant and the supporting transfer system. Successful reprocessing of fuel, axial and radial breeders and residues from PFR has been demonstrated. In 1996 a further 24 radial breeder and 4 core sub-assemblies were reprocessed; in total approximately 127 Kg Pu and 3654 U was separated.

European Commission continued its fast reactor research activities along the same lines as previously, but with the main emphasis on partitioning and transmutation (P&T) of long-lived radionuclides and reactor safety. The P&T work was carried out by research institutions in the Member States and by the EC Joint Research Centre (JRC) as cost-sharing actions. The JRC has also been performing its own programme through institutional and competitive research activities. The JRC institutes involved in these studies are the Institute of Systems, Informatics and Safety (ISIS) in Ispra, the Institute for Transuranium Elements (ITU) in Karlsruhe and the Institute for Advanced Materials in Petten. The implementation of P&T involves research in three areas: (i) partitioning of long-lived radionuclides from the high level waste, (ii) development of fuel and targets containing these long-lived elements in view of their (iii) transmutation of various burners (fission reactors and accelerator-driven transmutation devices). The European Commission has partly supported experimental work on partitioning both in the framework of the shared-cost programme and at the JRC/ITU, fuel and target development at the JRC/ITU and an overall strategy study on the potentialities of P&T for nuclear waste management as a shared-cost action. After the high pressure load of the European Fast Reactor (EFR) Project, the interest for LMFBR severe accidents is somewhat reduced. Actually, some savings were made in the funding of computer code developments for whole core accident calculations thanks to the coordination activities of the Whole-core Accident Calculation (WAC) Group which managed to converge all calculational developments in the EU towards one single code, namely the European version of SAS/4A - which has been adopted also by Japan. As an alternative solution - especially for the modeling of preirradiation and fuel pin mechanics phenomena - the computer codes TRAFIC (UK) and GERMINAL (F) have been made available to some of the WAC Group activities in view of specific calculations in parallel with SAS/4A. As a result of a common interest for developing severe accident analysis codes, a second common follow-up benchmark exercise concerning

Russian BN-800 in its nearly-zero void reactivity version has been proposed jointly by EU and IAEA/IWGFR with the aim of also including India and Japan besides Russia. The main goal of this new comparative calculation of BN-800 encompasses these items: (i) to establish a basis so as to evaluate the characteristics of a BN-800 type reactor with a nearly zero sodium void reactivity core design under hypothetical severe accident conditions, as far as an energetic unprotected loss of flow (ULOF) is concerned, and (ii) 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.

10. ANY OTHER BUSINESS

The mode of operation and role of the IWGFR Annual Meetings were discussed. It was agreed that the presentations on the National status reports should continue, since this is the only way in which the Members can learn about all the world's fast reactor programmes; however the time for each presentation should be restricted to a maximum of 30 minutes. To facilitate this, the National status report papers are to be sent to the IWGFR Secretariat, for onward distribution to Members, at least 3 weeks before the Annual Meeting. It was also agreed that detailed discussion of any topics of particular concern to any Member shall take place at the Annual Meetings in future. Any Member wishing to raise such a topic is to send details to the Secretariat at least 2 months before the Annual Meeting.

11. DATE AND VENUE OF THE 31st ANNUAL MEETING OF THE IWGFR

It was agreed that the 31st Annual Meeting of the IWGFR should be held in Vienna from May 12 to 14, 1998. It was proposed that the meeting be chaired by the Japanese representative, H. Yamashita.

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Attachment 2

30th Annual Meeting of the International Working Group on Fast Reactors
Beijing, China, 13 - 16 May 1997

AGENDA

1. 09:30 Welcome Address CIAE

2. Opening of the meeting IAEA, Meeting's Chairman

3. Introduction of the participants

4. Review and approval of the agenda

5. Report on the activities of the Working Group

   5.1. Technical Committee Meetings

   a) Creep Fatigue Damage Rules to Be Used in Fast Reactor Design
   Manchester, United Kingdom, 11-13 June 1996

   b) Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors
   O-arai, Japan, 11-14 November 1996

   5.2. Consultancies

   a) IAEA/EC Comparative Calculation for Hypothetical Severe Accident (ULOF) in a BN-800 Reactor with near Zero Void Core Design
   Brussels, Belgium, 25-28 June 1996 and VIC, 4-6 December 1996

   b) Harmonization of International Assistance to Ensure Stable Operation During the Remaining Lifetime and the Development of the Effective Decommissioning Programme for the BN-350 Fast Reactor in Kazakstan
   VIC, 24-25 October 1996

   5.3. Consideration of joint research activities

   a) Inter-comparison of Analysis Methods for Seismically Isolated Nuclear Structures
   RCM, St. Petersburg, Russian Federation, 27-31 May 1996

   b) Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes Using Experimental Data
   RCM, Lyon, France, 25-27 November 1996

6. Consideration of conferences

IAEA Symposium **Fast Reactors and Related MOX Fuel Cycle Technologies for Medium and Long Term Deployment: Experience, Advances, Trends**
IPPE, Obninsk, Russian Federation, Spring 1999

7. Meeting arrangements for 1997

7a. TCM on Influence of High Dose Irradiation on Advanced Reactor Core Structures and Fuel Materials
Obninsk, Russian Federation, 16-21 June 1997

7b. TCM on Sodium Removal and Disposal from LMFR During Normal Operation and within Framework of Decommissioning
Aix-en-Provance, France, 3-7 November 1997

8. Selection of topics for meetings to be held in 1998

9. Consideration of activities

a) Status Report on LMFR
b) Fast Reactor Database
c) Other Publications

10. National reports on status of fast reactor development

Brazil	Kazakstan
China	Korea, Republic of
France	Russian Federation
Germany	Switzerland
India	United Kingdom
Japan	

11. Other business

12. Consideration of date and venue of the 31st IWGFR meeting

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Technical Committee Meeting on "Progress in FBR Developments"
(30th Annual Meeting of the IWGFR)
13-16 May 1997, Beijing, China

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Summary of the TCM on Influence of High Dose Irradiation on Advanced Reactor Core Fuel and Structure Materials, held from 16 to 21 June 1997 in Obninsk, Russian Federation.

General Background

The main purpose of the meeting was to provide a forum in which to review the world-wide experience on the effect of high dose irradiation on the material properties of MOX fuel elements and assemblies, with an eye toward achieving high burn-up of fissile materials and thus provide a path for the more effective incineration of plutonium.

It is convenient for this summary to focus initially on the fuel aspects of this review, and then to concentrate on the structural materials separately. Of course there can be no simple complete separation of these two categories, since the evolution of the fuel will influence the environment experienced by the cladding, and thereby contribute to the degradation of the cladding's integrity. When appropriate, such interconnections will be noted.

Fuel materials

Within the span of 40 years' of intensive multinational development, great experience on highly irradiated MOX fuel was accumulated. At this meeting there were reviews presented on the experience that has been acquired on the operation with uranium oxide fuel in BN-350 and BN-600, MOX fuel in pellet form in BOR-60, BN-350, DFR, PFR, RAPSODIE, PHENIX and SUPERPHENIX, and uranium nitride in BR-10, UC, UNC, UPuC, in BR-10, BOR-60. In Europe, about 8,000 fuel pins have reached burn-up values of 15 at %. Some experimental pins have attained burn-up levels of 23,5 at % in PFR and 16,9 at % in PHENIX. More than 4,000 fuel pins with pellet MOX fuel were irradiated in BN-350 and BN-600, maximum burn-up in BN-600 is ~12 at %. Experience is being acquired with MOX fuel with high Pu content for its use as actinide burner. From the irradiation results obtained from BOR-60, BN-350, and BN-600, it may be concluded that there are no significant differences in performance among UO_2, PuO_2 and MOX fuels, but additional experimental support of such a conclusion is required.

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In addition, MOX fuel has been studied in vibro-packed form in Russia and in the UK, where vibro-packed fuel was used extensively in DFR, principally in granular form at very high burn-up levels, often in excess of 20 at %. While this fuel was found to give satisfactory performance at reference ratings up to 450 -500 W/cm , its use at high ratings in excess of 500 W/cm was often associated with abnormally high cladding corrosion rates. The latter were attributed to loss of fuel stability at high power ratings, and a parametric study of the data failed to reveal any sensitivity to O/M ratio, clad composition or Pu content. After the repetition of these observations in PFR vibro fuel, the UK abandoned further development of this fuel type in favor of the use of annular MOX pellet fuel, which was eventually taken successfully to burn-ups in excess of 23% in PFR without problems attributed to the fuel itself. Their excellent performance allowed high irradiation exposures of advanced cladding material and wrappers to be made.

In Russia, for BOR-60 applications, techniques have been found to maintain stable MOX vibro-packed fuel densities and to reduce the O/M ratios in this fuel type, primarily through the use of on uranium getter; two features identified as contributing to high cladding corrosion values. Use of these modifications has allowed some vibro-packed fuel pins, clad in ferritic-martensitic material, to reach record burn-ups of about 35 at % over a 10 year period, with ratings of ~ 450 W/cm at the start of irradiation. This performance must be maintained when the fuel is exposed under commercial conditions at high flux/ratings.

In India, the use of mixed carbide fuel with 70 at % PuC used as driver fuel FBTR Mark 1 cores will continue, following the successful operation of FBTR Mark 1 carbide fuel to 30000 MWD/t without fuel failure. The post-irradiation examinations of this fuel indicates that exposure could be comfortably increased to 50 000 MWD/t, with linear ratings as high as 400 W/cm. However, MOX fuel will be used for India's prototype fast reactor with power output of 500 MWe. The meeting participants noted that until the mid-1980s, research programmes in the United States dominated the base technology with respect to MOX fuel high burn-up and reprocessing. Unfortunately, these programmes were terminated when the US LMR programme was discontinued.

Structural materials

It was pointed out by the participants at the outset that the primary limitation for attainment of high burn-up is associated with the integrity of the cladding, and not any inherent properties of the fuel. When the cladding is breached to allow release of fission gas and solid isotopes, as well as fuel to the reactor coolant, this requires that further burn-up be terminated. During discussions at this TCM, a number of persistent themes emerged as common observations and developments from all participating countries. Each of these themes are presented separately in the following sections.

Variability of response in nominally similar materials and environments

There was often a sharp contrast in the amount of data scatter from various types of experimental sources. In well-controlled and often single-variable, the data scatter was minimal

and the fundamental dependencies were easy to identify and model. In other data collections from real operating components there was often a wide amount of scatter that sometimes precluded the confident identification of the principle variables and degradation mechanisms that were responsible for the materials' response to the irradiation environment. While the results of the controlled single-variable experiments can be used to help analyse such complex data fields, there was much discussion of the origin of such large scatter. Three points of view were discussed.

First, many of the simultaneously contributing processes involved in fuel pin evolution are inherently stochastic in nature and cannot be defined with precision. Crack propagation and corrosion fit this description. Second, there are large spatial and temporal variations in all important environmental variables across each structural and fuel component, and also between components. Therefore, a complex interaction in stress and other variables arises between each of the components in any given structural ensemble. Results of analysis of BN-350 creep and swelling data by Karaulov were interpreted with an eye toward such variations. Finally, it was noted that the larger amount of data scatter often was directly correlated to the variability of microstructure and microchemistry in the steels arising from insufficiently well-controlled production processes. **Therefore, one of the major conclusions of the TCM was that the highest priority be given to development of both stringent material specifications and production process control.**

Paths toward improved structural material

It was obvious, with one significant exception, that most countries were following one or both of branched paths in the development of improved materials. On the first branch, the attempt was being made to extend the duration of the incubation period of void swelling in low (≤18 %) nickel austenitic steel. In all countries with such programmes, this approach has resulted in significant delay of the onset of swelling often to doses on the order of 80 -100 dpa (displacement per atom). Given the inherent phase instability of these steels under irradiation, such an approach entails some uncertainty, however, since the microchemical and microstructural evolution is often found to be very sensitive to details of reactor operational histories, all aspects of which cannot always be anticipated. Once the transient regime of swelling is terminated, austenitic steels have the potential to swell at ~1 %/dpa and failure arising from swelling-related problems.

The second branch of material development has often been pursued simultaneously with the first branch in many countries and relies on the significantly lower swelling rate inherent in ferritic and ferritic/martensitic alloys. In this path, the major limiting property now becomes radiation-induced embrittlement rather than void swelling.

In both branches, the major focus of research and development activity has been the development and maintenance of stable phases and microstructures, a requirement that has been found to be strongly associated with maintenance of desirable properties. There was some concern that in incompatibility might develop at very high burn-up if austenitic cladding was to be included in a martensitic or ferritic-martensitic wrapper.

The single most significant exception to these two branches was that of the use of the high

nickel Nimonic PE16 alloy in the UK. Based on a long prior history with this alloy, the UK fortunately found a microstructurally-stable, swelling-resistant alloy early in their material development effort. In the UK, they have optimized this material and its production, and have successfully reached very high burn-up (~25%) levels with very low failure rates at doses approaching ~150dpa.

Voids are the dominant concern

Once voids swelling values levels in excess of several percent, it dominates not only the dimensional stability, but controls the creep rate, phase stability, embrittlement characteristics, and to a lesser extent, properties such as elastic modules, and the thermal and electrical conductivity. Some investigations showed that the electrical conductivity could be used to measure the void swelling. **Discussions led to the conclusion, that deviations of the resistance change from the theoretically predicted values could be used to estimate the microchemical contributions to resistivity change as well.**

Most significantly, the primary concern resulting from void swelling was the of void-related extreme embattlement, designated as EES. In many cases the life-limiting factor for fuel pins and other structural components was the EES phenomenon. The EES was most likely also going the be a major end-of-life concern for austenitic near-core internal components. It was shown that the irradiation creep of martensitic steels was lower than that of austenitic steels, primarily due to the lower swelling rate of these steels. The creep-swelling relationship, however, appeared to be independent of the differences in the crystal structure.

Environmental interactions with EES

A number of investigations were devoted to the deleterious influence of corrosion and intergranular attack resulting from fission products that accumulate during high neutron exposure. A Chinese study explored such effect in a simulated environment. Russian studies showed that such attack would provide extreme cracks on the inner surface of cladding, and that these cracks can initiate the failure mechanism provide by EES. Of particular concern was whether such corrosion attack problems would accelerate at higher burn-up levels.

The inherent superiority of martensitic steels

A number of investigations concentrated on the performance for ferritic-martensitic steels Although the compositional paths taken in such development varied from country-to-country, the performance to high dpa dose was overall superior to that of austenitic steels. Several qualifications apply to such a statement, however. First such steels must be limited to somewhat lower temperature ranges since their strength falls strongly at high temperatures. The "Achilles Heel" of martensitic steels in the development of large shifts in ductile-to-brittle transition temperature at lower irradiation temperatures. This can be partially avoided by the use of higher inlet temperature, but in BN-350 such steel have served very well in spite of the low inlet temperature. One study showed that in one such steel there was an initial period in such the steel was most brittle and then some recovery ensured at higher exposure. This behavior suggests that the steel in not completely phase-stable under irradiation. Smaller levels of instability observed in

- In Europe, more than 7,000 pins have reached burn-up values of 15 at %; in addition, some experimental pins (with solid or annual pellets) have attained burn-up levels of 23.5 at % in PFR and 16.9 at % in PHENIX

- In the USA, more than 63,500 pins (with solid pellets) have been irradiated in FFTF under prototypic conditions, more than 3,000 at 15 at % at least with a maximum burn-up level of around 24.5 at %

- In Japan, about 50,000 pins (with solid pellets) have been irradiated in JOYO and foreign fast reactors, with a maximum burn-up level of around 13 at % in JOYO and 15 at % in FFTF

- In the former USSR (and recently in Russia), a great experience was gained in vibro-compacted MOX fuel, a record burn-up level of about 35 at % has been sucessfully reached with an experimental subassembly in BOR-60 (6 fuel pins) and about 260 standard fuel pins have attained burn-up levels of 25-30 at % (June 1997). More than 4,000 fuel pins with pellet MOX fuel were irradiated in BN-350 and BN-600; maximum burn-up in BN-600 is 11.8 at % (June 1997).

2. It is apparent that the cladding, rather than the fuel or the wrapper material, provides the greatest potential limitation to reaching high dpa levels, and thereby hinders high fuel burn-up. The major degradation problems are void swelling for austenitic, and to a lesser extent, embrittlement at low temperatures for martensitic and ferritic-martensitic steels. The maintenance of desirable properties is directly coupled with maintaining a stable microstructure against the action of neutron-induced displacements. The most important requirement for such stability is well-defined specifications and well-controlled production methods.

3. There are three paths toward achievement of high fuel burn-up: 1) low nickel austenitic steel; 2) martensitic and ferritic-martensitic alloys, and 3) high-nickel nimonic PEI6 alloy. Of these, the lower-nickel austenitics are inherently unstable during irradiation, and eventually must swell. If fuel burn-ups of >20% are to be achieved in fast reactors, the combination of austenitic cladding and the use of one of the other two swelling-resistant materials for wrappers may pose a problem.

Recommendations and proposed actions

1. It is recommended that the IAEA arranges a special meeting on MOX and other fuels' behaviour with the agenda covering the following issues: fuel swelling, gas release and fuel behaviour under transient and accidental conditions. The TCM participants support the IAEA Symposium on Fast Reactors and their related MOX and other Fuel Technologies for Medium and Long-term Utilization: Experience, Advances and Trends, which will be held in Spring 1999 in Obninsk, Russian Federation.

2. It is recommended that the fast reactor community recognize their unique role as the major source of high dose data required for other reactor programmes such as fusion, light water and spallation devices; and that special attention be paid to reach out to these programmes to invite continued irradiation in fast reactor facilities. IAEA and other organizations should

incorporate such outreach effort as part of their goals.

3. A special IAEA sponsored workshop on the application of both fast reactor and thermal reactor data to other nuclear system concepts should be held in the near future.

4. A somewhat greater emphasis on fundamental studies and theoretical understanding should be included in future workshops on structural materials.

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Summary Report of the Sixth Consultancy on "IAEA/EC Comparative Calculations for the ULOF (Unprotected Loss-of-Flow) in the BN-800 Reactor"

June 30 - July 3, 1997, Brussels, Belgium

PURPOSE AND ORGANIZATION OF THE CONSULTANCY

The purpose of the consultancy was to discuss the progress of the IAEA/EC sponsored comparative calculations and to clarify and agree on the aspects of the exercise which were left open during the last consultancy meeting held in Vienna from 4-6 December 1996.

The specific objectives of this consultancy were :

- to discuss and approve summary reports on neutron physics input data, steady state fuel pin characterization and pre-boiling transient analysis,
- to discuss the results of boiling transient analyses for ULOF upto first fuel pin failure for base cases and parametric case,
- to discuss results of post-failure ULOF analyses for base and parametric cases,
- to discuss scope and time table for finalization of the ULOF analyses,
- to discuss status of UTOP transient calculations and UTOP-ULOF calculations, and
- to define work programme upto the next consultancy meeting.

Four countries and two international organizations took part in the meeting, i.e.: France, Germany, India, Russia, IAEA and EC being represented by the following six participants :

(1)	J.M. FRIZONNET	IPSN, Cadarache, France
(2)	D. STRUWE	FZK, Karlsruhe, Germany
(3)	OM PAL SINGH	IGCAR, Kalpakkam, India
(4)	I. CHVETSOV	IPPE, Obninsk, Russia
(5)	A. RINEJSKI	IAEA, Vienna, Austria
(6)	G. VAN GOETHEM	EC, Brussels, Belgium

Mr. NIWA from Japan, though did not participate in the meeting, sent his report to Dr. STRUWE for presentation in the meeting.

Mr. RINEJSKI from IAEA welcomed the participants, outlined the importance of the comparative study, presented status of nuclear power in the world in 1996 including the plants connected to grid in 1996 and the plants to be connected to grid in 1997 and the projected nuclear power growth upto 2050.

Mr. VAN GOETHEM from EC also welcomed the participants and pointed out that the Commission has proposed for the next RTD framework programme to further cooperate with CEEC and CIS countries on priority basis. He also pointed out that EC has a concern on decreasing number of nuclear expertise and is supporting member states to maintain expertise on fast reactor technology.

Mr. RINEJSKI proposed Dr. D. STRUWE to be the chairman and Dr. OM PAL SINGH to be the co-chairman of the meeting and provided the agenda of the meeting (see Appendix A). The participants agreed for this.

The chairman mentioned at the outset that the inputs of the project are confidential and if any participant parts away any information of the project to any third party, all other participants should be informed about it.

The following are the highlights of the meeting:

1. THE MEMBERS APPROVED THE MINUTES OF THE 5TH CONSULTANCY MEETING.

2. LIST OF ACTIONS OF THE 5TH CONSULTANCY MEETING WERE ALSO REVIEWED AND APPROVED.

3. SUMMARY REPORT ON NEUTRON PHYSICS INPUT DATA

The report prepared by IPPE was discussed. The report was approved with the comments that the changes suggested in the text be incorporated and the modifications as indicated in the list of actions (see Appendix B) be implemented and final version of the report be submitted to IAEA within 3-weeks after the meeting.

4. SUMMARY REPORT ON STEADY STATE FUEL PIN CHARACTERIZATION

The report prepared by FZK was discussed. By and large, the report is found to be good and is approved by the participants with some minor changes marked in the text.

5. SUMMARY REPORT ON PREBOILING TRANSIENT ANALYSIS

The report prepared by IGCAR was discussed and found to be too brief. A list of actions (see Action 4 in Appendix B) was suggested to modify the report. It was agreed upon that IGCAR will modify the report and circulate it to all the participants before August 15, 1997 and final version of the report will be prepared by August 31, 1997 after incorporating the comments of the participants.

6. RESULTS OF BOILING AND POST FAILURE ULOF ANALYSES

Four organizations (PNC, IPSN, FZK and IPPE) presented the results. The results of PNC, Japan were presented by Dr STRUWE in the absence of Mr. NIWA.

6.1 PNC Japan [1]

The results are based on SAS.4A code as used by Japan. The code is only slightly different than which is used by IPSN and FZK.

The present results are based on the modified two phase boiling in which the heating of sodium vapour after dry out of clad surfaces is considered by heat transfer from clad to sodium through convection and radiation is considered. The input data were provided by FZK in 31 channel representation and 17 channel representation. The 31 channel representation of core was followed and two cases were analysed :
Reference case : In this case, expansion of radial core and control rod is ignored. Plenum gas blow out is also neglected but the molten cladding motion is considered.
NOCLAP case : In this case, motion of molten cladding is also neglected but heat capacity of the molten cladding is taken into account. This case is studied to find the time of fuel melting and pin failure. In reference case, calculation breaks down because of melting of wrapper before fuel melts.

The main conclusions are :
In reference case, boiling takes place in the upper part of the fissile zone in channel 5/1 at 19.88s and then it spreads rapidly to upper sodium plenum giving negative reactivity and therefore the power reduction. Subsequently, the boiling propagates towards the central fissile zone giving positive reactivity and power increases from 0.40 Po to 0.69 Po; Po being nominal power. In the mean time, boiling is initiated in other channels also in the upper part of the core which spreads to sodium plenum and gives negative reactivity. This reduces the power. Thus, power oscillations are observed. Negative reactivity from sodium plenum is not enough to quench the boiling. First clad melting is discovered to occur in 4/1 channel at 26.115s at 3/4 height of the fissile zone. At 29.755s, subassembly wrapper melts due to radial heat transfer from fuel and calculations breaks down because of the limitation of the code. No fuel failure is observed upto 29.755s. For NOCLAP case, the progression of the accident is same as for reference case upto 26.115s : the time of first clad melting. First fuel melting occurs at 35.85s and first fuel pin failure takes place at 39.818s.

6.2 IPSN France [2]

The contribution from France covered also the results of a base case calculation as well as a parametric case where the clad motion was not taken into account. The calculations were performed with the new release of the SAS.4A code (REF96.R1) on the basis of input data set provided by Germany.

Due to the evolution of the JOG modelling, the steady state results for the low burnup channels were slightly modified : lower fuel-clad heat transfer coefficients were obtained leading to slightly higher fuel temperature and fission gas release. The increase power to flow ratio led to sodium boiling initiation in channel 5/1, 19.2s after LOF initiation. Then, sodium boiling spread over 17 channels. During the boiling period, sodium reactivity oscillations were observed due to the voiding of the upper sodium layer

(negative effect) followed by the downwards boiling propagation into the core (positive effect) successively in the different groups of low burnup channels (highest power channels). Clad motion was initiated in channel 5/1, 6s after boiling onset and later on in channel 1/1. This induced first a limited steel accumulation at the upper core fissile end (negative reactivity effect) followed by steel drainage to the lower end of core fissile height (positive reactivity effect). At the end of the calculation, melting temperature of the hexean was reached and thus the limitation of the use of SAS.4A code was met. Further core material relocation would become 2-dimensional which cannot be simulated by quasi-one dimensional approach followed in SAS-4A code. The accident is expected to evolve slowly towards continuation of sodium voiding and clad relocation and to fuel melting and slumping. The accident sequence enters into a transition phase.

The parametric case was performed in order to delay the achievement of applicability limits of the code and to have a look on the tendancy of the continuation of the calculation. Boiling propagation in the core is rather slower than in the reference calculation due to the lower power in the absence of positive reactivity effect due to the clad motion. But as expected, sodium reactivity oscillations continued and amplified due to the expansion of sodium voiding radially (26 channels) and axially. Fuel melting occured and pin failure was calculated in channel 1/1 at the end of the calculation (18.9s after boiling onset). It was noticed that, at that time, the hexean temperature reached 2820 K and thus the SAS-4A code was used well beyond its applicability limits. Nevertheless, this non-conservative parametric case (neglecting the positive reactivity due to clad motion) have shown the tendancy of the core to enter into fuel melting and slumping and thus to the transition phase.

Thus, the results are similar to as those as provided by Japan.

6.3 FZK Germany [3]

FZK analyzed two phase boiling using SAS-4A Ref96.Re10.1-mod FZK code which has a modified two phase flow model when compared to the original code provided by ANL. In addition, this code version applies modified DEFORM-C model when compared to the one used by IPSN and PNC. The calculations are done in 17 channels and 31 channels representation and using scaled down LSPB-PRIMAR-4 representation of the primary and secundary circuit. The conclusions of the study are the following :

- The negative reactivity feedback contribution due to the voiding of the upper sodium layer in a quite efficient measure to mitigate the impact of the positive reactivity feedback contribution when the fissile core height becomes voided.

- As a consequence of the ULOF, the normalized power becomes considerably reduced and is stabilized at a low value after boiling onset. However, this power reduction is not sufficient to prevent clad dry-out and clad melting to occur in several subassembly groups. This holds for

both cases i.e. without and to a lesser amount with consideration of radial core expansion.

- In subassembly groups where clad relocation is calculated blockages of the coolant channels at the upper fissile core end and more massive ones at the lower fissile core end are calculated to become established in a few seconds. Clad relocation results in a considerable positive reactivity feedback.

- Extended steel blockage formation at the upper fissile core end modifies the materials configuration significantly as compared to the initial core configuration taken as basis for the determination of reactivity worth curves. Therefore it becomes necessary to evaluate whether the negative reactivity feedback contribution associated to the voiding of the upper sodium plenum is still maintained (streaming correction of the reactivity worth distribution allocated to sodium voiding of the upper sodium plenum after clad relocation onset necessary).

- A few seconds after clad relocation, onset of melting of the inner hexcan surface is expected due to the radiative heat transfer from the heated-up outer fuel pellet surfaces to the inner hexcan surface. Subsequently a mechanical collapse of the concerned subassemblies is to be expected. The positive reactivity feedback effect as a consequence of such a solid state fuel fragments compaction can hardly be calculated reliably with a deterministic model. Therefore consequences of such events should be analyzed parametrically (external positive reactivity ramprate and total amount to be defined).

- The ULOF-transient is not at all finished at the end of the SAS-4A calculations. Establishment of longterm coolability of the partially destroyed core is not to be expected neither in the case (A) nor in the case (B) calculation. A quite complicated transition phase will follow where material relocation phenomena will deviate considerably from the quasi-one-dimensional behaviour in the initiation phase due to the calculated hexcan melting.

- It is hardly possible to predict the behaviour of the core design during the transition phase. It is most probably characterized by a slowly progressing core melt-down interrupted by multiple recriticality events. Whether these might lead to a mechanical loading of the heated-up structures of the primary system cannot be predicted, because termination of the ULOF-transient is only possible if fissile core material is efficiently removed from the original core region. Relocation paths are difficult to become accessible and material relocation is rather limited as long as the fuel stays practically solid.

- Present results of the comparative exercise does not allow yet to conclude that ULOF-transients in a BN-800 like reactor core design leads only to benign consequences in terms of the thermal and mechanical loading of structures of the primary system.

6.4 IPPE Russia [4]

IPPE participant informed that the sodium boiling results for base case reported in 5th consultancy were not changed and remains valid today. The importance of simulation of melted steel relocation phenomena for correct prediction of ULOF accident sequence was realised. Russia presented a brief description of "Candle model" developed for simulation of the transient clad melting and relocation under the "soft" boundary conditions low power level. In the frame of the model the simulation of refreezing process of the molten steel due to the contact with relatively cold structures and the crust formation on the outer surface of the liquid steel film is possible. Some preliminary results of code testing were presented. Participants were informed that "Candle" model is already implemented into the code GRIF-SM and the calculations of the ULOF accident for BN-type reactor was already started.

7. STATUS OF UTOP TRANSIENT ANALYSIS

Participants analyzed the UTOP transient for 0.05$/s and 0.5$/s reactivity ramp; the total reactivity added restricted to a maximum value of 3.9$. France, Germany, India and Russia reported the results of analysis.

7.1 IPSN France [5, 6]

SAS-4A calculations were presented by France [5] for slow UTOP (O.05$/s) as well as the fast one (0.5$/s). In the slow UTOP calculations, the imposed reactivity was mainly counterbalanced by the Doppler effect. The negative effect of the fuel axial expansion (linked to the balance of force model used) as well as the negative effect due to sodium heating also played a role. Net reactivity was then rapidly stabilized to a small positive value. Thus a quasi-linear power ramp was initiated and led to sodium boiling in channel 5/1, 72.5s after reactivity ramp muation. Six channels were calculated to boil but as in the LOF calculation, sodium voiding extended downwards into the core and led to sodium reactivity increased which pushed the relative power to higher values and to pin failure in channel 5/1, 4.9s after boiling initiation. Then after a limited fuel accumulation around the failure location which increased slightly the relative power, a significant fuel dispersal due to fission gas release led the relative power to 0.6 at the end of calculation.

In the fast UTOP, the calculated scenario of the accident was more rapid but very similar slow UTOP. Boiling onset was reached in channel 5/1, 7.7s after reactivity ramp initiation. Thirteen channels were predicted to boil due to the higher power reached during the boiling phase and which was consecutive to the higher reactivity ramp rate. It was noticed that, due to hydraulic coupling via the inlet plenum, the increase of the inlet plenum pressure during the boiling phase allowed some boiling channels to recover single phase sodium condition. But as for the slow TOP calculation, in the leading channel 5/1, the core was voided leading to an increase of the sodium reactivity and thus of the power. Pin failure was calculated 1.3s after boiling onset in channel 5/1. Then, as for the slow TOP calculations, limited fuel accumulation is calculated to occur and to increase the relative power, but it is followed by a

significant fuel dispersal, again by fission gases which led the relative power to 0.7 at the end of calculation.

It was concluded that for these two UTOP initiators, the integrity of the fuel pin could not be maintained and that the accident was expected to enter into the transition phase.

After discussions, it was noticed that the failure mechanism (pin break-up of mechanical clad failure) as well as the data used for the failure evaluation (15-15 titanium stabilised or 316 stainless steel cold work cladding, temperature and strain rate dependencies) have to be investigated in more detail.

Then, the basic phenomena [6] which are expected to occur in reactor condition due to a possible fuel squirting effect were presented. The new associated model (EJECT) was developed and implemented in the SAS-4A new code version (REF96R1) by IPSN. Negative reactivity effect can occur prior to the failure due to in-pin fuel squirting in the central hole towards the plena, while positive reactivity effect can occur after the failure due to in-pin fuel motion towards the axial failure location. Validation of this new modelling is being performed on CABRI experiments, but there are still some difficulties to be solved linked to the mechanical modelling which are coupled to squirting calculation and thus cavity pressure oscillations may be obtained and can stop the calculations. These difficulties were met in the UTOP calculations using the fuel squirting option. In the slow UTOP, calculations stopped 1.7s after the beginning of fuel melting ans squirting in channel 5/3. For the fast UTOP, calculations stopped 0.11s after onset of fuel melting and squirting. Up to the end of the calculation, no reactivity feedback effect due to fuel squirting was observed due to the very low fuel melt fraction.

It was pointed out that EJECT modelling refinements as well as experimental validation are needed prior to continue these calculations.

7.2 FZK Germany [7]

The calculations have been done in 17-channel representation of the core, ignoring radial core expansion, scaled down LSPB-PRIMAR-4 representation of the primary and secondary loop and using SAS-4A.Ref96.Re10.1-mod FZK, an improved DEFORM-4C model and different assumptions to determine axial fuel pin expansion. It is observed that for both the ramp rates, fuel pin fails in a few subassemblies at 80 % core height when clad strain reaches a value of 1 %. Subsequent to the clad failure, calculated fuel relocation lead to a rapid reactor shutdown and partial blockage of concerned subassemblies. Blockage formation results in a configuration which can hardly be cooled in-place on a long time scale. A slow core meltdown with a progressive core destruction is likely to take place.

7.3 IPPE Russia [4]

Russian GRIF-SM code was applied for analysis of two cases of UTOP accidents with different reactivity ramp rates : 0.05$/s and 0.5$/s. For both cases the insertion of external reactivity leads to power growth and fuel pin

melting, but for the second one in a shorter time scale. Cladding stays intact at the moment of fuel melting. Boiling of the coolant starts at the moment when approximately 50 % of fuel is melted in the pin belonging to the "hottest" channel. Fuel pin failure conditions were not evaluated as there is no coupling to a transient fuel pin mechanics code package available.

7.4 IGCAR India [8]

IGCAR presented results based on the improved algorithm to calculate fuel melt fraction and by accounting reactivity feedback from sodium plenum more accurately. Results are based on 2 criteria of fuel pin failure following transient over power accident : 1) when the fuel melt fraction is 50 % and 2) when damage parameter defined based on TREAT tests on oxide fuel, exceeds 0.0038. Two sets of calculations were performed, i.e. by ignoring and accounting the load pad reactivity feedback. The conclusions of the study are :

- Fuel pin failure occurs in the upper part of the core.
- Fuel pin failure propagation is incoherent in space and time.
- Load pad feedback delays the initiation of meltings but is not sufficient to prevent fuel melting.

8. STATUS OF UTOP-LOF CASE CALCULATIONS

In the 5th Consultancy meeting, Russian participants introduced into the discussion a new proposal to analyze UTOP-LOF with the following conditions :

- Reactivity ramp rate of 0.05 $/s for 10 s at full flow
- At 1.15 times nominal power, a coolant mass flow reduction to be initiated to a value of 25 % of the nominal flow.
- At the same time, an additional reactivity ramp rate of 0.05$/s is also to be initiated.

Russia, India and Germany reported the results of analysis for this initiator.

8.1 IPPE Russia [4]

Russia presented preliminary results of transient calculations of UTOP-ULOF accident for three different sets of boundary conditions. Dr. CHVETSOV informed participants that the accident scenario that was presented at the last meeting is very close but not fully corresponds to the real scenario that was adopted for BN-800, and in which one second delay between scram signal formation and triggering primary mass flow reduction is available.

So two variants of accident scenario were analyzed by Russians: one with one second time delay and the second without any delay. Only the first stage of the transient including core heat up and initial part of boiling phase was analysed.

As it follows from the calculations, the sequence of events is generally similar to that for ULOF analysis but for UTOP + ULOF case due to additional reactivity insertion, sodium boiling starts earlier on 11th second and at a time when reactor power is higher: 1.67 times of initial power.

The calculations of case 2 (without 1 second time delay) show that transient behaviour of main parameter did not change significantly. The boiling starts approximately one second later.

8.2 IGCAR, India [8]

The results were reported as time profile of power, reactivity components and fuel, clad and coolant temperature. It is observed that power rises to 15 % after 4.39s of the start of the transient and so at this time LOF and second TOP is starts. Sodium boiling takes place at 10.5s in the upper part of the core which gives negative reactivity, and that wipes out the external positive reactivity.

The continued external positive reactivity and negative reactivity from voiding in other channels lead to oscillatory behaviour in net reactivity and power. Ultimately, clad dry out and fuel pin failure is to occur and lead to consequences similar to ULOF.

8.3 FZK Germany

- The UTOP/ULOF case initiates coolant boiling first at the level of the upper sodium layer. Negative reactivity feedbacks due to the initial voiding of the upper sodium layer mitigates efficiently the positive reactivity feedback due to further voiding of fissile core height. The event sequence is quite similar to the one calculated for the pure ULOF case without radial core expansion. However, the power level stays higher due to the super-imposed external reactivity ramp rate. Fuel pin failure conditions are not calculated because the calculation stops when applicability limits of SAS-4A are exceeded.

- Results of the 17 channel calculations need to be verified by the 31 channel calculations. Radial core expansion should be considered as a parametric case.

9. WORK PROGRAMME

A. ULOF

The present analyses results are to be checked against the changes in neutronic parameters like sodium, clad and fuel worth and Doppler coefficient due to voiding of sodium plenum and clad relocation. Under this background, the following is decided for the next round of calculations.

– IPPE will repeat the calculations of sodium and if possible for steel and fuel worth and Doppler coefficient for the following configuration of the core:

- Sodium vo id fraction 100% in LEZ and MEZ, 80% in sodium plenum, B_4C, shield SA-outlet and lower axial blanket, HEZ and 0% elsewhere.

- Clad material from LEZ and HEZ relocated with a layer of 5 cm in lower axial blanket (75% of clad) and in fuel pin end area (25% of clad).

- Sodium vapor temperature, liquid sodium temperature, clad temperature in the core, clad temperature elswhere, and fuel temperature may be taken respectively as 1300 K, 1150 K, 1500 K, 1150 K and 2200 K. Sodium around compensating rod may be taken as 900K and old geometrical structure may be retained in the code.

The results of the calculations shall be reported in the next meeting.

- IGCAR, India has also been invited to perform the revised neutronic calculations and make the results available for the next meeting.

- FZK and IPSN will check if there is possibility of performing SIMMER-II and SIMMER-III calculations for the transition phase and report in the next meeting.

- In the next meeting it will also be decided whether to attempt SIMMER type calculations or the reactivity effects of dry fuel pellet slumping in the concerned subassemblies is to be simulated by a pre-specified externally supplied reactivity ramp rate.

B. UTOP

Regarding UTOP analyses the following decisions were taken :

(1) Corrections of neutronic data applicable to voided / relocated fuel and clad material, will also be applied to the UTOP calculations after fuel failure.

(2) Incoherence in procedure followed by participants to determine fuel pin failure was observed. Therefore, keeping in view the calculational capabilities of participating organizations, the following was decided for the next round of calculations.

- FZK and IPSN will follow the balance of force (base case) model for fuel axial expansion and clad expansion as a parametric case. For calculating cavity pressure, pressure in melt cavity alone will be the base case and as a parametric case, instantaneous equilibrium and EJECT application will be used.

- IPPE and IGCAR used fuel expansion alone for fuel pin axial expansion effect. Both the organizations are advised to use damage parameter model of BAARS for fuel pin failure calculation.

(3) IPPE will collect experimental data base concerning temperature dependence of BN-800 clad steel for yield strength, tensile strength and failure strain and send to participating organizations within 3-months after the meeting.

C. UTOP-ULOF

The following was decided for the next round of calculations :

(1) In the initiating event of UTOP one second delay will be provided in the initiation of LOF once the power is reached 1.15 times the nominal power.

(2) Maximum reactivity to be added in the initiation of TOP at 1.15 times the nominal power, will be provided by IPPE within 3-weeks after the meeting.

DATE AND PLACE OF NEXT MEETING

It was decided that the next consultancy meeting will be held in Vienna on December 10-12, 1997.

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[2] J.M. FRIZONNET and C. RONGIER, "SAS-4A Calculations (Ref96.R1) for ULOF, Base Case Results upto Hexcan Melting and Parametric Case Results Without Clad Motion upto Fuel Pin Failure" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[3] D. STRUWE, B. PORSCHA, W. PFRANG and M. EIGEMANN, "Results of ULOF-Analysis for BN-800 like Reactor with Near Zero Void Worth Core Design: Revision 1" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[4] I. CHVETSOV, I. KOUZNETSOV and A. VOLKOV, "Preliminary Results on EC/IAEA Comparative Calculations for ULOF, UTOP and ULOF/UTOP" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[5] J.M. FRIZONNET and C. RONGIER, "SAS-4A Calculation (REF96.R1) for UTOP Reference Case" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[6] J.M. FRIZONNET and C. RONGIER, "SAS-4A Calculations (REF96.R1) for UTOP, Status of Parametric Case Calculation with Fuel Squirting" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[7] D. STRUWE, B. PORSCHA, M. EIGEMANN and W. PFRANG, "Results of UTOP Analysis for a BN-800 like Reactor with Near Zero Void Worth Core Design: Revision 1" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

[8] R. HARISH and OM PAL SINGH, "Analysis of Unprotected Transient Over Power and Loss of Flow Accidents in BN-800 like Reactor" 6th Consultancy Meeting, Brussels, 30 June to 2 July 1997.

APPENDIX B

LIST OF ACTIONS

ACTION 1 : IPPE to modify the summary report on "Neutron physics data" as follows:

- Modify cover page and skip the table of contents.
- Acknowledge in footnote that the work was partly funded by the European Commission under contract number COSU-CT94-0094.
- Replace the word "disturbance" by "perturbation" in the text.
- Replace on page 4, last line by "Russia, with active participation from Dr. STRUWE (FZK, Germany) and Dr. OM PAL SINGH (IGCAR Kalpakkam) on all stages of work.
- Add one paragraph on uncertainty in physics parameters by referring to intensive discussion held in fourth consultancy meeting held in Brussels in June 1996.
- Final version of the report to be submitted to IAEA within 3-weeks after the meeting.

ACTION 2 : IPPE is invited to inform partners about the status of validation of the code, KONDOR with respect to fuel, burnup, temperature, etc at one of the next meetings.

ACTION 3 : FZK to modify the report on "Steady state fuel characterization" as marked in the text during the discussion on the report.

ACTION 4 : Summary report on "Pre-boiling transient analysis", IGCAR to modify the report as follows:

- Wherever there are differences in the results of different countries, physical explanation is to be provided.
- Explain why IGCAR reports onset of boiling in channel 2/1 and not in channel 5/1 as reported by other participating organizations.
- Provide plots of individual reactivity feedbacks like Doppler, fuel axial expansion and sodium expansion feedbacks. Also add the plot of flow vs. time.
- Document axial discretisation (mesh size in fissile zone, sodium plenum etc.), coupling of gap conductance with thermohydraulics during transient and the different high thermal output of 1725 MWt in Russian and Japanese studies and low thermal output of 1525 MWt by others.
- Include discussion on inter-wrapper flow considered by IPPE in their calculations.
- The revised report to be submitted to IAEA by August 15, 1997 for obtaining comments from other participants and the final report is to be submitted to IAEA by August 31, 1997 after incorporating the comments from others.

INTERNATIONAL ATOMIC ENERGY AGENCY

Second Research Co-ordination Meeting of the IAEA Research Co-ordination Programme on *Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures*

Summary Report

1. INTRODUCTION

The RCM was held in Taormina, Italy, from 25 to 29th August 1997, in conjunction with the Post-SMiRT Conference Seminar on "Seismic Isolation, Passive Energy Dissipation and Active Control of Vibrating Structures". The RCM participants included the Chief Scientific Investigators (CSIs) - or their alternates - from India, Italy, Japan, the Republic of Korea, the Russian Federation, the United Kingdom, the USA and the European Commission.

The purpose of this RCM was to review the benchmark results obtained by various organizations for the analysis of experimental data on seismic isolators. One key objective of the meeting was to discuss the seismic analysis codes and methods developed in various countries.

The welcome address was made by Dr. Martelli on behalf of the ENEA, and followed by opening remarks by Mr. A. Rinejski, the Scientific Secretary of the meeting, on behalf of the IAEA. Presentations of representatives from India, Italy, Japan, the Republic of Korea, the Russian Federation, the United Kingdom, the USA and the European Commission followed.

2. SUMMARY OF THE RCM's TECHNICAL SESSIONS

India

The results of numerical prediction of the static deformation characteristics of HDRB of Italy and NRB of Japan have been discussed. For Italian HDRB, three types of modeling have been adopted viz. 3-D model of 180° sector, axisymmetric model and single layer model. The computer code used is ABAQUS 5.6.1. The hybrid CD8H element and reduced integration element, C3D84 are used to model rubber and steel shims respectively, for the 3-D and single layer models. For the axisymmetric model, CAXA8H1 and CAXAR1 of ABAQUS have been used. The vertical and horizontal loads are applied at the top layer of the HDRB. The rubber material is characterized by the polynomial (N=2) and Ogden (N=2) forms of the strain energy function. The required material constants are obtained by fitting the experimental data provided by Italy. The following three types of analyses have been completed: 1) combined compression, with vertical load of 50kN, and shear up to 400% (120mm); 2) combined compression, with vertical load of 100kN, and shear up to 250% (75mm) and 3) compression up to 1200% of design vertical load (600kN). From the results of the analyses, horizontal and vertical displacements have been extracted and load-displacement curves plotted. The numerical predictions are good under combined compression and shear in the normal working range. However, at large loads marginal differences between analysis and experiment have been observed. The numerical predictions under pure compression are also good up to 100kN (200% design vertical load), but about 25% less than the experimental value when the vertical load is 1100% of design value. The predictions of all the three models (3D, axisymmetric and single layer) are virtually the same.

The single layer model can thus be adopted effectively for the purpose of validating the constitutive model of rubber as it takes less CPU time.

For Japanese NRB, only the 3-D and single layer models are studies. The modeling features are similar to the Italian HDRB, and ABAQUS has again been used for the numerical analysis. The only difference is the characterization of the rubber material. Since Japan has supplied the derivative of the strain energy density function, the material behaviour of rubber is described through the user subroutine UHYPER available with ABAQUS. The following analyses have been completed: 1) combined compression and shear up to 300% and 2) compression with different offset shear strain. The comparison between numerical and experimental results is good for the case of combined compression and shear. The single layer model for NRB also compares very well with experimental data. The poor agreement between numerical predictions and experiment for compression with different offset shear strains is attributable to the assumption of incompressible rubber behaviour in the constitutive equations. From the above analyses, it is concluded that FEM can be regarded as a useful tool in predicting deformation characteristics of laminated rubber bearings used for seismic isolation.

Italy

In order to assess the accuracy of both the Finite Element (FE) model meshing and FE solver, the torsional benchmark problem presented by TARRC was analysed by the Italian participants. Prediction of the couple and axial load acting on the cylinder was obtained from FE calculations using ABAQUS, version 5.6, and was compared to the analytical solutions for two types of material model, Mooney-Rivlin and Ogden. An extensive numerical work was performed by considering meshes with different refinements and different element types. The results, presented ruing the RCM, showed an excellent agreement. Compression and combined compression and shear behaviour of the Italian High Damping Rubber Bearing (HDRB), CRIEPI Natural Rubber Bearing (NRB) and ALMR HDRB were analysed, using different FE models and different hyperelastic strain energy functions, by means of the ABAQUS code. The results of the extensive numerical activities performed by the Italian participants were presented during the RCM technical sessions, and are reported in more detail in three different papers provided to the Agency. Elastomers' constitutive models, currently available in the ABAQUS package to represent the non-linear and hysteretic behaviour of bearings, were used to solve the problem. Mooney-Rivlin and Ogden forms with different order of the strain energy function, and, alternatively, user-defined strain energy function, were used in modeling the rubber. Results from different meshes, with different discretizations and different types of elements (axisymmetric, axisymmetric with non-axisymmetric deformation, shell and solid) were compared to the experimental data. The choices of the appropriate strain energy function, element selection and FE modeling were examined and their ability to model the behaviour of elastometic bearings presented and discussed. The Italian participants provided the Agency with the data concerning a steel frame mock-up which was subjected to a wide-ranging experimental campaign in the framework of a project funded by the EC. Tests results concerning different seismic excitations for both the fixed-base and isolated

configurations have been selected and proposed as the benchmark exercise. Data included seismic excitations, recorded displacements and accelerations as well as bearing characterization. The data was made available by ENEL on a dedicated website, and the necessary instructions on how to get the data were explained to the RCM participants.

Japan

Central Research Institute of Electric Power Industry (CRIEPI) has conducted various rubber bearing tests within the research project on applying seismic isolation to planned fast breeder reactor. Of these test results, data on natural rubber bearing (NRB) and lead plug rubber bearing (LRB) were used for the comparison with numerical simulation. Test data are those obtained by shear loading test under design compressive load and compressive loading test under given shear deformations. Numerical simulations were conducted for these test results as well as for the high damping rubber bearing (HDRB) tests for which data was supplied by Italy via ABAQUS, using strain energy density functions based on bi-axial tensile test of rubber strip. As for the shear loading tests, simulation results agreed with experimental ones, in particular for NRB. However, in the case of LRB, discrepancy between experiment and simulation was observed in the region of small shear displacement, where lead plug has an effective role in the mechanical characteristics of LRB. Also in the case of HDRB, discrepancy between the experiment and the simulation was perceived where shear displacement is large. As for the compressive test results, numerical results showed good agreement with the experimental ones. Outline of shaking table test for the base isolated rigid mass, conducted by CRIEPI was shown. The rigid mass is supported by eight LRBs and the input motions of various intensity were used for excitation. In the case where design earthquake motion was used, base isolation effect, i.e. decrease of response compared with input motion was shown per each type. For the input motion three times as large as the design earthquake, where the rubber bearing entered its hardening region, base isolation effect was not shown. Numerical simulation was conducted in the case where design earthquake motion was inputted, modelling mechanical characteristics of LRB with bilinear model, where parameters specifying the model were evaluated from the rubber bearing test, and both test and simulation results showed good agreement.

Republic of Korea

The combined shear and compression behaviors of the ENEA HDRB(High Damping Rubber Bearing) with low shear modulus and the CRIEPI LRB(Lead Rubber Bearing) are analyzed using the hyperelasticity material option of the ABAQUS computer program. The purpose of the analyses is to predict the quasistatic force-deflection behavior of laminated rubber bearing from rubber test data. Several kinds of strain energy density function representing the hyperelastic characteristics of rubber are tried. For ENEA HDRB, using the 2D model the solution accuracy is investigated by changing the number of the elements for a rubber layer in the rubber thickness direction. The Ogden model is used in this analysis. The horizontal force and displacement relations show no differences with respect to the number of the elements for a rubber layer. According to the material models of rubber, however, there are some different results. The Mooney-Rivlin model, with a linearity of material properties, gives a large discrepancy from the test results, but the Polynomial (N=2) and the Ogden (N=3) methods give results which compare accurately with the test results, up to a shear displacement of 60 mm. The 3D solid model, the results using the Polynomial and the Ogden models are close to the test data in the lower shear strain region up to 200% which equals the shear displacement of 60 mm. The Seki model calculates a higher shear force than the test data, and the Mooney-Rivlin model shows much difference from the test data. In computing efficiency, the 3D model is taken to be above 10 times longer than the 2D model. In vertical deformation analyses, a minimum of two elements in thickness direction for a rubber layer is required to get accurate results. The Polynomial and Ogden models are much better than the Mooney-Rivlin model, and the Seki model is the worst because the volumetric tests data are not considered in the formulation. For CRIEPI LRB, the finite element analyses using the 2D element are performed. The two kinds of stress- strain curves of the lead plug are used in these analyses, where the initial yield stresses are assumed by the static and the dynamic yield stresses representing the soft and hard properties respectively. For combined compression and shear analyses, theresults modeled by the hard lead properties agree well with the test ones until a 200% shear strain, at which the convergence problem occurs. The shear force using the soft lead in this analysis is little lower than that of the test results. The calculated horizontal forces according to the rubber material models such as Ogden and Polynomial agree well with experimental ones up to a 200% strain. There is little discrepancy of the solution accuracy between the 2D and 3D models but the computation time using the 3D model is taken to be over 10 times longer than that using the 2D model.

Russian Federation

Development of the Russian multicomponent low-frequency seismic isolation system (SIS) for reactor buildings for VVER - 640 NPPs has passed to the stage of practical realization. As a consequence, choice of the optimal engineering solution is now the pressing problem as regards the seismic isolators, construction of the seismically isolated building, and the seismic isolation system arrangement. The selection procedure is based on the static and dynamic analyses of seismic-isolated structures, design requirements and evaluation criteria, and data from long-term trial operation of a part of the full-scale system. Comparison of methods of analysis of the various isolator types. in the context of meeting the requirements and criteria formulated, is of immediate interest. Indices of dynamic and other technical performance aspects and of costs of seismic isolation are formed, in accordance with the specifications and type of applications, for comparative analysis of the various SIS at the feasibility report phase and for evaluation of the isolation system design. These indices characterize the structural SIS design from the viewpoint of its conformity with the following:-

(a) requirements of the overall seismic isolation system, which cover its dynamic behaviour, other technical characteristics and costs
(b) requirements of the SIS individual components and
(c) requirements of the seismic isolation system to fulfil its protective functions.

United Kingdom

It was noted that the prediction of the force-deformation behaviour of elastomeric base-isolators using finite element analysis requires not only a suitable model for the elastomer, the parameters of which can be readily determined from material characterisation tests, but also accurate finite element solvers. The analytical solution to the problem of the torsional deformation of a cylinder is presented as a "tool" for establishing the degree of accuracy of the finite element solvers used in packages such as MARC and ABAQUS. The analytical solution is expressed as derivatives of the strain energy function used to model the material. It is therefore applicable to any form of the function. Prediction of the couple and axial load acting on the cylinder are obtained for two types of material model (Mooney-Rivlin and Ogden) both from finite element calculations using MARC and the analytical solution. The results generally agree reasonably well. Other work is concerned with the fitting of experimental data to the parameters of the material models.

The finite element (FE) results for the force-deformation behaviour of base-isolators using MARC finite element program have been discussed. These analyses are for both a low damping natural rubber bearing (NRB) and high damping rubber bearings (HDRB). The influences of the rubber material model and the type of element are being studied. Comparisons between the FE results and experimental data have been made.

United States of America

An extensive programme of experimental work on seismic isolators for nuclear application has been carried out at the Earthquake Engineering Research Center (EERC) of the University of California at Berkeley over the past several years. Recent work has focused on isolators for the Advanced Liquid Metal Reactor (ALMR). The reactor platform of the ALMR is supported on a seismic isolation system comprising 66 high-damping rubber bearings. This design is intended to limit the lateral shear transferred to the reactor and associated equipment under severe earthquake ground motions by shifting the fundamental period of the structure away from the most damaging portion of the input spectrum. As part of the design validation of the seismic isolation scheme for the ALMR, a procurement specification was developed, reduced-scale isolation bearings were purchased from two different manufacturers, and an extensive series of tests were performed on these bearings at EERC. Studies were also carried out on high-damping rubber bearings (of the same designs as those evaluated under a previous ALMR bearing test program) manufactured by Unison Industrial Company, Seoul, Rep. of Korea, and provided by the Korean Atomic Energy Research Institute (KAERI). The test program undertaken for evaluating the KAERI high-damping rubber bearings was similar to the ALMR bearing test program. The objective of the reduced-scale bearing test program was to establish bearing mechanical properties and failure characteristics that can be used for later experimental and analytical evaluation of seismic isolation concepts using these bearings. In particular., emphasis was placed on determining the variation in observed bearing shear properties as a function of shear strain amplitude, loading frequency, applied axial stress, and load history. Vertical tests of the isolators were also performed, both with zero horizontal displacement and with one applied. Finally, shear failure tests were performed to establish margins for use in analyses of beyond-design level earthquakes. Two different bearing designs, with scale factors of one-quarter and one-eighth, were tested to investigate the influence of size on isolator properties. It was noted that this aspect is of particular interest, because reduced-scale bearings will be required for earthquake simulator tests of a system model.

European Commission (EC)

The European Laboratory for Structural Assessment (ELSA) of the Joint Research Centre (JRC) has powerful and accurate means for carrying out Pseudo-dynamic (PsD) tests to simulate the behaviour of large scale structures subjected to earthquake loading. The ELSA laboratory is at present engaged in international consortia to develop, optimise and test innovative anti-seismic devices based on passive vibration control. To validate a standard procedure to test base-isolated structures by the PsD method, a collaboration was set up with the Italian Working Group on Seismic Isolation, which includes the national research centre ENEA, the national electricity board ENEL, the industrial research centre ISMES and a producer of isolators ALGA. In the framework of this collaboration it was decided to test at the ELSA laboratory a scaled 5-storey frame structure (provided by ENEL), isolated by means of high damping rubber bearings (HDRBs), a set-up previously tested on the shaking-table of ISMES. This experimental activity aimed to compare the results obtained by means of the PsD testing technique with those obtained

by means of a truly-dynamic test on a shaking-table. The PsD test procedure includes the following steps:

(1) characterisation of the isolators for different frequencies to evaluate the stiffening effect due to the reduced strain rate and the corresponding correction that must he applied to the shear force
(2) comparison between a dynamic snap-back and a PsD snap-back and
(3) PsD tests with seismic inputs and comparison with shaking-table tests.

PsD testing is, by virtue of the expanded time-scale of the test with respect to real seismic events, normally restricted to materials assumed to behave in a rate-independent manner. As regards seismic isolation, though the strain rate effect is present in the isolators, it can be taken into account in the numerical part of the method. Moreover, the quality of the PsD tests described in the paper has been improved by applying an innovative technique called continuous PsD testing, which considerably reduces the test duration and opens the way to accurate testing of the earthquake response of structures with velocity-dependent devices. In the conventional PsD test procedure, the actuator motion is stopped (hold period) when the test specimen reaches the target displacement so that the reaction force can be measured and the next target displacement computed. In continuous PsD testing instead of stopping the actuators they move continuously, and the reaction force is measured when the specimen passes the target displacement. Recognising this reaction force as the restoring force at the target displacement, the equations of motions are integrated "on the fly" (without hold period), the next target displacement is determined and the motion proceeds without any interruption.

3. RECOMMENDATIONS OR ACTIONS PROPOSED

A. Seismic Isolators

It was agreed that the follwoign single isolator types be analysed by the CRP participants:

Table 1

No.	Type of isolator	Country supplying data for analysis
1	Natural rubber bearing (NRB)	Japan (produced in 1995)
2	Lead rubber bearing (LRB)	Japan (produced in 1995)
3	Pneumatic 3D isolation device (3DIS)	Russian Federation, Nov. 1997
4	High Damping Rubber Bearing (HDRB) G=0.4MPa	Italy, produced in 1996
5	High Damping Rubber Bearing (HDRB) G=1.4MPa	USA
6	High Damping Rubber Bearing (HDRB) G= 0.9 MPa	Rep. Of Korea, December 1997

The first step of the analysis concerns single isolators. The Italian and Japanese data have been

produced and have been partly analysed by the partners. Data from Russian Federation have also been distributed but their analysis requires some additional information which will be provided by November 1997. The USA has already sent a set of data to Italy in the framework of previous cooperation. Italy will transmit such data to IAEA for distribution amongst the other partners. The Republic of Korea will provide their isolator data by December 1997 together with material data obtained by ENEA from tests on specimens fabricated from the Korean rubber. A question arose as to whether test data on the Rolling Ball Rubber Layer systems can be made available to the participants for the analysis. Such data will be obtained in the framework of a project funded by the European Commission. The Partners in this project which attended the RCM agreed and will seek agreement from the other Partners and the European Commission to make the data available. ENEL will take care of the formal steps as the prime contractor of the aforesaid project. The experimental data should become available by the end of 1998.

TABLE 2: UPDATED LIST OF PARTICIPANTS CONTRIBUTIONS

	India	Italy	Japan	Korea	Russia	UK	USA
By the end of 1st phase	1,2,4,5,6 3*	1,2,4,5,6 3*	1,2,4,5	1,2,4	1,2,3 4,5,6	1,2,4,5 6	**
By the next meeting	2,5,6 3*	2,3*	2[1], 4[1],5*	5,6	4,5,6	2,5,6	**
Code	CASTEM 2000 & ABAQUS	ABAQUS	ABAQUS	ABAQUS	SEISM & Construction	MARC	**

[1] continuation
* decision to be made later
** ANL to confirm contribution

List of participants' contributions updated as regards both the type of isolators analysed by each country and the work to be done before the next meeting (see Table 2). The decision whether to analyse the Russian data will be taken after the necessary supplementary information has been distributed. Dr. Chung will check whether ANL will be able to perform analytical work on behalf of the USA. The isolator parameters on which the inter-comparison of the results of the analyses are to be based will be selected at the next RCM meeting after the results have been obtained. The torsion benchmark problem proposed by TARRC with the support of ENEL was illustrated and other participants with computer codes other than ABAQUS and MARC were invited to participate in the benchmark analysis. A document outlining the problem was distributed to be included in the minutes as an annex.

B. Seismic Isolated Nuclear Structures

The Italian data concerning the MISS mock-up were sent by e-mail to the IAEA who will distribute them by 15th September. The Agency will also repeat the distribution of Korean data concerning the spent fuel storage pool. The European Commission will provide the Agency with data on the MISS

mock-up subjected to Pseudo-Dynamic tests by the end of September. Professor J. Kelly proposed as the USA contribution to provide by the end of the year data on a reinforced concrete structure model supported by both soft high damping rubber and lead-rubber bearings. The mock-up data will be provided with all the necessary isolator characteristics. The proposal was agreed by the participants. Dr. Hirata informed the meeting that the data concerning the rigid mass mock-up supported by scaled lead-rubber bearings will be provided before the end of 1997, together with data on the characteristics of the isolators. The preliminary agreement on which type of isolated nuclear structures will be analysed in the second phase is summarized in table 3.

TABLE 3: STRUCTURES TO BE ANALYSED: UPDATED PRELIMINARY PLAN

Type	Country	First possible date of supply
MISS* pseudo dynamic	EC**	Sept. 97
MISS* shaking table	Italy	July 97
Rigid mass mock-up	Japan	end 97
Pool	Korea	provided in Dec. 96
VVER-640	Russian Federation	end 97

* Model of Isolated Steel Structure
** Joint Research Centre

Next Meeting

The next meeting is proposed to take place during the last week of May 1998 at TARRC in Hertford, UK. TARRC will confirm the availability of the facilities by the end of September 1997.

Purpose of the next meeting:

1. Presentation of the results of the remaining analyses of isolators and intercomparison of the results of the various countries.

2. Presentation of the experimental test data concerning the isolated structures, detailed discussion of such data and presentation of the preliminary analyses, if any.

Summary Report of the Second RCM of the IAEA's Research Coordinated Programme on "Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes and Relations Using Experimental Data"

Obninsk, Russia, 6-10 October 1997

1.Technical Background

A benchmark exercise on "Tee junction of LMFR secondary circuit" was proposed by France in the scope of the subject 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 calculate, because of the complexity of the phenomena involved. This issue has been encountered in the Phenix reactor on the secondary loop, where initial defect in a tee junction zone was detected during a campaign of inspections after an operation of 90,000 hours of the reactor.

The input data which were provided by Framatome-Novatome (France) come consequently from the actual operation of the reactor. Necessarily, because of the complexity of these data, some have been simplified where possible (i.e. when it was certain that this simplification would not influence the results). Also, the comparison of thermohydraulic results with the actual phenomenon is possible, owing to an instrumentation installed on site during the campaign of inspection. The number of available measurements, although less large than mockup tests could have provided, is nevertheless satisfactory. In addition to these measurements, a lot of visual inspections and metallurgical examinations have been performed and are available. They will make it possible to make comparisons with thermomechanical evaluations. This benchmark provided the opportunity to test numerical approaches on a real problem, not idealized but industrial, involving a lot of parameters and different aspects of the phenomenon (thermohydraulic, thermomechanical, fracture mechanics). It will permit to compare the results of numerical approaches with the actual phenomena observed on site, in order to validate and to verify the adequacy of these approaches to simulate an actual phenomena.

This was the second RCM. The first one was held in November 1996, to discuss the status and scope of contributions to the CRP from participating organizations, to review the case setup data provided by Framatome and to define additional requirements, to review and discuss first results of thermohydraulic and thermal calculations provided by the participating organizations, and to define a detailed work programme up to the next RCM.

2. Conduct of the RCM

The second RCM was held at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russian Federation, from 6 to 10 October 1997. Meeting participants came from France, India, Japan, the Republic of Korea and Russian Federation. The Chief Scientific Investigator from the United Kingdom, although not participating at the meeting in person, sent his findings for presentation at the meeting.
The purpose of this meeting was to perform a first comparison of these results of thermomechanical and fracture mechanics analysis of the tee junction of the LMFR coolant pipes. The key objective of the meeting was to quantify the major differences in the calculation procedures used by the participants and to confront the results with the actual phenomena observed on site (Phenix reactor).

On behalf of the IPPE, Professor Yu. Bagdasarov, the Deputy Director of the Fast Reactor Department, welcomed the participants. The meeting was opened by Mr. A. Rinejski, Scientific Secretary of the RCM, IAEA.

Ten presentations have been made by the Chief Scientific Investigators or their substitutes and a technical tour of the IPPEs thermohydraulic laboratories was organized. During the meeting, Dr. L.A. Kotchetkov, Advisor of the Board of Directors of the IPPE delivered a lecture on Fast Reactor Development in the Russian Federation. Presentations of countries represented at the meeting followed.

3. Summary of the RCM's Technical Sessions

France: A thermomechanical analysis is performed using two kinds of thermalhydraulic loadings: 1)The thermal fluctuations are assumed being in the form of a sinusoidal signal with a constant amplitude equal to the maximum temperature variation in the fluid. The random feature of the fluctuations is not taken into account in such approach. The frequency of the sinusoidal signal is taken within the range (0,1 Hz ; 5 Hz) which is characteristic of turbulent movements inside the fluid and 2) The thermal fluctuations are provided by thermalhydraulic calculations which attempt to reproduce the physical phenomenon. The results of the AEA thermalhydraulic evaluation are considered. In particular, it is considered the temperature history calculated at TC01A location (80 mm downstream of the tee junction) which displays the maximum amplitude (calculation duration : 32.5 s). This loading history is applied everywhere in the tee area, in the circumferential weld and in the base metal. In addition to the fluctuating loadings, the main pipe is submitted to stationary loadings which are the following: a) Hot spot induced by the impact of the jet on the pipe and b) Forces and moments in the pipe. Different computer codes are used to evaluate the stress level inside the base metal and the circumferential weld. Regarding stationary stresses, it is found that: I) -membrane axial stresses vary from 30 MPa to 40 MPa along the circumferential weld due to the presence of the hot spot; ii) low stresses (about 10 MPa) are induced by forces and moments in the pipe. Fluctuating temperatures generate high stress ranges in the wall: a) maximum 310 MPa for a 0.5 Hz sinusoidal signal, and b) maximum peak-to-peak 340 MPa for the predicted random signal. The fatigue damage is estimated using the RCC-MR Code methodology, taken into account plasticity effects and stress concentration factor in the weld. For the random signal, the Rainflow method allows a cycle counting associating each value of stress range with the counted cycle number. Finally it is found that cracks may be initiated in a few hours in the circumferential weld (inner side) of the main pipe for both kinds of fluctuating loadings, sinusoidal or random. On the contrary, no initiation of crack can occur in the base metal during 90 000 hours. Then, a fatigue crack growth is performed in the weld. The Paris law is considered. It is demonstrated that the cracks propagate rapidly through the thickness of the wall and stop at about 1 mm from the outer side of the pipe due to the lack of significant stresses either to propagate the cracks or to break the remaining ligament.

Russian Federation: Using the results of a thermohydraulic analysis, peak-to-peak temperature difference and frequencies of temperature fluctuations on the wall are determined. On the basis of the obtained data, the analysis of stresses and cyclic strength calculation for the pipeline element under consideration was performed by using computer code ANSYS. On the grounds of the analysis performed, with the assumptions accepted, the following is stated: 1) amplitude of temperature fluctuation on inner surface wall decreases with increase of frequency of temperature fluctuations. Also, a penetration of the thermal wave into the wall falls with increase of frequency of temperature fluctuations and, therefore, stresses also decrease; 2) a variation of cyclical damageability is not monotone function of frequency of temperature fluctuations. The cyclical damageability increases in the frequency range from 10^{-2} to 0.2 Hz and it decreases with raising of the frequency, and 3) perturbation at frequency of 0.2 Hz gives the most damageability. In these conditions, the allowable lifetime of the T-junction is equal ~9000 hours as per the of Russian Norms for Strength Calculation.

United Kingdom:[1] A crack initiation assessment of the PHENIX pipe failure has been carried out using methods developed in the UK and approved by AGT9b.[2] Using the maximum temperature amplitudes from the computed short timescale temperature-time histories for the wall, crack initiation is not predicted at any location, including the weld. However, crack initiation is predicted at TC1A for fluid temperature fluctuations for Cases P2 and P3 (fatigue strength reduction factors of 1.6 and 1.9 respectively), and also at TC2 for Cases W2 and W3 (fatigue strength reduction factors of 2.0 and 2.4 respectively). Weld failure can therefore only be suggested if the wall temperature fluctuations are higher, perhaps due to the interaction of the inner weld bead (which stands proud into the flow) with the fluid flow. The assessment method used shows the sensitivity of the results to several factors: 1) The fatigue strength reduction factors actually appropriate for the material under consideration, 2) the computed maximum temperature amplitude for a short time period; 3) the method of estimation of maximum

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[1] Presentation has been made by Ms. Gelineau (France)

[2] Working group to study and produce common design rules in the field of structural integrity (was commissioned for EFR - European Fast Reactor).

temperature amplitude in the component lifetime (or infinite time period); 4) maximum temperature amplitude calculated from Rainflow cycle counting (generally less than the max/min temperature difference; 5) the scaling of Rainflow cycle counts in a short time period to represent those over the component lifetime. The method of scaling Rainflow cycle counts using the Gumbel distribution is recommended in preference to direct scaling of the counts by the ratio of time periods. This method also seems to give a better prediction of maximum temperature amplitude in component lifetime, compared to the maximum predicted by Gumbel analysis using the individual maxima in sub- blocks of the computed short time period temperature-time history. The studies produced the following conclusions: I) crack initiation in component lifetime is not predicted using wall temperature fluctuations; ii) crack initiation is predicted at thermocouple TC1A for parent metal AISI 304 for Cases P2 and P3 using fluid temperature fluctuations and iii) crack initiation is predicted at thermocouple TC2 for weld metal for Cases W2 and W3 using fluid temperature fluctuations. It makes the recommendation that consideration is given to the effect of the inner weld bead upon the magnitude of temperature fluctuations on the weld surface.

Japan: Under the boundary condition provided from thermohydraulic analysis, preliminary 1D and 2D thermoelastic analysis by FINAS code and crack growth analysis with CANIS code, rather focusing on high frequency temperature fluctuation have been carried out. Calculated results indicate that cracks hardly initiate at base metal portions. On the other hand, strength of welded joint is reduced by peak stress concentration at penetration beads and other factors. When taking these strength reduction factors into pessimistic account, and also low frequency temperature fluctuation is assumed, there is possibility of crack initiation within 90000 hours operation time at the welded joint portion. According to fracture mechanics analysis, it is estimated that through-wall cracks may not occur at the welded joint portion, even if initial cracks exist. However, period of crack growth from 0.1 mm to through-wall is sensitive to mechanical loads. If additional mechanical loads exist or low frequency temperature fluctuation is assumed, cracks will propagate with accelerated rate and will pass through the wall. In the next phase, it is planned that influence of low frequency temperature fluctuation will be evaluated by 3D structural analysis model under thermal boundary conditions calculated from long time thermohydraulic analysis.

Republic of Korea: The evaluation of thermomechanical fatigue and fracture of Phenix tee-junction under thermal striping load has been done using an efficient numerical method based on Green`s function concept. The Green`s function approach with Duhamel integration theorem enables to calculate the stress intensity factor as well as thermal stresses very efficiently under arbitrary change of the fluid temperature. The present approach has been verified by standard FEM. The ABAQUS code V5.6 was used for heat transfer, thermal stress and fracture analysis. In heat transfer analysis, the thermohydraulic data received from AEA at the location of TC1A which is 22 mm apart from TC1, were used in the present study. Two-dimensional axisymmetric model at heat affected zone was used. The evaluation result of the fatigue damage for the case of boundary layer attenuation (BLA) considered. Showed no fatigue usage contribution. However, when BLA is to be considered, the fatigue usage may be cumulated. For fracture assessment, the stress intensity factor values determined by Creen`s function approach did not cause crack initiation. However, it seems that Crack propagation will occur for the BLA considered. As for future work, 3D analysis near the welded part considering reaction force moment and evaluation of residual stress distribution is to be carried out.

India: Detailed investigations of the problem of thermal striping in a T-Junction from the fracture mechanics view point have been made. Most of the data required for the fracture mechanics analysis have been given in the technical specification report of IAEA. Due to unavailability of some data other literature and document have been referred to. Fatigue crack initiation have been reported to occur when the stress intensity factor is more than the threshold value of the material. It is also found that surface crack will grow more in the length direction because of the existence of high surface stresses (peak+membrane+bending) on the inner wall surface than in the depth direction because of the sharp decay of stress distribution in the depth direction. This fatigue crack will grow when the stress intensity factor comes to be more than the threshold value. It is also investigated that the stresses caused by the mean temperature distribution along the height of the pipe may cause local ligament rupture after some fatigue crack growth of the initial defect. This way it is a threat to the LBB justification.

Thermal striping of PFBR components is to be considered in the design of LMFBR. The vicinity of bottom portion of the control plug just above the core and also in the mixing tees of the heat transport sodium circuit are

of concern other. It is shown that in case of high frequency thermal striping, the effects of plate thickness and the amount of constraints offered by the surrounding structure are not significant. This kind of effect are significant only for the low frequency cycles. The acceptable temperature limits in case of thermal striping have been established by the RCC-MR approach and the fracture mechanics approach considering the associated low cycle fatigue damage. In addition to above, allowable sodium free level oscillations have been reported corresponding to different associated frequencies calculating the allowable stress ranges by the RCC-MR procedure.

4. Conclusions and Recommendations

4.1. Thermohydraulic (T/H) calculations

1. The purpose of the calculation was to obtain spatial distribution of averaged on time liquid temperatures and velocities, tube wall temperatures, and the temperature fluctuation frequencies and amplitudes. Each participant developed separately his calculation model and chose computer code. In spite of different approaches to the task statement, there was qualitative agreement between the results obtained by different participants for the averaged on time characteristics of flows mixing in the T-junction. Average temperature values for the inner tube wall, obtained for the thermocouple locations are different for different authors. This can be explained by that the flow pattern of ejected jet is mainly determined by the boundary conditions in the lateral nozzle: velocity profile, thin wall elements of tube junction, the extent of jet compression at the nozzle outlet and its turn inside the nozzle. Conclusion has been made on that it is not permissible to restrict the calculation model by the boundaries of area where the change of boundary conditions causes solution changing. Since such requirements to the determination of calculation model boundary were not taken into account, this is the principal drawback of the whole performed work, which caused the differences in evaluation of the average temperatures of the piping wall. Comparison of calculated results with the experimental data presented by Framatome-Novatome has shown that the temperature difference between the neighbouring thermocouples is constant. This is observed if the ejected jet is kept against the piping wall without separation.

2. Pulsation characteristics of the liquid temperature profiles, obtained by the participants of the benchmark, differ much from each other. This is mainly because of quite different methods of obtaining of pulsation characteristics. In some papers, steady state or averaged on time values of the liquid velocity and temperature are calculated first. Then, on the basis of these data, the amplitudes of temperature fluctuation on the piping wall and their frequency range were evaluated using different codes and models. Such temperature fluctuations are caused by the turbulent flow, they have statistical character and high frequency spectrum up to 1 kHz. Their maximum is observed in the vicinity of thermocouple 1, and value decreases downstream. In other works, liquid transient flow and temperature profile are calculated and thus periodical changes of liquid velocity and temperature in the vicinity of the pipe wall. Frequency of these changes is less than 10 Hz. These fluctuations are caused by the interaction of ejected jet and the main flow in the whole volume of the pipe. The span of temperature fluctuations induced on the inner surface of the pipe wall reaches 19-20K in the vicinity of 3-15 thermocouples. In order to obtain the comprehensive picture of temperature fluctuations on the pipe wall, these two types of distribution should be summed. The result should be as follows: liquid temperature fluctuation span near the pipe wall is approximately constant along the wall axis above the ejection point, while the dominant frequency of these fluctuations decreases downstream.

4.2. Thermomechanical (T/M) calculations

The fatigue assessment results are sensitive to various parameters. The results are scattered among the participants due to the different parameters used. The relevant items are the following:

- Low frequency fluctuation of temperature
- Damping of temperature fluctuations
- Mean stress
- Strength reduction factors of weld joints

- Fatigue damage under random cycle.

Low frequency fluctuation of temperature: T/H calculation, because of too short calculation duration and of the complexity of the phenomena involved in this circuit, fail to give complete information needed for T/M analysis. However, low frequency fluctuation shall be considered in T/M analysis as they are able to damage the structure, comparing with high frequency ones. Having an experimental data base including a great number of results could be useful.

Damping of temperature fluctuation: Quantification of the damping ratio due to both boundary layer attenuation and heat transfer by convection is of great importance. Also, damping ratios used in T/M analysis are different between participants. However, it appears that if large damping ratio is considered (>20%), no crack initiation can be proved with small strength reduction factor. An issue may be to take into account the weld bead which could be able to disturb the flow inside the boundary layer and as a consequence to reduce the attenuation.

Mean stress: Mean stresses have a strong effect on the crack propagation assessment, so they cannot be neglected. The question to consider manufacturing residual stresses is asked as most of the participants do not take them into account. A sensitive analysis may be useful to judge of the influence of such stresses. Moreover, even if no participant considered them, the effect of mean stress on the crack initiation would have been discussed.

Strength reduction factor of weld joints: The presence of the weld bead shell be taken into account either by means of geometrical discontinuity factor, or by means of reduction factor applied on the fatigue curve, or using both of these. Due to the large scatter-band of fatigue rupture data, the point is to choose the most correct coefficient regarding the actual geometry and past operating conditions. An issue of this exercise could be to show the better way, in a design approach, to introduce the presence of weld like in the Phenix tee-junction into the fatigue assessment

Fatigue damage under random cycle: The problem raised concerns the combination of high frequency fluctuations to low frequency fluctuations The use of reduction fractions may be recommended in such case. However, no participants have used such a route.

5. Next meeting

The final meeting is proposed to take place at Framatome-Novatome, Lyon, France from 20-24 April 1998.

Purpose of the final meeting: (1) presentation of the results of the remaining thermohydraulic and thermomechanics analyses and intercomparison of the results of the various countries (2) presentation of the final reports and (3) preparation and discussion of the CRP summary.

Technical Committee Meeting on

Sodium Removal and Disposal from LMFRs in Normal Operation and in the Framework of Decommissioning"

3-7 November 1997, Aix-en-Provence, France

Considerable R&D on different aspects of LMFR technology has been done within the span of 40 years and LMFRs are in a better position than generally appreciated. The sodium coolant technology and mixed oxide fuels are largely demonstrated in a number of experimental, prototypes and near commercial LMFRs which have been built and operated.

Some of the existing LMFR plants need of effective maintenance and/or decommissioning technology, especially: i) sodium removal and decontamination of reactor circuits, equipment and structural materials aimed at their reutilization; and ii) bulk disposal of sodium in the framework of reactor plant decommissioning. To this end IAEA's International Working Group on Fast Reactors (IWGFR) at its meeting in May 1996 recommended to convene a Technical Committee Meeting on Sodium Removal and Disposal from LMFRs in Normal Operation and in the framework of Decommissioning for the exchange of information in this field of LMFR technology. The IWGFR and Consultants' meeting on ".....Effective Decommissioning Programme for the BN-350 LMFR in Kazakhstan", which was held in the VIC in October 1996 has recommended that the Scientific Secretary of the IWGFR should act as coordinator and undertake appropriate measures which guarantee that information on sodium removal and disposal technology in leading fast reactor Member States can be provided to all countries.This information would be used in existing plants and/or applied to the decommissioning issues during the design of advanced LMFRs.

The necessary steps have been undertaken and the above mentioned meeting was organized by IAEA/IWGFR in cooperation with the Cadarache Nuclear Research Centre, CEA, France with participation from all IWGFR Member States. The TCM was held in Aix-en-Provence, from 3-7 November, 1997. Eleven Member States (Brazil, China, France, Germany, India, Italy, Japan, Republic of Korea, Russian Federation, United Kingdom, and USA) sent their experts, who presented 29 papers on the technology of sodium removal and disposal and LMFR decommissioning options.

2. SCOPE OF PRESENTATIONS AND DISCUSSIONS

The technical parts of the meeting were divided into three major sections as follows:

- Sodium removal (cleaning) processes
- Decontamination processes
- Bulk disposal of sodium in the framework of LMFR plants decommissioning .

During the meeting, papers were presented by the delegates on behalf of their countries or organizations. The papers were either in the form of a general survey of the subject or on specific technical subjects. In each subject area, presentations appropriate to the subject were made from the submitted papers. The presentations were followed by discussions of the questions raised and summaries were made. On November 6 and 7, 1997, tours of the Cadarache Research Sodium Center facilities and the LMFRs Phenix and Superphenix were organized for the participants of the meeting and a final meeting was held to discuss the resulting summaries, conclusions and recommendations which were agreed upon by the participants and which will be published in the TCM proceedings.

A comprehensive discussion was devoted to sodium disposal process options. After discussion 6 options were identified as being potentially suitable for the bulk of the sodium:

(a) Reaction with caustic
(b) Burning in air (spray fire)
(c) Burning in air (pool fire)
(d) Reaction with steam
(e) Reaction with CO_2
(f) Reaction with water in oil

These 6 options were then evaluated (table 1) against the following criteria:

(a) **Development status** - how well proven is the process for bulk sodium disposal?
(b) **Throughput** - what would be the best disposal rate which could be reliably achieved?
(c) **Ease of control** - is the process stable and could it be quickly shut down in an emergency?
(d) **Effluent suitability** - how easy would it be to condition the effluent for sea disposal via the low-active drain?
(e) **Hazards of other reactants** - what are the hazards (and thus handling difficulties) associated with any other reactants?
(f) **Sodium handling arrangements** - is the sodium well contained throughout the process?
(g) **Manning requirements** - how many Operators would be required and how intensive would that involvement be?

3. CONCLUSIONS

3.1 Sodium removal (cleaning) processes

Two sodium removal processes have been discussed : 1) water process (dispersed water or steam in N_2 or CO_2; cold water under vacuum) and 2) alcohol process (methanol -ethanol and "heavy" alcohol (ethylcarbitol)). It was concluded that use of alcohol has to be restricted in quantity, due to the thermal runaway that occurred in France and Germany with ethylcarbitol.
For large components, the water cleaning process is recommended considering the efficiency and the type of waste produced. Due to the production of hydrogen by water/sodium reaction, the hydrogen risk has to be taken into account in the choice of the adopted method. The cold water vacuum process is recommended as a final method to complete a cleaning operation, in order to eliminate all sodium traces, even in crevices.

3.2. Decontamination processes

The optimization of decontamination processes must be based upon efficiency and corrosion effects and also upon environmental considerations (release of chemical (phosphates) or radiochemical (tritium) effluents). The decontamination effects on the component are strongly linked to its history and the operational conditions of the reactor. Thus, to apply a decontamination process it is recommended to have a good knowledge of the history of the component. Moreover, the best is to have a representative sample of each material to test the acid bath influence before application to the whole component.

A variety of reactor components in France have been successfully decontaminated, and returned to service using a mixture of sulfuric acid and phosphoric acid as a decontamination bath. Russia used an acetic acid bath after immersion in KM_nO_4.

3.3. Bulk disposal of sodium in the framework of decommissioning

Four main technological operations have been identified: 1) Sodium draining; 2) Sodium bulk processing; 3) Main vessels and circuit cleaning and 4) Effluents treatment. Technical options and

tasks for each have been discussed and recommendations in forms of improvement of existing techniques or development of new ones, have been made. Six sodium disposal technical options were identified as being potentially suitable for bulk disposal; a) reaction with caustic, b)burning in air (spray fire), burning in air (pool fire), reaction with steam, e) reaction with CO_2 and f) reaction with water in oil.

In the treatment of active sodium, the liquid effluent produced is active sodium hydroxide. The treatment of this type of effluent depends on regulations of each country defining what is acceptable or not. For disposal, the treatment of liquid active effluents has to transform sodium hydroxide into a salt that can be discharged after decontamination (release of inactive salt to the sea for the United Kingdom), or stabilized into a form that can be stored in good conditions (sodium carbonate in barrels for the Argonne National Laboratory-West, USA; stabilization into cement or ceramic for the CEA).

4. RECOMMENDATIONS

1. To convene an IAEA/IWGFR TCM on LMFR decommissioning options (working title) in 1999. The USA delegate informally proposed to hold this meeting at ANL, USA in Spring, 1999.

2. To organize in 1998 (February-March upon completion of work by Russian experts) a consultants' meeting on "Technical options for the decommissioning of the BN-350 LMFR in Kazakhstan" in cooperation with other IAEA's divisions involved in technical assistance to Kazakhstan.

Table 1: Sodium Disposal Process Options

	Development Status	**Throughput**	**Ease of Control**	**Effluent Suitability**	**Hazards of other reactants**	**Containment**	**Manning Required**
Reaction with concentrated NaOH solution	Well proven technique. Minimal contamination of reaction vessel. *Not suitable for dirty Na.*	Has been demonstrated up to ~ 40 kg /hr. Development to achieve 100 kg/hr likely to be minimal.	Controlled by regulating Na supply. Low inventory of reactants & low temp, pressure process.	Aqueous solution of NaOH. Liberates hydrogen.	Concentrated hydrochloric acid, highly corrosive, tends to fume.	Sodium well contained in system.	Could be controlled remotely by one Operator. Likely to be minimal maintenance.
Burning in air (Spray fire)	Well proven technique but may need some development (e.g. addition of shielding) for use with active Na. *Not suitable for dirty Na.*	Has been demonstrated up to 100 kg/hr.	Controlled by regulating sodium supply. Very low inventory of reactants in reaction vessel.	Airborne oxide particles which need to be scrubbed out to give NaOH solution. Reaction vessel residues would need periodic washing.	Concentrated hydrochloric acid, highly corrosive, tends to fume.	Sodium well contained in system. Scrubber efficiency very important in reducing aerial discharge.	May need two Operators. Some operator involvement in periodic nozzle replacement & cleaning.

Table 1 (continued): Sodium Disposal Process Options

	Development Status	Throughput	Ease of Control	Effluent Suitability	Hazards of other reactants	Containment	Manning Required
Burning in air (Pool fire)	Very well proven. Would need development (e.g. addition of shielding) for use with active Na. *Also suitable for dirty Na.*	Has been demonstrated up to 100 kg/hr.	Reaction rate depends on surface area. Would need an inert gas purge for rapid shutdown.	Airborne oxide particles which need to be scrubbed out to give NaOH solution. Solid residues in tray.	Concentrated hydrochloric acid, highly corrosive, tends to fume.	Sodium poorly contained in system.	Would need 2 Operators. Work is "hands-on" in nature. Trays of residues would need to be steamed & washed.
Reaction with steam	Not proven for volume destruction. Would need development (e.g. addition of shielding) for use with active Na. *Not suitable for dirty Na.*	Has been demonstrated up to 100 kg/hr but likely to suffer serious nozzle erosion.	Controlled by regulating sodium supply. Very low inventory of reactants in reaction vessel.	Airborne oxide particles which need to be scrubbed out to give NaOH solution. Reaction vessel residues would need periodic washing. Liberates H_2.	Concentrated hydrochloric acid, highly corrosive, tends to fume.	Sodium well contained in system. Scrubber efficiency very important in reducing aerial discharge.	Would need two Operators. Likely to need regular nozzle replacement & cleaning.

Table 1 (continued): Sodium Disposal Process Options

	Development Status	**Throughput**	**Ease of Control**	**Effluent Suitability**	**Hazards of other reactants**	**Containment**	**Manning Required**
Reaction with CO_2	Well established process but likely to be complex plant. Requires very high operating temperatures. *May be suitable for dirty Na.*	Potentially 100 kg/hr.	Controlled by regulating supply of sodium.	Solid reaction product containing C & Na_2CO_3 . Needs dissolution & separation.	Large scale storage of carbon dioxide. Concentrated hydrochloric acid, highly corrosive, tends to fume.	Sodium well contained in system.	Would need 2 Operators. Likely to be significant operator involvement in effluent dissolution, nozzle replacement & cleaning .
Reaction with water in oil.	Not proven for volume destruction of sodium. May be useful for active Na. *Not suitable for dirty Na.*	Uncertain.	Likely to be slow emergency shutdown due to low reaction rate.	Likely to need complex separation to remove oil which would tend to poison caesium removal ion exchange resin. Liberates H_2.	Concentrated HCl acid, highly corrosive, tends to fume.	Sodium well contained in system.	Would probably need two Operators. Likely to be significant operator involvement in effluent treatment

Summary Report of the Seventh Consultancy Meeting on "IAEA/EC Comparative Calculations for Severe Accident (ULOF, Unprotected Loss-of-Flow) in the BN-800 Reactor"

December 11-12, 1997, Vienna, Austria

1. PURPOSE AND ORGANIZATION OF THE CONSULTANCY

The purpose of the consultancy was to discuss the progress of the IAEA/EC sponsored comparative calculations and to clarify and agree on aspects of the exercise which were left open during the last consultancy meeting held in Brussels from June 30 to July 3, 1997.
The specific objectives of this consultancy were:

- to discuss the results of comparative calculations, scope and time table for finalization of ULOF analyses,
- to discuss the results of UTOP-analyses, scope and time table for its finalization,
- to discuss scope and time table for preparation of the final report of the Comparative Calculations,
- to define work programme up to the next consultancy meeting, its date and place.

Five countries and one international organizations took part in the meeting: i.e: France, Germany, India, Japan, Russia and IAEA being representing by the following participants:

J.M. FRIZONNET	IPSN, Cadarache, France
D. STRUWE	FZK, Karlsruhe, Germany
B. PORSCHA	FZK, Karlsruhe, Germany
R. HARISCH	IGCAR, Kalpakkam, India
H. NIWA	PNC, Japan
Yu. CHVETSOV	IPPE, Obninsk, Russia
I. KUZNETSOV	IPPE, Obninsk, Russia
A. DANILYTCHEV	IPPE, Obninsk, Russia
A. RINEJSKI	IAEA, Vienna, Austria

Mr. Van Goethem of EC had sent apologies for not being able to participate in the meeting. Mr. M. Jankowski of Dept. of Nuclear Safety was invited, but did not participate.

Mr. Kupitz, SH-NPTDS welcomed the participants on behalf of the IAEA and EC and outlined shortly the importance of the ongoing comparative exercise against the IAEA position to support the LMFR development work. He welcomed the connection to the grid of India Experimental Fast Reactor in July 1997. In view of a significant role of investigation of safety characteristics of advanced reactors he pointed out that there is considerable interest on the outcome of the said comparative calculations completion is to be scheduled on a short term basis. He wished all participants success in the endeavor. J. Kupitz turned the meeting over to the Scientific Secretary of the IWGFR A. Rinejski who opened the consultancy, presented the Agenda and detailed the purpose of the meeting, A.Rinejski surrendered the Chair to Dr. D. Struwe of Germany who chaired the remainder of the meeting.

The meeting participants approved the minutes of the 6th consultancy (Brussels, June 1997) with small corrections. The Russian representative Mr. Kuznetsov presented information on the status of BN-800 NPP which is on the way to be built.

2. DISCUSSION OF CALCULATIONS RESULTS, CONCLUSIONS

2.1. UPDATING OF ULOF ANALYSES

The detailed analysis of neturonic parameters has been presented by the Russian participants for core configurations, simulating different stages of accident development. The following approaches and codes were used: diffusion theory, transport theory and Monte Carlo.

1. The results of calculations show that the diffusion approach underestimates the efficiency of sodium in the core and especially in the sodium layer both for the initial state (intact core) and for the voided core with melted and relocated steel. Therefore rising the sodium void reactivity coefficients that were presented in the "input data" can lead to conservative results.

2. Doppler coefficients depend significantly on the temperature range and total amount as well as spacial distribution of sodium and steel in the core. In order to take this dependence into account it is recommended to enter into a formula for calculation of the Doppler constant with additional multipliers. The values of that multiplier can be estimate by the Russian side on the base of their neutronic calculation.

3. Significant interdependence of sodium and steel efficiency due to their redistribution is found. For more correct simulation of the proper reactivity feedbacks it is desirable to take this phenomena into account. The meeting participants after heated discussion proposed to recalculate some data.

The Japanese representative, mr. H. Niwa formulated some proposals on updating neutron physics data as follows:

In order to clarify the change of sodium void worth due to streaming cut-off effect, it was recommended to calculate the sodium worth at the sodium layer in the core state 5 (plugged geometry) and to add these results in Table 9 in the IPPE's report. In the original sense, the streaming cut-off effect means a reactivity change caused by the geometrical change from a heterogeneous configuration to a homogenous one. This is actually caused by fuel break up at the fissile top (without axial material motion). Therefore, it was recommended to calculate the following cases with MNCP code: (1) removal of sodium (core state 3) the fuel being intact, and (2) removal of sodium, fuel being broken up at the fissile top and the end plug. These results should be added to Table 1 of the IPPE's report. It was commented that, based on the Japanese data for higher temperature region, the Doppler coefficient could be reduced to about half, but it does not come almost to "zero". Therefore Mr. Niwa did not agree with the conclusion (1) of Table 5 given in the IPPE's report.

During discussion of the conclusion of IPPE that the change in the Doppler-coefficient is responsible for the modified neutron physics characterization of the destructed core configuration concern arose that this result might be due to an uncertainty of the used cross-section library for high temperatures. Mr. Niwa from Japan offered to provide IPPE with information from analyses performed during respective evaluations for the MONJU-reactor.

2.1.1. IPSN, France

The base case calculation (balance of force and clad motion options) was re-run. Basically, the results presented at the last meeting are not significantly modified. The increase of power to flow ratio leads to sodium boiling initiation in channel 5/1, 19.2s after LOF initiation. Then, sodium boiling spreads over 16 channels. During the boiling period, sodium reactivity oscillations are calculated due to the voiding of the upper sodium layer (negative effect) followed by the downwards boiling propagation into the core (positive effect) successively in the different groups of low burnup channels (highest power channels). Clad motion is initiated in Channel 5/1, 6s after boiling onset and 0.7s later in channel 1/1. This induces first a limited

steel accumulation at the upper core fissile end (negative reactivity effect) followed by steel drainage to the lower end of the core fissile height (positive reactivity effect). At 26.9s, fuel melting is not reached, but hexcan melting is calculated in channel 5/1 and thus, the limitation of the use of the SAS4A code is reached. Later, core material relocation would become 2-dimensional and cannot be stimulated by the quasi-one-dimensional approach followed in the SAS4A code. Continuation of sodium voiding and clad relocation as well as fuel melting and slumping is expected to occur. Thus, the accident enters into the transition phase. As these results are very similar with those presented at the last meeting, the parametric case (without clad motion) was not re-run because the conclusion presented at the last meeting is still valid (similar accident sequence on a longer time scale).

2.1.2. FZK, Germany

- Analytical investigations show, that the negative reactivity feedback contribution due to the voiding of the upper sodium layer is a quite efficient measure to mitigate the impact of the positive reactivity feedback contribution when the fissile core height becomes voided.
- As consequence of the ULOF the normalized power becomes considerably reduced and is stabilized at a low value after boiling onset. However, this power reduction is not sufficient to prevent clad dry-out and clad melting to occur in several subassembly groups. This holds for both cases i.e. without and to a lesser amount with consideration of radial core expansion.
- In subassembly groups where clad relocation is calculated blockages of the coolant channels at the upper fissile core end and more massive ones at the lower fissile core end are calculated to become established in a few seconds. Clad relocation results in a considerable positive reactivity feedback.
- Extended steel blockage formation at the upper fissile core end modifies the materials configuration significantly when compared to the initial core materials configuration taken as basis for the determination of reactivity worth curves. Therefore it becomes necessary to evaluate whether the negative reactivity feedback contribution associated to the voiding of the upper sodium plenum is still maintained (streaming correction of the reactivity worth distribution allocated to sodium voiding of the upper sodium plenum after clad relocation onset necessary).
- A few seconds after clad relocation onset melting of the inner hexcan surface is to be expected due to the radiative heat transfer from the heated-up outer fuel pellet surfaces to the inner hexcan surface. Subsequently a mechanical collapse of the concerned subassemblies is to be expected. The positive reactivity feedback effect as consequence of such a solid state fuel fragments compaction can hardly be calculated reliably with a deterministic model. Therefore consequences of such events have been analyzed parametrically (external positive reactivity ramp rate of 2$/s and total amount up to 8 $ starting at 8.5s after boiling onset.
- The ULOF-transient is not at all finished at the end of the SAS4A calculations. Establishment of long-term coolability of the partially destroyed core is not to be expected neither in the case (A) nor in the case (B) calculation. A quite complicated transition phase will follow where material relocation phenomena will deviate considerably from the quasi-one-dimensional behaviour in the initiation phase due to the calculated hexcan melting.
- It is hardly possible to predict the behaviour of the core design during the transition phase. It is most probably characterized by a slowly progressing core melt-down interrupted by multiple recriticality events. Whether these might lead to a mechanical loading of the heated-up structures of the primary system cannot be predicted without detailed analyses.
- Termination of the ULOF-transient is only possible if fissile material is efficiently removed from the original core region. At the end of the initiation phase relocation paths are difficult to become accessible and material relocation is rather limited as long as the fuel stays practically solid.
- Results of the comparative exercise does not allow yet to conclude that ULOF-transients in a BN-800 type reactor core design leads only to benign consequences in terms of the thermal and mechanical loading of structures of the primary system. However, the results have shown, that the upper sodium layer is a quite efficient design measure to prevent the net reactivity to approach prompt criticality in the initiation phase of a ULOF-transient.

- It was concluded, that thermal and mechanical loading of structures of the primary system can only be determined by analyses of the early and late transition phase. In this sense it is not yet demonstrated that this innovative core design leads to less severe consequences core designs as the ones of the CAPRA project.

2.1.3. PNC, Japan

Studies included: *(1). a parametric modeling of the neutron streaming cut-off effect in SAS4A, and (2). a parametric calculation of an ULOF accident considering streaming cut-off in a BN-800 type core.* The objective of this study is to look briefly and quickly at the effect of the streaming cut-off in the scenario of ULOF in such a core with a sodium layer. In the first part, the modeling of the treatment of streaming cut-off, which was introduced to SAS4A, is explained. This model is rather simple and treats the effect in a parametric manner. The streaming cut-off effect is treated in each channel independently, and reduction of the negative void worth is considered with one input variable. Sample results of the void reactivity behaviour in Ch. 13 of BN-800 are presented for the cases with the reduction coefficient of 0.1, 0.5 and 0.9 compared with the case neglecting streaming cut-off effect. In the second part, a parametric calculation of an ULOF accident considering streaming cut-off is presented. Based on the reference case which was reported in the previous meeting at Brussels in June, 1997, three parametric cases with three reduction coefficients (0.1, 0.5 and 0.9) are shown. If the reduction coefficient is 0.1, then the scenario does not largely changed from the original case. If the reduction coefficient is 0.5 or 0.9, the void reactivity of the fissile region and sodium layer becomes positive (if the streaming cut-off effect is fully disappeared). Therefore the scenario could be changed and even a power burst could be expected. In the concluding remarks, it is stressed, that more refinement of the modeling is needed if this effect is important to evaluate a BN-800 type core, and also that the scenario could be changed if the effect of streaming cut-off is not negligible or if the effect of negative void worth at the sodium layer disappears due to this effect.

2.1.4. IGCAR, India

Only the preboiling calculations were done. The code PINCHTRAN was modified to include finer spacial descredization in the sodium plenum and B_4C shield for heat transfer calculations. The calculations indicate a slightly earlier boiling onset of 16.6s against 17.4s reported in the earlier meetings for the base case. For the parametric case the boiling onset is advanced to 28.1 s. The earlier onset of boiling times is attributed to the reduced plenum feedback of the fineness introduced into the calculations. Post boiling calculations were not performed.

2.2. UTOP ANALYSIS

The clad material properties of the BN-800 reactor distributed by IPPE, have been evaluated by FZK (B. Porscha, D. Struwe). It was noted that these data showed some uncertainties, especially no strain rate dependencies, higher ultimate tensile strengths and yield stresses for irradiated than for non-irradiated material, and a large scattering for failure strains. Measurements only are available for non irradiated material. The experimental data are partially taken into account for SAS4A calculations. They were compared with those given by FZK modified modeling for material properties (Fabianelli, CEA). As a conclusion from the comparison of the experimental data of BN-800 and well evaluated data for VIGGEN-4 experiments, calculations for the BN-800 safety analysis with the material properties of a 15/5 Ti - stabilized steel were performed with a high and a low strain rate to bound the scatter band of the experimental data[1] of ultimate tensile strength. Yield strength data are consistently below measured values, but the difference between the experimental yield stress and ultimate tensile strength was kept constant. Failure strain data represent well the average of the scatter band of measurements for the BN-800, taking into account that failure strains for irradiated material are lower than those for non irradiated material.

[1] The calculations with different strain rates show similar results.

2.2.1. IPSN, France

The slow UTOP (0.05 $/s) base case calculation (balance of force option) was re-run. Basically, the results presented at the last meeting are not significantly modified. The imposed reactivity is mainly counterbalanced by the Doppler effect. The negative effect of the fuel axial expansion (linked to the use of the balance of force option) as well as the negative effect due to sodium heating also play a role. Net reactivity is then rapidly stabilized to a small positive value. Thus a quasi-linear power ramp is initiated and lead to sodium boiling in channel 5/1, 73.2s after reactivity ramp insertion. Six channels are calculated to boil but as in the LOF calculation, sodium voiding extends downwards towards the core and leads to sodium reactivity increase which pushes the relative power to higher values and to clad failure in channel 5/1, 4.2s after boiling initiation. Then, after a limited fuel accumulation around the failure location which increases slightly the relative power, a significant fuel dispersal due to fission gas release leads the relative power to 0.8 at the end of the calculation. At 78.s hexcan melting is calculated to occur in channel 5/1. As these results are very similar with those presented at the last meeting, the fast UTOP (0.5 $/s) base case was not re-run because the conclusion presented at the last meeting is still valid (similar accident sequence on a shorter time scale). An attempt was made to perform a parametric calculation taking into account the internal fuel motion effect (squirting effect) but without success. Fuel squirting is supposed to start at fuel melting onset (31.5s in channel 5/1), but the code stopped when only 5% molten fuel fraction is reached. The EJECT coupling with SAS4A still needs refinements. It is concluded that, as for the ULOF initiator, for the UTOP initiators, the integrity of the fuel pin could not be maintained and the accident is expected to enter into the transition phase.

2.2.2. FZK, Germany

- The slow reactivity ramp rate cases 10.05$/s lead to fuel pin failures in one subassembly group at 92% core height when a failure strain of 2.4% becomes exceeded. The subsequently calculated fuel relocation initiates a rapid shutdown. Partial blockage formation in the concerned subassembly group results in a configuration which can hardly be cooled in-place on a long time scale. The accident most probably will enter into a slow core melt down with a progressive core destruction by a thermally induced subassembly to subassembly propagation.
- The rapid reactivity ramp rate case (0.5 $/s) leads firstly to boiling onset in several subassembly groups with fuel pin failures calculated to occur about half a second later at 92% core height when a failure strain of 1.2% becomes exceeded. The subsequently calculated fuel relocation initiates a rapid shut-down. Partial blockage formation, the high fuel enthalpies and the heat-up of the coolant hardly allows long-term in-place coolability of the partially destructed core configuration. A more or less rapid core melt down is to be expected. It was concluded that the recriticality issue needs to be analysed carefully.

2.2.3. IGCAR, India

The UTOP cases have been carried out for 0.05$/s without the radial core expansion feedback and the 0.05$/s with the radial core expansion feedback. The 0.05 $/s case indicates that the fuel pin failure at 39.02s is based on the 50% melt fraction criterion in the 5/3 channel. Pin failure based on the damage parameter criterion of Bars occurs slightly earlier at 38.7s in 5/1 channel where the power to flow ratio is highest. An optimistic fuel coolant interaction drives the fuel in the region outside the core at 7 m/s giving a rapid negative reactivity insertion rate of -5$/s shutting down the reactor. For the 0.5$/s case, the pin failure based on the 50% melt fraction criterion is predicted at 5.05 in channel 5/3 as in the 0.05$/s case. The damage parameter criterion indicates the pin failure at a relatively earlier time of 4.3s in channel 5/1. As in the 0.05$/s case, the reactor shuts down due to fuel sweep out in the region.

2.3. UTOP-ULOF TRANSIENT CALCULATIONS

The calculations have been done for the following accidental scenario:(1) t = o, Reactivity ramp rate

initiation (0.05 $/s for 10 s) due to control rod withdrawal at full power; (2) t=3.8 s; P/Pn - 1.15, Additional reactivity ramp rate initiation (0.05 $/s for 100 s) due to 6 compensating rods withdrawal, and (3) t=4.8 s (1s later). Coolant mass flow reduction is initiated to a value of 25% of the nominal flow.

IPSN, France

Single phase calculation results: Before 3.8s, due to the first power ramp initiation, the net reactivity and thus the power increase. At 3.8s the relative power reaches 1.15 value. Between 3.8s and 4.8s, due to the second power ramp initiation, the net reactivity and hence the power increases more rapidly. At 4.8s (1s after the second power ramp initiation), ULOF is initiated. Between 4.8 s and 10 s, additional negative coolant reactivity decreases slightly the net reactivity and thus limits the power increase. After 10 s, due to the end of the first power ramp, the net reactivity and hence the power decrease.
Sodium boiling calculation results: Sodium boiling occurs in channel 5/1 at 13.16s. At this time, the relative power reaches 1.23 value.
Post boiling calculation results: Sodium boiling extends both radially in the upper part of the core and downwards in the high power channels (1/1, 2/1, 3/1/ 4/1, 5/1) leading as for the ULOF case to sodium reactivity oscillations and therefore to net reactivity and power oscillations. But downwards voiding extension leads finally to reactivity and power increase. Voiding extends to 23 channels within about 5s. Clad motion is calculated to occur first in channel 5/1 at 16.62s (3.46s after first B.O.). Then, clad motion occurs in channels 1/1 and 2/1. Steel blockage and hexcan melting are then calculated in channel 5/1 at 18.12s. At this time, the relative power is calculated to be 3, but fuel melting is not reached.

Parametric calculation (without clad motion): Compared to the base case calculation, the sodium voiding is more extended radially (28 channels) as well as downwards. Thus, a large reactivity increase is calculated and a power pulse which reached $5P_N$, is obtained. Fuel melting is calculated in channel 5/1 at 19.57s and first fuel break-up occurs at 19.58s (6.42s after first B.O.) and at $3.1P_N$. Fuel break-up is calculated to occur in 3 channels. During the first 20ms following the first failure, reactivity increases due to molten clad motion effect and thus, the power increases to $4.2P_N$. Later, extended voiding of the core leads to sodium reactivity decrease and thus to the decrease of the power. Hexcan melting is calculated to occur in channel 5/1 at 19.92 s. At this time, the relative power is calculated to be 2.2. These 2 ULOF-UTOP calculations are not basically different. At the end of the calculation (limitation of the use of the SAS4A code after hexcan melting or clad blockage), significant residual power is calculated to be available (2.2-3 P_N). Core melting and degradation are anticipated and the integrity of the hexcan and of the fuel pin cannot be maintained. The accident is expected to enter into the transition phase.

2.3.2. FZK, Germany

The calculations show that the UTOP/LOF case initiates coolant boiling first at the level of the upper sodium layer. Negative reactivity feedback due to the initial voiding of the upper sodium layer mitigates efficiently the positive reactivity feedback due to the further voiding of fissile core height. The event sequence is quite similar to the one calculated for the pure LOF-case (A) without radial core expansion. However, the power level stays higher due to the superimposed external reactivity ramp rate. Fuel pin failure conditions are not calculated because the calculation stops when applicability limits of SAS4A are exceeded. At this time the net reactivity stays 7 c below prompt critical, but increases with about 5 $/s shortly after fuel pin break-up is expected to occur in the hottest subassembly group reducing the net reactivity significantly due to the fuel relocation. Therefore it is not expected that prompt criticality will be exceeded in the initiation phase of this transient. The following transition phase needs to be analyzed carefully.

2.3.3. IGCAR, India

The incident specified by the IPPE is followed by its analysis. Initially 0.05$/s reactivity is introduced at 115% power; a second ramp of 0.05$/s is introduced. Normal flow reduction is introduced one second later, radial feedback being ignored. The power of 1.15 times of the rated value is reached at 1.97 s. The loss of flow and the combined addition of reactivity lead to boiling onset in ch. 13 at 9.8s. Boiling takes place in the bottom of the sodium plenum propagating into the plenum and to the top of the core. Void propagation is not observed in the middle of the core where large, positive reactivity is available. However, void propagates radially into the top two zones of the core. The results are at variance with those reported by SAS4A and GRIF-SM codes.

2.3.4. IPPE, Russia

The new version of the combined code GRIF-SM-CANDLE have been presented. The basic new feature of the code is an ability to simulate relocation and freezing of molten steel during a severe accident. The code was applied to the numerical analysis of the ULOF-UTOP accident. It was shown that this scenario of the accident leads to sodium boiling, pin clad melting, steel relocation and fuel melting at the end of considered time interval. The net reactivity becomes positive but relatively small (much below beta eff) at final stage of the accident but changes rather gradually and slowly. It means that most probably the entering into transition phase will occur without prompt critically.

3. SCOPE AND TIME OF TABLE OF ANALYSES COMPLETION

3.1. ULOF ANALYSIS

After an in-depth discussion of the results of new neutron physics data for the partially destructed core configuration it was unanimously agreed, that transient calculations being available up to now should form the basis for writing the summary report on boiling transient analysis and post-failure transient analysis up to the end of the initiation phase. In parallel IPPE accepted to document the results of the neutron physics analysis, giving information on the background and meaning of the different calculations and to complement the evaluation by comparing Keff results of MCNP calculations with those for sodium removal and sodium & steel removal. In addition IPPE will evaluate the reasons for the difference in the steel reactivity with between the "final core state" and the "initial core state" (p.15 of the distributed viewgraph). Independently from these activities IPPE offered to provide information on the dependency of the Doppler coefficient from the mixture composition and here especially from the steel volume fraction before the end of March 1998. Then will be decided by each participant of the exercise wether they will perform complementary calculations simulating the reduction in the Doppler reactivity feedback effect in a simplified manner.

Mr. Niwa from Japan accepted to prepare a summary report about calculations for the boiling and post-failure transient up to the next consultancy meeting. It was agreed upon that only those results becoming available up to end of March 1998 will be considered in this summary report. Each participant of the meeting was asked to support Mr. Niwa's effort to his best means.

3.2. UTOP ANALYSIS

Results presented for the UTOP- and UTOP/LOF transients were discussed. It appeared that the different contributions offer the possibility to develop a clear picture on generic features of the core behaviour, though details are simulated slightly differently in the different calculations. Therefore it was unanimously agreed to finish the exercise and summarize the major aspects of the results. Mr. Frizonnet from France accepted to prepare a summary report about calculations presented by the different participants to the exercise. It was again agreed upon, that only those results becoming available up to the end of March 1998 will be considered in this summary report. Each participant of the meeting was asked to support Mr. Frizonnet's effort to his best means.

4. CONTENT AND SCOPE OF THE FINAL REPORT

Participants of the meeting discussed intensively the scope and contents of the final report. It was of common interest to focus the report structure on physics and phenomena and not on the different contributions from the different nations contributing to the exercise. In addition it was agreed to prepare one report only containing a specific look of details with regard to physics and phenomena on the one side and an executive summary on the other side. On the basis of a proposal made by the chairman the structure of the report should be as follows:

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List of participants

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Thus the report will consist of five major chapters from which two and a half are already available. Chapters 3.2 to 3.4 will be prepared by Mr. Niwa from Japan, Chapter 4.1. by Mr. Porscha from Germany, Chapter 4.2 to 4.3 by Mr. Frizonnet from France, Chapter 5 by Mr. Danilytchev from Russia and the executive summary and Chapters 6 by Mr. Struwe from Germany. The respective draft proposals have to be made available hopefully prior but at the latest at the next consultancy meeting.

5. PLACE AND DATE OF NEXT MEETING

The next consultants' meeting will be the last in this endeavor. Participants thought that it is appropriate to consider the possibility to have this meeting in Obninsk, Russia. This way it would be possible to finalize the consultancy activity in due time and in an efficient manner and it would provide the possibility to present the main findings of the activity to a broader community of interested engineers, physicists and managers from Russia. The idea was principally accepted and supported by the Secretary of the IWGFR. So tentatively it was decided to have the 8th and last consultancy in Obninsk, Russia on 29 June - 3 July, 1998[2]. The work programme will cover review and approval of the early prepared summary report chapters, discussions and approval of "Conclusions and Recommendations" and the "Executive Summary". These discussions might result in a proposal of topics of interest for further initiatives in the fast reactor domain to be followed by the IAEA.

[2] Agreed with IPPE's Division of International Relations

INTERNATIONAL ATOMIC ENERGY AGENCY

Consultants' Meeting on "LMFR Operating Experience and Feedback for Future Projects"

17 - 19 December 1997

Summary Report

Participants	Mr. V. I Bolgarin	Kazakhstan
	Mr. O. M Saraev	Russian Federation
	Mr. A. M Judd	United Kingdom
	Mr. A. A Rinejski	IAEA
	Mr. V. Iarovitsine	Russian Federation

Introduction

The participants were welcomed by Dr J Kupitz (IAEA, Head of the Nuclear Power Development Section). He reminded them that, LWRs had achieved good reliability, and costs when their respective large scale introduction had taken place and we cannot say that this will not happen in the case of LMFR. In order to assist nations which needed to introduce fast reactors the Agency wished to exchange information on reactor operating experience and to provide expert advice and guidance. He defined the purpose of the present meeting to be

- To review operational experience on fast reactors and to identify and characterise the related main design and technology problems, and
- To make recommendations of the scope of work of the next year's TCM on "Unusual Occurrences during LMFR Operation: Review of Experience and Consequences for Reactor Systems". The intention of the planned TCM is to provide a forum for face-to-face discussion of the details of operating experience which had not always been reported completely in the literature.

Mr Rinejski, the Scientific Secretary of the International Working Group on Fast Reactor (IWGFR) emphasised the need to concentrate on specific issues and to produce results which, on publication, would indicate that some LMFRs have demonstrated good operational characteristics.

Mr Judd was nominated to take the chair for the meeting.

Presentation - BN 600

Mr Saraev emphasised the good performance of BN 600 over nearly 18 years, during which period it had operated for 77 % of the time, with 22 % shut down for planned maintenance and only 2 % lost to unplanned incidents. Of the lost time 33 % was due to manufacturing failures, 34% to design errors, 17% to personal errors and 11 % to maintenance malfunctions.

He reviewed the performance of various plant components as follows.

Fuel: Steady increase of the burnup from 6.1 % to 11.8 % currently had been made possible by reducing the linear rating from 54 to 48 kW/m and increasing the core height. There are plans for further increase to 16 - 17 %. The number of enrichment zones had been increased from 2 to 3. MOX fuel is currently being tested.

Fuel Handling Equipment: This has caused no delays to refuelling operations. It had been necessary to modify the position control equipment.

Spent Fuel Route: There have been no delays. Teflon valve seals have been replaced by rubber.

Rotating Plugs: There has been some tendency for the rotation torque to increase because of deposition of sodium oxide. The cause has been identified and modifications to correct the problem are in hand.

Control Rods: There have been no problems. The design life is 120000 hours.

IHX: These are stable in operation. Inspection in situ is not possible. It is planned to replace one IHX in 1999.

Primary Sodium Pumps: A vibration problem which caused failures in the couplings has been corrected. The impellers have a short lifetime and require frequent replacement.

Primary Cold Trap: There have been two sodium leaks due to temperature cycling (see below), and as a result the pipework has been modified.

Steam Generators: The design was modified at a late stage in the light of BN 350 experience. There have been a total of 12 leaks, but none has caused overpressurisation of the secondary loop, and only two have caused a secondary loop to be shut down.

Turbo-alternator: In common with other equipment from the same manufacturer there has been a problem of hydrogen leakage.

Protective System: There have been 18 plant trips initiated by the automatic protective system in 18 years. The last was in 1993.

Sodium Leaks: There have been several sodium leaks, involving radioactive sodium, but due to effective operation of the leak suppression system there was no excess radiation exposure of operators or the public. The effect of the leaks on plant operation has been very small.

Operating Statistics: Dr Saraev presented a series of graphs which compared several aspects of BN 600 operating experience (load factor, trips, personnel dose, waste arisings, radioactive release)with world data. These show that BN 600 has in many cases become better than the world median.

Discussion: In discussion the importance of well-validated design rules and manufacturing procedures to the success of BN 600 was emphasised. Simple and continuous control functions are also important.

It was suggested that the following items should be featured in next year's TCM;

- Design rules
- Sodium fires (leak detection, passive damage prevention, etc)
- Rotating shield operation
- Steam generators, and
- Operating statistics.

Presentation - PFR

Mr Judd summarised the main unusual events over the 20 years of operation of PFR. These are:

Contamination of Control Rod Magnet Faces 1974 - 1994 (due to deposition of sodium aerosols):
Shut-off Rod 4 Friction 1987 (due to swelling of the wrapper of an adjacent subassembly):
DHR Heat Exchanger Leaks 1975 - 1982 (due to excess thermal stresses cased by unstable parallel flow in horizontal heat exchanger tubes with natural convection):
Superheater and Reheater Weld Leaks 1974 - 1976 (due to manufacturing defects, leading to extensive cracking of the tube plates):
Evaporator Weld Leaks 1974 - 1982 (due to incorrect design of the tube-to-tubeplate welds, leading to major loss of operation and repair of all the tube-to-tubeplate joints):
Superheater 2 Under-sodium Leak 1987 (due to tube vibration, leading to failure of a total of 40 tubes and replacement of the tube bundle):
Sodium Fire in Steam Generator Cell 1 1984 (due to malfunction of trace heating on sodium pipework):
Steam Generator Cracks 1987 - 1993 (due to delayed reheat cracking in weld repaired during manufacture):
Primary Sodium Pump Oil Leak 1991 (due to overpressurisation of the oil reservoir, leading to breakage of sodium filters, reduction of coolant flow in parts of the core and loss of 15 months' operation):
Seizure of the Primary Cold Trap Pump 1984 (due to blockage of gas pipes in the pump tank sodium level control system):
Blockage of the Secondary Cold Trap 1980 (due to incorrect precipitation of impurities).

He also described the following events which gave concern but in the event did not lead to failures.

Excessive Temperature Gradients in the Reactor Vessel 1975 (due to inadequate cooling):
Hypothetical Damage to the Above-Core Structure Supports (due to thermal shock):

Hypothetical Damage to the Above-Core Structure (due to thermal striping):
Hypothetical Damage to IHX Tubeplates (due to asymmetrical operation):
Hypothetical Damage to the Core Support Structure (due to possible degradation of the material).

Presentation - BN 350

Mr Bolgarin presented an operation histogram showing the reliable performance of BN 350 up to 1995. Currently it is operating at 420 MWth, 50 MWe, and producing 45000 tonnes of distilled water per day. He commented on various plant components as follows.

IHXs: Integrity if a circular weld is a continuing concern. The rate of change of power is limited to protect it. An account of damage accumulation is kept.

Protective System: Initially there were many plant trip signals, but the system has been simplified and the number of spurious trips has decreased.

Fuel: The burnup has steadily increased from the design value of 5 %, and should reach 14 % soon. A further increase to 16 % is desirable for commercial reasons.

Rotating Plugs: Operating rules had been introduced to prevent sticking.

Plant Components: In general the major plant components had been tested prior to assembly, and this had resulted in reliable operation in the main. However it had been decided not to test the steam generators, for reasons of expense, and this had shown in poorer performance of these items.

Steam Generators: There were manufacturing defects in the original re-entrant evaporator tubes, and all had been replaced. There had been two further large leaks, of 360 and 800 litres of water respectively. As result 2 of the 6 evaporators had been replaced by micro-modular units of a different design. The superheaters had caused no problems, but when some of the peripheral tubes were found to be vibrating they were plugged as a precautionary measure.

Sodium Leaks: There have been 16 leaks, of which only one very small leak at a ball-and-cone joint in the primary cold trap involved radioactive sodium. The joint was welded. The worst leak, of 100 - 150 kg of sodium, occurred at the time of one of the steam generator leaks. A residue was left in a horizontal pipe after incomplete draining. Reaction with moisture caused stress-corrosion damage at a transition weld. There was a spray fire. Some of the sodium, which remained in the insulated containment vessel, was frozen by diverting the air supply to the vessel and removing the insulation, and removed by hand.

Life Extension: The plant had a 20 - year design life, but continued operation is needed for commercial reasons. A review of suitability for continued operation has been made. All components have been inspected and found in good condition, and operation to 2003 has been approved. A decision on the need to clean the steam generator tubes has yet to be made.

In discussion the point was made that the steam generator leaks and associated sodium leaks had little effect on plant performance because it was possible to operate on 5 out of the 6 loops. The value of multiple steam generators had been demonstrated. In addition the importance of well-trained staff, able to make their own decisions within the safety rules, had been shown.

It was suggested that the following items should appear in next year's TCM;

- Design rules,
- Comprehensive component testing, ideally to internationally-accepted standards, of plant components,
- Modular steam generators, and
- Staff training and flexible operation.

The Case for Investment in Fast Reactors

The meeting participants concluded that the advantages of fast reactors were not properly understood, and that they should be stressed to make the case for greater investment. After discussion the advantages were listed as follows.

- The coolant undergoes no phase transition.
- The whole-core reactivity coefficients are negative (at least if the plutonium inventory is low).
- Internal breeding permits operation with a small reactivity margin.
- The power profile is stable, so the control system can be simple.
- The low coolant pressure allows the use of tough steels which protect against uncontrolled leakage.
- The coolant is chemically compatible with the structural and fuel materials.
- The heat capacity of the coolant is high (for a pool reactor).
- In design, construction and operation the objective of complete elimination of leaks is possible.
- The high quality of the design and the nature of the plant lead to good safety culture.
- Plutonium of any origin can be used.
- Decommissioning is relatively simple because a narrow range of technologies are involved.
- The environmental impact, in terms of routine and accidental releases, personnel dose, waste production, consumption of minor actinides, and mining, is low.
- Series production will reduce costs.

There are disadvantages as follows.

- Costs are high, for demonstration and/or prototype LMFRs.
- Sodium is flammable.

However solutions for all these perceived disadvantages are known and can be demonstrated.

Seminar:

The conclusions of the meeting were presented at a Seminar to Dr. Murogov and other IAEA personnel, together with a brief summary of the status of the CAPRA project in Western Europe . There was a valuable discussion of the needs and opportunities for international cooperation in this and related areas of advanced fast reactor technology in applications related to the management of the entire nuclear fuel cycle.

Conclusions

The meeting recommends strongly to the Agency that it is important to contine support of the development of fast reactors. It is pārticularly necessary that the advantages of fast reactors (listed above) should be made clear, and that an accurate account of the experience of operating fast reactors, much of which has not been published, should be made available to the nations which forsee an urgent need to deploy the technology in the near future. It is essential to counteract the impression that LMFRs have not been successful by disseminating accurate data on reliability and on the work which has been done to master the perceived disadvantages and take advantage of the great potentialities of the technology.

The participants in the meeting agreed to submit proposals for subjects to be addressed at the TCM. These proposals should be supported by statements of the reasons why they are important, and the lessons to be learnt from them.

DRAFT

Consultancy on "Technical Option for the Decommissioning of the BN-350 LMFR in Kazakhstan"
23-27 February 1998
Obninsk, Russia
(622-I3-CT-01531)

SUMMARY REPORT

Opening Session

Five countries participated in the meeting: France, Kazakhstan, Russia, UK and USA ~~(list of participants - see Attachment I)~~.

Mr. A. Tsiboulya (IPPE, Department of Nuclear Power Plants) welcomed the participants. Mr. E.Adam (UKAEA) undertook the chairmanship of the meeting at the request of Mr. A.Rinejski (IAEA, Division of Nuclear Power), who opened the meeting.

The agenda ~~(see Attachment 2)~~ proposed by the Scientific Secretary was agreed by the participants accepting that the timing of the presentation could be adjusted if necessary. Mr. Rinejski defined the purpose of the meeting to be the following: 1) to present information on and review current technical knowledge and experience in the decommissioning technology of sodium cooled fast reactor plants, and; 2) to discuss and review technical options for the decommissioning of the BN-350 LMFR in Kazakhstan.

The Chairman and each of the participants introduced themselves and gave a short presentation on their background and experience.

In the opening session the first presentation on the "Current Status and Decommissioning Issues of the BN-350 NPP" was made by Mr. P. Nazarenko (MAEK, Kazakhstan), who described the current status of the BN-350 plant and went on to discuss the issues that arise when decommissioning of the plant starts in 2003. Currently it is operating at 420 MWth, 50 MWe, and producing 45,000 tonnes of distilled water per day. Although the plant is in Kazakhstan it was stated that Russian regulatory documentation is used, which requires the preparation of a decommissioning document 5 years before shutdown, i.e. in 1998. Although work on decommissioning started in 1990 not much progress has been made due to lack of financing. It was noted that a proposal to increase the unit price for all power produced in Kazakhstan to cover the BN-350 decommissioning cost had been rejected by the Government although a modest increase in the BN-350 unit price had been agreed. It was also noted that a meeting sponsored by the IAEA in 1996 had been very helpful. Mr. P. Nazarenko went on to describe work supported by the USA on fuel options and work being undertaken by NUKEM on liquid waste management. He also highlighted the issues related to the handling of the 500 t of primary coolant and 480 t of secondary coolant.

The second presentation in the opening session on "Decommissioning of the UK Reactors DFR and PFR" was made by Mr. E. Adam (UKAEA). He started by describing how decommissioning was approached in the UK, i.e. in accordance with the policy that redundant nuclear facilities would be decommissioned in a safe, timely and cost effective manner. Mr. Adam did however point out that it was not policy to dismantle in the early stages if there is no safety or economic justification. The way in which the decommissioning programmes had been divided into three active stages separated by periods of core defuelling and maintenance were described. It was also highlighted that the aim during Stage 1 was to have staff with the necessary expertise available to operate, maintain and monitor the plant. Normally these staff will be the former operators. Mr. Adam went on to describe the current

status of the DFR decommissioning programme and tasks presently being undertaken. He emphasized the work being done to refurbish and decommission the NaK disposal plant, used for the secondary coolant, to enable its use for the disposal of the primary coolant and the need to obtain special equipment to remove the remain fuel (997 breeder elements) from the primary circuit following treatment of the primary coolant. The current programme which schedules completion of stage 3 at the end of the next century was also discussed. Mr. Adam continued his presentation by discussing the current status of the PFR decommissioning programme. He described the successful completion of the defuelling of the reactor and removal of the turbine and auxiliary equipment to enable the liquid metal disposal plant to be sited in the turbine hall. He went on to highlight the current main task i.e. the preparation for disposal of the primary and secondary sodium coolant and intimated that commission of the plant and transfer system was scheduled to start in October 1998. Finally the costs of the decommissioning programmes to date and the projected cost to the completion of studies were provided.

Second Session

The presentation at the start of the second session was given by Mr. G .Streiff (CEA-France) who described the decommissioning of the Rapsodie reactor to level 2. Following a detailed description of the work and the showing of a short video which covered the work undertaken Mr. Streiff provided the following statistics: total dose uptake - 312 mSv; total cost - 170 M (FF): 33% - Maintenance & Safety; 25% - Dismantling; 13% - Organization; 11% - Equipment; 10% - Waste; 1% - R&D. The presentation was concluded with the statement that the estimated cost for level 3 decommissioning is 200 M(FF) in a period of 6 years. However it was deemed not necessary to start level 3 work for 30 years.

The next presentation was made by Mr. J. Michelbacher (Argonne National Laboratory-West, USA) addressing the "Deactivation of the EBR-II Complex". He pointed out that deactivation of a liquid metal fast reactor presents unique concern. Residual amounts of sodium remaining in the primary and secondary systems must be either reacted or inerted to preclude future concerns with sodium-air reactions that generate explosive mixtures of hydrogen and leave corrosive compounds. Residual amounts of sodium on components will effectively "solder" components in place, making future operation or removal unfeasible. The sequence of events used to "lay-up" sodium wetted systems is thus crucial in assuring these concerns do not occur. After removal of selected components, unique methods, potentially employing unique robotic applications, will be employed to assure complete reaction of the residual sodium. Specialized components/systems exist within the primary system, include primary cold traps, a cover gas condenser, and miscellaneous systems containing a sodium-potassium (NaK) alloy. The sodium or sodium-potassium alloy in these components must be reacted in place or the components physically removed in order to achieve a radiologically and industrially safe condition. The Sodium components Maintenance Shop, a facility at ANL-W, provides the capability for washing primary components, removing residual quantities of sodium or NaK while providing some decontamination capacity. Finally Mr. Michelbacher provided information on reactor D&D (decontamination and decommissioning). Major primary components will be removed from the primary tank for cleaning and disposition. An integral part of the primary component removal is for providing the necessary access to primary tank internal components for D&D activities, removal of hazardous materials, and removal of stored energy sources. ANL-W's plans for the deactivation of the EBR-II address these issues, providing for an industrially and radiologically safe complex, and requiring minimal surveillance during the interim period between D&D.

The final presentation of the second session was given by Mr. J.-L. Garcia (CEA, France) on "Sodium Disposal at Rapsodie using the DESORA facility". The NOAH process developed by CEA is a continuos process for disposal of large quantities of sodium coolant both primary and secondary from shutdown FBRs. Aqueous caustic soda is produced which can be sent to the appropriate effluent

treatment plants. Mr. Garcia went on to describe how the NOAH process had been applied successfully with the DESORA equipment used to dispose of the primary sodium from the Rapsodie reactor. It was also noted that the process has been selected for the plant to be used to dispose of all the coolant from the PFR in the UK.

The questions/comments period at the end of the first and second session concentrated mainly on the presentations by Mr. Adam and Mr. Garcia.

Although not part of the presentation gave Mr. Adam was asked about the plant at Dounreay to dispose of the KNK II secondary and primary coolant Mr. Adam was also questioned on the break down of the decommissioning costs he had given as global figures.

There were many detailed technical questions on Mr. Garcia's presentation of the NOAH process and the DESORA facility. Particular emphasis in the question related to the handling of wastes and the alternative options.

The USA delegation was questioned on information on the decommissioning experience from EBR-I to which they had to respond that neither they had experience of the project and had not come prepared for such a question.

There were two particular comments which related to learning for future reactor design requirement from the experience from decommissioning to ensure that in future everything is being done to minimise the cost of decommissioning.

In response to a general question of future nuclear plant requirements Mr. Rinejski provided an overview of the current status of nuclear plants under construction in the world and the programme currently being undertaken in India, China and the Republic of Korea to develop/build fast reactor facilities/plants. He also commented on the difficulties in reducing costs of fast reactor plants, particularly the decommissioning cost when there is no on-going programme.

Third Session

Mr. A. Tsykunov (IPPE, Russia) presented the paper entitled "Evaluation of accumulated activity in the cells of BN-350 primary and secondary systems". Information on the allowable number of damaged fuel pins in core and waste classification as of Rules for nuclear safety was discussed. In addition examples of the activity of the primary circuit, secondary circuit and cold traps were presented along with estimates of surface contamination and tritium levels in the secondary circuits. By confirming that the figures were backed on an operating programme up to 2003, he was sure that there was a need to the operating limits for damaged fuel pins.

The next presentation was made by Mr. E. Popov (IPPE, Russia) on the "Evaluation of accumulated activity in structural materials of the BN-350 reactor and in the reactor cavity". A considerable amount of tabular data was presented and discussed. The figures had been produced using the computer code "RISPACT". Steel structure activity was dependent on the level of impurities, e.g. decay times of manganese ~10 years, europium ~100 years, and cobalt ~150 years. In concrete calcium dominates for ~100 years and in graphite C14 dominates for 10-12 years. Mr. Popov concluded by suggesting that radioactive waste in future reactors can be reduced by reducing or eliminating certain impurities, e. g. replace nickel with manganese in standard steels, reducing amount of cobalt to 0.01ppm and reducing the niobium and europium content.

The final presentation of the session addressing the "Radiological conditions in the cells of the BN-350 NPP primary and secondary systems and in the reactor cavity" was made by Mr. V. Sergeyev (IPPE, Russia). After discussing the conditions that were anticipated at the start of decommissioning in 2003 and measuring that would have to be taken to reduce radiation beam through empty pipes from the reactor vessel and replacement of safety and monitoring equipment Mr. Sergeyev concluded that

it was not deemed possible to disassemble the primary circuit without robotic in under 100 years although limited access to structures should be possible after 50 years. Limited access could be gained after 10 years provided chemical decontamination was carried out on the primary circuit.

Fourth Session

Ms. F. Masse (CEA, France) presented the paper on caesium removal from LMFBR primary sodium. Primary sodium of French fast reactors is mainly contaminated by caesium. Other radio contaminants as ^{56}Mn, ^{60}Co or tritium have shorter radioactive periods and a lower solubility in sodium (in deposed contamination). In Rapsodie, ^{137}Cs was the most important radioactive isotope. In order to reduce the activity of the primary sodium, it is possible to trap caesium on carbonaceous materials. This method was used for Rapsodie but also on CEA experimental loops. The activity can be lowered by a factor of about 12 for each removal operation (at 200 °C). For lower temperature, the efficiency decreases. The purpose of the Rapsodie caesium removal operation was to obtain ^{137}Cs activity at the same level as 22Na activity. Due to the fact that sodium destruction occurred ten years after caesium removal operation, ^{22}Na activity decreased by a factor of ten whereas ^{137}Cs activity decay was negligible. The best is then to reduce as much as possible the caesium activity. In order to verify if a total removal is possible and to be able to design such device, basic studies and modelling works are still carried out in CEA. The aim is to limit biological shieldings and dose rates on sodium destruction devices, to reduce activity level of sodium hydroxide produced and optimize the type of final waste, and to make easier carriage or release of the sodium hydroxide.

Mr. Michelbacher (ANL, USA) presented the paper on "The Sodium Process Facility at Argonne National Laboratory - West". The Sodium process facility (SPF) was designed to react elemental sodium to sodium carbonate through two-stages involving caustic process and carbonate process steps. The sodium is first reacted to sodium hydroxide in the caustic process step. The caustic process step involves the injection of sodium into a nickel reaction vessel filled with a 50 wt% solution of sodium hydroxide. Water is also injected, controlling the boiling point of the solution. In the carbonate process, the sodium hydroxide is reacted with carbon dioxide to form sodium carbonate. His dry powder, similar in consistency to baking soda, is a waste form acceptable for burial in the State of Idaho as a non-hazardous, radioactive waste. The sodium will be processed in three separate and distinct campaigns: the 290,000 litres (77,000 gallons) of Fermi-1 primary sodium, the 50,000 litres (13,000 gallons) of the EBR-II secondary sodium, and the 330,000 litres (87,000 gallons) of the EBR-II primary sodium. The Fermi-1 and the EBR-II secondary sodium contain only low-levels of radiation, while the EBR-II primary sodium has radiation levels up to 0.5 mSv (50 mrem) per hour at 1 meter. The EBR-II primary sodium will be processed last, allowing the operating experience to be gained with the less radioactive sodium prior to reacting the most radioactive sodium.

The final presentation was on "The Handling of Radioactive Water Solutions with High Salt Content" by Mrs. N. Bogdanovich (IPPE, Russia). It was devoted to the experimental substantiation of R&D works on new technology of rendering harmless of medium level liquid radioactive waste (LRW), based on sorption extraction of caesium and strontium radionuclides, determining LRW activity, and formation of new type of cement, namely geocement, by means of radioactive sorbent immobilization into the mineral-like alkaline and earth-alkaline hydroaluminosilicates. Studies have been made on the principal laws of sorption of caesium and strontium radionuclides by the natural zeolite - clinoptilolite under the steady state and transient conditions from low and medium LRW, and immobilization of sorbent, containing Cs-137, Cs-134 and Sr-90 into the geocement stones. It has been shown that clinoptilolite, modified by the sodium ions is an effective sorbent for the extraction of Cs-137, Cs-134 and Sr-90 radionuclides from the medium LRW, and capable, provided that the prescribed parameters of the process are observed, of assuring high quality of LRW cleaning from caesium and strontium (cleaning degree up to 99.98%, distribution coefficient up to 1x106 ml/g) and transfer of

medium LRW to the non-radioactive category with respect to Cs-137, Cs-134 and Sr-90 content with their concentration lower than that permissible in water. In the process of immobilization of the sorbent containing caesium and strontium radionuclides, by grouting under normal humidity conditions, samples of geocement were obtained with the sorbent content up to 40 mass.%, high moisture resistance (10-6-10-7 g/cm2day value, which is close to that for the glass-like materials), and high compressive strength (10-25 Mpa). The process has been developed on the basis of accessible, low price natural and technogeneous materials, such as clinoptilolite as a sorbent, and metallurgical slag, clays and solutions of compositions of alkaline elements as components of binding system. Stability of the slag-alkaline cement stone (geocement), produced as a result of hydration solidification of the binding system is determined to considerable extent by the likeness of the sorbent and new formations of the binding system with respect to element composition and structure forming groups, thus allowing creation of hydroaluminosilicates - the analogues of the rock forming minerals, fit for the environmentally safe disposal or long term storing. The proposed method makes it possible to convert LRW, containing caesium and strontium radionuclides, into the solid form with more than two orders reduction of their volume, the consumption of the clinoptilolite being relatively low (2-3 kg/m3). After caesium and strontium radionuclides are extracted from the LRW salt solutions, the radioactivity of these solutions is decreased down to zero or insignificant level (if some other radionuclides are present), thus making much more easy their further management.

Fifth Session

At this session, three papers were presented:

-*Purification and utilization of the BN-350 reactor spent primary and secondary sodium coolant* (V. Matveev, IPPE, Russia),

- *Processing of the BN-350 reactor radioactive sodium waste into a state safe for burial* (A. Karpov, IPPE, Russia)

- *Washing from sodium and decontamination of circuits, equipment and structural materials of the BN-350 reactor* (Yu Nalimov, IPPE, Russia).

The following issues were discussed: At decommissioning of the BN-350 reactor there arises a problem of utilization of ~500 t of primary sodium with a specific activity of ~300 MBq/kg and of ~480 t of secondary sodium with a specific activity of 0.5 Bq/kg. Sodium can be used in the nuclear industry for the newly constructed BNs, and in chemical, metallurgical, engineering industries as well. In this case some limitations both on radioactive and non-radioactive impurities are imposed upon sodium.

The rules of radioactive materials transportation limit the -measurement level on the container surface (0.55 Sv/s) and at a distance of 1 m from it (0.028 Sv/s). On the basis of the calculations performed it has been shown that at using for sodium transportation of tank cars these requirements are met by sodium with a specific activity in $Na^{22} \leq 6 \cdot 10^5$ Bq/kg and in $Cs^{137} \leq 3.7 \cdot 10^5$ Bq/kg. Taking into account the initial sodium activity on these isotopes of $3 \cdot 10^7$ Bq/kg and $2.5 \cdot 10^8$ Bq/kg, respectively, the BN-350 primary sodium should be purified from cesium nuclides by a factor ~700; to reduce Na^{22} activity, its decay storage during ~15 years is required.

Another approach to coolant activity limitation is connected with non-admission of NPP personnel overirradiation in the course of equipment maintenance at start-up and adjustment activities on a new reactor. Calculations have shown that these requirements are met by sodium with Na^{22} activity $\leq 4.5 \cdot 10^6$ Bq/kg and with Cs^{137} activity $\leq 2.5 \cdot 10^6$ Bq/kg. For this purpose the spent sodium should be cleaned from cesium by a factor of 100 and stored during ~6 years for decay of Na^{22}.

For primary sodium utilization in the chemical industry more rigid requirements are posed: purification from cesium by a factor of 20 000, from -nuclides - by 10 times and decay storage during 32 years.

For inactive impurities the following requirements are posed: carbon $\leq$ 30 ppm, oxygen $\leq$ 10 ppm, nitrogen $\leq$ 10 ppm, potassium $\leq$ 200/1000 ppm, hydrogen $\leq$ 0.5 ppm. The spent coolant from the BN-350 satisfies these requirements, except for potassium impurity (in the I circuit-400 ppm and in the secondary one ~2000 ppm). Cleaning from potassium impurity is possible by means of a single distillation at 250°C or by dilution with pure sodium.

As a result of the analysis it has been shown that coolant activity reduction can be achieved by the following means:

- on cesium isotopes - by means of purification with the use of carbon-graphite filters;
- on tritium, hydrogen and oxygen - by means of purification using the cold filter traps;
- on corrosion products - by means of purification with the use of the getter traps.

The spent sodium coolant of the BN-350 secondary circuit can be used in a new BN reactor without purification.

For burial, only small (~20 t) volumes of the BN-350 radioactive waste sodium coolant unsuited for re-use, a so-called "sub-standard sodium", can be dispatched. It is sodium strongly polluted with the products of interaction with air or water, with fire extinguishing means, sodium removed from cold traps and cesium traps.

An analysis of methods for processing of this sodium for safe burial has been carried out. The analysis has revealed that the methods of waste sodium processing by means of sodium injection into water or alkali developed in France and UK cannot be used for the processing of the "sub-standard" sodium. The development of a new safe technology is required for which the experimental rig tests should be performed. To choose an optimum procedure of processing on the basis of the analysis it is recommended to test the four methods of sodium processing: burning in air, gas-phase (O_2 or CO_2), solid-phase (metal oxides), liquid-phase (with water under vacuum). The experimental studies performed at IPPE on a laboratory scale confirm the feasibility of these methods for the BN-350 "sub-standard" sodium radioactive waste. Semi-industrial rig tests will allow to develop the technology to be used at the BN-350 reactor.

There are presented results of the analysis and from the determination of sodium coolant volumes remaining in the circuits after draining of sodium from the primary, secondary circuits and of NaK from the cold traps cooling system. Their total volume is ~100 t. Recommendations are given on the mode of sodium removal from various parts of the plant down to volumes allowing safe conduction of operations on washing and decontamination of the circuits and equipment (~150 l in the reactor vessel, ~15 l in the cold trap).

As the main method for residual sodium removal there is recommended the steam-gaseous procedure mastered at thrice-repeated washing of the BR-10 reactor primary circuit. Preliminarily, for the I and II circuits it is recommended to carry out vacuum distillation of sodium.

For decontamination of the reactor primary circuit there are recommended solutions and procedures that have proved their effectiveness at carrying out decontamination of equipment of reactors in operation:

- 0.5% $KMnO_4$ at t=70°C;
- 5% HNO_3+1% $H_2C_2O_4$ at =70°C.

Cleaning of cold traps from sodium si recommended to be carried out using the technology developed by IPPE and used at reprocessing of the BN-350 reactor secondary circuit cold traps.

The main conclusions of the Consultancy can be summarized as follows:

1. For future reactor systems design:

- All materials used for the construction of the primary circuit and equipment should have a minimum level of constituents that could become activated. When such materials must be used preference should be given to those constituents that have short decay times.

- Wherever possible modular construction methods (i.e. readily assembled/disassembled) should be used for equipment and shielding.
- A bank of samples of all the types of materials used in construction and primary circuit equipment manufacture be established before the first criticality and that this should be complemented with a programme of samples being installed in the coolant circuit in appropriate places.
- Designers must address the draining of the coolant circuits and as appropriate engineering the necessary means to maximize drainage.
- Designers must always address the D&D issues that will relate to the specific equipment or system before manufacture is completed.

2. *For BN-350 sodium purification and disposal the goal is to minimize the disposal and maximize reuse of sodium:*

- Techniques for obtaining purity levels shall be developed or existing techniques refined;
- Storage options for reusable sodium shall be investigated, including the use of the BN-350 dump tank, barrels, etc.;
- Uses for sodium other than current and future LMFRs should be investigated;
- Continue to elevate disposal possibilities, comparing different options with different technologies. Costs and schedules associated with deployment of these options need to be established;
- BN-350 has a NaK/mineral oil mixture. Technologies need to be identified/developed for reprocessing this mixture. Applications for use of the clean NaK need to be considered.
- Identify and/or develop a technology for dealing with liquid waste. The goal in dealing with liquid waste is waste volume minimization and final waste form stabilization. Methods of achieving these goals should be investigated, such as radionuclide sorption from liquid radioactive waste and radioactive sorbent immobilization in the geocement based on the alkaline slag binders.

The meeting participants made the following recommendations:

1. The IAEA should co-ordinate and advise on all aspects relating to the preparation for the D&D and arrange no less frequently than annually a meeting of appropriate international experts to discuss and advise on the D&D preparations and programme, and use its best endeavours to obtain funds from the international community to ensure the most urgent tasks are undertaken in preparation for the D&D of BN-350.

2. A R&D programme supporting the D&D of BN-350 be set up by Kazakhstan in co-operation with other international organizations.

3. The identified failed fuel assembly should be withdrawn from the BN-350 reactor coolant as soon as practicable.

4. International collaboration should be set up to address the disposal of cold & caesium traps.

5. The development of robotic equipment and special instruments is considered particularly for the primary vessel and associated circuit of BN-350 NPP.

6. Kazakhstan, the United Kingdom, France, Russia, and the United States should collaborate through the IAEA to share experience and technologies for dealing with sodium residuals disposition in reactor and circuits.

Attachment 1

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Attachment 2

Consultant's Meeting on
"Technical Options for the Decommissioning of the
BN-350 LMFR in Kazakhstan"
****
IPPE - Obninsk, Russia
23-27 February 1998
Programme

Monday, 23 February 1998

09:30-10:10	Welcome address (IPPE) Opening of the meeting (IAEA) Introduction of participants by themselves Review and approval of the Programme (Chairman, all)
10:10-11:10	*"Current status and decommissioning issues of the BN-350 NPP"*, P. Nazarenko (BN-350, Kazakhstan), L. Kochetkov (IPPE Russia)
11:10-11:30	Coffee break
11:30-12:30	*"Decommissioning of the UK reactors DFR and PFR"*, E. Adam (UKAEA)
12:30-14:00	Lunch
14:00-15:00	*"Dismantling of Rapsodie reactor at level 2"*, G. Streiff (CEA, France)
15:00-15:40	*"The deactivation of the EBR-II complex"*, J. Michelbacher and P. Planchon (ANL, USA)
15:40-16:00	Coffee break
16:00-16:40	*"Sodium disposal at DESORA installation"*, J.-L. Garsia (CEA, France)
16:40-17:30	Questions, discussion

Tuesday, 24 February 1998

09:00-09:40	*"Evaluation of accumulated activity in the cells of the BN-350 NPP primary and secondary systems"*, A. Tsykunov (IPPE)
09:40-10:20	*"Evaluation of accumulated activity in the structural materials of the BN- 350 reactor and in the reactor cavity", E. Popov (IPPE)*
10:20-11:00 *secondary*	*"Radiological conditions in the BN-350 NPP cells primary and systems and in the reactor cavity"*, V. Sergeyev
11:00-11:20	Coffee break
11:20-12:00	*"Sequence of works of NPP decommissioning spent fuel"*, L. Kochetkov (IPPE)
12:00-12:30	*"Cleaning and decontamination of components and piping of the BN-350 NPP primary and the secondary sodium systems and auxiliary NaK circuit"*, Yu Nalimov (IPPE)
12:30-14:00	Lunch
14:00-14:40	*"Purification and utilization of the BN-350 primary and secondary sodium and sodium-potassium alloy"*, V. Matveyev (IPPE)
14:40-15:20	*" Disposal of sodium coolant waste"*, A. Karpov (IPPE)
15:20-15:40	Coffee break
15:40-16:20 *West"*,	*"The sodium process facility at Argonne National Laboratory-* J. Michelbacher and P. Planchon (ANL, USA)
16:20-17:00	*"Handling of radioactive water solutions with high salt"*, N. Bogdanovich (IPPE)

Wednesday, 25 February 1998

09:30-10:20	*"Sodium decontamination from caesium"*, F. Masse (CEA, France)
10:20-11:00	Questions, discussions (Chairman, all)
11:00-11:20	Coffee break

11:20-13:00	Conclusions and recommendations of the Consultant's Meeting (Chairman, all)
13:00-14:30	Lunch
15:00-18:00	Sightseeing tour of Borovsk Pafnutyev Monastery
19:00	Reception

Thursday, 26 February 1998

09:30-12:00	Technical tour of the IPPE
12:30-14:00	Lunch
14:00-17:00	Preparation of the Meeting Summary Report (Chairman, all)

Friday, 27 February 1998

09:30-11:00	Review and approval of the Meeting Summary
11:00-11:20	Coffee break
11:20-12:00	Final conclusions and recommendations of the Meeting
12:00-13:30	Lunch
13:30-14:00	Adjourn Chairman, IPPE, IAEA

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IAEA DOCUMENTS OF THE INTERNATIONAL WORKING GROUP ON FAST REACTORS

Ref. Number	Title of Meeting, place and date, title of IAEA-TECDOC, data of publication
(no ref. no.)	Special Meeting (SM) on Sodium-Water Reactions Argonne, USA 5-6 November 1968
(no ref. no.)	SM on the Value of Plutonium Alpha Winfrith, United Kingdom 30 June - I July 1969
(no ref. no.)	SM on Core Instrumentation for Sodium-Cooled Fast Reactors Karlsruhe, Federal Republic of Germany 9-10 October 1969
(no ref. no.)	SM on the Detection and Location of Cladding Failures in Fast Reactors Cadarache, France 5-6 October 1970
(no ref. no.)	SM on Operational Safety of Sodium Circuits Risley, United Kingdom 17-19 March 1971
(no ref. no.)	SM on Sodium-Water Reactions Melekess, USSR 18-21 May 1971
(no ref. no.)	SM on Fission and Corrosion Product Behaviour in Primary Systems of LMFBRs Bensberg, Federal Republic of Germany 20-22 September 1971
(no ref. no.)	SM on Handling and Transportation of Spent Fuel Elements for LMFBRs Rome, Italy 26-28 April 1972

(no ref.no.) SM on Sodium Combustion and its Extinguishment-Techniques and Technology
Richland, Washington, USA
22-25 May 1972

(no ref. no.) SM on Sodium Impurity Measurements and Control Cadarache, France
14-17 November 1972

(no ref. no.) SM on Decontamination of Plant Components from Sodium and Radioactivity
Caithness, Scotland
9-12 April 1973

(no ref. no.) SM on Decontamination ... / Appendix to Summary Report

(no ref. no.) SM on Handling of the Design for and Mitigation of Thermal Transients in LMFBR Plants
Canoga Park, California, USA
17-21 June 1974

(no ref. no.) SM on Operating Experience and Design Criteria of Sodium Valves
Richland, Washington, USA
23-27 September 1974

(no ref. no.) Sodium Rig Engineering Safety Philosophy and Code of Practice by P.H. Delves, UKAEA, 1975

(no ref. no.) Summary of the Main Results of the Different Hot Spot Factor Calculations by F. Hofmann, Karlsruhe Nuclear Centre, 1975

(no ref. no.) IAEA Advisory Group Meeting on LMFBR Fuel Reprocessing
Leningrad, USSR
17-21 May 1976

(no ref.no.) IWGFR First Annual Meeting
Vienna, Austria
5-7 March 1968

(no ref. no.) IWGFR Second Annual Meeting
Vienna, Austria
18-20 March 1969

(no ref.no.) IWGFR Third Annual Meeting
Cadarache, France

	18-20 March 1970
(no ref. no.)	IWGFR Fourth Annual Meeting Vienna, Austria 5-7 May 1971
(no ref.no.)	IWGFR Fifth Annual Meeting (Part 1) Vienna, Austria 19-21 April 1972
(no ref. no.)	IWGFR Fifth Annual Meeting (Part 11)
(no ref. no.)	IWGFR Sixth Annual Meeting (Part 1) Vienna, Austria 9-11 May 1973
(no ref. no.)	IWGFR Sixth Annual Meeting (Part 11)
(no ref. no.)	IWGFR Sixth Annual Meeting (Part 111)
(no ref. no.)	IWGFR Seventh Annual Meeting (Part 1) Winfrith, United Kingdom 18-20 March 1974
(no ref. no.)	IWGFR Seventh Annual Meeting (Part 11)
(no ref. no.)	IWGFR Senventh Annual Meeting (Part 111)
IWGFR/1	Study Group Meeting on Steam Generators for LMFBRs Bensberg, Federal Republic of Germany 14-17 October 1974
IWGFR/1/Suppl.l	Supplement to IWGFR/1
IWGFR/2	IWGFR Eighth Annual Meeting (Part 1) Vienna, Austria 15-18 April 1975
IWGFR/3	IWGFR Eighth Annual Meeting (Part 11)
IWGFR/4	SM on the Reliability of Decay Heat Removal Systems for Fast Reactors Harwell, United Kingdom 28 April - 1 May 1975

IWGFR/5	SM on Fuel Failure Mechanisms Seattle, Washington, USA 12-16 May 1975
IWGFR/6	IWGFR Eighth Annual Meeting (Part 111)
IWGFR/7	SM on Fission and Corrosion Product Behaviour in Primary Circuits of LMFBRs Dimitrovgrad, USSR 8-11 September 1975
IWGFR/8	IWGFR Ninth Annual Meeting (Part 1) Vienna, Austria 30 March - 2 April 1976
IWGFR/9	SM on Cavitation in Sodium and Studies of Analogy with Water as Compared to Sodium Cadarache, France 12-16 April 1976
IWGFR/10	In-Service Inspection and Monitoring of LMFBRs Bensberg, Federal Republic of Germany 9-11 March 1976
IWGFR/11	SM on High Temperature Structural Design Technology of LMFBRs Champion, Pennsylvania, USA 27-30 April 1976
IWGFR/12	IWGFR Ninth Annual Meeting (Part 11 and Part 111)
IWGFR/13	Profiles of Facilities Used for FBR Research and Testing
IWGFR/13-1	Supplement to IWGFR/13
IWGFR/14	LMFBR Plant Parameters
IWGFR/14/Rev.1	LMFBR Plant Parameters
IWGFR/15	Technical Committee Meeting on Aerosol Formation, Vapour Deposits and Sodium Vapour Trapping Cadarache, France 13-17 December 1976
IWGFR/16	Technical Committee Meeting on Fuel and Cladding Interaction Tokyo, Japan

	21-25 February 1977
IWGFR/17	IWGFR Tenth Annual Meeting (Part 1) Vienna, Austria 29 March - 1 April 1977
IWGFR/18	IWGFR Tenth Annual Meeting (Part 11)
IWGFR/19	SM on Role of Fission Products in Whole Core Accidents AERE, Harwell, United Kingdom 28 June - 1 July 1977
IWGFR/20	IWGFR Tenth Annual Meeting (Part 111)
IWGFR/21	SM on LMFBR Flow Induced Vibrations Argonne, Illinois, USA 20-23 September 1977
IWGFR/22	SM on Properties of Primary Circuit Structural Materials Including Environmental Effects Bergisch Gladbach, Federal Republic of Germany 17-21 October 1977
IWGFR/23	SM on Sodium Removal and Decontamination Richland, Washington, USA 14-16 February 1978
IWGFR/24-1	IWGFR Eleventh Annual Meeting (Part 1) Bologna, Italy 17-20 April 1978
IWGFR/24-2	IWGFR Eleventh Annual Meeting (Part 11)
IWGFR/24-3	IWGFR Eleventh Annual Meeting (Part 111)
IWGFR/25	SM on Leak Detection and Location in LMFBR Steam Generators Dimitrovgrad, USSR 6-9 June 1978
IWGFR/26	SM on Equations of State of Materials of Relevance to the Analysis of Hypothetical Fast Breeder Reactor Accidents AERE, Harwell, United Kingdom 19-23 June 1978
IWGFR/27	Compilation of Piping Benchmark Problems -

	Cooperative International Effort June1979
IWGFR/28	SM on Sodium Fires and Prevention Cadarache, France 20-24 November 1978
IWGFR/29	SM on Thermodynamics of FBR Fuel Subassemblies under Nominal and Non-Nominal Operating Conditions Karlsruhe, Federal Republic of Germany 5-7 February 1978
IWGFR/30-1	IWGFR Twelfth Annual Meeting (Part 1) Vienna, Austria 27-30 March 1979
IWGFR/30-2	IWGFR Twelfth Annual Meeting (Part 11)
IWGFR/30/3	IWGFR Twelfth Annual Meeting (Part 111)
IWGFR/31	SM on Theoretical Modelling of LMFBR Fuel Pin Behaviour, Fontenay-aux-Roses, France, 28 May - 1 June 1979
IWGFR/32	SM on Bellows for Sodium Systems Tokyo, Japan 5-9 November 1979
IWGFR/33	SM on Carbon in Sodium Harwell, United Kingdom 27-30 November 1979
IWGFR/34-1	IWGFR Thirteenth Annual Meeting (Part 1) Vienna, Austria 9-11 April 1980
IWGFR/34-2	IWGFR Thirteenth Annual Meeting (Part 11)
IWGFR/34-3	IWGFR Thirteenth Annual Meeting (Part 111)
IWGFR/35	SM on In-Service Inspection and Monitoring of LMFBRs Bensberg, Federal Republic of Germany 20-22 May 1980

IWGFR/36	SM on Demonstration of Structural Integrity under Normal and Faulted Conditions Chester, United Kingdom 3-5 June 1980
IWGFR/37-1	IWGFR Fourteenth Annual Meeting (Part 1) Vienna, Austria 31 March - 3 April 1981
IWGFR/37-2	**IWGFR** Fourteenth Annual Meeting (Part 11)
IWGFR-38 KfK 3203 Kernforschungs- zentrum Karlsruhe	Fuel Failure Detection and Location in LMFBRs Karlsruhe, Federal Republic of Germany 11-14 May 1981
IWGFR/39	SM on Design Features Affecting a Dynamic Behaviour of Fast Reactor Cores Rome, Italy 2-5 June 1981
IWGFR/40	SM on Sodium Boiling Noise Detection Chester, United Kingdom 9-11 June 1981
IWGFR/41	IWGFR Fifteenth Annual Meeting Obninsk, USSR 30 March - 2 April 1982
IWGFR/42	IWGFR Fifteenth Annual Meeting (Status of National Programmes on Fast Breeder Reactors) Obninsk, USSR 30 March - 2 April 1982
IWGFR/43	SM on Sodium Fires, Design and Testing Richland, USA 24-28 May 1982
IWGFR/44	SM on Advances in Structural Analysis for LMFBR Applications Paris, France 11-15 October 1982
IWGFR/45	SM on Thermal Stratification in Sodium Grenoble, France 18-21 October 1982

IWGFR/46	IWGFR Sixteenth Annual Meeting Vienna, Austria 12-15 April 1983
IWGFR/47	IWGFR Sixteenth Annual Meeting (Status of National Programmes on Fast Breeder Reactors) Vienna, Austria 12-15 April 1983
IWGFR/48	SM on Absorber Materials and Control Rods for Fast Breeder Reactors Obninsk, USSR 7-10 June 1983
IWGFR/49	SM on Mechanical Properties of Structural Materials including Environmental Effects Vol. 1 and Vol. 2 Chester, United Kingdom 10-14 October 1983
IWGFR/50	SM on Theoretical and Experimental Work on LMFBR Steam Generator Integrity and Reliability with a Particular Reference to Leak Development and Detection The Hague, The Netherlands 9-11 November 1983
IWGFR/51	LMFBR Plant Parameters
IWGFR/52	IWGFR Seventeenth Annual Meeting (Status of National Programmes on Fast Breeder Reactors) Vienna, Austria 3-6 April 1984
IWGFR/53	SM on Maintenance and Repair of LMFBR Steam Generators O-Arai Engineering Center, PNC, Japan 4-8 June 1984
IWGFR/54	SM on Predictions and Experience of Core Distortion Behaviour Wilmslow, United Kingdom 1-4 October 1984
IWGFR/55	SM on Methods and Tools to Detect Thermal Noise in Fast Reactors Bologna, Italy 8-10 October 1984

IWGFR/56	IWGFR Eighteenth Annual Meeting (Status of National Programmes on Fast Breeder Reactor Development) Vienna, Austria 16-19 April 1985
IWGFR/57	SM on Heat and Mass Transfer in the Reactor Cover Gas Harwell, United Kingdom 8-10 October 1985
IWGFR/58	SM on Cavitation Criteria for Designing Mechanisms Working in Sodium Application to Pumps Interatom GmbH., Bergisch Gladbach, Federal Republic of Germany 28-29 October 1985
IWGFR/59	SM on LMFBR Fuel Rod Behaviour Under Operational Transients Kalpakkam, India 3-6 December 1985
IWGFR/60	IWGFR Nineteenth Annual Meeting Kalpakkam, India 11-14 March 1986
IWGFR/61	SM on Cover Gas Purification in LMFBRs Richland, USA 24-26 September 1986
IWGFR/62	SM on Flow Induced Vibrations in FBRs Paris, France 22-24 October 1986
IWGFR/63	Status of National Programmes on FBRS; IWGFR 20th Annual Meeting Vienna, Austria 24-27 March 1987
IWGFR/64	SM on Fission and Corrosion Products Behaviour in Primary Circuits of LMFBRs Karlsruhe, Fed. Rep. of Germany 5-8 May 1987
IWGFR/65	SM on LMFBR Block Antiseismic Design and Verification Bologna, Italy 12-15 October 1987 (to be published)
IWGFR/66	Status of National Programmes on Fast Breeder Reactors,

	Twenty-First Annual Meeting, Seattle, USA, 9-12 May 87
IWGFR/67	SM on Sodium Fires, Obninsk, USSR, 6-9 June 1988
IWGFR/68	Final Report of the Co-ordinated Research Programme on Signal Processing Techniques for Sodium Boiling Noise Detection 1985-1988
IWGFR/69	Proceedings of Specialists' Meeting on Methods for Reactor Physics Calculations for Control Rods in Fast Reactors, Winfrith, UK, 6-8 December 1988
IWGFR/70	Status of National Programmes on Fast Breeder Reactors, 22nd Annual Meeting, Vienna, 18-21 April 1989
IWGFR/71	Specialists' Meeting on Advanced Controls for Fast Reactors, Argonne, Ill., USA, 20-22 June 1989
IWGFR/72	SM on Instrumentation for Supervision of Core Cooling, Kalpakkam, India, 12-15 December 1989 (to be published).
IWGFR/73	Status of National Programmes on Fast Breeder Reactors, 23rd Annual Meeting, Vienna, 24-27 April 1990.
IWGFR/74	Review of IWGFR Activities in 1989.
IWGFR/75	Final Report of a Co-ordinated Research Programme on Intercomparison of LMFBR Core Mechanics Codes, Verification and Validation of LMFBR Static Core Mechanics Codes, Part I
IWGFR/76	Verification and Validation of LMFBR Static Core Mechanics Codes, Part II, RCM on CRP on Intercomparison of LMFBR core Mechanics Codes, VIC, 1986-89
IWGFR/77	Summary Report on 23rd Annual Meeting of the IWGFR Vienna, 24-27 April 1990

IWGFR/78	SM on Steam Generator Failure and Failure Propagation, Aix-en-Provence, France, 26-28 September 1990
IWGFR/79	SM on Steam Generator Acoustic/Ultrasonic Detection in Sodium Water Leaks on Steam Generators, Aix-en-Provence, France, 1-3 October 1990
IWGFR/80	LMFBR Plant Parameters, 1991
IWGFR/81	Review Report on the IWGFR Activities for the Period since the 23rd Annual Meeting of the IWGFR, March 1991
IWGFR/82	Summary Report of the 24th Annual Meeting of the International Working Group on Fast Reactors, Tsuruga, Japan, 15-18 April 1991
IWGFR/83	Status of National Programmes on Fast Breeder Reactors 24th Annual Meeting, Tsuruga, Japan, 15-18 April 1991
IWGFR/84 KfK 4935	Proceedings of an IAEA Specialists' Meeting on Properties of Structural Materials in Liquid Metal Environment, Karlsruhe, Germany, 18-20 June 1991
IWGFR/85	Specialists' Meeting on Passive and Active Safety Features of LMFBS. Japan, 5-7 November 1991
WORKING MATERIALS IAEA-TC-385.53	Status of National Programmes on Fast Breeder Reactors, Summary Report of the 25th Annual Meeting of the IWGFR, Vienna, 27-30 April 1992
IWGFR/87	Specialists'Meeting on Seismic Isolation Technology, San Diego, CA., 18-20 March 1992
IWGFR/88	Specialists'Meeting on Evaluation of Decay Heat Removal by Natural Convection, Oarai Engineering Centre, PNC, Japan, 22-23 February '93
IWGFR/89	Material-Coolant Intaraction and Material Movement and Relocation in LMFR TCM, O-arai, Japan, 1994
IWGFR/90	Correlation Between Material Properties and Thermohydraulics Conditions in LMFRs, SM Aix-en-Provence, France, 1994

IWGFR/91	Status of National Programmes on Fast Reactors, 1995-1996 IWGFR, 1996
IWGFR/92	Evaluation of Radioactive Materials Release and Sodium Fires in LMFRs, TCM O-arai, Japan, 1996
TECDOC-154	Gas-Cooled Fast Reactors, Mnisk, 24-28 July 1972 Proceeding of the IAEA Study Group Meeting sponsored by the USSR State Committee on the Utilization of Atomic Energy
IWGFR/93 (94)	Intercomparison of Analysis Methodes for Seismically Isolated Nuclear Structure RCM, St. Petersbourg, Russia, 1996
TECDOC-166	Principles of Hot-Channel Factor Calculations for Fast Reactors, 1974
Technical Report Series No. 246	Status of Liquid Metal Cooled Fast Breeder Reactors 1985
TECDOC-466	Advanced Fuel for Fast Breeder Reactors: Fabrication and Properties and Their Optimization Proceedings of a TCM, VIC, 3-5 November 1987
TECDOC-514	Seismic Analysis of Liquid Metal Fast Breeder Reactors, Report prepared by R.J. Gibert, CEA, France, and A. Martelli, ENEA, Italy, 1989
TECDOC-687	Fission and Corrosion Product Behaviour In Liquid Metal Fast Breeder Reactors, 1993
TECDOC-693	Use of Fast Reactors For Actinide Transmutation Proceedings of a Specialists Meeting, Obninsk, Russian Federation, 22-24 September 1992
TECDOC-730	Fast Reactors Steam Generators with Sodium on the Tube Side: Design and Operational Parameters, 1993
TECDOC-731	Evaluation Of Benchmark Calculations On A Fast Power Reactor Core With Near Zero Sodium Void Effect, Final Report Of An International Benchmark Programme Supported By The International Atomic Energy Agency And The Commission Of The European Communities, 1992-1993

TECDOC-741	Status of National Programmes On Fast Reactors IWGFR, 1993
TECDOC-791	Status of Liquid Metal Fast Reactor Development IWGFR, 1993
TECDOC-798	Intercomparison of Liquid Metal Fast Reactor Seismic Analysis Codes Vol. 1: Validation of Seismic Analysis Codes Using Reactor Core Experiments, 1995
TECDOC-817	Influence of Low Dose Irradiation on the Design Criteria of Fixed Internals in Fast Reactors, 1995
TECDOC-819	Earthquakes: Isolation, Energy Dissipation and Control of Vibrations of Structures for Nuclear and Industrial Facilities and Buildings, 1995
TECDOC-829	Intercomparison of Liquid Metal Fast Reactor Seismic Analysis Codes Vol. 2: Verifications and Improvements of Reactor Core Seismic Analysis Codes Using Core Mock-up Experiments, 1995
TECDOC-866	Fast Reactors Data-Base, 1996
TECDOC-876	Progress in Liquid Metal Fast Reactor Technology IWGFR 1995
TECDOC-882	Intercomparison of Liquid Metal Fast Reactor Seismic Analysis Codes Vol. 3: Comparison of Experimental Results with Computer Prediction for Reactor Cores, 1996
TECDOC-884	Absorber Materials Control Rods and Design of Shutdown Systems for Advanced Liquid Metal Fast Reactor, 1996
TECDOC-907	Conceptual Design of Advanced Fast Reactor, 1996
TECDOC-908	Fast Reactor Fuel Failures and Steam Generator Leaks: Transient and Accident Analysis Approaches, 1996
TECDOC-933	Creep-Fatigue Damage Rules for Advanced Fast Reactor Design, 1997
TECDOC-946	Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium-Water Reactions in Liquid Metal Fast Reactors, 1997

<u>to be published in 1998:</u>

Advances in Fast Reactors Technology (IWGFR-30, Beijing, China, 13-16 May 1997)

Influence of High Dose Irradiation on Advanced Reactor Core Structural and Fuel Materials (TCM, Obninsk, Russia, June 16-19, 1997)

Sodium Removal and Disposal from LMFRs in Normal Operation and in the Framework of Decommissioning (TCM, Aix-en-Provence, France, November 3-7, 1997)

INTERNATIONAL ATOMIC ENERGY AGENCY

International Working Group on Fast Reactors

Technical Committee Meeting

on

"Methods and Codes for Calculations of Thermohydraulic
Parameters for Fuel, Absorber Pins
and Assemblies of LMFRs
with Traditional and Burner Cores "

Obninsk, Russia,

27-31 July, 1998

Organized in Co-operation with the Institute of Physics

and Power Engineering (IPPE), MINATOM, Russia

INFORMATION SHEET AND CALL FOR PAPERS

1. INTRODUCTION

The Technical Committee Meeting (TCM) on "Methods and Codes for Calculations of Thermohydraulic Parameters for Fuel, Absorber Pins and Assemblies of LMFRs with Traditional and Burner Cores" will be held in Obninsk, Russian Federation from 27 to 31 July 1998.

The issues which will be discussed at this TCM have general scientific value, since the approaches discussed will be useful for LMFRs and also for other types of nuclear systems cooled by liquid metals (e.g., subcritical blankets of fission/fusion hybrids).

2. TECHNICAL BACKGROUND AND OBJECTIVES OF THE MEETING

2.1. The core is the most loaded component of nuclear reactor, its reliable performance largely determines operation of the nuclear power plant. Consideration of the processes in the reactor core requires an understanding of mutual correlation between neutron, thermohydraulic and mechanical parameters of the core.

The purpose of the meeting is to provide a forum to review the methods and codes for calculation of fuel and absorber pins and of fuel assemblies thermohydraulic parameters in LMFR with traditional and burner cores, and to discuss the thermohydraulic advances and design approaches providing a reliable, safe and robust reactor core, capable of routinely achieving high burnup of plutonium.

2.2. The temperature behavior of the fuel and absorber pins is influenced by many factors , such as non-uniform power generation, deformation of fuel and cladding due to swelling and thermal stresses, inter-channel and inter-assembly heat exchange, etc., which affect core performance.

Previous investigations on thermal hydraulics of the above mentioned core components have allowed the influence of many factors on temperature to be taken into account. It should be noted that these studies have brought the thermohydraulic substantiation of the reactor core performance to a higher qualitative level.

The purpose of the present meeting is to review the results of experimental and analytical research on the effect of various geometrical and flow parameters on temperature fields and heat transfer in fuel and absorber pins, assemblies and the reactor core as a whole.

2.3. By now a lot of numerical procedures and codes have been developed to maintain thermohydraulic validation of thermal processes in the components of the reactor core. The approaches developed vary in type of equations, numerical scheme, closing relationships.

The availability of numerous approaches raises the problem of verification of the codes and numerical methods.

As applied to reactor core operational safety, of special importance are the features of core transient modeling and the problems of the optimization of the core parameters from the point of view of thermohydraulics.

The exchange of information on the results obtained, on analytical approaches and codes to support projects of experimental and power LMFRs, as well as experience of their use seems to be of interest and it would allow to define lines of further investigations.

The purpose of the meeting is to exchange accumulated experience in different countries on using the codes of thermohydraulic calculation, to indicate the methods of verification of the codes available for the calculation of reactor core thermal hydraulics under nominal and non-nominal conditions and to discuss the issues requiring further research and development.

3. OVERVIEW AND PROPOSED TOPICS OF THE MEETING

During the period of Liquid Metal Cooled Fast Reactor (LMFR) development, thermohydraulic investigation has been one of the issues among the physical and mechanical ones. This topic has been a key research subject of the IAEA WGFR Specialists Meeting (SPM) on Thermodynamics of Fast Breedser Reactor (FR) Fuel Subassemblies Under Nominal and Non-Nominal Operating Conditions, Karlsruhe, Germany, February 1979; and SPM on FR Fuel Rod Behaviour under Operational Transients, Kalpakkam, India, December 1985.

Unfortunately, the current research programmes in the world in this field are not as active as they used to be. Therefore, it is believed to be highly beneficial to take the opportunity of IAEA Technical Committee Meetings to review and compile the state-of-the-art understandings and key remaining issues on this subject. It is hoped that the meeting proceedings would serve as a present status document, on which future research programmes in Member States can be based.

The provisional list of topics of the meeting is proposed below. Participants are encouraged to prepare papers describing their views freely, based on their studies and literature survey, without strict adherence to the proposed topics. In spite of the nature of the meeting, concentrating more on general review of the current status, and discussions on individual specific subjects may be admissible. It is recommended to provide an adequate and sufficient objective review of the current status on the subject ,so that the papers would be useful to all the IWGFR members. It is also desirable not only to discuss direct results of the studies but to describe their implication in core designs and reactor safety.

Proposed topics of the meeting

a) Thermohydraulics modeling, methods and codes for:

- fuel and absorber pins;
- fuel assembly (traditional and burner);
- control rod;
- reactor core (traditional and burner) under steady-state and transient conditions;
- conjugate issues on pin/coolant heat and momentum transfer;

- deformed pin bundles, non-uniform power distributions, swelling expansion;
- special types fuel assembly design, e.g. ductless fuel assembly;

b) Heat transfer and fluid flow in fuel assemblies and cores under natural convection decay heat removal conditions:

- interassemblies heat transfer;
- intraassembly heat transfer;
- plenum-core thermohydraulic coupling.

c) Thermohydraulic modeling and simulation of reactor core and assemblies under abnormal and accident conditions:

- liquid metal boiling heat transfer;
- local blockage;
- total inlet blockage.

d) Influence of various factors on heat transfer process and temperature behaviour of fuel and absorber pins including:

- non-uniform power distributions;
- inter-subchannel and inter-assembly heat exchanges;
- type of spacers;
- uncertainties and sensitivity analysis in experimental techniques and computational models.

e) Inter comparison of different methods (subchannel, porous model ect)

f) FR core thermohydraulic design to increase negative reactivity feedback under abnormal operating conditions.

Meeting sessions will be organised based on the number and specific topics of paper abstracts submitted, and the final programme will be announced to designated participants.

The place of the meeting is:

State Scientific Centre of the Institute of Physics and Power Engineering
Bondarenko sq., 1, Obninsk, Kaluga Region, 249020 RUSSIA
Fax:(007) 095 2302326 or (007) 095 8833112
e-mail: sorokin@obninsk.ru
sorokin@lab22.obninsk.su

4. REQUIREMENTS FOR ABSTRACTS PAPERS AND PRESENTATIONS

Each national delegation is required to send abstracts of not more than 400 words of all the papers to the address shown in item 6, by 20 June 1998, and bring 30 copies of each paper to be distributed at the beginning of the meeting. The abstracts will be used to organise the technical sessions; therefore, they should contain sufficient detail to allow' the organisers to finalise the meeting programme.

Each paper must begin with an abstract, typed on one side in single-space with a column width not exceeding 160 x 250 mm, and should be prepared in a suitable form for publication in proceedings. At least one original or camera-ready copy of each paper should be given to Mr. Anatoli Rinejski at the meeting.

Two overhead projectors will be available at the meeting for the oral presentations of papers. The use of other standard equipment, such as a 35 mm slide projector, a 16 mm film projector and a video film projector, can also be arranged; such requests should be made when the abstract is sent. The time allocated to each presentation and the final programme of the meeting will be determined later.

5. SECRETARIAT

The Scientific Secretary of the meeting is Mr. A. Rinejski of the Division of Nuclear Power (NENP), Nuclear Power Technologies Development Ssection, Mr. V. Yarovytsin of the IPPE International Division is the local co-ordinator, and Mr. A.Sorokin of the IPPE is the chairman of the meeting.

6. PARTICIPATION

Each participant should complete the attached forms A (participation form) and B (accommodation booking) and send them to the competent official authority (Ministry of Foreign Affairs or National Atomic Energy Authority) for transmission to the International Atomic Energy Agency and IPPE before 20 June 1998:

Address:

Mr. A. Rinejski
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E-mail: a.rineiskii.@iaea.org

Mr. V. Yarovytsin
International Division of the Institute of
Physics and Power Engineering
Bondarenko square, 1
Obninsk, Kaluga Region,
249020 RUSSIA
Tel: 007 08439 98815
Fax: 007 095 8833112
Telex: 911509 URAN SU

(for **urgent** communication only).

The designation of a participant will be considered for acceptance only if forwarded by the Government of an IAEA Member State or International Organisation invited to participate.

7. WORKING LANGUAGE

The Meeting will be conducted in English only. All communications, abstracts and papers must be in this language.

8. VISAS

Designated participants who need a visa for entering Russia should submit the necessary application to the nearest diplomatic or consular representative of the Russian Federation as soon as possible. For the issuance of visas in time the Organising Committee must have the following information from all specialists and their accompanying persons, arriving at the Meeting from outside Russia:

- Full name,
- Date and place of birth,
- Passport number (issuance and expire data), place of work, position or title.

This information should be sent to the Organising Committee before 20 June 1998.

9. EXPENDITURES

No registration fee is charged to participants. Travel and subsistence expenses of participants will not be borne by the Agency. However, assistance may be offered to a limited number of participants from developing countries provided they make an important contribution.

Please note that compensation is not paybale by the Agency for any damage to or loss of the experts' personal property. However, for the period of the experts' engagement with the agency including travel between their residence and the duty station, they 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 for medical expenses up to an amount of US $ 10,000 in case of illness, injury or death attributable to their relationship with the Agency, subject to the terms, of the insurace policy.

10.LOCAL ARRANGEMENT

An optional technical tour of IPPE will be arranged, following the meeting.

Any correspondence concerning matters, including abstracts and papers, should be sent to Prof. A. Sorokin, Chairman of the TCM at the following address:

Prof. A. Sorokin
Fast Reactor Division
Federal Scientific Center of Russian Federation
Institute of Physics and Power Engineering (IPPE)
Bondarenko Square 1
249020 Obninsk, Kaluga region
Russia
Tel: 007 - 084 399 8447
Fax: 007 - 095 23 02326 or 007 - 095 88 33112
E-mail: sorokin@obninsk.ru
sorokin@lab22.obninsk.su

and copy to Mr. A. Rinejski (the address is given in item 6).

FORM A

INTERNATIONAL ATOMIC ENERGY AGENCY

Technical Committee Meeting
on

METHODS AND CODES FOR CALCULATIONS OF THERMOHYDRAULIC PARAMETERS FOR FUEL, ABSORBER PINS AND ASSEMBLIES OF LMFRs WITH TRADITIONAL AND BURNER CORES

from 27 to 31 July 1998
Obninsk, Russian Federation

To be sent to the competent official authority
(i.e. Ministry of Foreign Affairs or to the National Atomic Energy Authority)
for transmission to: Mr. A. Rienjski, Division of Nuclear Power,
the International Atomic Energy Agency
P.O.Box 100, Vienna International Centre, A-1400 Vienna, Austria
Fax: ++43 1 20607
with copy to the Technical Coordinator, Mr. Y. Yarovytsin

PARTICIPATION FORM

Family name:	All initials of given names:	Mr. Ms.
Institution:	Full address:	
Position/title:	For urgent communications please indicate: Telephone No.: Telex No.: Telefax No.: Cable address: E-mail:	
Nationality:	Designating government or organization:	
Mailing address (if different from address of institution):		
Date of departure from mailing address:		

Do you intend to present a paper? YES ☐ NO ☐

Title of paper
(A one page abstract (400 words) of each paper offered should be attached to this form before forwarding):

FORM B

INTERNATIONAL ATOMIC ENERGY AGENCY

TCM on

METHODS AND CODES FOR CALCULATIONS OF THERMOHYDRAULIC PARAMETERS FOR FUEL, ABSORBER PINS AND ASSEMBLIES OF LMFRs WITH TRADITIONAL AND BURNER CORES

Obninsk, Russia
27-31 July 1998

ACCOMMODATION BOOKING

(To be sent before 20 June 1998)

Mr. V. Yarovytsin
International Division of the Institute of
Physics and Power Enginnering (IPPE)
Bondarenko Sqr. 1
249020 Obninsk, Kalaluga Region
RUSSIAN FEDERATION
Tel: 007 08439 98815
Fax: 095 8833 112 or 095 239 23 26

Name:_____ Initials:_____

Institute/Organization/Company:

_____

Full Address:

_____
_____
_____

Telex:_____ Tel:_____

Fax: _____ E-mail:_____

Please reserve single room:

Arrival date in Obninsk:_____ Departure date from Obninsk:_____

Flight no:_____ Flight no:_____

Arrival time:_____ Departure time:_____

Date:_____ Signature:_____

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Annex A

LIST OF MEMBER STATES AND INTERNATIONAL ORGANIZATIONS TO BE INVITED TO THE

Technical Committee Meeting (TCM) on "Methods and Codes for Calculation of Thermohydraulic Parameters for Fuel, Absorber Pins and Assemblies of LMFRs with Traditional and Burner Cores"

Obninsk, Russian Federation
27-31 July 1998

MEMBER STATES

Brazil

China

France

Germany

India

Italy

Japan

Republic of Kazakhstan

Republic of Korea

Russian Federation

Switzerland

United Kingdom

United States of America

INTERNATIONAL ORGANIZATIONS

European Commission (EC)

OECD Nuclear Energy Agency

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INTERNATIONAL ATOMIC ENERGY AGENCY

Advisory Group Meeting
on
Propulsion Reactor Technologies for
Civilian Applications
Obninsk, Russia

20 - 24 July 1998

Organized in cooperation with the Institute of Physics and Power Engineering (IPPE), MINATOM, Russia

INFORMATION SHEET

AND CALL FOR PAPERS

1. INTRODUCTION

The Advisory Group Meeting (AGM) on "Propulsion Reactor Technology for Civilian Applications" will be held in Obninsk, Russian Federation, from 20 - 24 July 1998.

There is an increasing interest, especially from developing countries, in the deployment of small and medium sized power reactors. This was confirmed in the last General Conference which resulted in a resolution requesting the Agency to assign appropriate priority to the development of the small and medium sized reactor programmes.

2. PURPOSE AND SCOPE OF THE MEETING

The purpose of this AGM is to provide a forum for Member States to advise the Agency on the different types of water and liquid metal-cooled ship propulsion reactors, barge mounted reactors, power reactors which do not require on-site refuelling, and other similar reactor types presently in existance or under consideration. The purpose of the meeting is to also receive advice from Member States on their needs and interests for development so that the Agency can support them within the small and medium reactor (SMR) programme.

The AGM will provide an opportunity to review the following general topics:

- operational experience on differnt types of ship propulsion nuclear plant, lessons learned from experience;
- reactor block systems and components (overall design, core design, main heat transport systems, safety systems, instrumentation and control);
- status of SMR concepts which do not require on-site refuelling;
- development in advanced ship propulsion nuclear reactors and their general design features, anticipated performance, cost estimates;
- status of R & D programmes, findings and conclusions, future R&D needs and trends;
- prospects of civilian applications of ship propulsion reactors;
- barge mountaned propulsion power reactors;

- phases and possible executors for propulsion nuclear power plant development and deployment in developing countries: design development - construction - owenership - operation - decommissioning-radioactive waste disposal;

- needs and interest from developing countries in the deployment of the above-mentioned reactor systems;

- the role of the Agency in this programme.

3. REQUIREMENTS FOR ABSTRACTS, PAPERS AND PRESENTATIONS

Each national delegation is required to send abstracts of not more than 400 words of all the papers to the address shown in item 6 by 5 June1998 and bring 25 copies of each paper to be distributed at the beginning of the meeting. The abstracts will be used to organize the technical sessions; therefore, they should contain sufficient detail to allow' the organizers to finalize the meeting programme.

Each paper must begin with an abstract, typed on one side in single-space with a column width not exceeding 160 x 250 mm, and should be prepared in form for publication in proceedings. At least one original or camera-ready copy of each paper should be given to Mr. Anatoli Rinejski at the meeting.

Two overhead projectors will be available at the meeting for the oral presentations of papers. The use of other standard equipment, such as a 35 mm slide projector, a 16 mm film projector and a video film projector, can also be arranged; such requests should be made when the abstract is sent. The time allocated to each presentation and the final programme of the meeting will be determined later.

4. SECRETARIAT

The Scientific Secretary of the meeting is Mr. A. Rinejski of the Division of Nuclear Power (NENP), Nuclear Power Technology Development Section (NPTDS). Mr. V. Yarovytsin of the IPPE International Division is the local co-ordinator and Mr. A. Zabudko of the IPPE is the Chairman of the meeting.

5. PARTICIPATION

Each participant should complete the attached forms A (participation form) and B (accommodation booking) and send them to the competent official authority (Ministry of Foreign Affairs or National Atomic Energy Authority) for transmission to the InternationalAtomic Energy Agency and IPPE before 20 June 1998:

Address:

Mr. A. Rinejski
Scientific Secretary, IWGFR,
IAEA, NEPF-NPTDS
A2575, Wagramerstrasse 5
P.O. Box 100, A-1400 Vienna
Tel: 43 1 2060 ext. 22812
Fax: 43 1 20607 or
2060 29598
e-mail: a.rineiskii@iaea.org

Mr. V. Yarovytsin
International Division of the Institute of
Physics and Power Engineering
Bondarenko square, 1
Obninsk, Kaluga Region,
249020 RUSSIA Tel: 007 08439 98815
Fax: 007 095 8833112

The designation of a participant will be considered for acceptance only if forwarded by the Government of an IAEA Member State or International Organization invited to participate.

6. WORKING LANGUAGE

The working language of the meeting will be English. All communications, abstracts and papers must be in this language.

7. VISAS

Designated participants who need a visa for entering Russia should submit the necessary application together with the invitation that they will have received from Russia to the nearest diplomatic or consular representative of the Russian Federation as soon as possible.

For the issuance of invitation letters from Russia in time the Organizing Committee must have the following information from all specialists and their accompanying persons, arriving at the Meeting from outside Russia:

- Full name,
- Date and place of birth,
- Passport number (issuance and expire data), place of work, position or title.

This information should be sent to the Organizing Committee, Mr. Yarovytsin, (address see item 5) before 20 June 1998. Mr. Yarovytsin will then send an official invitation to the participants to visit Russia.

8. EXPENDITURES

No registration fee is charged to participants. Travel and subsistence expenses assistance may be offered to a limited number of participants developing countries provided they make an important contribution.

Please note that compensation is not payable by the Agency for any damage to or loss of the expert's personal property. However, for the period of the experts' engagement with the Agency including travel between their residence and the duty station, they 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 anf for medical expenses up to an amount of US$ 10,000 in case of illness, injury or death attributable to their relationship with the Agency, subject to the terms of the insurance policy.

9. LOCAL ARRANGEMENT

An optional technical tour of IPPE will be arranged, following the meeting.

Any correspondence concerning technical matters, including abstracts and papers should be sent to Dr. A. Zabudko, Chairman of the AGM.

Dr. A. Zabudko
Director of Department
Federal Scientific Center of Russian Federation
Institute of Physics and Power Engineering (IPPE)
Bonarenko Square 1
249020 Obnins, Kaluga Region
Russian Federation
Fax: +7-095-2302326 or +7-095-8833112

and copy to Mr. A. Rinejsi (address see item 5)

FORM A

INTERNATIONAL ATOMIC ENERGY AGENCY

AGM on
Propulsion Reactor Technologies for
Civilian Applications
Obninsk, 20 - 24 July 1998

To be sent before 20 June *1998 to the competent official authority (Ministry of Foreign Affairs or National Atomic Energy Authority) for transmission to: Mr. A. Rinejski, Division of Nuclear Power, the International Atomic Energy Agency, P.O. Box 100, Vienna International Centre, A-1400 Vienna, Austria, Fax: +43 1 20607 with copy to Technical Coordinator Mr. Y. Yarovytsin.*

PARTICIPATION FORM

Family name: All initials of given names: Mr./Ms.

Institution: Full address:

Position/title: Tel:
Fax:
E-mail:

Nationality: Designating government or organization:

Mailing address (if different from address of institution):

Date of departure
from mailing address:

Do you wish to present a paper? YES ___ NO ___
Title of paper
(A one page abstract (400 words) of each paper offered should be attached to this form before forwarding):

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1998-02-05

FORM B

INTERNATIONAL ATOMIC ENERGY AGENCY

AGM on
Propulsion Reactor Technologies for
Civilian Applications
Obninsk, 20 - 24 July 1998

ACCOMMODATION BOOKING

(to be sent before 20 June 1998)

Mr. V. Yarovytsin
International Division of the Institute of
Physics and Power Engineering
Bondarenko Sq. 1
249020 Obninsk, Kaluga Region
RUSSIAN FEDERATION
Tel: 007-08439 98815
Fax: 007 095 8833112

Name: _____ Initials: _____

Institute/Organization/Company:

_____

Full address:

_____

_____

_____

Telex: _____ Tel: _____

Fax: _____ E-mail: _____

Please reserve single room:

Arrival date in Obninsk: _____ Departure date from Obninsk: _____

Flight no: _____ Flight no: _____

Arrival time: _____ Departure time: _____

Date: _____ Signature: _____

INTERNATIONAL WORKING GROUP ON FAST REACTORS (IWGFR)
Technical Committee Meeting - 31st Annual Meeting
of the Working Group on Fast Reactors
IAEA, Vienna, 12-14 May 1998

List of Proposed Topics for the IWGFR Meeting

No:	*Title*	*Country*	*Proposed for consideration at the IWGFR meeting in:*	*Remarks*
1.	Experimental fast reactors: experience, main scientific and engineering results, new projects	China	1993	
2.	Impurities mass behaviour in LMFR sodium coolant circuits (primary and secondary) and its possible influence on the reactor safety	Russia	1994	
3.	Neutron flax prediction and measurement in LMFR peripheral location	India	1998	
4.	Evaluation of core physics tests of power reactors	Japan	1998	Proposed to be held in Tsuruga, Japan
5.	AGM to organize a Symposium on MOX fuel cycle technologies for medium and long-term deployment: experience, advances and trends (1999, jointly with NFCMS)	IAEA	1998	

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1. 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 got negative interferences in some developed countries due to the fact that: 1) at the moment and in the near future, they do not urgently need more electricity; 2) they are facing strong anti-nuclear actions; and 3) they have enough money to buy or import oil for electricity production. Developing countries, which are almost not using oil, they should develop FBRs at a higher economic level, as of their limited petroleum resources. So it is gradually becoming more interesting to develop FBRs in developing countries.

Objectives:

a) Review and exchange the national FBR development strategy and projects.
b) Exchange operating experiences and main scientific and engineering results on experimental FRs
c) Review experiences on enhancing experimental FBR's safety
d) Exchange experiences on multi-utilization of experimental FRs.

2. Impurities Mass Behaviour in Fast Reactors Sodium Coolant Circuits (Primary and Secondary) and its Possible Influence on the Reactor Safety, Russia

In the frame of the meeting, the following items can be considered:

- Impurities sources in FRs sodium coolant (including accidental ones such as oil ingress, air penetration, etc.);
- Distribution of impurities between the different zones of the 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 their localization and influence of the 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.

3. Neutron Flux Prediction and Measurement in LMFR Peripheral Locations, India

The accurate prediction and measurement of neutron flux and activation and radiation damage in peripheral components of fast reactors is very important. Problems of component maintenance, life extension and decommissioning depend on these parameters. Typical components of concern are: grid plate, core support, reactor vessel, inner vessel, upper internals, IHX primary pumps, etc. The flux values at expile detector locations are important for proper design of monitoring instrumentation. The

calculation of deep penetration flux requires challenging methods of analysis involving multidimensional neutron transport and suitable multi group cross section libraries. A meeting is proposed to discuss the status of prediction and comparison with measurements in FBRs.

4. Evaluation of Core Physics Tests of Power Reactors, Japan

Evaluation of core physics tests has been conducted in each country, based on the data obtained from each real reactor, such as Phenix, Super Phenix, FFTF, EBR-II, PFR, BR-10, BOR-60, BN-350, BN-600, etc. It is beneficial to summarize those data and investigate the coincidence of the results for the estimation of future core design and characteristics. Major candidate issues to be discussed are as follows:

a) Criticality and control rod reactivity worth
b) Reaction rate, flux-power distribution and breeding ratio
c) Sodium void worth and material worth
d) Power coefficient and burn-up reactivity swing.

SELECTION OF MEETINGS for 1998

SELECTION OF 1 TOPIC WITH PRIORITY (the highest priority 5 points)

TOPICS FOR FURTHER SELECTION

Country	Brazil	China	France	Germany	India	Italy	Japan	Kazakhstan	Korea, Republic of	Russia	Switzerland	UK	USA		Priority
1															
2															
3															
4															
5															

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NEW CO-ORDINATED RESEARCH PROJECT (CRP)
PROPOSED FOR INCLUSION IN THE 1999-2000 PROGRAMME AND BUDGET

Project A2.03 Liquid Metal Cooled Reactors, Task No. 4

Title of CRP: Updated Codes and Methods to Reduce the Calculative Uncertainties of the LMFR Reactivity Effects

Specific objective: This CRP will provide the basis to quantify and decrease the uncertainty in the Doppler, sodium void and other reactivity coefficients used in LMFR safety analysis by verification and improvement of calculation codes and methods using analytical and experimental data.

Nuclear component: The LMFR is a proven technology capable of providing nearly unlimited energy supply from the world's ample resources of depleted uranium, low-grade natural uranium and thorium. Significant development efforts on LMFR have made France, Russia, Japan and India. Fast reactor programmes have recently been initiated in China, Republic of Korea and Brazil. CSIs would pool scientific and technical information for the common availability to all countries.

Relationship to previous and ongoing CRPs: The activities and technical documentation from the previous CRP on *Intercomparison of LMFR Core Mechanics Codes* (1986-1990) and the IAEA/EC benchmark programme on *Evaluation of Benchmark Calculations on FR Core with Near-Zero Sodium Void Effect* (IAEA-TECDOC 731) would be made available to the CSIs of this new CRP in support of their activities.

Duration: 1999-2003

Number of contracts: ~3

Number of Agreements: ~5

Financial resources: Cost of contracts:
Cost of RCMs:
Other:

Source of funds: Regular budget

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NEW CO-ORDINATED RESEARCH PROJECT (CRP)
PROPOSED FOR INCLUSION IN THE 1999-2000 PROGRAMME AND BUDGET

Project A2.03 Liquid Metal Cooled Reactors, Task No. 2

Title of CRP: Power Reactor Option for Effective Incineration of Radioactive Waste

Specific objective: It is estimated that by the year 2010, there will be more than 300,000 tons of spent fuel (including 3,000 tons of plutonium, 100 tons of ^{237}Np and of the americium amounts depend on the individual decay times). There will also be about 250 tons of ^{99}Tc, 90 tons of ^{135}Cs and 60 tons of ^{129}I. The extent to which these quantities can be reduced by incineration and transmutation in reactors or accelerator-driven systems is being widely studied. The objective of this project will be to clarify the scientific issues related to the various options for effective transmutation and incineration of transuranics and radioactive waste.

Nuclear component: The recycling of both the actinide isotopes and selected fission products in fast reactor in which all transuranics are fissionable in the fast neutron spectrum, is a feasible objective. Consideration will also be given to Accelerator Driven System (ADS), in which a higher number of neutrons per fission (from fission plus spallation) could improve the efficiency of waste incineration, if spallative radioactive products pose less of a problem. Molten Salt Reactor (MSR) systems have some important features from the point of incineration of transuranics.

Relationship to previous and ongoing CRPs: The results from an ongoing CRP on *Use of Th-based Fuel Cycle in ADS to Incinerate Pu and to Reduce Long-term Waste Toxicities* (I3.30.08) and within the framework of project A2.03, investigations (IAEA-TECDOC 693, Use of Fast Reactors for Actinide Transmutation) would be made available to CSIs of this new CRP in support of their activities.

Duration: 2000-2003

Number of contracts: ~3

Number of Agreements: ~5

Financial resources: Cost of contracts:
Cost of RCMs:
Total cost of RCMs:
Other:

Source of funds: Regular budget

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Project A.2.03. Liquid Metal Reactors

	Tasks planned for 1999-2000	Related Actions (Meetings, etc.)	Completion
1.	Prepare a Symposium on MOX Fuel Cycle Technologies for Medium and Long-term Deployment: Experience, Advances and Trends (1999), jointly with NFCMS	AGM 99 SYM 99	1999
2.	Prepare a TECDOC on Advanced Power Reactor Options for Effective Incineration of Radioactive Waste (intercomparison of LMFR, ADS and Molten Salt Reactor Systems)	CRP2000-2003 RCM2000	2004
3.	Prepare a TECDOC on Design and Performance of Reactor and Subcritical blanket Systems with Lead and Lead-Bismuth as Coolant and/or Target Maerial *	AGM2000	2000
4.	Prepare a TECDOC on Updated Codes and Methods to Reduce the Calcualational Uncertainty of the LMFR Reactivitiy Effects	CRP 99-2003 RCM 99, 2000	2003
5.	Prepare a TECDOC on Coupling of LMFR's with Heat Application Processes *	CS 99	2001
6.	Prepare a progress report on LMFR Developments (IWG)	TCM 99, 2000	2000
7.	Prepare a TECDOC on Intercomparison of Analysis Methods for Predicting the Behaviour of Seismically Isolated Nuclear Structures (I3.20.02) (in co-operation with NSNI)	CRP 96-99 RCM 99	2000
8.	Prepare a technical report on Experience with LMFRs * (in co-ordination with NSNI)	CS 2000	2001

** Final titles to be recommended by IWGFR*

TERMS OF REFERENCE

INTERNATIONAL WORKING GROUP ON FAST REACTORS
(IWGFR)

1. The International Working Group on Fast Reactors will be a continuity working group within the framework of the International Atomic Energy Agency.

2. In appointing the members of the IWGFR, the Director General will be guided by the following considerations:

   a. Each member will be a scientist or expert actively working in the field of fast reactors or having broad responsibilities for programmes in this field.

   b. The Working Group will include one member from each Member State that has a major programme in the field of fast reactor

   c. Each member will be appointed by the Director General after consultation with the member's Government.

   d. Each member may be acoompanied by advisers or specialists.

   e. The Director General may from time to time co-opt members and invite observers from other Member States on an ad hoc or continuing basis.

   f. As a means of ensuring continuity, members from States having major programmes in the field of fast reactors would normally serve on the Working Group for a period of not less than three years.

3. Functions

   The main functions of the working Group will be:

   a. To provide to the Director General of the IAEA advice and opinions on the Agency's activities in this field.

   b. To assist in the coordination of conferences in the field of fast reactors and of proposals of topics for specialist meetings and topical conferences in this field.

   c. In particular to advise the Director General of the IAEA on proposals for reviews and studies in this field.

   d. To serve as a means of exchanging information on the progress of national programmes in this field.

   e. To review the terms of reference after a period of three years and advise the Director General of the IAEA on any desirable modifications.

4. Methods of Work

Except as otherwise herein stated, the Working Group will determine its own methods of work, including preparation of its Agenda, establishment of special groups, keeping of records and other procedures.

The chairmanship of the Working Group will be rotated amongst the members.

5. Secretariat

Subject to the availability of funds, the Director General will provide the administrative and secretariat services required by the Working Group including interpretation and translation services, meeting facilities, maintenance of records and distribution of documents. The IAEA would provide the services of a scientific secretary.

6. Meetings

The IWGFR would normally meet once a year. It may be desirable for the IWGFR to meet from time to time away from the IAEA headquarters to familiarize itself with fast reactor activities in particular IAEA Member States. Special arrangements will be made to provide secretariat services for such meetings in co-operation with the host country.

7. Relations with other Organizations

The Director General may invite international or regional organizations which have programmes in the fast reactor field to be represented at meetings of the IWGFR or at particular sessions during such meetings.

I W G F R - TERMS OF REFERENCE

DISTRIBUTION OF REPORTS

IWGFR/17

Appendix Preparation and Distribution of Minutes and Reports.

a. To issue the Summary Report of the Annual Group Meetings in three parts: Part I - Minutes of the Meeting, Part II - National Programmes in the Field of Fast Reactors, Part III - Discussions of the National Programmes.

b. The Summary Reports of Annual Group Meetings should be restricted to member organisations of the IWGFR and should not be, in principle, generally available because of their narrow professional contents and the absence of special treatment of the appropriate perception by the general public. The Agency should reproduce and distribute the report among the IWGFR membership and among the specialists interested in it. The members of the IWGFR can distribute such reports among various interested organisations of their respective countries, and the representatives of the international organisations can distribute these reports among the Member States of their respective international organisations.

c. The Summary Reports of Specialists' Meetings (Technical Committee Meetings) should be distributed among the same organisations described in (b) above.

d. The distribution of the Summary Reports of Annual Group Meetings and Specialists' Meetings(TCMs) was agreed to as 25 copies to each member of the IWGFR, 3 copies to each representative of international organisations to the IWGFR and one copy to each alternate and to each Mission of the IWGFR member-country.

e. The US member requested to have five copies of each summary report sent directly to him by airmail plus 20 copies transmitted to him via the US Mission

The Status of Fast Reactor Technology Development in China

For The Thirty First Annual Meeting of
the International Working Group
on Fast Reactors. IAEA

by XU MI
China Institute of Atomic Energy
May 12-14, 1998, Beijing, China

THE STATUS OF FAST REACTOR TECHNOLOGY DEVELOPMENT IN CHINA

Abstracts

The paper describes recent status and trends on Chinese national economy, electrical power capacity and nuclear power development.

The preliminary design of the CEFR has been approved by the State Science and Technology Commission. Now it is in the detail design stage. It is planned that the first pot of concrete will be in April of 1999, in the end of 2000 the reactor building construction will be finished and the first criticality of the reactor will be envisaged in July 2003.

The brief of preliminary design, analysis results of some beyond design basic accidents and design basic accidents, CEFR research works, and international cooperation are presented in the paper.

1. INTRODUCTION

China, as a developing Country, has been keeping her economy growth trends for more than ten years. From 1992 to 1996 the average annual increasing rate of Gross National Products was 12%. In 1996 and 1997 the annual increasing of GNP were 9.7% and 8.8%[1] respectively. The goal of national economy development planned by Chinese Government is to maintain the annual growth rate at 8 ~ 9% before the year 2000[2]. Correspondingly, the electricity generation capacity also got a rather quick development and the development trends have been envisaged as shown in Table 1[1,2]

Table 1 CHINA's POWER GENERATION AND TRENDS

year	Total capacity (GWe)	Total production (TWh)
1993	183	836
1994	199	905
1995	210	990
1996	225	
1997	250	
2000	300	1400
2010	590	2700
2020	800	3700

But, 800 GWe by 2020 which means that the power installation capacity and power production per capita only about 0.5kW and 2500kWh, roughly equal to the world's current average level.

In this country, three PWRs are in operation, four PWRs in construction and four nuclear power reactors are planned. Their main properties and schedule are presented in Table 2. And Table 3 shows their load factors recently.

Chinese Experimental Fast Reactor (CEFR)with the thermal power 65MW is going along. Its preliminary design has been reviewed and agreed by the experts organized by China National Nuclear Corporation in August, 1997, and then approved by the State Science and Technology Commission in November. Now the detail design of CEFR has been started. The additional site land about 10 ha has been bought from the local government. It is planned to have first pot of concrete in April next year. The construction of reactor building will be completed in the end of 2000. And its first criticality is envisaged in July 2003.

Table 2 MAIN PROPERTIES AND SCHEDULE OF CHINA's NUCLEAR POWER REACTORS

Reactor	Type	Capacity,(MWe)		Operator	NSSS Supplier	Construction Start	First Criticality	Grid Connection	Commer. Opertion
		Net	Gross						
in Operation									
Qinshan-1	PWR	279	300	QNPC	CNNC	1985.03.20	1991.10.31	1991.12.15	1994.04.01
Daya Bay-1	PWR	944	984	GNPJVC	FRAM	1987.08.07	1993.07.28	1993.08.31	1994.02.01
Daya Bay-2	PWR	944	984	GNPJVC	FRAM	1988.04.07	1994.01.21	1994.02.07	1994.05.07
Under construction									
Qinshan2-1	PWR	610	642	NPQJVC	CNNC	1996.06.02	2000.11	2002.2	2002.6
Qinshan2-2	PWR	610	642	NPQJVC	CNNC	1997.04.01	2002.9	2002.12	2003.4
Lingao-1	PWR	935	984.7	LANPC	FRAM	1997.05.15	2002.2	2002.4	2002.7
Lingao-2	PWR	935	984.7	LANPC	FRAM	1997.11.28	2002.10	2002.12	2003.3
Planned									
Qinshan3-1	PHWR	664.9	728	TQNPC	AECL	1998.6	2002.10	2002.10	2003.2
Qinshan3-2	PHWR	664.9	728	TQNPC	AECL	1998.12	2003.7	2003.7	2003.11
Lianyungang-1	PWR	999.5	1060	JNPC	AEE&ZAES	1999	2003	2003	2004
Lianyungang-2	PWR	999.5	1060	JNPC	AEE&ZAES	2000	2004	2004	2005

Table 3 LOAD FACTOR OF CHINA's NPPs %

Reactor	Power	1994	1995	1996	1997
Qinshan-1	300	68	84	84.7	76.55
Daya Bay-1	900		45.22	76.1	75.31
Daya Bay-2	900		77.92	64.1	68.62

2. CEFR DESIGN

After the collection and preparation of necessary computer codes and the decision of main technical selections, the conceptual design of the CEFR was started in 1990 and completed in 1993 including the confirmation and optimization to some important design characteristics. Having spent almost whole 1994 for its preparation, the CEFR preliminary design was started in the early of 1995 and finished in August 1997. After about another half year for its necessary modification, the detail design is now just started.

Following presentation to this reactor is based on its preliminary design.

2.1 Introduction

The CEFR is a sodium cooled experimental fast reactor with$(Pu,U)O_2$ as fuel, but UO_2 as first loading, bottom supported pool type, two main pumps and two loops for primary and secondary circuit respectively. The water-steam tertiary circuit is also two loops but the superheat steam is incorporated into one pipe which is connected with a turbine. The following are General Engineering Characteristics:

Thermal Power	65 MW
Electrical Power(T.G.Capacity)	25 MW
Electrical Power(net)	20 MW
Acreage of Reactor Site	15 ha
Floor Space of Buildings	43731 m^2
First Loading (UO_2 64.4%)	417 kg
Water Supply	4500 t/d
Power Supply	3000 kW

2.2 Reactor Core

The reactor core, as shown in Fig.1, is composed of 81 fuel subassemblies(Fig.2), 3 safety subassemblies, 3 compensation subassemblies and 2 regulation subassemblies, then 336 stainless steel reflector subassemblies and 230 shielding subassemblies and in addition 56 positions for primary storage of spent fuel subassemblies are included. 92%B-10 enrichment B_4C is used for safety and compensation absorber. but natural boron B_4C (B-10 enrichment about 20%) for regulation and shielding subassemblies.

The compensation and regulation subassemblies are also as first shutdown system.

The design limitation to the core which was fixed before the core design is as following:

Thermal Power	65 MW
Linear Power	430 W/cm
Operation Cycle(min.)	73 d

Inserting Rate of Reactivity(max.)	0.07	β/s
Positive Reactivity for Each Inserting(max.)	0.4	β
Reactivity Equivalent of Each Regulation Subassembly(max.)	1	β
Cladding Temperature in Nominal Operation (max.)	700	℃

Some results about core neutronics and thermohydraulics are listed in Tables 4-7.

Table 4 CEFR FUEL LOADING (EQUILIBRIUM)

Fuel		$(Pu,U)O_2$	UO_2
PuO_2	wt%	29.4	
U-235 enrichment	%	36	64.4
U-235	kg	92.33	236.6
Pu-239	kg	65.76	
Total Pu	kg	141	

Table 5 MAXIMUM HEAT RATE*

		Cycle	
		Beginning	End
Power of Fuel Subassembly	(MW)	0.980/0.957	0.978/0.952
Linear of Fuel Pin	(W/cm)	405/395	396/388
Volume Power Density	(MW/m^3)	756/738	744/725

*:$(Pu,U)O_2/UO_2$

Table 6 MAXIMUM NEUTRON FLUX RATE*(10^{15} $n/cm^2 \cdot s$)

	Cycle	
	Beginning	End
Center (steel)	3.73/3.13	3.76/3.15
First Row of Fuel S.A**	3.70/3.10	3.73/3.13
First Row of Reflector S.A	2.20/1.90	2.26/1.92
First Row of Shielding S.A	0.293/0.261	0.298/0.264
Storage Position	0.0269/0.0252	0.271/0.0254

*:$(Pu,U)O_2/UO_2$

**:S.A-Abbreviation Subassembly

Table 7 CLADDING AND FUEL TEMPERATURE* max.℃

Flow Region	Cladding**		Fuel***	
	Cycle		Cycle	
	Beginning	End	Beginning	End
1 (15 Fuel S.As)	665.68	670.44	2511.36	2497.30
2 (18 Fuel S.As)	665.02	670.76	2373.06	2370.10
3 (21 Fuel S.As)	659.68	662.24	2219.16	2213.50
4 (27 Fuel S.As)	656.66	656.68	2039.46	2030.60

* $(Pu,U)O_2$

** Including 2σ deviation of average square root related to the nominal value.

*** Using 3σ for uncertainty of design parameters.

2.3 Reactor Block

The CEFR block is composed of main vessel and guard vessel which is supported from bottom on the floor of reactor pit, reactor core and its support structure, internal structures on which main pumps, intermediate heat exchangers and DHRS heat exchangers are supported and reactor top roof and double rotation plugs on which control rod driving mechanisms, fuel handling machine and some instrumentation structures are supported. The CEFR block is shown in Fig 3.

The main vessel has an outside diameter of 8010mm and a diameter gap of 175mm toward guard vessel. The narrow gap design permits the core still immersed in sodium when the main vessel leak accident has unexpected happened. Using 2D Sn computer code, the calculation shows that the neutron fluence during 30 years is about 1.0×10^{22} n/cm^2. The anti-seismic analysis is under going. For main vessel only temperature, strain and sodium leak detection are considered.

In the design a core catcher is equipped even though there is no any serious accident in which the large part of core damage could be happened based on the analysis of all the beyond design basic accidents.

2.4 Main Heat Transport System

The primary circuit is composed of main pumps, four intermediate heat exchangers, reactor core support diagrid plenum, pipes and cold and hot sodium pools. In cold pool, two primary loops are separated each other, but in hot pool they are linked up. In normal operation the average sodium temperature in cold pool is 360 ℃ and in hot pool it is 516 ℃.

The secondary circuit has two loops each one is equipped with one secondary pump, two intermediate exchangers(IHX), one evaporator, expansion tank, one superheater and valves. The outlet sodium temperature of secondary circuit from IHX is 495 ℃. When it leave evaporator it will decrease up to 310 ℃, and in outlet of superheater it is 463.3 ℃.

The tertiary water steam circuit has one turbine gererator, three low pressure heaters, one deoxygen heater, one demineralization facility and feed water pumps.

Each evaporator is connected with one release-to-air valve and two safety release valves are equipped, but for superheater, one release-to-air valve and one safety release valve. And the by-pass de-temperature de-pressure valves are equipped for discharging the steam to the condenser when the turbine is not in operation.

2.5 Safety Characteristics

2.5.1 Active Safety Measure

The CEFR is equipped with 8 control subassemblies of which 3 for safety , 3 for compensation and 2 for regulation. Their reactivity values are given in Table 8.

Table 9 and 10 gives the reactivity balances of the first and second shut down system. They all could meet the regulation of the nuclear safety criteria even considering the control rod stuck criteria.

Table 8 CONTROL SUBASSEMBLY REACTIVITY VALUE
$(Pu,U)O_2$ core %ΔK/K

Shut Down System	Control Subassembly	Material	Cycle	
			Beginning	End
first	Comp.-1	B_4C Enriched	1.85	1.84
	Comp.-2		1.82	1.81
	Comp.-3		1.82	1.81
	Regu.-1	B_4C Natural	0.14	0.13
	Regu.-2		0.14	0.13
	3 comp.s		5.59	5.63
	Total		5.83	5.84
	Total except comp.1		3.91	4.00
second	SA-1	B_4C Enriched	0.98	0.99
	SA-2		0.99	1.00
	SA-3		0.98	0.99
	Total		3.10	3.13
	2 SA s		2.01	2.03
Total	8 Subassemblies		9.13	9.25
	7 Except comp.1		7.19	7.18

Table 9 REACIVITY BALANCE OF FIRST SHUT DOWN SYSTEM %ΔK/K

	Reactivity
Maximum excess reactivity to be compensated	2.81
Control subassemblies value	
Total	5.83
Total except comp.-1 (In real position)	4.39
Subcritical value at 250 ℃	
Total action	3.03
Total except comp.-1	1.58

The decay heat will be removed by main heat transport system when the reactor is shut-down, but the flow rate of the main pump will be changed to 15% which power is supplied by power grid, stand-by grid and emergency power.

Table 10 REACTIVITY BALANCE OF SECOND SHUT DOWN SYSTEM
%ΔK/K

	Reactivity
Temperature-power effect to be compensated	1.03
Control subassemblies value	
Total	3.10
Total except SA-1	2.01
Subcriticality	
Total action	2.07
Total except SA-1	0.98

2.5.2 *Passive Safety Properties*

It was intended at early stage to design the CEFR having a perfect passive safety property.

Firstly, it has rather deep negative feedbacks when the temperature or the power is increasing. These feedback properties are shown in Tables 11-14.

Table 11 TEMPERATURE REACTIVITY EFFECT
$(Pu,U)O_2$ CORE, 250-360 ℃, %ΔK/K

	Cycle	
	Beginning	End
Sodium Density	-0.169	-0.170
Axial Expansion of Reactor Core	-0.037	-0.038
Axial Expansion of Lateral Reflector	-0.019	-0.020
Radial Expansion (diagrid plenum)	-0.204	-0.205
Doppler Effect	-0.032	-0.033
Total	-0.461	-0.466

Table 12 POWER REACTIVITY EFFECT, HOT STANDBY–FULL POWER
$(Pu,U)O_2$ core %ΔK/K

	Cycle	
	Beginning	End
Sodium Density	-0.107	-0.112
Sodium Volume Fraction Exchange	-0.029	-0.034
Axial Expansion of Reactor Core	-0.339	-0.311
Axial Expansion of Lateral Reflector	-0.011	-0.011
Radial Expansion (S.A. Winding)	-0.02	-0.02
Doppler Effect	-0.063	-0.063
Total	-0.569	-0.556

Table 13 DOPPLER CONSTANT OF CEFR CORE
$(Pu,U)O_2$ 10^{-3}ΔK/K

	Kw(with Na)	Kd(without Na)
Fuel:633-1593K Cladding: 633-1000K	-0.878	-0.618
All Materials 523-633K	-1.574	-1.048
All Materials 373-523K	-1774	-1.142

Table 14 REACTIVITY EFFECT OF SODIUM LOST
$(Pu,U)O_2$ CORE %ΔK/K

Region Lost Sodium	Cycle	
	Beginning	End
Core (Fuel Section)	-2.366	-2.404
Core (Fuel section+upper section of S.A)	-3.067	-3.110
Core (Whole Section of S.A)	-3.737	-3.788
Whole Core (including central stainless steel rod, and stainless steel reflector)	-5.196	-5.194

Secondly, two independent passive residual heat removal systems as one of specific safety installations are designed for the reactor. Each one includes one Na-Na heat exchanger immersed in primary sodium and one sodium-air cooler with a high stack. The normal power of each system at its service condition is 0.525MWth, and at its standby condition the power lost will be 0.052MWth.

2.6 Main Design Parameters

Parameter	Unit	Conceptual Design	Preliminary design
Thermal Power	MW	65.5	65
Electric Power, net	MW	20	20
Reactor Core			
Height	cm	50	45.0
Diameter Equivalent	cm	58.5	60.0
Fuel		$(Pu,U)O_2$	$(Pu,U)O_2$
Pu,total	kg	121.6	141
Pu-239	kg	93.2	65.76
U-235(enrichment)	kg	97.6(30%)	92.33(36%)
Linear Power max.	W/cm	430	430
Neutron Flux	$n/cm^2 \cdot s$	2.97×10^{15}	3.7×10^{15}
Bum-up, target max.	MWd/t	100000	100000
Bum-up, first load max.	MWd/t	50000	60000
Inlet Temp. of the Core	℃	360	360
Outlet Temp. of the Core	℃	530	530
Diameter of Main Vessel(outside)	m	8.0	8.010
Primary Circuit			
Number of Loops		2	2
Quantity of Sodium	t	~ 300	~ 300
Flow Rate, total	t/h	1400	1328.4
Number of IHX per loop		2	2
Secondary Circuit			
Number of loops		2	2
Quantity of Sodium	t	48.2	48.2
Flow Rate	t/h	986.4	986.4
Tertiary Circuit			
Steam Temperature	℃	480	480
Steam Pressure	MPa	10	14
Flow Rate	t/h	93.5	96.2
Plant Life	a	30	30

3. PRELIMINARY SAFETY ANALYSIS REPORT

For the application of the reactor construction, the preliminary safety analysis report has been almost completed. The RG1.70 of US-NRC is taken as reference for the compilation format and contents. The most important typical accidents are selected to analyze. Here Six Beyond design Basic Accidents(BDBA) and three Design Basic Accidents (DBA) are as example.

3.1 DBDA-1 Power lost and shut down signals failure

Conditions and initial case

(1) Reactor is in operation at 100% nominal power;

(2) Grid power and stand-by power lost;

(3) All shut-down signals failed;

(4) Primary and secondary pumps stopped in 40 and 60 seconds respectively after grid power lost;

(5) All windows of the air coolers are open in 10 minutes after grid power lost;

(6) Up to 2700 seconds after grid power lost, the reactor shut down manually by operator.

Results:

(1) Before reactor shut down manually, reactor power decreased by the negative feedback due to the temperature raising;

(2) The maximum core outlet temperature and maximum cladding temperature will reach 830 ℃ and 850 ℃ respectively during very short period;

(3) The integrity of fuel pins is kept.

3.2 BDBA-2 The guillotine or leakage of the primary sodium purification pipe happened and simultaneously the isolation valves failed.

Results: thanks to the special siphon automatic interference device, the leaked sodium less is than 60 l. But, if the leak from Φ2.5 mm hole happened and the siphon effect destroyed manually by operator after 7 hours since leak has happened, the maximum leak less than 1372 l.

For the second case the calculation shows that if extinguishing system failed and with following conditions, the released aerosol of sodium will be 150 kg.

Floor surface	115 m^2
Height of room	3 m
Flow rate of ventilation	3850 m^3/h.

3.3 BDBA-3

Grid power and stand-by power lost and all windows of the air cooler of DHRS can't be open.

Conditions and initial case:

(1) Reactor is in operation at 100% nominal power;

(2) Reliable power supply is O.K.;

(3) Shut down systems is O.K.;

(4) Primary and secondary pumps inert running in 40 and 60 seconds respectively after the power is lost;

(5) The air flow rate of the DHRS is only 10% of the design value..

Results:

Maximum outlet sodium temperature of the core will be less than 520 ℃.

3.4 BDBA-4 One regulation subassembly unexpected raised and shut down systems are effectiveless.

Conditions and initial case:

(1) Reactor is in operation at 100% nominal power;

(2) DHRS is in hot stand-by.

Results:

(1) The total reactivity inserted is 0.0014 ΔK/K;

(2) The maximum power is only 117% due to the negative feedback;

(3) The maximum temperature of sodium outlet of the core and cladding are 570 ℃ and 712 ℃ respectively. And maximum temperature of fuel pin center in 2500 ℃.

3.5 BDBA-5 The check valves in two primary pressure pipes are closed to the limited position(it still has 20 degree) simultaneously during 20 minutes.

Conditions and initial case:

(1) Reactor is in operation at 100% nominal power;

(2) DHRS is at hot stand-by;

(3) Shut down systems are OK.

Results:

The maximum temperature of the core outlet sodium and cladding are 570 ℃ and 710 ℃ respectively.

3.6 BDBA-6 Main vessel and guard vessel leaked successively

Conditions and initial case:

(1) The Reactor is in operation at 100% nominal power;

(2) The crack equivalent diameter is 3.2mm at main vessel, when 23m^3 sodium filled in the gap, the guard vessel started to leak (about 165 hours from main vessel leakage).

Results:

About 2000kg sodium aerosol will release to the environment.

3.7 DBA-1 Main vessel leaked

Conditions and initial case:

(1) Same as 3.6;

(2) The crack equivalent diameter is 3.2mm, 23m^3 sodium filled in the gap during 165 hours.

Results:

Radioactivity gas of 1×10^9Bq will release from 23m^3 sodium to the environment.

3.8 DBA-2 Primary sodium leaked at auxiliary pipe without protection sheath

Conditions and initial case:

It is assumed that there is a crack equivalent to 2.5mm diameter at the pipe. and the check valve is closed during 20 seconds in 10 minutes after leakage has happened.

Results:

The leakage quantity will be 40 kg on the floor in the room in which 2kg sodium will release to the environment as aerosol.

3.9 DBA-3 The cover gas of primary sodium leaked

Condition and initial case:

(1) The pressure of the cover gas is 0.05MPa;

(2) Reactor is in operation at 100% nominal power .

Results:

When the pressure of cover gas decreased to 0.04MPa. The accident signal will be appeared. Reactor will be shut-down manually. Totally 43kg Ar (20kg Ar from reactor and 23kg Ar from compensation tank) will be released to the environment.

3.10 Environment Effect Evaluation

The personal maximum dose equivalent values to the residents during abovementioned accidents have been calculated and results are given in Table 15.

Table 15 MAXIMUM DOSE EQUIVALENT VALUE mSv

no. of accidents	Grid lost	Shut down failed	DHRS failed	Dose calculated	Dose limitation
BDBA-1	v	v		0.12	5
BDBA-2	Purification pipe leaked and check valve failed,1372kg Na			0.0692	5
BDBA-3	v		v(10%kept)	0	5
BDBA-4	Regulation S.A. unexpected raised v			0	5
BDBA-5	Two check valves in primary circuit closed			0	5
BDBA-6	Main and guard vessels leaked successively			0.00414	5
DBA-1	Main vessel leaked			negligible	0.5
DBA-2	Purification pipe leaked and check valve O.K. ,40kg Na			0.0013	0.5
DBA-3	Cover gas leaked			negligible	0.5

4 RESEARCH WORK

4.1 Material

Based on the experience of 316(Ti) stainless steel trail production and tests started in the early of 70s 316(Ti) stainless steel which will be really used has been trial produced in the Shanghai Fifth Iron-Steel Plant 3 years ago. Its mechanics performance at room temperature is stable and its creep and endurance performance at high temperature is under carrying on.

4.2 Sodium test loops

Thanks to the ENEA, Italy the components of two large scale sodium loops — 110m^3/h ESPRESSO and 320m^3/h CEDI and other facilities ceded to our Institute as a gift of technical cooperation more than 4 years ago. The redesign and reconstruction of the two loops and other facilities-core subassembly dimension meter and washing plant were completed in June 1997. The cold adjustment of ESPRESSO and CEDI was finished in October, 1997 A new sodium purification facility which was designed to offer sodium in the fast reactor grad at 200kg per day is nearly completed.

It will be started, to clean these two loops with Argon and FR grad sodium from June, 1998. It is half year delayed to the schedule due to the offering strategy changement of FR grad sodium.

4.3 Design Demonstration

Following design demonstration tests have been or are going to be arranged:

(1) Demonstration test on water pool facility for 3-D thermohydraulic codes for DHR.

(2) Natural convection capability demonstration on CEFR pool water simulation mockup.

(3) CEFR core flow distribution demonstration .

(4) Materials and its welding zone corrosion test in sodium.

(5) Demonstration of extinguishing sodium fire.

(6) Leakage sodium receiver demonstration.

(7) Potassium and chlorine analysis methods demonstration as impurities in sodium.

(8) H_2-meter and other instrumentation demonstration.

(9) CEFR core nuclear design demonstration with zero power mockup.

Besides above demonstrations, a number of approving tests of components will be put in the duration of fabrication.

5 ENGNEERING PROGRESS

(1) The additional site land about 10 ha has been bought from the local government.

(2) The road out-site which is connected to the main road has been started modified.

(3) The ordering of the components of CEFR has been started.

6 INTENATIONAL COOPERATION

During past years, China has had her cooperation with some countries on the fast reactor technology fields.

As technical presents, the Italian Energy, New Technology and Environment Agency has ceded some research facilities and components including ESPRESSO and CEDI, sodium purification facilities, core subassembly dimension meter etc.

Under the framework of the reactor R&D cooperation between China National Nuclear Cooperation and Commissariat a l'Energie Atomic de France, we have good cooperation with French colleagues on fast reactor technology including computer codes cooperation, information exchange on sodium technology and on neutronic physics which are all going fluently.

The cooperation with Russian FBR Institutions have been conducted during past years including training CIAE' engineers, experts' lectures, CEFR design consultancy and zero power facility experiments etc. Some design concept and experiences for fast reactors is gained from this cooperation.

IAEA, as the International Agency stimulating peaceful use of atomic energy, has given much helps to China in the fields of FBR technology development including the informations and financial support to the participation of technical committee meetings.

Chine has a rather weak bases for the development of fast reactor technology, it is intended to have more cooperation with other countries to share the experiences.

7. CONCLUSION

The CEFR right now is at detail design stage, and its engineering work has been started. The construction schedule has been planned in which the first criticality will be realized in July, 2003. In fact, China is lack of the experience on fast reactors, so the schedule may be changed in future.

Even though China has conducted fast reactor technology development for years, but not so enough for the systems and components of fast reactors. So we have a prescience of difficulties during its construction, and need more cooperation with other countries.

Acknowledgement

The Author thanks deeply to his colleagues for providing the informations used to this paper.

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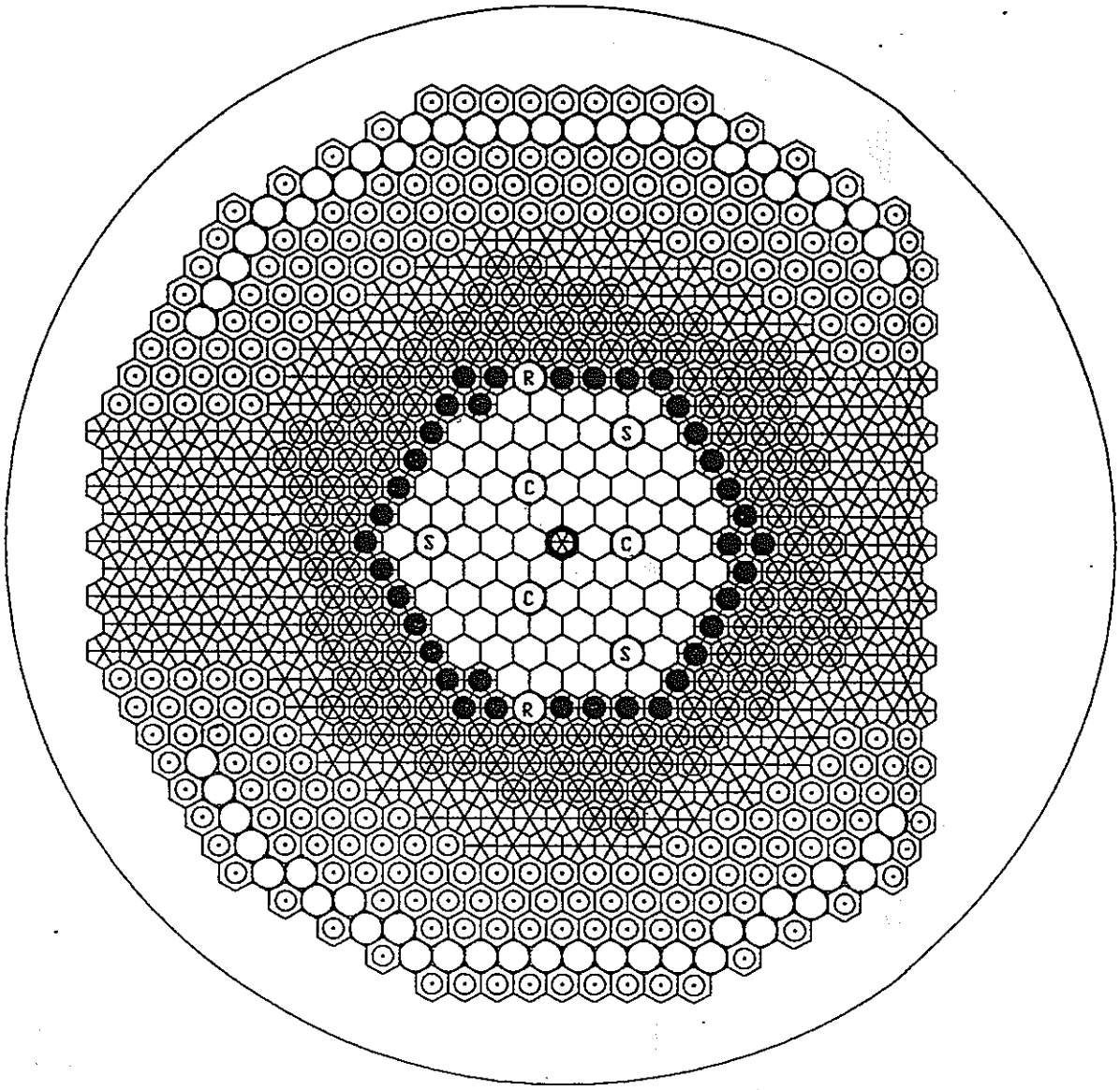

Fig.1 CEFR Core

⬡	Fuel subassembly	81
⊗	Stainless steel rod	1
⊗	Stainless steel reflecter subassembly	37
⊗	Stainless steel refleetor rod	132
⊗	Stainless steel refleetor rod	167
⊙	Sheilding subessembly	230
○	Storage position for spent fuel subessonbly	56
Ⓢ	Safety subassombly	3
Ⓡ	Regulation subassombly	2
Ⓒ	Compensation subassembly	3

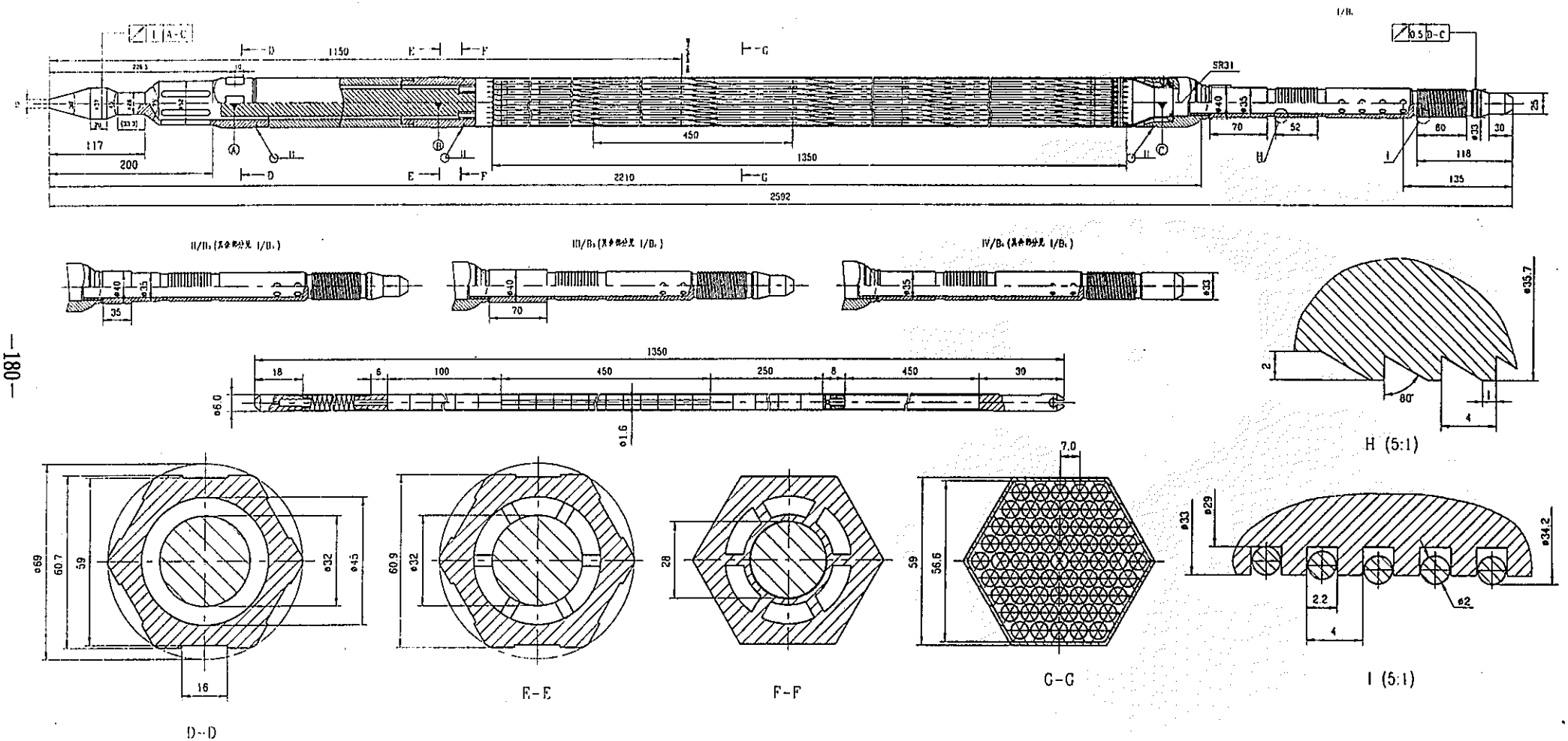

Fig.2 CEFR Fucl Subassembly

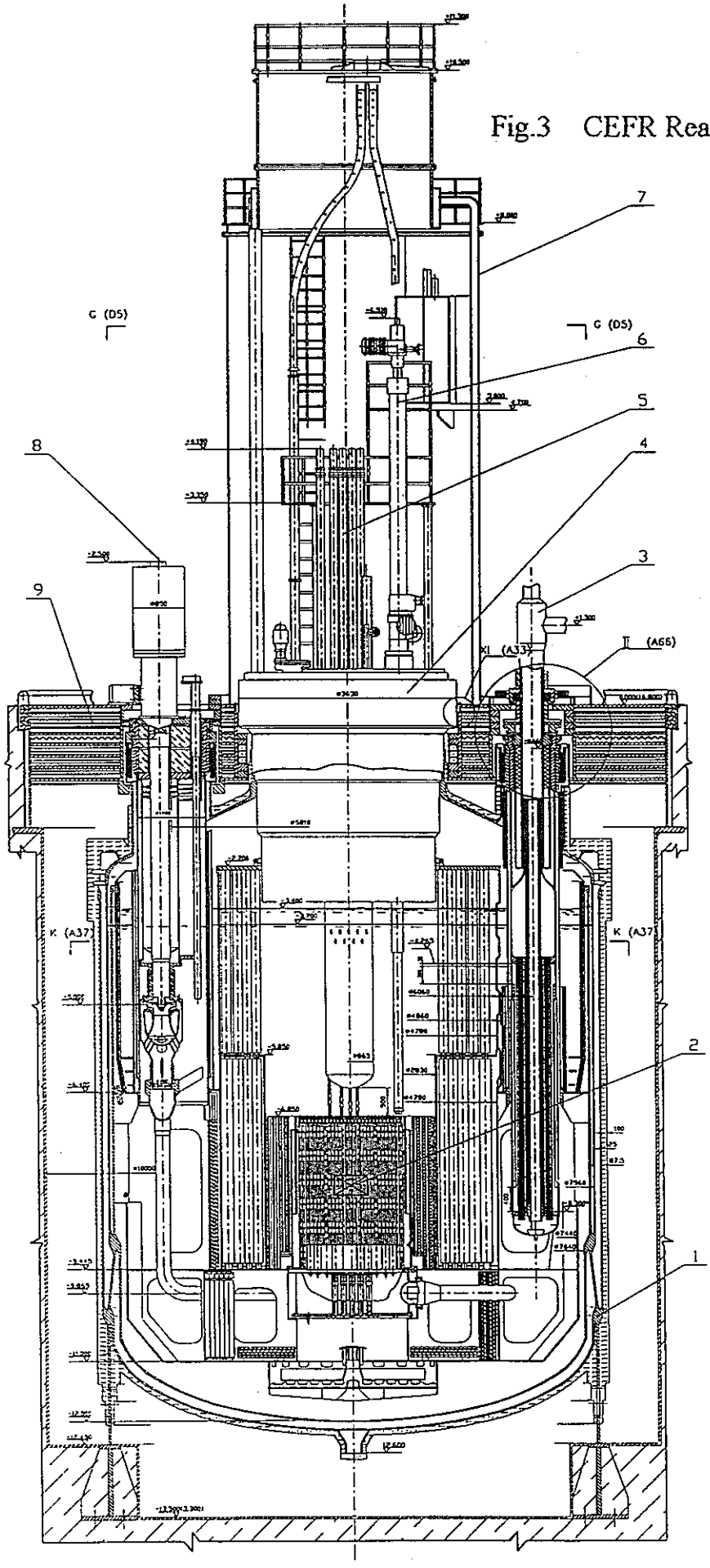

Fig.3 CEFR Reactor Block

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The Status of Fast Reactor Technology Development in China

For The Thirty First Annual Meeting of the International Working Group

by XU MI
China Institute of Atomic Energy
May 12-14, 1998, Beijing, China

1. INTRODUCTION

Table 1 CHINA's POWER GENERATION AND TRENDS

year	Total capacity (GWe)	Total production (TWh)
1993	183	836
1994	199	905
1995	210	990
1996	225	
1997	250	
2000	300	1400
2010	590	2700
2020	800	3700

Table 2 MAIN PROPERTIES AND SCHEDULE OF CHINA's NUCLEAR POWER REACTORS

Reactor	Type	Capacity, (MWe)		Operator	NSSS Supplier	Construction Start	First Criticality	Grid Connection	Commer. Operation
		Net	Gross						
in Operation									
Qinshan-1	PWR	279	300	QNPC	CNNC	1985.03.20	1991.10.31	1991.12.15	1994.04.01
Daya Bay-1	PWR	944	984	GNPJVC	FRAM	1987.08.07	1993.07.28	1993.08.31	1994.02.01
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2 CEFR Design

General Engineering Characteristics:

Thermal Power	65 MW
Electrical Power (T.G. Capacity)	25 MW
Electrical Power (net)	20 MW
Acreage of Reactor Site	15 ha
Floor Space of Buildings	43731 m^2
First Loading (UO_2 64.4%)	417 kg
Water Supply	4500 t/d
Power Supply	3000 kW

Table 4 CEFR FUEL LOADING (EQUILIBRIUM)

Fuel		$(Pu, U)O_2$	UO_2
PuO_2	wt%	29.4	
U-235 enrichment	%	36	64.4
U-235	kg	92.33	236.6
Pu-239	kg	65.76	
Total Pu	kg	141	

Table 5 MAXIMUM HEAT RATE*

	Cycle	
	Beginning	End
Power of Fuel Subassembly (MW)	0.980/0.957	0.978/0.952
Linear of Fuel Pin (W/cm)	405/395	396/388
Volume Power Density (MW/m^3)	756/738	744/725

*: $(Pu, U)O_2/UO_2$

Table 6 MAXIMUM NEUTRON FLUX RATE* (10^{15} n/cm^2 • s)

	Cycle	
	Beginning	End
Center (steel)	3.73/3.13	3.76/3.15
First Row of Fuel S.A**	3.70/3.10	3.73/3.13
First Row of Reflector S.A	2.20/1.90	2.26/1.92
First Row of Shielding S.A	0.293/0.261	0.298/0.264
Storage Position	0.0269/0.0252	0.271/0.0254

*: $(Pu, U)O_2/UO_2$

**: S.A-Abbreviation Subassembly

Table 7 CLADDING AND FUEL TEMPERATURE* max. ℃

Flow Region	Cladding**		Fuel***	
	Cycle		Cycle	
	Beginning	End	Beginning	End
1 (15 Fuel S. As)	665.68	670.44	2511.36	2497.30
2 (18 Fuel S. As)	665.02	670.76	2373.06	2370.10
3 (21 Fuel S. As)	659.68	662.24	2219.16	2213.50
4 (27 Fuel S. As)	656.66	656.68	2039.46	2030.60

* $(Pu, U)O_2$

** Including 2σ deviation of average square root related to the nominal value.

*** Using 3σ for uncertainty of design parameters.

Safety Characteristics

(1) Active Safety Measure

Table 8 CONTROL SUBASSEMBLY REACTIVITY VALUE

$(Pu, U)O_2$ core %ΔK/K

Shut Down System	Control Subassembly	Material	Cycle	
			Beginning	End
first	Comp.-1	B_4C Enriched	1.85	1.84
	Comp.-2		1.82	1.81
	Comp.-3		1.82	1.81
	Regu.-1	B_4C Natural	0.14	0.13
	Regu.-2		0.14	0.13
	3 comp.s		5.59	5.63
	Total		5.83	5.84
	Total except comp.1		3.91	4.00
second	SA-1	B_4C Enriched	0.98	0.99
	SA-2		0.99	1.00
	SA-3		0.98	0.99
	Total		3.10	3.13
	2 SA s		2.01	2.03
Total	8 Subassemblies		9.13	9.25
	7 Except comp.1		7.19	7.18

Table 9 REACIVITY BALANCE OF FIRST SHUT DOWN SYSTEM

%ΔK/K

	Reactivity
Maximum excess reactivity to be compensated	2.81
Control subassemblies value	
Total	5.83
Total except comp.-1 (In real position)	4.39
Subcritical value at 250 ℃	
Total action	3.03
Total except comp.-1	1.58

Table 10 REACTIVITY BALANCE OF SECOND SHUT DOWN SYSTEM

%ΔK/K

	Reactivity
Temperature-power effect to be compensated	1.03
Control subassemblies value	
Total	3.10
Total except SA-1	2.01
Subcriticality	
Total action	2.07
Total except SA-1	0.98

(2) Passive Safety Properties

Table 11 TEMPERATURE REACTIVITY EFFECT

$(Pu, U)O_2$ CORE, 250-360 ℃, %ΔK/K

	Cycle	
	Beginning	End
Sodium Density	-0.169	-0.170
Axial Expansion of Reactor Core	-0.037	-0.038
Axial Expansion of Lateral Reflector	-0.019	-0.020
Radial Expansion (diagrid plenum)	-0.204	-0.205
Doppler Effect	-0.032	-0.033
Total	-0.461	-0.466

Table 12 POWER REACTIVITY EFFECT, HOT STANDBY--FULL POWER

(Pu, U) O_2 core %ΔK/K

	Cycle	
	Beginning	End
Sodium Density	-0.107	-0.112
Sodium Volume Fraction Exchange	-0.029	-0.034
Axial Expansion of Reactor Core	-0.339	-0.311
Axial Expansion of Lateral Reflector	-0.011	-0.011
Radial Expansion (S.A.Winding)	-0.02	-0.02
Doppler Effect	-0.063	-0.063
Total	-0.569	-0.556

Table 13 DOPPLER CONSTANT OF CEFR CORE

$(Pu, U)O_2$ $10^{-3}\Delta K/K$

	Kw(with Na)	Kd(without Na)
Fuel:633-1593K Cladding: 633-1000K	-0.878	-0.618
All Materials 523-633K	-1.574	-1.048
All Materials 373-523K	-1774	-1.142

Table 14 REACTIVITY EFFECT OF SODIUM LOST

$(Pu, U)O_2$ CORE %ΔK/K

Region Lost Sodium	Cycle	
	Beginning	End
Core (Fuel Section)	-2.366	-2.404
Core (Fuel section + upper section of S.A)	-3.067	-3.110
Core (Whole Section of S.A)	-3.737	-3.788
Whole Core (including central stainless steel rod, and stainless steel reflector)	-5.196	-5.194

Main Design Parameters

Parameter	Unit	Conceptual Design	Preliminary design
Thermal Power	MW	65.5	65
Electric Power, net	MW	20	20
Reactor Core			
Height	cm	50	45.0
Diameter Equivalent	cm	58.5	60.0
Fuel		$(Pu, U)O_2$	$(Pu, U)O_2$
Pu, total	kg	121.6	141
Pu-239	kg	93.2	65.76
U-235 (enrichment)	kg	97.6 (30%)	92.33 (36%)
Linear Power max.	W/cm	430	430
Neutron Flux	$n/cm^2 \cdot s$	2.97×10^{15}	3.7×10^{15}
Bum-up, target max.	MWd/t	100000	100000
Bum-up, first load max.	MWd/t	50000	60000
Inlet Temp of the Core	℃	360	360

Outlet Temp of the Core	℃	530	530
Diameter of Main Vessel (outside)	m	8.0	8.010
Primary Circuit			
Number of Loops		2	2
Quantity of Sodium	t	～300	～300
Flow Rate, total	t/h	1400	1328.4
Number of IHX per loop		2	2
Secondary Circuit			
Number of loops		2	2
Quantity of Sodium	t	48.2	48.2
Flow Rate	t/h	986.4	986.4
Tertiary Circuit			
Steam Temperature	℃	480	480
Steam Pressure	MPa	10	14
Flow Rate	t/h	93.5	96.2
Plant Life	a	30	30

Table 15 MAXIMUM DOSE EQUIVALENT VALUE mSv

no. of accidents	Grid lost	Shut down failed	DHRS failed	Dose calculated	Dose limitation
BDBA-1	∨	∨		0.12	5
BDBA-2	Purification pipe leaked and check valve failed, 1372kg Na			0.0692	5
BDBA-3	∨		∨(10%kept)	0	5
BDBA-4	Regulation S.A. unexpected raised	∨		0	5
BDBA-5	Two check valves in primary circuit closed			0	5
BDBA-6	Main and guard vessels leaked successively			0.00414	5
DBA-1	Main vessel leaked			negligible	0.5
DBA-2	Purification pipe leaked and check valve O.K. , 40kg Na			0.0013	0.5
DBA-3	Cover gas leaked			negligible	0.5

4 RESEARCH WORK

Material
Sodium test loops
Design Demonstration

5 ENGNEERING PROGRESS

6 INTENATIONAL COOPERATION

Italy
France
Russia
IAEA

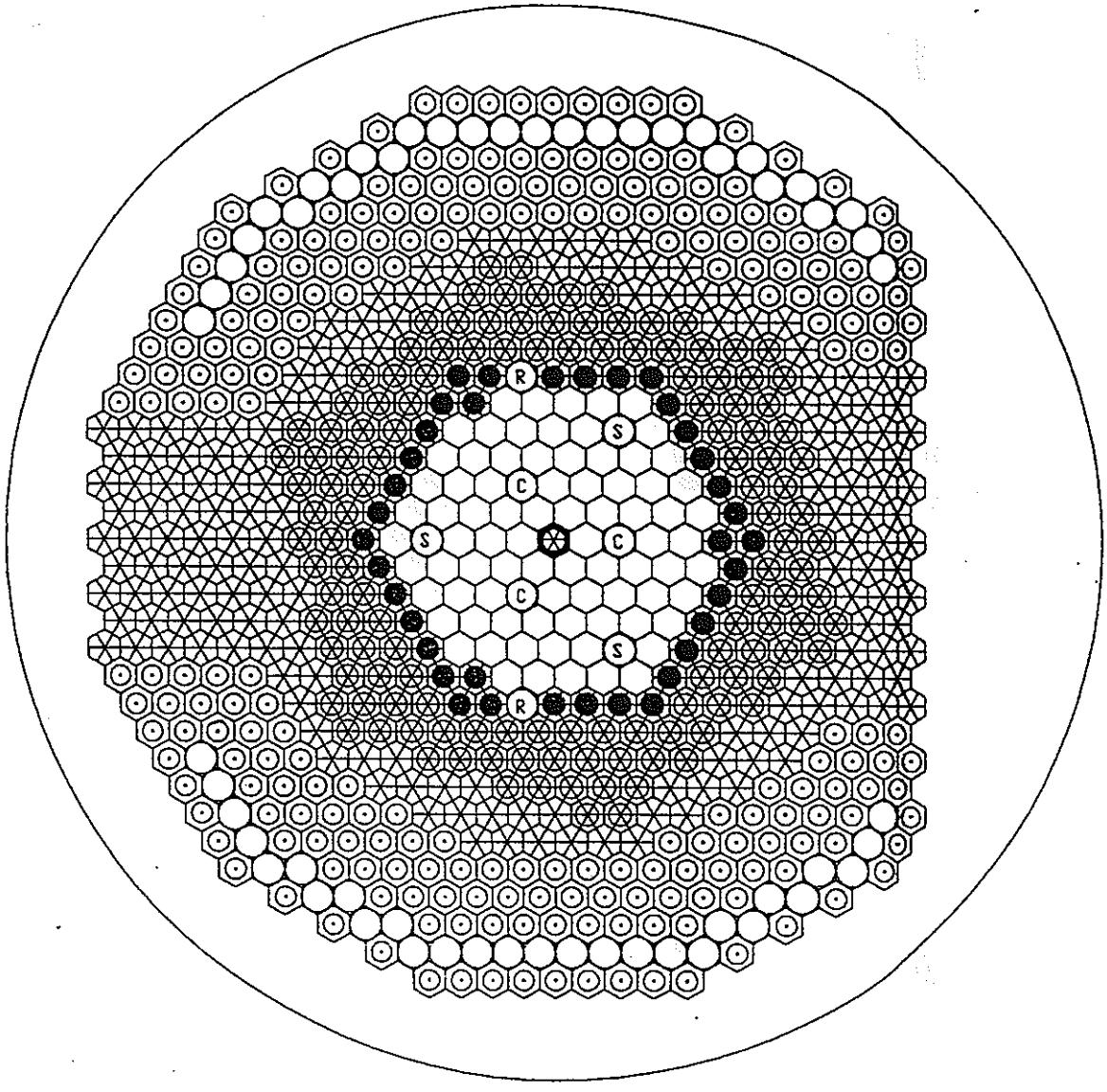


Fig.1 CEFR Core

⬡	Fuel subassembly	81
⊗	Stainless steel rod	1
⊗	Stainless steel reflecter subassembly	37
⊗	Stainless steel refleetor rod	132
⊗	Stainless steel refleetor rod	167
⊗	Sheilding subessembly	230
⬡	Storage position for spent fuel subessonbly	56
Ⓢ	Safety subassombly	3
Ⓡ	Regulation subassombly	2
Ⓒ	Compensation subassembly	3

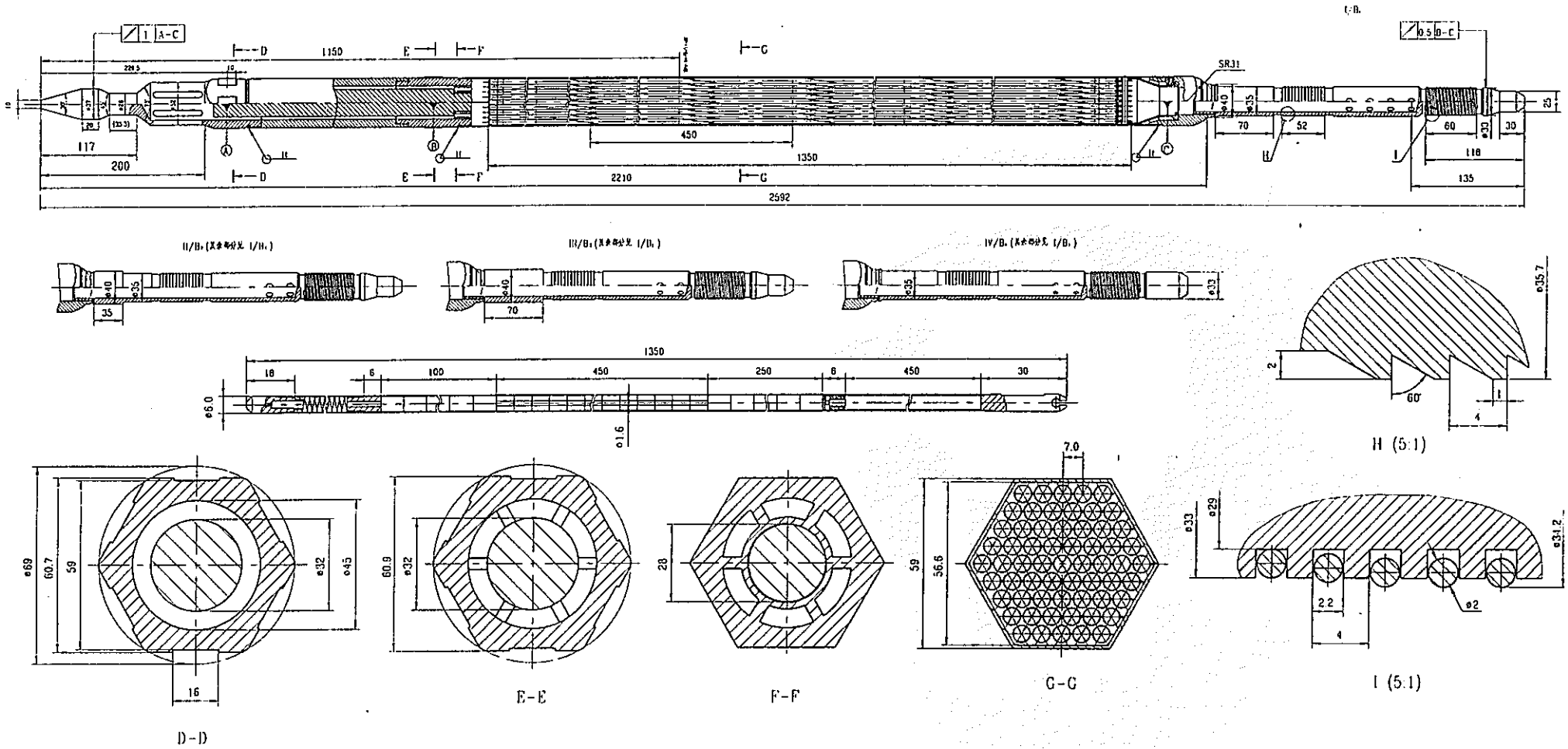


Fig.2 CEFR Fuel Subassembly

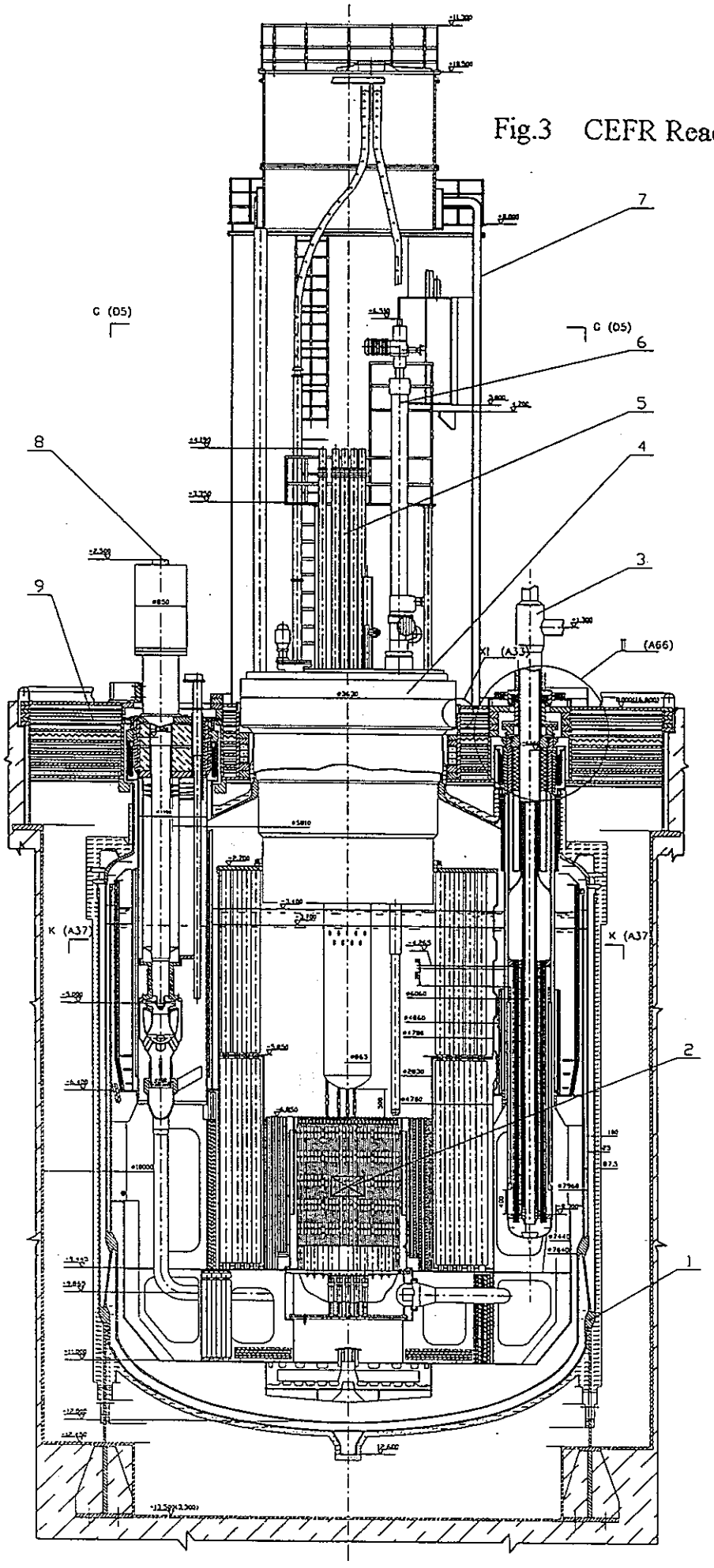


Fig.3 CEFR Reactor Block

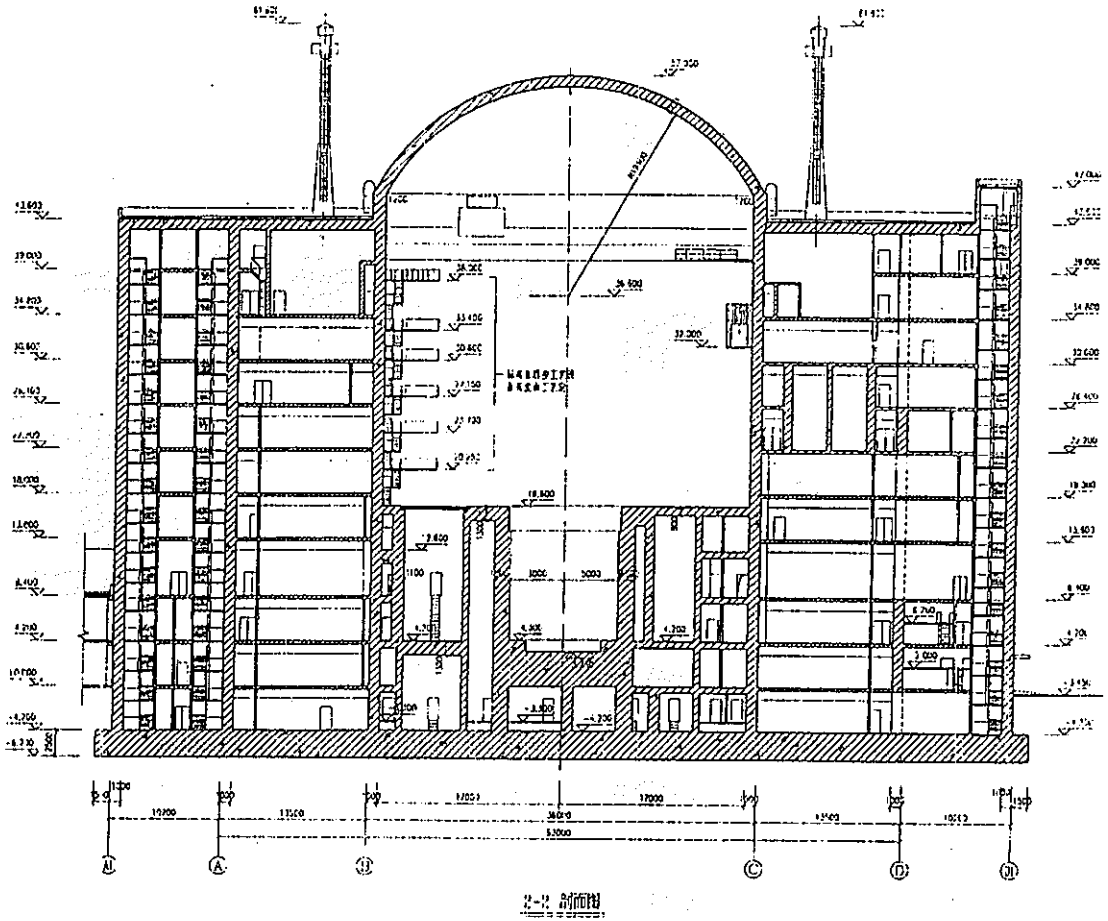
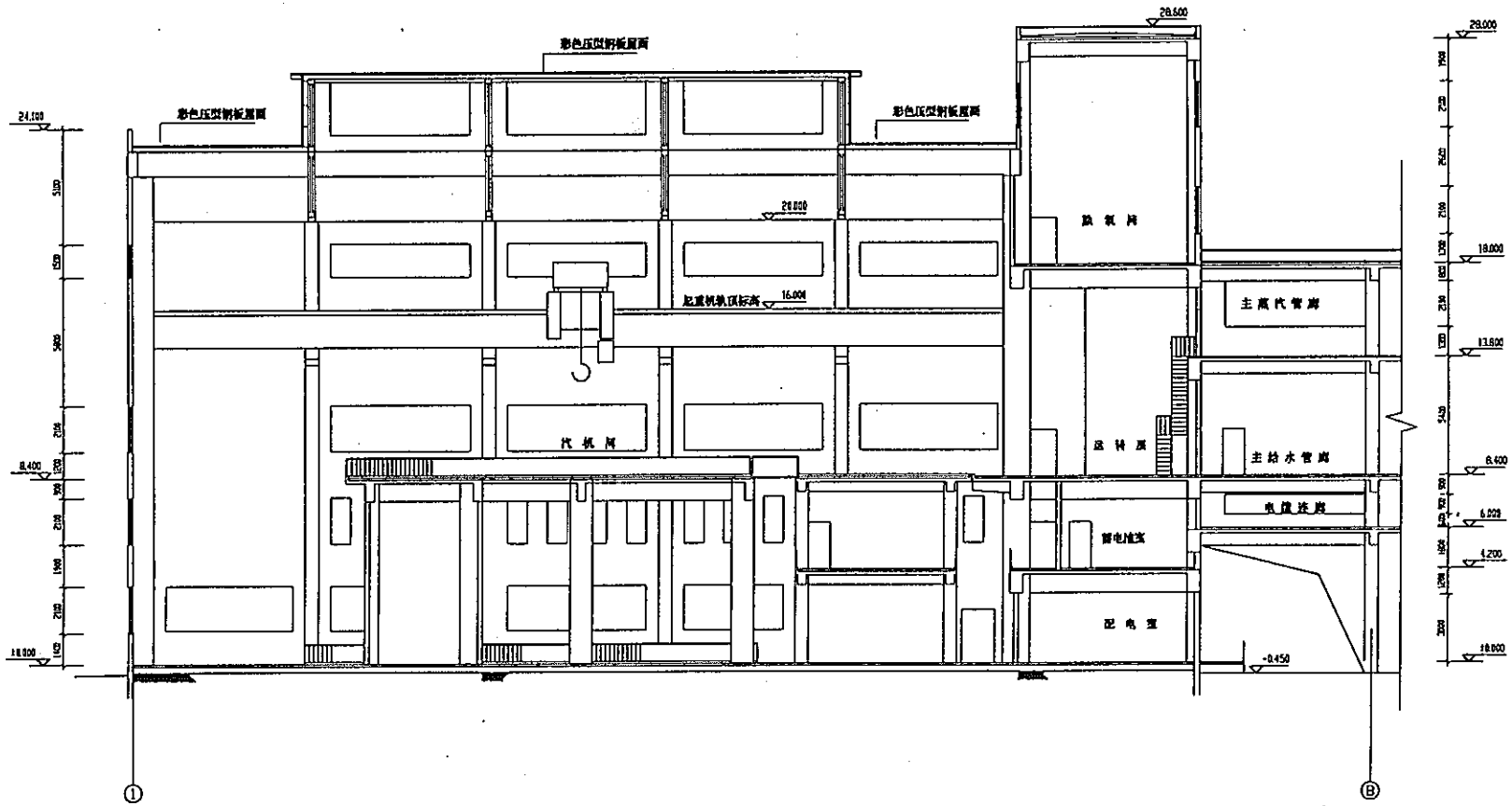


Fig. 4 Reactor Building



I—I 剖面图 1:100

图例

加气混凝土

防火墙

混凝土墙

1:100

会签栏		CCTRIC—JZC01025			
专业	总工程师	工程号 9046	子项号 CC01	核工业第二研究设计院	
审定	总设计师				
	主任设计人	汽轮发电机厂房及连廊		比例 1:100	设计
审核	组长设计人	I—I 剖面图		日期	专业 审核
	工种负责人				
校对	校对员	中国实验快堆		图号 904603-JZC01-005	
设计	设计员				

Fig.5 Turbine - Generator Building

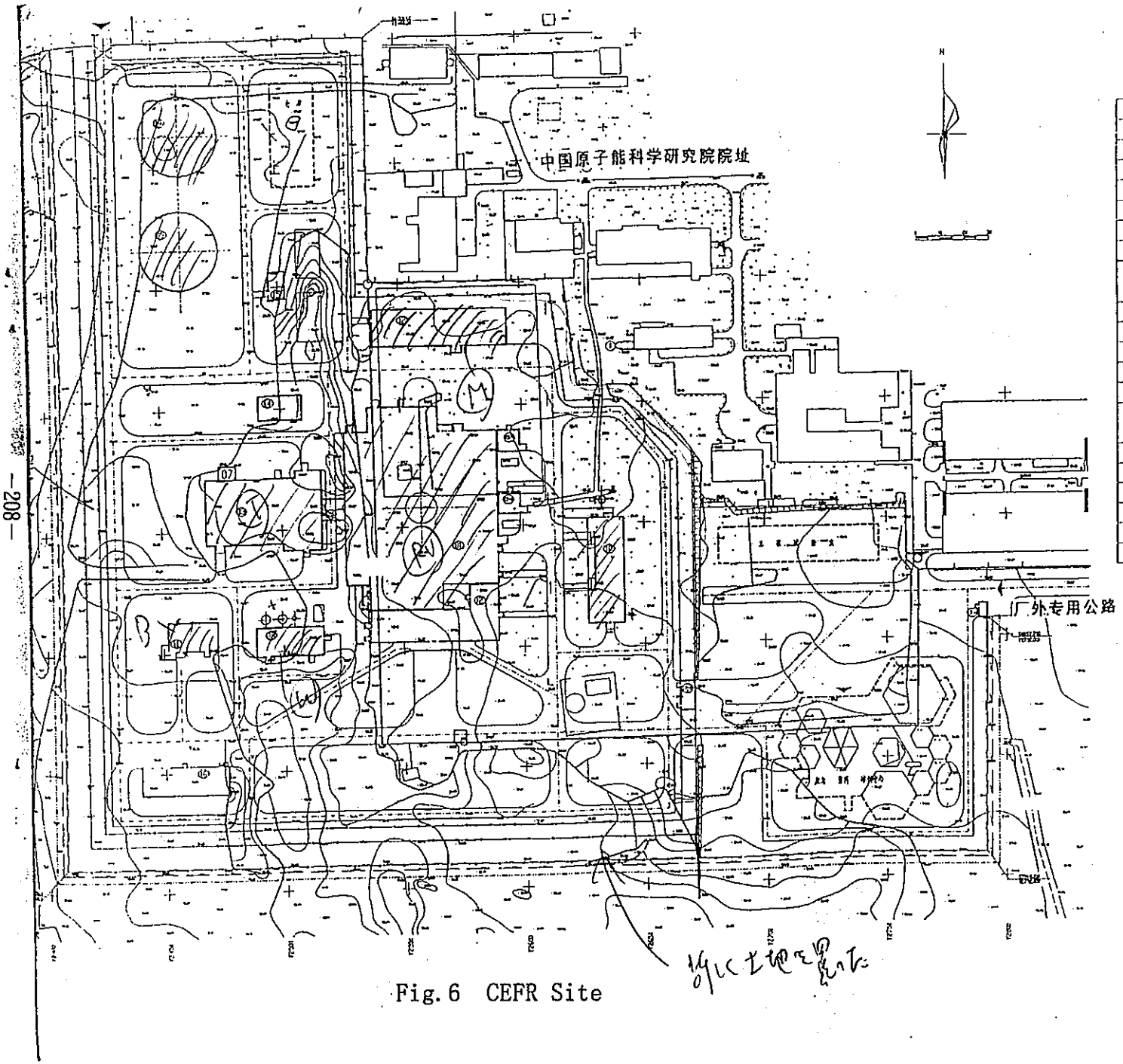


Fig. 6 CEFR Site

建构筑物名称表

序号	代号	建构筑物名称	建筑面积(m²)	占地面积(m²)	备注
1	01	核岛厂房	36206.00	4592.33	
2	02	核岛专用厂房	1926.00	935.00	
3	03	汽轮发电机厂房	2380.00	1425.00	
4	04	泵房	208.00	96.00	
5	05-1	1# 柴油发电机厂房	90.25	90.25	
6	05-2	2# 柴油发电机厂房	90.25	90.25	室外油池地下油箱
7	07	开关开关站	--	54.00	——
8	08	核岛厂房（包括冷却塔护栏）	356.00	*590.00	*包括塔 中和水及建筑物占地
9	09	冷却塔	--	1815.04	
10	10	氢气 压缩空气制冷机房	607.00	607.00	
11	11	启动锅炉房及除盐站	301.00	292.00	
12	12	设备检修车间	868.00	868.00	
13	14	气体罐区	--	180.00	
14	15	消防水池及泵房	84.50	*484.50	*包括水池及建筑物占地
15	16	纳天火消防车库	432.00	432.00	
16	17-1	反应区警卫室	36.00	36.00	
17	17-2	保护区警卫室	36.00	36.00	
18	18-1	反应区围墙			
19	18-2	保护区围墙			
		合 计	43721.00	12014.47	

主要技术经济指标表

序号	指标名称	单位	指标	备注
1	厂区占地面积	m²	84349.75	*厂区围墙内总面积
	总规划区（包括）面积	m²	*502041.36	总规划区外厂区
2	建筑面积	m²	43721.00	
3	建构筑物占地面积	m²	12014.47	
4	建筑厂区绿化面积	m²	17556.36	
5	室外管线占地面积	m²	13496.00	
6	建筑系数	%	14.2%	建构筑物建筑物占地面积
7	利用系数	%	51.77	*
8	绿化系数	%	25.00	
9	土石方工程量（挖方）	m³	--	
	其中 填方	m³	262176.38	
	弃方	m³	502167	
10	防洪堤长度	m	1965.00	
11	护岸长度	m	2200.00	

26

FAST REACTOR DEVELOPMENT PROGRAM IN FRANCE IN 1997

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ABSTRACT

The operating performance of the French nuclear plants in 1997 were again satisfactory, confirming the good 1996 results : 78% of the national electricity production was of nuclear origin (376/481 TWh) ; the nuclear safety indicators were even better with a slight decrease in the number of reported incidents ; the availability factor was higher than 82,5 % ; the kWh production cost continue to decrease by about 1,7 % per year; thanks to an ALARA approach, individual and collective doses are still decreasing ; the liquid and gaseous releases are also still decreasing and the volume of nuclear wastes is equivalent to last year.

The major event of 1997 was the confirmation for the start up of the most advanced nuclear plants, N4 type, with Chooz B1 reaching full power on May 9, 1997 and Chooz B2 following soon with first criticality on March 10 and full power on September 18. As a confirmation and enjoying the experience of Chooz B the first unit of Civaux was synchronised to the grid end of 1997 after fuel loading in Sept 1997.

The status of fast reactors was deeply modified, with the governmental decision to abandon Superphenix, which led to decide to close also EFR studies in 1997, and the perspective of operating Phenix till 2004, mainly for studies on radioactive wastes incineration

Three meetings were held during the second half of the year by the « Groupe Permanent » (GP - Permanent Committee) on Phenix ; after reviewing all the safety aspects, they concluded to the possibility to resume operation according to the general following scheme : one cycle in 1998, about one year for works, then 6 cycles between 2000 and 2004.

Of course, the main event concerning Superphenix is the decision of abandon announced in June 1997. The decree allowing operation had been cancelled in February. The works foreseen in the frame of the planned « 320 EFDD » shutdown had been conducted satisfactorily, during the first-half of the year. At the beginning of 1998, it was stated that the reactor would not operate again, in spite of the availability of one and half core.

This decision leads to drop the « Knowledge Acquisition Programme ».

The Capra program was reoriented. Now, emphasis is put on burning the minor actinides (MA) and long-lived fusion products (LLFP), which answers the French 1991 law on radioactive wastes ; as a consequence, the experimental programme proposed for Phenix is mainly devoted to this objective.

This report presents also the status of Rapsodie, and the main achievements on the R&D programme on fast reactors.

(*) with contributions from :
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P.MARMONIER, R.DEL NEGRO, G.SAINT-ARROMAN for RAPSODIE chapter
P.COULON for PHENIX chapter
B.MESNAGE for SUPERPHENIX chapter
S.PILLON for CAPRA chapter
G.HUBERT, JC.LEFEVRE (FRAMATOME-NOVATOME) for EFR chapter

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1. GENERAL SITUATION

1.1. OPERATING REACTORS

In 1997, the electricity produced in France was shared as follows :
Nuclear power plants 376 (78 %)
Conventional power plants 38 (08 %)
Hydraulic plants 67 (14 %)

Total 481 TWh, among which 454 from EDF.

69,5 TWh were exported, 4,2 TWh imported, 5,2 used for pumping and line losses came to 29,8 TWh.
So, the net electrical power consumption was 380,7 TWh, i.e. a decrease of 0,6 % in comparison of the 1997 value.
At the end of 1997, Electricité de France had 57 PWR units in operation, the operating performance of which had been again satisfactory, confirming the good 1996 results :

- the nuclear safety indicators were even better with a slight decrease in the number of reported incidents, but vigilance is maintained on this top priority issue for EDF; (7.8 significant incidents per unit versus 8.1 in 1996; 1.4 scrams per unit per 7000h of operation, versus 1.5 in 1996);
- the availability factor was higher than 82,5 %. This good result comes from further progress in management of the anticipated outages;
- the kWh production cost continue to decrease by about 1,7 % per year; (4.1% drop in electricity rates for customers in 1997, in the frame of a commitment for an average decrease of 14% in 4 years)
- thanks to an ALARA approach, individual and collective doses are still decreasing : minus 50% for the first (individual), minus 10% for the second (collective) ; (average individual dose 2.65 mSv);
- the liquid and gaseous releases are also still decreasing and the volume of nuclear wastes is equivalent to last year.

These results confirm the benefits of operating standardised nuclear plants (57 PWRs).

1.2. THE START UP OF N4 PLANTS IN CHOOZ AND CIVAUX

The major event of 1997 was the confirmation for the start up of the most advanced nuclear plants, N4 type, with :

- Chooz B1 reaching full power on May 9, 1997
- Chooz B2 followed soon with first criticality on March 10 and full power on September 18.

EDF's Chooz B has been awarded the « Power Plant Award 1997 » by a joint committee of two American trade press publications « Electric power International » and « Electrical World ».

Chooz B has been selected for its « advancing commercial nuclear power by bringing on-line the world's most advanced nuclear unit » and the « largest single power generation unit of any type » (1500MW).

As a confirmation and enjoying the experience of Chooz B the first unit of Civaux was synchronised to the grid end of 1997 after fuel loading in Sept 1997.

1.3. STATUS OF FAST REACTORS

Three meetings were held during the second half of the year by the « Groupe Permanent » (GP - Permanent Committee) on Phenix ; after reviewing all the safety aspects, they concluded to the possibility to resume operation according to the general following scheme : one cycle in 1998, about one year for works, then 6 cycles between 2000 and 2004.

Of course, the main event concerning Superphenix is the decision of abandon announced in June 1997. The decree allowing operation had been cancelled in February. The works foreseen in the frame of the planned « 320 EFDD » shutdown had been conducted satisfactorily, during the first-half of the year. At the beginning of 1998, it was stated that the reactor would not operate again, in spite of the availability of one and half core. This decision leads to drop the « Knowledge Acquisition Programme ».

The Capra program was reoriented. Now, emphasis is put on burning the minor actinides (MA) and long-lived fusion products (LLFP), which answers the French 1991 law on radioactive wastes ; as a consequence, the experimental programme proposed for Phenix is mainly devoted to this objective.

The decision about Superphenix led to decide to stop the EFR programme, which constitutes an other major event of 1997. This year was in consequence the last for technical achievements, ant it is foreseen to close the programme in 1998.

2. RAPSODIE

2.1. INFORMATION ON THE ACCIDENT THAT OCCURRED AT CADARACHE ON MARCH 31, 1994

The first items of information were given during the IWGFR meeting in May 1995 ; further details were supplied during the following meetings in May 1996 and May 1997.

It should be recalled that this accident took place in a bunker next to the RAPSODIE reactor containment during the final dismantling operations of the reactor's sodium system. An explosion occurred during the destruction, using ethylcarbitol, of a pool of 100 to 150 kg of sodium that remained at the bottom of a 55 m^3 metallic tank. A CEA agent was killed and four others seriously injured.

During the past year, the CEA in-house inquiry commission set up in 1994 continued its work in parallel with the judicial inquiry.

The metallurgical examination confirmed and clarified the initial conclusions of the inquiry:

- The bursting of the tank was of pneumatic origin; the pressure reached at the time of rupture can be estimated as being between 13 and 16 bar, which is very much higher than the design pressure of the tank ;
- No dysfunction occurred in the cleaning systems, the sweeping systems, or the facility in general.

The chemical examination has not yet been completed due to the fact that seals have been affixed as required by judicial procedure.

However, laboratory tests have confirmed that :

- in the event of such a destruction, the reaction: $EC + Na \Rightarrow ECNa + \frac{1}{2} H_2 + 168$ kJ/mole is not the only one to occur ;
- at the very beginning of the destruction, several other reactions intervene and not only produce hydrogen, but also large quantities of other gases ;
- the energy released by the destruction of sodium depends on the concentration of alcohol in the environment.

It is still not possible to clearly identify :

- the cause(s) that triggered the runaway of the decomposition reactions in this environment ;
- The reactions that led to a far greater gas release than that corresponding to the initial basic reaction.

In view of these uncertainties, the CEA is continuing to prohibit the use of ethylcarbitol for sodium destruction.

2.2. WORKS ON THE PLANT

In 1997, work progressed in two main directions :

2.2.1. Dismantling and cleaning

The relevant work prepare the dismantling of the plan towards level 2.

- Evacuation of soda produced by DESORA ;
- Preparation of evacuation of the secondary sodium ;
- Evacuation of TFA[1] wastes ;
- Evacuation of TFA, FA[2] and HA[3] wastes coming from LDAC (adjacent test lab for dismantling and post-irradiation examination) ;
- Dismantling of the air coolers building ;
- Dismantling of the small (so called "light") cell ;
- Revamping of the protection against lightning of the buildings of the nuclear installation ;
- Revamping of the control command of LDAC and Rapsodie.

2.2.2. Reconstruction of the plant

The objective is to get rid of the consequences of the accident of 1994, by rebuilding and cleaning the affected zones. The 1997 results are :

- For the reactor containment building: restarting the ventilation and qualifying it ;
- Repair of the control command of the lack of materials ;
- Storage of the prices of the RENA 302 tank in an annex building ;
- Sending to wastes the alcohol tank ;
- Cleaning of the peripheral gallery and the Southern zone of Rapsodie ;
- Evacuation of TFA wastes.

After all these preparatory works, the construction began on 09.03.98.

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[1] *TFA : very low activity*

[2] *FA : low activity*

[3] *HA : high activity*

3. PHENIX

3.1. INTRODUCTION

The dominant event for Phénix in 1997 was the preparation of the meetings with the Regulatory Authority (permanent committee), to which the authorisation to resume operation had been asked. Three meetings took place during the second half of the year, after which the permanent committee gave its agreement to restart the Phénix power plant for the 50th cycle in 1998 and for continuation until 2004 once the plant has been brought up to standard. The return to operation was expected in April 98, and a comprehensive experimental irradiation programme prepared till 2004.

3.2. LIFE EXTENSION PROJECT

This project, conducted over several years, came to an end with the completion of safety files for safety review ; it allowed a thorough examination of the plant ; a final synthesis is being prepared.

Above core structure - tests on crack initiation on circumferential welds were made on two mock-ups ; elastic calculations appear conservative.

FLUTE tests on T junctions allowed to improve calculation methods for creep fatigue, by accounting for relaxation.

Intergranular cracking was observed on 304SS at high temperature ; for comprehension a test consisting in thermal cycling a valve cap was prepared, whereas residual stresses induced by the welded junction of this cap were calculated ; the study should be completed in 1998.

3.3. STATUS OF THE RENOVATION WORKS

They had been split into 23 parts. Two have been abandoned :

- Hydrogen detection : it will be ensured by the existing system
- Helium injection circuit (power limited to 350 Mwth)

Nine parts have been completed, which were required before the 50th cycle.

Twelve parts are in progress or are scheduled to be done during the shutdown between the 50th and 51st cycles.

Works completed

- Replacement of 321 steel by 316LN (secondary circuits) ;
- Replacement of steam generators sodium collectors ;
- Improvement of auxiliary circuits ;
- Partial protection of the steam generators building against sodium fire (between 0 and –8 m) ;
- Study of thermal cracking risks ;
- Installation of a complementary shutdown system ;
- Seismic reinforcement of buildings (annexes and control/offices building) ;
- Feed water tanks in the electricity production installation ;
- Increased seismic reinforcement for the emergency cooling circuit (open.)

Works in progress or to be done during the shutdown between 50th and 51st cycles :

- Manufacture of 3 intermediate heat exchangers (only 1 necessary for the 51st cycle) ;
- Manufacture of a hot collector pole (temperature, movements) ;
- Manufacture of components for primary or secondary sodium pumps ;
- Increased seismic reinforcement of reactor building ;
- Installation of new "closed" emergency cooling circuit ;
- US inspection of the core support structure ;
- Partial emptying of the vessel / core cover plug checking ;
- Sodium-fire protection of steam generators building ;
- Protection against rupture of water/steam pipes ;
- Reinforcement of steam generators housings : earthquakes, sodium fire, rupture of steam pipes ;
- Improvement of auxiliary sodium circuits
- Increased seismic protection for steam generators, turbine hall and handling buildings.

Preparation of the shutdown between cycles 50 and 51 : planning

In January 98 was conducted a first-time overview of all the renovation works, which led to evaluate a 18 month planning.

In February 98, a further review identified and included the works required by 10 year outage, important operations such as core shell monitoring (fig 3.1), partial emptying of vessel, and the necessary requalifications, still inside a18 month planning.

From March 98 is developed a critical path analysis and work on the necessary means for minimising outage time (2 shift teams, ...), with the objective of reducing the shutdown time to 12 to 14 months.

3.4. EXPERIMENTAL PROGRAMME 1998 – 2004

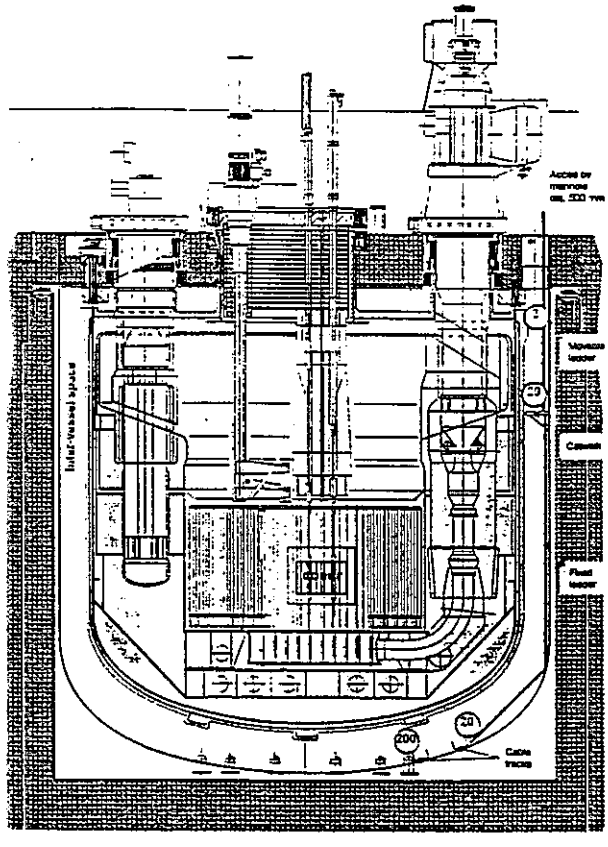
Taking in consideration the planning of Phénix till 2004, and the reassessment of the main requirements for fast reactor programmes, the experimental programme till the end of exploitation was defined ; hereafter are given the guidelines.

End of fuel cycle becomes a strong priority ; to answer the 1991 law on radioactive wastes treatment, it is necessary to study the ability of fast reactors to destroy by transmutation or fission the LLFP and MA ; this leads to a set of experiments on fuel and targets including MA, on specific matrices, on some LLFP, which constitute the most important part of the programme.

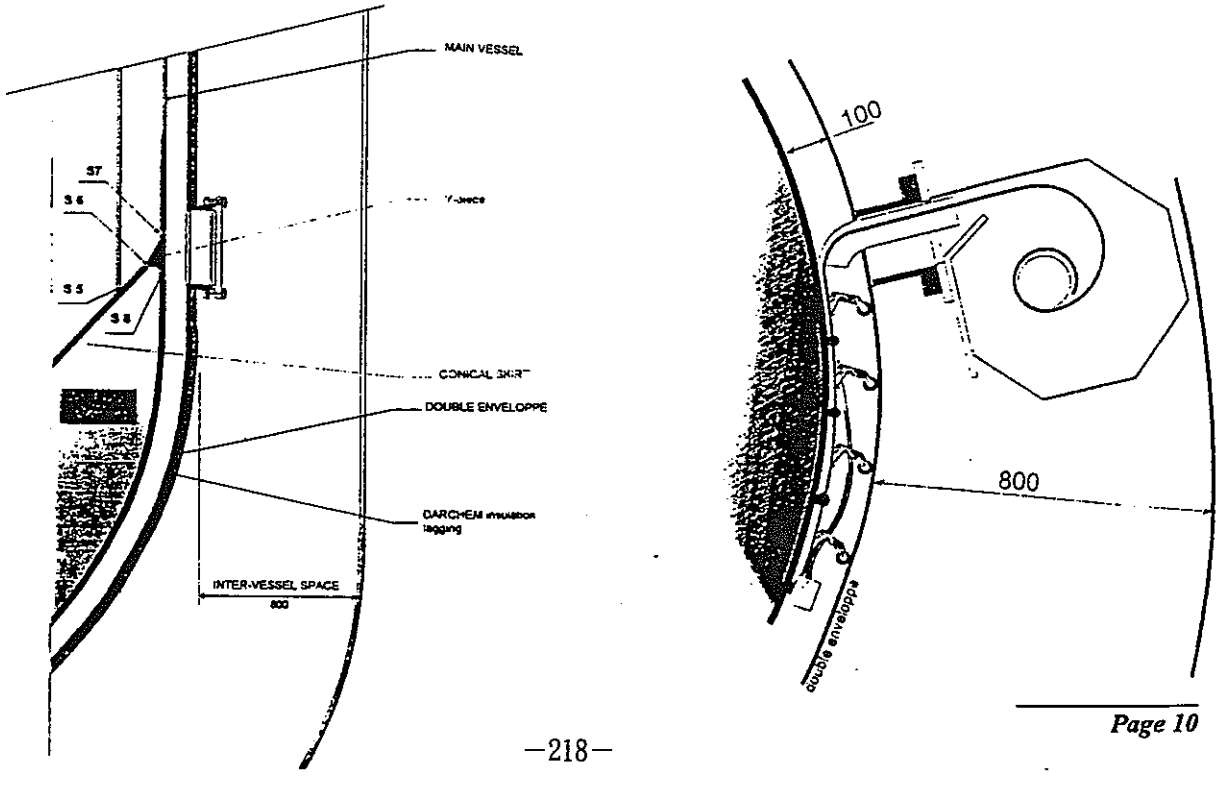
Some experiments remain on the study of high Pu content oxide fuel, in the frame of the programme on Pu management in a reactor park.

Another field is the development of core materials for high irradiation doses (objective more than 200 dpa), concentrated on a limited number of promising solutions ; it is based mainly on experimental S/A associating standard fuel and innovative claddings and/or wrappers, most of which are already in the reactor, and irradiation of samples in capsules.

Fig 3.1 - Phénix - ultrasonic inspection of the conical shell



Access port to welds of core support skirt



4. SUPERPHENIX

4.1. GENERAL OVERVIEW OF 1997

The reactor did not operate in 1997.

After operation throughout 1996, at the beginning of the year the plant started a scheduled six month outage with a view to :

- Maintenance work necessary for later plant operation - replacement of secondary cold traps and revision of the 10 year type for numerous pieces of equipment,

- Work and controls associated with safety - tube condition control on a steam generator inspection of in-sodium apparatus, control of diagrid seat, etc.,

- And lastly, a handling campaign for adapting Superphenix to its research missions ; i.e. replacement of the first fertile subassembly row by steel elements, and loading of the three experimental KAP fuel subassemblies (Knowledge Acquisition Programme) delivered on-site in November 1996.

1997 was, for Superphenix, to be the year of the first unloading of subassemblies, which was to enable the plant to validate the handling line as a whole up to storage in the APEC pool and, in addition, the year of the beginning of the KAP relating to plutonium burning and minor actinide destruction.

This plant outage was to have been followed in the summer by a resumption in power of the plant in the framework of operating approval to 90% NP, obtained in October 1996, and this operating period was to have continued until the spring of 1998 when a new scheduled short shutdown was planned for maintenance and rearrangement of the core.

But two events upset this programme :

- First of all, the cancellation on 28th February by the State Advisory Board of the Licensing Decree dated 11th July 1994 as the Board considered that the Order which entrusted Superphenix with a research and demonstration mission was not in keeping with the contents of the file submitted for the Public Inquiry in 1993,

- And then, the announcement by the newly-appointed Prime Minister following elections in May, in the National Assembly on 19th June, that Superphenix was to be abandoned. This decision was one of the points in the election manifesto agreed between the Socialists and the Greens.

Issued whereas the plant outage ensued in compliance with the programme planned, cancellation of the Order prohibited any modification of the core and any fuel unloading including the zone preparations of the APEC, and therefore led to a practical suspension of the handling campaign, scheduled for March, in the expectation of a new Order promised within the shortest possible time by the then Government.

The consequences of postponing this handling campaign to a later date were however limited, thanks to the anticipation of scheduled in-service inspection works following on in the reactor and, in spite of delays in the provisional schedule due to various technical set-backs and to certain operations taking longer than planned, plant shutdown works in mid June were nearly completed.

On the other hand, the Prime Minister's declaration of 19th June changed everything, and marked the beginning, for the plant and its personnel, of a long waiting period to obtain the exact definitive shutdown procedures by the Ministers concerned. This waiting period was extended on several occasions and came to an end with the Governmental decision on 2nd February 1998 for

definitive shutdown for Superphenix, without any resumption of power even for a limited period of time, as NERSA proposed, to burn the two cores which had already been provided and paid.

This long period therefore saw, on the one hand after a temporary stoppage of operations, technical continuation of plant outage activities, at a less intensive rhythm, with a view to bringing the installation back to its normal status for resumption if applicable; this was the status practically reached on 2nd February 1998 and, on the other hand, the beginning of studies to define the stages of definitive plant shutdown which are to be the main activity of 1998.

But this long period was to be, above all, marked by very considerable number of protests voiced from June onwards, not only in the Rhône-Alps but also in Paris on 3rd July, by the plant personnel as a whole grouped in their Trade Unions or in the Superphenix Support Committee created at the beginning of June with most of the local businesses, shopkeepers, craftsmen, councillors and inhabitants of the Morestel region. It was to be marked too by the demonstration of the environmentalists on 2nd August to celebrate the 20th Anniversary of the 1977 protest. Lastly, this long period saw also a succession of numerous actions in favour of Superphenix by diverse personalities from the scientific, economic and political scenes - all in a continuous whirlwind of media attention.

In fact, all these events did not prevent the plant from pursuing, or even intensifying its policy of information through its three-monthly journal, numerous lectures given in schools in the region, welcome for a considerable number of visitors including a high percentage of foreigners, more particularly industrialists and students, but also politicians, researchers and, of course, journalists. The plant also pursued its international co-operation with Russia and especially with Japan by inviting a second Japanese engineer for all the year.

4.2. PLANT OUTAGE ACTIVITIES

Considering the events, the plant outage programme initially planned was finally extended over the whole year to the beginning of February 1998 as the schedule in Figure 4.1 shows, giving the most noticeable activities carried out at the same time as numerous upkeep operations and periodical tests planned within the maintenance programmes and general operating rules.

After characterisation at 180° C of the reactive status of the core following reactor shutdown on 24th December 1996, outage activities started on 6th January after draining secondary loops C and E on which the following work was carried out simultaneously :

- Replacement of the cold traps by new improved traps (Figure 4.2). Draining and filling operations on the NaK thermal seals were carried out without particular difficulty.

- Practical application of a considerable non-destructive testing programme concerning welds on the sodium circuits (X-rays), the expansion tanks (ultrasonic inspection) and the water-steam piping (X-rays and ultrasonic) before SG hydrotests.

- Hydrotests at 268 bar successfully carried out on SG C on 14th March, then on SG E on 3rd April - these tests were preceded by placing a blind flange on the SG discharge circuits and replacing one of the rupture disks so as to resist any build-up in SG pressure in the event of tube break during the hydrotests (Figure 4.3).

- Standard replacement of snubbers and upkeep of secondary pumps with a standard replacement of one of the electric motors in accordance with the preventive maintenance programme.

The two loops were then successively sodium-filled during April while the other two secondary loops D and F were drained; in turn the same operations were carried out although limited as far as the hydrotests were concerned to SG F as the SG D test had already been completed in 1990.

Benefiting from the previously acquired experience, replacing the two D and F cold traps occurred under excellent conditions as well as the SG F hydrotests. The latter took place on 6th June 1997 and the F secondary loop was then sodium refilled on 20th September.

Performed for the first time in industrial environment, control on the condition of 36 preselected tubes of SG D had started at the beginning of May according to the procedure developed by Novatome and the CEA which, after cutting the introduction nozzle situated between the bundle and the steam collector (Figure 4.4), comprised four main stages on each tube :

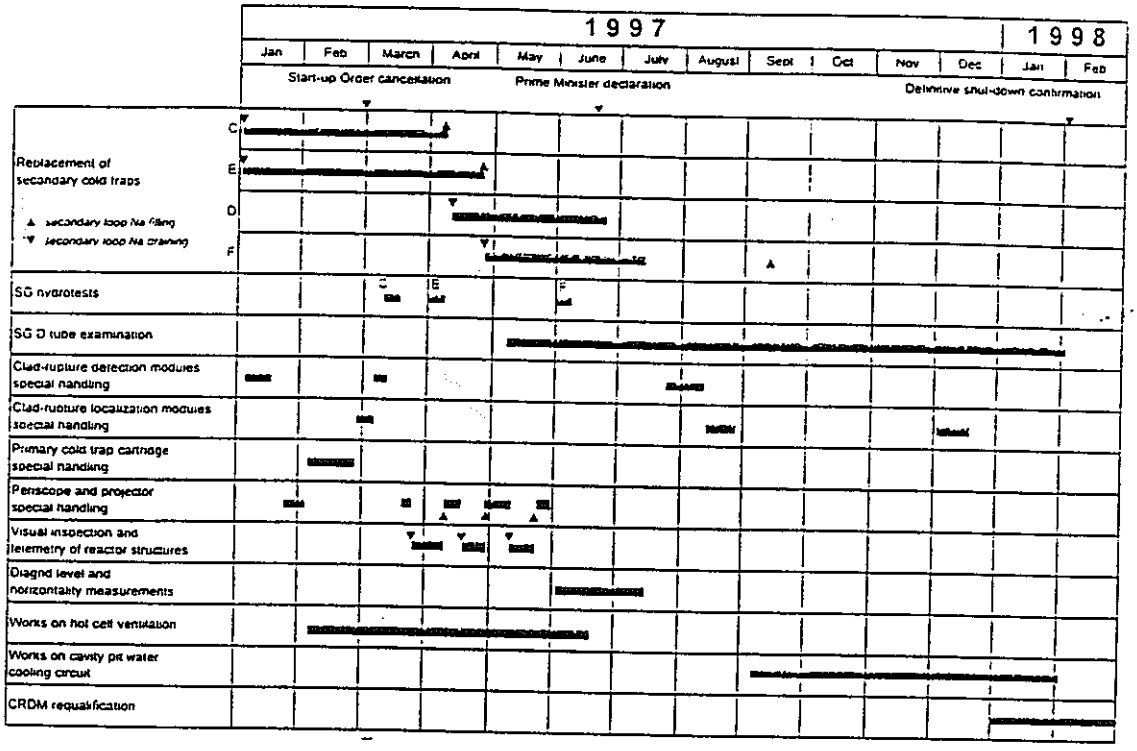


Fig 4.1 - Maintenance outage schedule

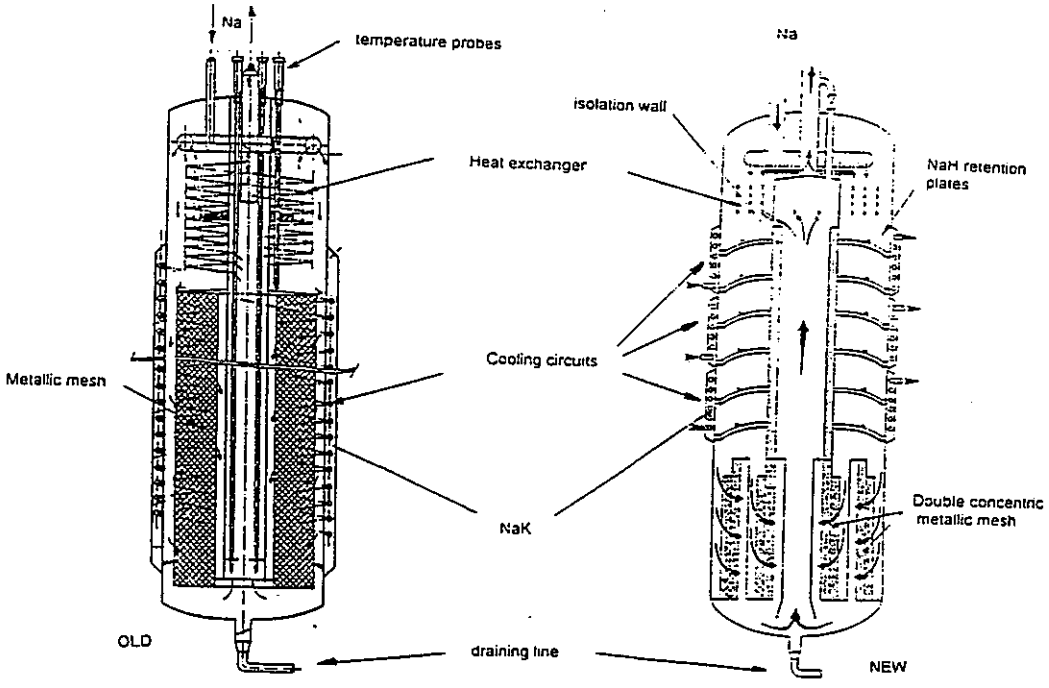


Fig 4.2 - Superphenix secondary cold traps

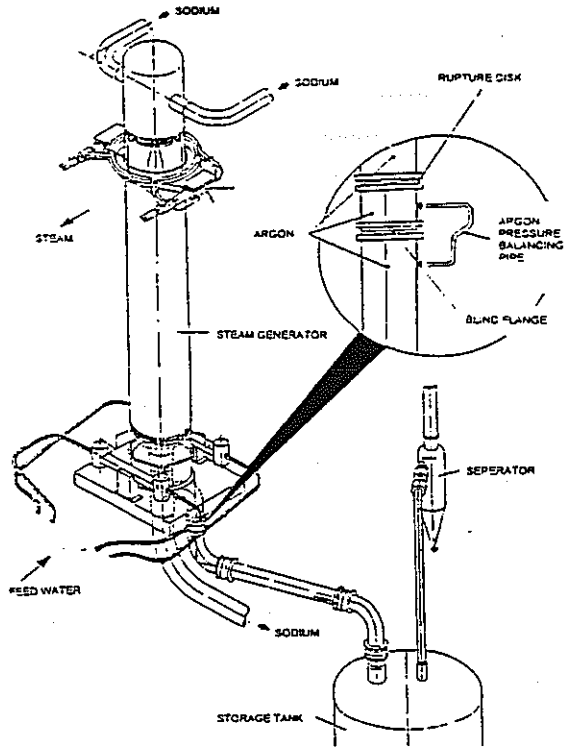


Fig 4.3 - Superphenix SG preparation for hydrotest

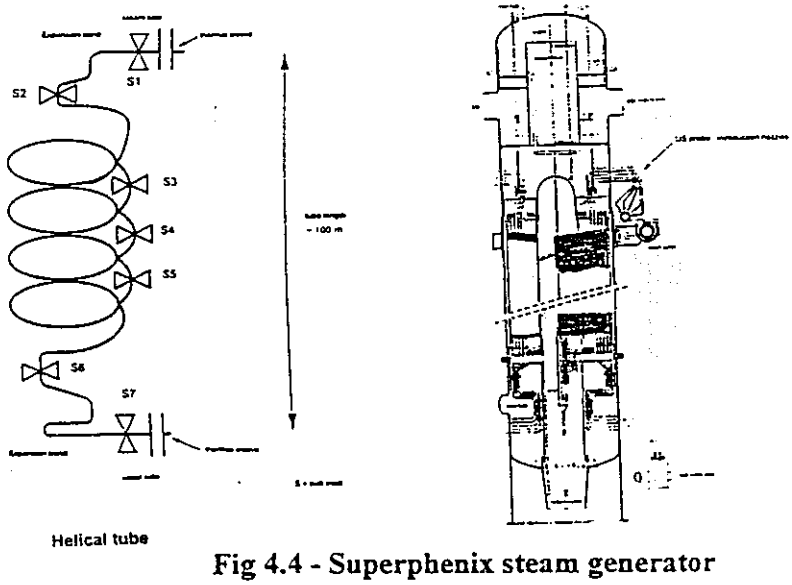


Fig 4.4 - Superphenix steam generator

Cleaning the tube with a brush propelled in the helical tube by water circulation,

- A check on the internal diameter of the tube by a gauge equally propelled by water circulation,
- Teleinspection of the tube by introducing a camera propelled by pushing its electricity supply cable by a special device installed near the SG (Figure 4.5),
- And lastly, control of the tube by ultrasonic (US) by three specialised probes (Figure 4.5) successively introduced by this same system of displacement. These three probes, of the multi-element type which ensures ultrasonic beam rotation by electronic sweeping, allow controlling the acceptable condition of the SG helical tube welds, search for possible circumferential cracks at the edge of these welds and enable measurement of tube thickness.

The main pieces of equipment developed for these operations (brush, gauge, camera and probes) were completely satisfactory, in spite of fairly laborious implementation with at first some difficulty in obtaining the circulating water quality required, then numerous problems of connection which, apart from the upheavals following the declaration of 19th June, have caused considerable delays in the schedule.

Analysis of ultrasonic testing acquisitions did not reveal any notable defect. These controls covered on average 84 % of the area to be inspected, as certain tubes were not able to be checked over their entire length following a blockage of the probes before completing their travel. Interruption of the work after the declaration of 2nd February 1998, at a time when ultrasonic control was just beginning to prove successful, limited inspection to 28 tubes instead of the 36 planned.

Also at the beginning of January, work on the reactor was started, covering more specifically :

- A series of special handling campaigns which were successfully carried out without any special difficulty on the one hand to replace primary cold trap cartridges and on the other hand, extract, then reinstall in the reactor after intervention four clad-rupture detection modules and two clad-rupture localisation modules. As far as the former were concerned, modifications carried out previously on other modules were made with a view specifically to improving mobility of sample nozzles, and on the latter, to make necessary modifications to remove mechanical operating faults experienced in 1996.
- Removal, then repositioning after upkeep, of all the control rod drive mechanism motors imposing in return complete requalification of the mechanisms which was carried through at the beginning of 1998 following special handling campaigns.
- Standard replacement of the electric motor of one of the primary pumps in the framework of the corresponding preventive maintenance programme.
- And lastly, in the framework of in-service inspection programme, visual check and dimensional controls by triangulation on the in-gas internal structures of the reactor by periscope, followed by inspection of the diagrid seat by transfer machines used for core subassembly handling (Figure 4.6). These operations have, on the other hand, called upon special handling campaigns for the introduction and removal of the periscope and the associated projector in the corresponding penetrations of the slab, as well as placing the reactor in a handling configuration : use of rotating plugs, transfer machines and the VISUS.

The visual inspection that had enabled checking that there were no abnormal problems of deformation in the structures examined, was used to put together a set of photographs of the internals above sodium level. As for the dimensional controls carried out by periscope, the only objective was to create a reference base with a view during subsequent controls to detecting deformation or possible shift of certain structures, especially pump stacks and the cover core plug. In the same manner, diagrid seat control carried out with the transfer machines by feeling 33 subassemblies judiciously spread over the diagrid, was made with a view to creating a reference enabling subsequent detection of possible evolution of the diagrid vertical and horizontal position and of its supporting structure.

In preparation of the handling campaign, important activity was devoted to the subassembly handling line by satisfactory check, in January, of the biological protection of the rotating lock, transfer chamber and spent subassembly cells with an 18,000 Ci source, followed by an important operation on cell ventilation to improve proficiency in hydrogen given off during subassembly washing. Since the summer, diverse campaigns for upkeep and perfecting equipment were carried out mainly on the irradiated subassembly washing installation.

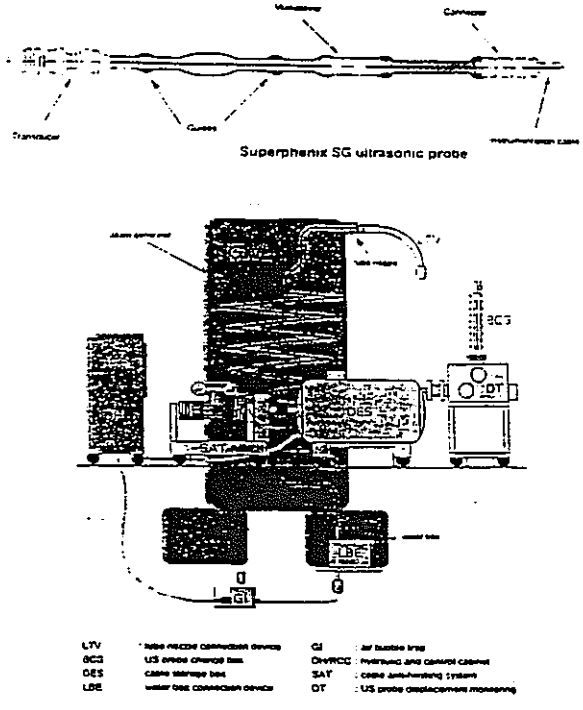


Fig 4.5 - Superphenix SG tube ultrasonic examination

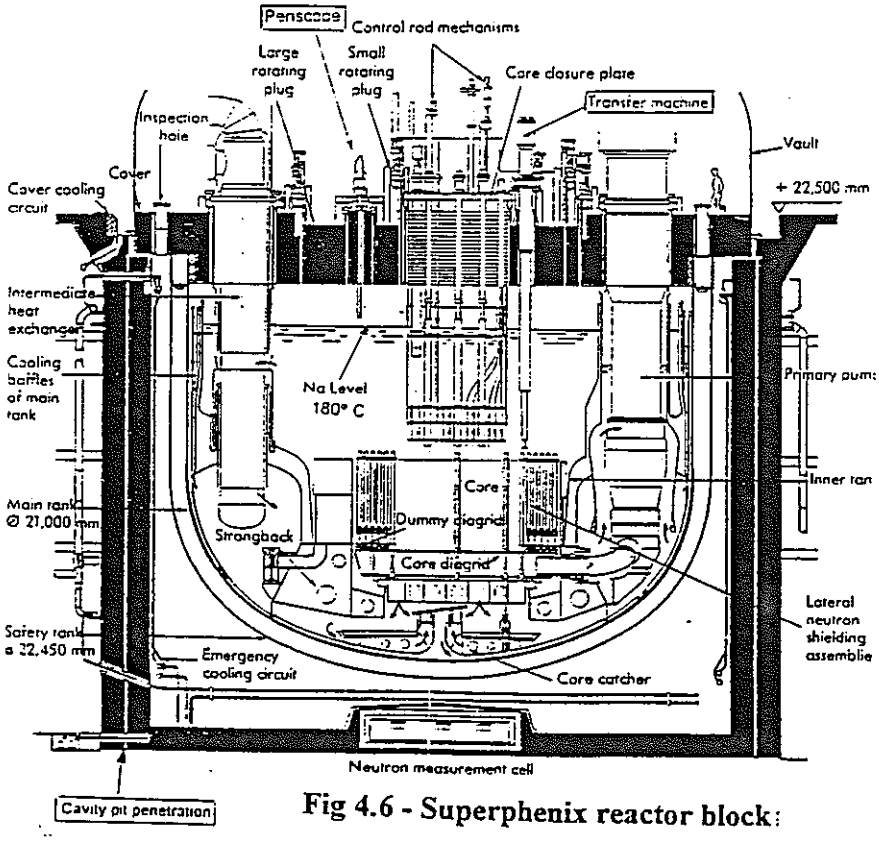


Fig 4.6 - Superphenix reactor block:

On the remainder of the installation, in the turbine hall where there were a number of hydrotests on pressure apparatus. in the water pumping plant, on diesel sets or electrical equipment cabinets, maintenance works progressed in line with the programme with a few marked interventions like factory overhaul of the last row of the HP turbine on train B. plugging on each condenser of about a hundred tubes that showed beginnings of wear, and replacement sections of piping on the cavity pit water cooling circuit where one had, on penetrations of the cavity pit rings (Figure 4.6), developed a slight leak following corrosion in damp environment.

Lastly, as far as radioprotection is concerned, the dose received by personnel on-site as a whole during this shutdown remained very low and did not exceed 0,046 hSv, considerably lower than the average yearly dose experienced on the French PWR park (≅ 1,3 hSv). Regarding work safety, there were 26 accidents for all EDF personnel and contractors as a whole with a total number of 2.200.000 hours worked.

DEFINITIVE SHUTDOWN PREPARATIONS

As soon as the declaration was made on 19th June to abandon Superphenix, the plant started studies to define the post-operative phase which will culminate in the definitive shutdown procedures for the installation corresponding to level 1 of the IAEA.

Generally speaking, these studies, which lead to the compilation of a definitive shutdown file under the control of the Safety Authorities prior to the publication of an Order, are scheduled five years before production stoppage.

In the case of Superphenix, the conditions of its stoppage and special character linked specifically to the presence of sodium have, on the other hand, imposed an adaptation of the procedural ruling. This has led to fixing a supplementary stage which consists in compiling a preliminary file to be presented to the Safety Authorities with a view to a preliminary Order giving the main steps of the post-operative phase and allowing core unloading and complete sodium draining. At Superphenix. these operations are not in fact covered by normal operating procedures.

As far as core unloading to the APEC pool is concerned, the studies are more particularly related to the two technical options such as core unloading and at the same time reloading a dummy core and, on the other hand, the unloading of the core without reloading a dummy core. This needs in return a study of seismic resistance in the course of unloading, as well as maintaining primary sodium temperature, considering the progressive modifications of the primary circuit hydraulics which no longer enables normal operation of the primary pumps.

As far as sodium draining, i.e. 3,500 tons of radioactive primary sodium, 1,550 tons of near zero radioactivity secondary sodium and 670 tons of inactive sodium from the old storage drum, are concerned, the studies relate initially to the resources available on-site for its storage. Then, they concern procedures to be implemented for primary sodium draining, maintaining it at temperature throughout the whole draining operation and elimination of retention fraction in the vessel.

An initial reflection is also under way on the future of the new fuel subassemblies of the second core already made and the spent subassemblies after their in-pool storage, as well as on the procedure concerning sodium destruction.

With a preliminary Order announced by the Government for the second half of 1998, the plant should be able to start core unloading in the course of 1999. It has been estimated that this operation should take 18 months, followed by sodium draining.

5. THE CAPRA PROGRAMME

5.1. INTRODUCTION

1997 was certainly a special year for the CAPRA programme, mainly for two reasons : the first one was the end of the demonstration phase aiming at proving the feasibility of burning plutonium in a fast neutron reactor. The second one was connected to the uncertainties on the SUPERPHENIX future there were throughout the year. The final decision of shutting down the SUPERPHENIX reactor finally stopped the experimental programme for demonstrating the plutonium consumption ability of a fast reactor and also the validation programme on the heterogeneous sub-assemblies (CAPRA 1 will not be irradiated), which represent one of the main innovations coming from the CAPRA studies.

However, another component of the CAPRA work programme was also to prove the ability of the CAPRA reactors to transmute significantly minor actinides. The long-lived waste management becoming world-wide a major concern (in France the law « Bataille » offers a very precise framework and schedule for the studies) it has been decided to reorient clearly the CAPRA programme not only on the minor actinides (MA) but also on the long-lived fission products (LLFP) burning, in addition (and probably to the detriment) of the plutonium consumption. Because the new CAPRA programme was defined recently (second half of 1997), no results will be presented in this paper, but only the detail of what we intend to do.

5.2. REORIENTATION OF THE CAPRA PROGRAMME

5.2.1. Historical account

At the beginning of the CAPRA project, launched in 1992 for demonstrating the ability of fast reactors to burn plutonium, the main working hypotheses were :

1. to burn plutonium as much as possible (Pu isotopic vector was that of a plutonium recycled twice in PWRs),
2. to contribute to the destruction of minor actinides,
3. to remain compatible with the existing technological know-how (EFR concept, oxide fuel, PUREX reprocessing)

This approach led to the conception of the reference core : the « CAPRA 4/94 core », based on a mixed oxide fuel. The plutonium content of this fuel was limited to 45%, in order to remain compatible with the PUREX reprocessing process, while ensuring the multi-recycling of the plutonium with an interesting burning rate of about 70 kg/TWhe. This new core was compatible with the EFR technology, so that it was possible to come back easily to the breeding, if need be.

To reach a higher burning rate, near the « theoretical » limit of 110-120 kg/TWhe, a pure plutonium nitride core (without uranium) had also been investigated. The feasibility of such a core has not been demonstrated yet, because of large uncertainties concerning the plutonium nitride fuel (especially the vaporisation behaviour under temperature transients) and the physics of this very innovative cores (without oxygen, nor uranium). However, this feasibility did not seem to be impossible. Furthermore, their use for burning very degraded plutonium should be a neat solution to offset the large Doppler effect loss due to the absence of uranium 238.

Scenario studies, performed with the CAPRA 4/94 layout, showed that less than 20% of the electrical power of the nuclear park had to be sustained by CAPRA reactors in order to multi-recycle all the plutonium stockpile coming from the PWRs.

5.2.2. The new CAPRA programme

5.2.2.1. MA management

To complete the previous CAPRA approach in order to burn MA, two different routes will be investigated :

1) The first consists of burning the MA and the LLFP in the CAPRA 4/94 core, which had previously been optimised for high Pu consumption. All the transuranian elements and LLFP are managed in the same type of reactor. MA can be burnt either by homogeneous recycling (i.e. with the MA diluted in the CAPRA fuel), or heterogeneous (i.e. with the MA concentrated in specific targets, separated from the CAPRA fuel). Depending on the type of MA recycling, different fuel and target cycles could be proposed.

2) The second scenario consists of separating the management of Pu and MA. Pu is burnt in CAPRA-type reactors, optimised for this function, and the MA are managed separately in dedicated reactors. This scenario means that there can be two different and independent fuel cycles, which have to be optimised for each function : one for U and Pu management and the second for MA management.

The second scenario leads us to conceive very innovative cores. The MA-based fuel with a MA content as high as possible (ideally 100%) is obviously completely new. Compared with the oxide CAPRA fuel, the MA compounds give it poor thermal properties, and the helium release, due mainly to decay of Cm^{242} produced by transmutation of Am^{241}, is very high. It should be also necessary not only to investigate various ways of improving the poor safety performance of a MA-enriched core cooled by sodium (low βeff, low Doppler effect, potentially high void reactivity effect,...) but also to investigate other types of fast neutron reactors perhaps more suitable for the function of MA burning. The results indicate that the use of lead or gas as coolant in place of sodium should be considered. A specific point is related to the fact that a 100% MA-core has in principle a high positive internal breeding gain, with a possible evolution of the reactivity towards higher values during the cycle.

5.2.2.2. LLFP management

The contribution of the LLFP to the radiotoxicité is important for the 100 first years, that is during the intermediate storage. Furthermore, if U, Pu and MA are recycled with low reprocessing losses (for example 0.1% of losses for each element), half of the waste radiotoxicité after 10^5 years is due to the LLFP. That shows how important is their management.

However, the quantity of LLFP involved in the PWR park is very high (#10 Kg/TWhe to be compared to the #4 kg/TWhe for the MA). Furthermore, their transmutation demands a significant neutron surplus because they are only « capturing » and not fissionable. So, their total management in fast reactors seems to be difficult without a large increase of the proportion of burners dedicated to their destruction. From this point of view, it is clear that the hybrid systems could have an advantage thanks to their ability of creating a large neutron fraction by spallation and their flexibility in terms of « safe » operating.

Another point to be noticed will be the necessity of a costly isotopic separation : the transmutable LLFP represents only 25% of the FP mass flow. It seems already difficult to introduce directly in the core all these elements and a choice must be made among all the LLFP to be transmuted : Zr^{93}, Tc^{99}, Pd^{107}, I^{129}, Cs^{135} and perhaps Se^{79}, Nb^{94}, Sn^{126}, Sm^{151}. For these latter nuclei, the residence time expected in fast reactor is close to the natural decay time. It is difficult to give surely priorities because of a compromise to be found between the global radiotoxicité (in this case Tc^{99}, I^{129} and Sn^{126} must be considered) and the risk of their release in the biosphere by migration (I^{129}, Tc^{99} and Cs^{135} are concerned). To be consistent with all our partners, the consequences of the management of the Tc^{99} and I^{129} will be firstly studied.

Preliminary calculations performed at CEA on the LLFP burning have shown that the best way for reaching high transmutation rates is to incinerate them in moderated targets S/As located within or just outside the core. Because the CAPRA 4/94 layout presents more than 50 unfuelled S/As in the core, the CAPRA heterogeneous core concept is thought to present very interesting features towards the LLFP transmutation.

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[4] *GLOBAL '97 : Future nuclear fuels for plutonium and minor actinides recycling. H. BERNARD et al.*

[5] *GLOBAL '97 : Strategies for the transmutation of americium. J.L. KLOOSTERMAN et al.*

[6] *GLOBAL '97 : Neutronic and fuel studies for americium target design. N. CHAUVIN et al.*

5.3. CONCLUSION

The new CAPRA programme, is now clearly reoriented to transmutation of MA and LLFP, in addition to the consumption of plutonium. In 1997 began the neutronic studies aiming at assessing the CAPRA 4/94 core conception changes in order to ensure the best transmutation rates for the MA and the LLFP. These calculations will continue in 1998 and will be followed by safety assessments on the new cores. A complete dossier should be provided in 2000.

In parallel, the first parametric studies aiming at assessing the feasibility of a fast reactor dedicated to the MA consumption have been launched. A large number of parameters like the core power, the MA and the Pu contents, their isotopic vector, the type of fuel, the interest of other coolant than sodium, ... are systematically tested vis-à-vis the core performance and the core behaviour under transients. The main objective is to orientate later the core conception studies. A preliminary dossier of core feasibility should be provided in 2000 and an experimental work programme aiming at validating the main conception options should be scheduled.

6. E. F. R.

6.1. ACHIEVEMENTS OF EFR STUDIES SINCE 1993, THE POSSIBILITY AND NECESSITY TO COME NOW TO EARLY CONCLUSIONS CONSIDERING THE EVOLVING SITUATION OF SPX.

6.1.1. Recall of the incentives for EFR studies since 1993

At the end of phase 2 of the EFR studies, in December 1993, EFRUG recorded the satisfactory progress of the EFR Associates work. EFRUG also pointed out the need to answer some open issues such as :

- responses to the recommendations expressed by a group of European Safety Experts (Ad Hoc Safety Club),
- the necessity of progress concerning the provisions for In Service Inspection and Repair,
- the need of further R&D validation of some of the options (e.g. Steam Generators),
- and the demonstration that a high load factor can be achieved in combination with ambitious operating and safety standards.

But at the same time EFRUG also came to the conclusion that the status of nuclear development in all the partner countries offered no prospect for an early development of Fast Reactors even if the basic incentives for developing the FR technology had been confirmed for the long term future.

Therefore the EFRUG Board defined a new strategy and in response EFR Associates launched a new work programme. This new strategy considered that the delay provided the opportunity to develop design further in response to open issues, to take benefit from the accumulated operating experience and experiments in reactors in operation such as SPX, and also to investigate the wider potential of the FR to manage the back end of the nuclear fuel cycle, in co-ordination with the CAPRA programme launched at the same time by the R&D organisations.

Outside EFR development, works and projects in Japan and Russia were also considered as opportunities to investigate alternative options thanks to the International Collaboration agreements between R&D organisations.

Although the assessment of potential alternative coolants was part of the EFR work programme, the reference remains the sodium technology. Therefore the main technical support of the EFR programme was the industrial validation through a commercial size reactor such as SPX, its Knowledge Acquisition Programme and associated R&D programmes.

SPX operation was also one of the incentives for the EFR work itself by participating in the preservation of the industrial design expertise to face unexpected events which might affect the availability of SPX.

The SPX operation together with other programmes such as EFR therefore formed a consistent scene for the future of the European FR development, scene which was expected to remain valid for several years.

6.1.2. The new conditions resulting from the SPX closure announcement

The above considerations explain why the announcement in June 1997 by the French Prime Minister of a premature closure of SPX operation led EDF to reconsider the usefulness of the EFR programme because of the loss of two main incentives :

- further operating experience and Knowledge Acquisition Programme of SPX,
- need of preserving specific expertise in support of SPX operation.

Consequently EDF, in close co-operation with BNFL which remain the only members of the former EFRUG currently funding EFR Associates work, have decided to close down the EFR work programme by the end of 1998 and have asked EFR Associates to come to conclusive synthesis of the work already performed. This decision was taken considering also that the current progress of the EFR work makes possible early conclusions in response to the EFRUG expectations.

Actually the early closure of SPX leaves open a number of questions concerning the FR technology :

- ability of the sodium technology to meet the industrial conditions of operation of a large nuclear plant,
- degree of validation of a number of options of the Reference EFR,
- flexibility regarding the breeding/burning capabilities of a large FR core.

However we expect that the EFR conclusions will make clear that beyond the specific events which have led to SPX closure, the basic sodium technology and Reference EFR design are not called into question from the design point of view regarding :

- its ability to respond to stringent operational and safety requirements,
- its potential for better economic performance than SPX making possible a future competitiveness considering missions of uranium savings and transmutation of nuclear wastes.

Therefore we expect that the conclusions on the Reference EFR will confirm it as being today the best digest of the European knowledge on Fast reactors, with identified ways of progress thanks to the innovative options which have been considered.

On the other hand as mentioned above, the industrial validation of the sodium technology is not fully confirmed today and therefore it is also important, as a precaution, to identify alternatives to the Reference technology. We expect from the preliminary evaluations of a number of alternative designs and coolants that recommendations could be made to select the most promising solutions. It seems that from this point of view lead - or lead-bismuth - or gas coolants are two possible candidates and it would be interesting to know which one appears to day as having the best potential (when Russian and others are promoting the lead).

Of course the previous position of EFRUG considering that there is no prospect for a utility decision of a new construction of a FR in Europe in the foreseeable future (at the scale of the usual industrial projects) remains valid. But also EDF confirm its interest in FR as the long term future of nuclear energy and therefore interest in the R&D work exploring the flexibility of the concept and further progress. The challenge will be to keep alive the knowledge together with great uncertainties concerning the time when industrial development will be necessary. In the frame of the R&D work and other specific agreements EDF expects that it should be possible to explore the capabilities of International Collaborations for an active survey of the technology progress.

This will be done without EFRUG which appears now no more suitable for the new conditions and has been dissolved with agreement of all partners in January 1998.

But first of all, the conclusive synthesis which is expected at the end of EFR studies will be kept as reference for this long term future. Therefore we attach great importance to the good quality of these documents which will have to reflect the considerable amount of work which has been performed since 1988 among all the associated partners.

6.1.3. Conclusions and summary of the decisions

Following the new conditions which result from the decision to close SPX and considering the large amount and good progress of work performed, EDF in agreement with BNFL decided to close EFR studies by the end of 1998 and ask EFR Associates to come to conclusive synthesis which will be kept as reference documentation for the future.

As a consequence the EFRUG Board have taken the decision for the dissolution of EFRUG.

Now interest of EDF, BNFL and other utilities will be mainly through association with the R&D programmes and corresponding R&D agreements.

6.2. PROGRESS OF EFR STUDIES IN 1997 AND FOLLOW-ON WORK PROGRAMME

6.2.1. Overview

The work has continued in 1997 focusing on the progressive evolution of the **EFR Reference Design**, which has remained the yardstick by which potential alternative designs and improvements are judged and the basis for participation in wider international co-operation.

Specific fast reactor issues have also been addressed in **Generic Studies** and **Innovative Concept Studies**, the results of which have been progressively fed back into the Reference Design where appropriate.

The **Feedback of Operating Experience** from fast reactors in Europe and world-wide as far as possible, has continued through systematic review of the design of systems and components ensuring that full benefit is taken in the EFR Reference Design.

Project **Specifications**, applicable **Codes & Standards** and **Computational Tools** are maintained to the latest state-of-the-art, so that penalising conservatism be eliminated progressively through the utilisation of advanced methods of analysis.

6.2.2. The EFR Reference Design update

6.2.2.1. Objectives

The major part of the work in 1997 has been devoted to the EFR Reference Design with the following objectives:

- To ensure compatibility of the design choices throughout all systems of the Nuclear Island and to establish / restore consistency of interface parameters,
- To achieve a consistent level of definition and validation of all systems and components as far as possible, and
- To introduce new features resulting from the Generic Studies and Innovative Concept Studies and the Feedback of Operating Experience, making design decisions based on the leading criteria of
  - competitive economics,
  - improved in-service inspectability and reparability,
  - better operational performance (availability, lifetime).
  -

6.2.2.2. Reactor structure studies

Detailed design features have been improved with regard to :

- The junction of the reactor vessel top flange with the steel ring embedded in the concrete vault, which carries the weight of the larger part of the reactor unit, including the reactor vessel and internals and the whole primary sodium coolant ;
- The peripheral bearings of the reactor top closure (the roof slab, made of solid steel plate) and the reactor cover gas sealing, aiming to improve access possibilities for in-service inspection ;
- The vault cooling system organisation, where the discharge of heat from the closed loop water system connected to cooling coils embedded in the vault concrete, is performed to the atmosphere through water / air heat exchangers in lieu of evaporating water from large sized buffer ponds.

6.2.2.3. Component studies

The steam generator units have been the subject of significant work this year, including :

- A reassessment of the technical-economical comparison of SGU variants, focusing on aspects of protection system and purification system requirements, which confirm that, overall, the EFR reference SGU with straight tube bundle in modified 9Cr 1Mo ferritic steel remains the lowest cost option, despite the fact that the A800 alloy (employed in a helical coiled tube bundle) is less sensitive to wastage and hydrogen diffusion ;

- The exploitation of test results from the ATLAS mock-up to improve the heat transfer models of the SGU thermal-hydraulic computer codes ;

- the confirmation of bursting disc reliability from tests performed on aged discs which have been utilised at Phénix.

6.2.2.4. Building and layout studies

The rectangular reactor building is now confirmed as the reference feature, replacing the previous cylindrical building. This follows on from generic studies on the containment performance which have demonstrated the effectiveness of a partially tight closure of the above roof area, by the so-called polar table, for mitigating the consequences of an assumed primary sodium fire on the roof *(see Section 6.2.3.1 below)*.

Detailed studies on roof layout have been carried out by a joint utility / designer working group, which has been established to investigate the issues. This allows to take benefit from the operator's experience e.g., to check the requirements for, and provision of, operator accesses with sufficient clearances and the removal of impeding obstacles.

6.2.3. Generic Studies

6.2.3.1. Containment performance enhancement

The analyses carried out for EFR show that the radioactive releases associated with the most severe accident conditions are compliant with the limits set in the European Utility Requirements (EUR).

Calculations show leaktightness of the building of 0.22% volume per day under 250mb overpressure, giving a large margin on the containment specification, which is 1% volume per day under 10mb. The ultimate pressure retaining capability of the reactor building is 600mb (g), whereas the maximum overpressure calculated for Plant State III is less than 250mb. Note: The Plant State III is an assumed load case for assessing the containment performance, corresponding to a large leakage of 1,500kg of primary sodium onto the reactor roof resulting in a sodium fire.

Retention chambers of reasonable size have been shown to be effective in further reducing the radiological releases and could be implemented should it be found beneficial in line with the ALARA principle.

The oxygen availability in the above roof zone limits the amount of sodium that can be burnt to 2,800kg. Further analyses accounting for the mitigating effects of high flame temperature on the sodium burning rate would result in demonstrating that the radiological consequences of such a primary sodium fire remain within EUR limits. This could remove the need to demonstrate that the sodium spillage hazard through the roof openings is less than the 1,500kg associated with Plant State III.

6.2.3.2. DHR function optimisation & reactor vault design

A preliminary study of an alternative decay heat removal system through an independent SGU feedwater system has shown the potential for improved operational flexibility at plant shutdown, with the associated advantage that the number of SGU dryouts could be reduced over the plant life due to the improved feedwater reliability. Further analysis is required to fully assess the cost / benefit balance of this design feature.

For the reactor vault, the floating liner design has been investigated as an alternative solution, but it is concluded that there is no cost advantage over the safety vessel itself anchored in the vault concrete (which is the reference feature), and it would require a large R&D demonstration programme, at least equal to that of the anchored safety vessel. In connection with the justification of the anchored safety vessel, progress has been made on the computational modelling of water migration in concrete and the mechanical analysis of the anchoring studs. However, additional experimental R&D support is required to enable further progress to be made.

6.2.3.3. In-service inspection and repair

The design studies and technique development are orientated towards in-situ ISI&R after the earlier studies on removability of internal structures and lay down areas above the reactor indicated the difficulty of such solutions.

The promising techniques for in-gas and under sodium inspection are agreed and their application to the EFR Reference Design confirms the design and R&D requirements. Various repair scenarios have been investigated to identify the needs for equipment and access possibilities.

6.2.4. Innovative Concept Studies

6.2.4.1. Features potentially integrable in the reference design

The preferred design is the reactor where the core support structure (strongback) is resting directly on the main vessel bottom head, instead of being connected to it through a shell. This design has significant advantages in reducing the ISI requirements on the core support load line.

The cylindrical inner vessel was also a feature proposed in view of potential ISI advantages due to the improved accessibility to the cold plenum structures. However, several complications, including the connections from the hot plenum to IHX inlets, need to be further investigated and considered against the ISI incentives.

6.2.4.2. Alternative reactor concepts

A comparative study of the Modular Fast Reactor (MFR, proposed by General Electric) with EFR has highlighted the key issue of the trade-off between the higher investment cost of the modular reactor, comprising a large number of smaller sized components, and the potential gains in availability due to the independent operation of the modules. However, a meaningful comparison of the economics can only be carried out against the background of identical specifications and regulations, which implies that some features of the modular reactor be altered to be compliant with the requirements imposed on EFR.

Former studies of the fast reactor without intermediate system have been re-evaluated, which confirm that the increased cost of the double wall tube steam generator compared to the single wall unit significantly reduces the potential economic advantage of the concept. This implies that few incentives remain for further studies.

With regard to the utilisation of alternative coolants to sodium, the on-going work concerns :

- A preliminary evaluation of the fast reactor cooled by lead and lead-bismuth alloy, which reveals a number of potential problems linked to corrosion, high freezing temperature and ISI difficulties, and
- A re-evaluation of the Existing Technology Gas-Cooled Breeder Reactor (ETGBR) concept, based on the UK AGR technology, which shows potential advantages in terms of inspectability and reparability, good flexibility for burning / breeding of Pu and minor actinides and the possibility of competitive economics. The studies continue on ISI&R issues related to the boilers in the context of a 60-year design life, and on the evaluation of severe accident behaviour.

6.2.5. Specifications, Codes and Standards

A joint working group has been formed between the utilities of EFRUG and the engineering companies of EFR Associates to carry out a comparative analysis of EFRUG Technical Specifications imposed on EFR with the European Utilities Requirements (EUR) for future NPPs. This aims to demonstrate that equivalence can be achieved, with due consideration of the technical peculiarities of the fast reactor, and to establish compliance of the EFR design with the EUR.

The development and application of design and construction rules dedicated to the fast reactor is a continuous process reflecting the experience being gained and improved understanding, with the aim to eliminate penalising conservatism through advanced methods of analysis.

The evolution of the RCC-MR code and production of topical reports by the DCRC (experts' committee on design and construction rules) continue. The DCRC reports recently issued or updated concern, e.g.:

- Design methods for flat and dished tubeplates.
- Design properties of the modified 9Cr 1Mo steel.
- Leak-before-break procedure for sodium retaining boundaries.
- Interaction diagram for ratchetting assessment.
- Design properties of carbon steels.

6.2.6. Feedback from Operating Experience

This part of the work programme consists of a systematic review of all EFR plant areas and components in the light of operating experience from European fast reactors. Experience from beyond Europe is used where available.

The reviews done so far have concerned the absorber rod drive mechanisms, the intermediate heat exchangers, the main sodium pumps, the primary auxiliary systems (sodium and argon cover gas), and the main sodium circuits of the intermediate heat transfer system. A technical synthesis with recommendations for the Reference Design evolution is issued for each area reviewed.

6.2.7. EFR Closing Programme

6.2.7.1. Rationale

As stated earlier in this document, the decision to close the Superphénix plant has resulted in the EFRUG's decision to close the EFR programme. Therefore, the EFR Associates have been requested to conclude the current studies in an orderly manner, by issuing a series of final synthesis reports which :

- Include a thorough account of the results obtained over the whole period of studies and the feedback from operational experience into the design,
- Identify outstanding issues and further studies and R&D support that would be required to complete the validation of the more advanced features of the Reference EFR, and
- Document the status of evaluation of innovative / alternative variants and the justification of design decisions.

The synthesis reports are to be completed by the end of 1998.

6.2.7.2. Organisation of final documentation

This will comprise three sets of documents :

- General documents, including e.g. :

- a comprehensive report on the Reference Design specification and justification,
- a synthesis on innovative concept studies.

- Overall synthesis reports per domains of general engineering, including e.g.:
  - an update of the 1993 edition of the non site specific safety report,
  - a synthesis on R&D needs for design validation,
  - a synthesis on feedback from operating experience,
  - a synthesis on plant performance,
  - a synthesis on in-service inspectability and reparability,
  - an update of the compendium of DCRC recommendations.

- A set of ca. 35 detailed synthesis reports per systems and components, covering all areas of plant, *i.e.*, core, primary & secondary systems, DHR, fuel and component handling, nuclear island building layout and civil works, I&C, electric's, auxiliaries, etc.

7. RESEARCH AND DEVELOPMENT

7.1. FUEL AND CORE MATERIALS

7.1.1. Materials for high burn up fuel

No further progress could be achieved this year in dose / burn up values in Phenix, and the maximum performances are those indicated last year.

Post-irradiation examinations conducted on highly irradiated S/A (Boitix 7 and 9) confirmed the excellent behaviour of the EM10 material for wrappers : very low deformation at 155 dpa gives confidence in the ability to reach at least 200 dpa ; on the contrary, the performance of 15/15 Ti cladding appears to be limited around 130 dpa. (fig 7.1).

So, the effort engaged on the development of ODS ferritic cladding remains of great interest ; significant steps were achieved on the fabrication process.

7.1.2. Modelling

A modulus on clad mechanics was added to Germinal 3, leading to Germinal 3M, the qualification of which was expected at the beginning of 1998.

In the frame of clad failure modelling, characterisation tests were conducted on $CsMoO_4$ compound taken from the fuel-clad gap ; tests on compatibility with sodium were achieved.

7.1.3. Capra fuel

The most noticeable achievement are the PIE of highly irradiated annular fuel pins with 32 % Pu content coming from PFR (examined in the frame of a PNC-CEA contract). The NDE are completed, and the destructive tests engaged on two pins. The first results indicate a general good behaviour.

7.1.4. Absorber element

No specific result is to be reported, the year activity consisted mainly in installing new equipments for post irradiation examination :

- Refurbishment of the cutting and sodium eliminating facility,
- Measurement of helium and tritium in irradiated B4C.

7.1.5. S/A/ Storage

A mock-up of a Superphenix S/A/ remained more than one year in the storage pool of « APEC » ; the first examinations after removal indicate a good behaviour of the EM10 wrapper.

7.1.6. Irradiation programme in Phenix

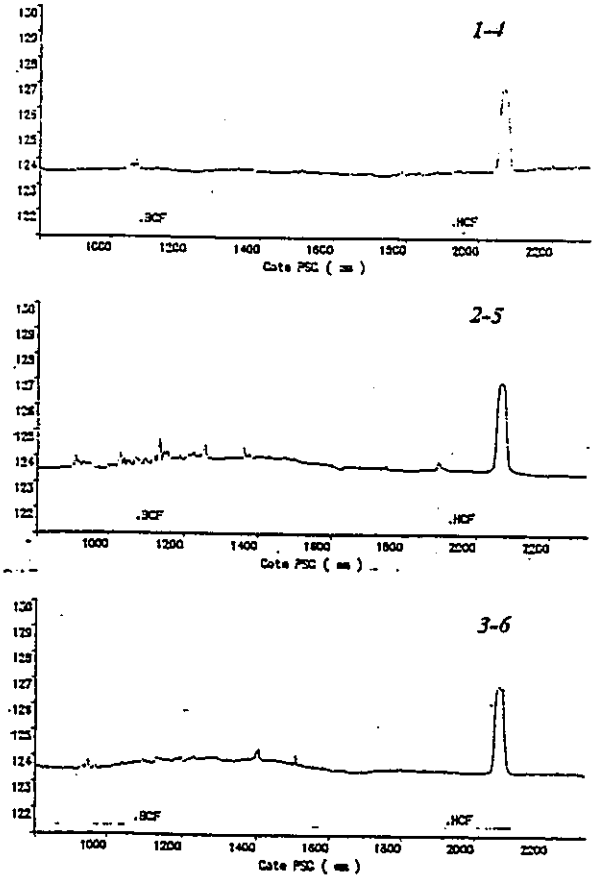
It was necessary to reassess the irradiation programme in Phenix, to take into account both the loss of Superphenix where a significant programme was foreseen, and the technical requirements coming from the 1991 law on radioactive wastes. This was done at the end of 1997 : a comprehensive set of experiments are planned for the last 7 cycles of Phenix ; the greatest part is devoted to studies of transmutation and incineration of MA and LLFP.

Fig 7.1 - NDE results of BOITIX 9

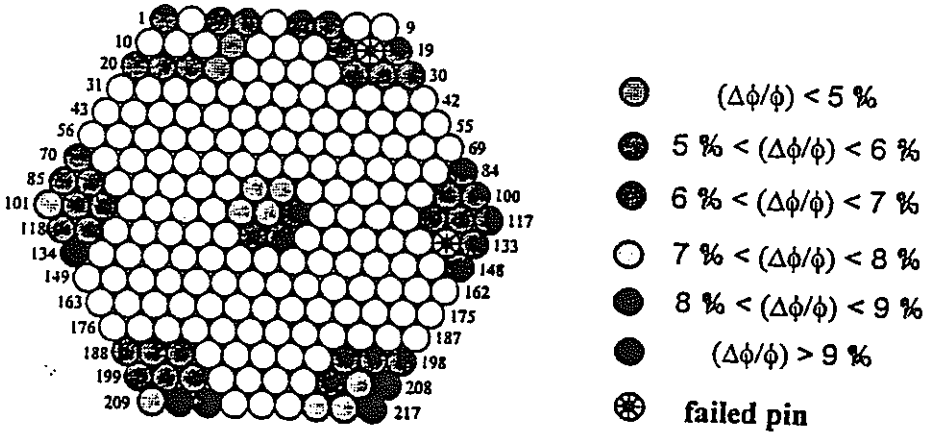
Hexagonal wrapper EM10

Cladding 15/15 Ti - Dose 155 dpa

distance across flats



Average cladding deformations



7.2. CORE PHYSICS

7.2.1. Codes

The outstanding fact is the completion of the ERANOS formulaire, the version 1.2. of which was delivered to EDF and FRAMATOME. It is a major step, concluding 10 years of development. ERANOS includes the ERALIB1 library, the cell code ECCO (2D) and a core code 3D transport, and takes into account the heterogeneity of control rods.

ERANOS was qualified on Superphenix results, and gave very good agreements on the main characteristics, without adjustment (fig.7.2).

7.2.2. Experiments

The « CIRANO » experimental programme in MASURCA was completed as foreseen, in October 97. The objective was to build an experimental database to qualify the fast reactor core formulaire for high Pu content (CAPRA programme) and to support the modifications foreseen for Superphenix and Phenix cores -mainly the removal of fertile subassemblies.

Two tests were conducted in Superphenix at the end of 1996, in the frame of the KAP programme. One was devoted to measuring the decay load power (fig 7.3), the other to measuring the sodium activation and the flux by irradiation of detectors in IHX, most of the interpretation was made in 1997.

7.2.3. Core Thermalhydraulics and Mechanics

The work consisted mainly in construction of code systems :

- ELOGE for thermalhydraulics of core and subassemblies, which replaces several different codes ;
- HARMONIE for core static mechanics, was qualified for configurations without friction, and the modelling of Phenix core is in progress.

7.3. SAFETY

7.3.1. Operating code

The OASIS code was developed to deal with normal and transient operating conditions ; its structure allows its utilisation for any kind of reactor.

A version qualified for Superphenix was issued in July. During the second half of the year, in the frame of the co-operation with CNNC, a Chinese attachee in Cadarache built a version for CEFR, demonstrating the feasibility of the code.

7.3.2. Cabri and Scarabée programmes

The abandon of Superphenix induced a important modification of these programmes. The new Scarabée cell, which was ready mid 97 will finally not be installed and the RAFT tests foreseen in Scarabée are dropped. The RAFT agreement with PNC was to be renegociated.

Two tests were conducted in Cabri during the first half of 1997 : LT4 test intended to study, on an annular fuel at intermediate burnup (6,4 %) the effect of a loss of flow followed by an overpower (LOF + TOP), ; the test was successful, PIE were not realised in 1997 ; a special attention will be paid to fuel squirting and fuel stub motion effect. RB1 test had for objective to assess what is the maximum allowable molten fraction in case of control rod withdrawal, in order to avoid fuel ejection if there is already a clad failure ; this may has important consequence on the maximum operating power.

Fig 7.2 - Comparison between calculated (C) and measured (E) Superphenix power distribution

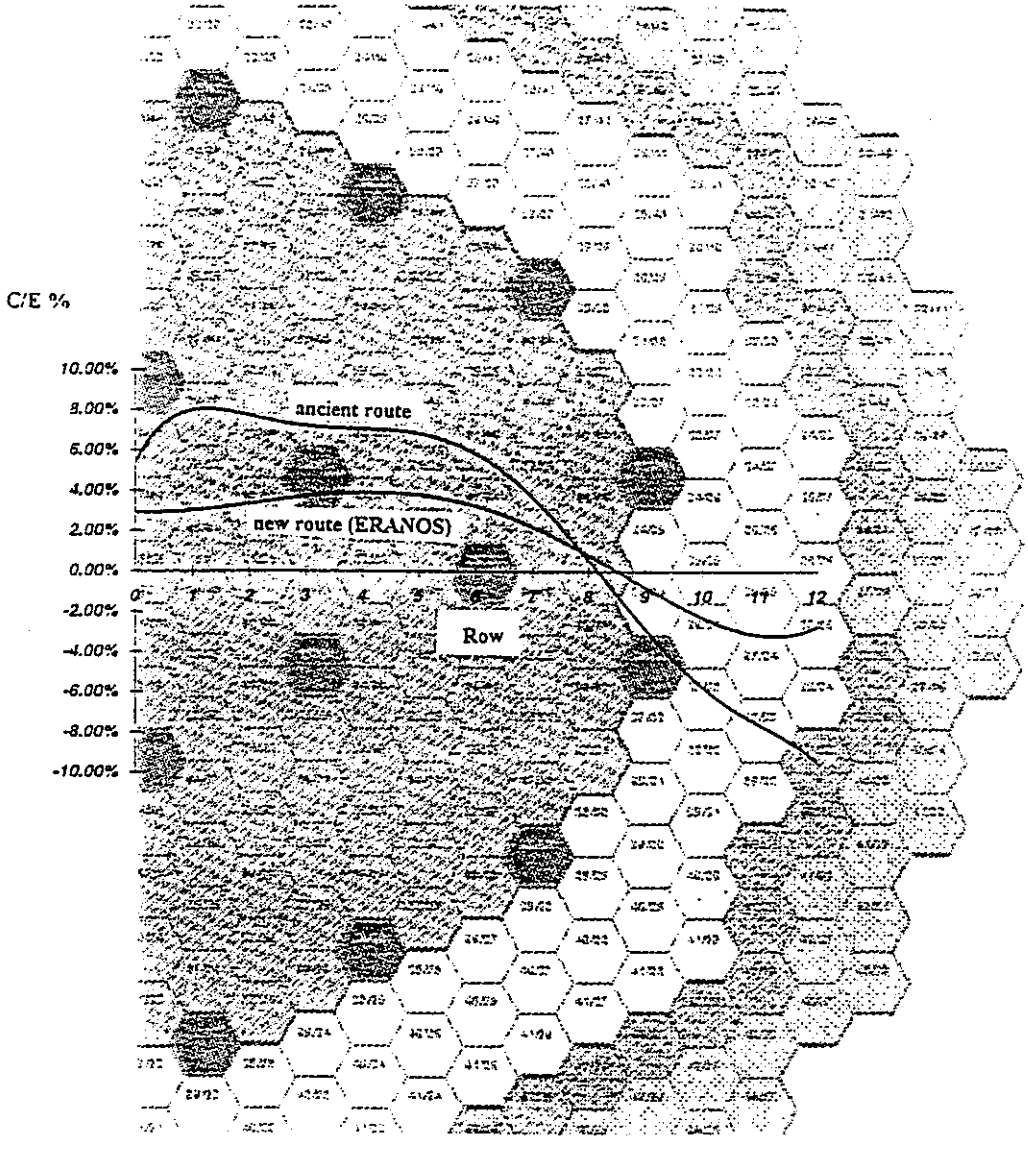
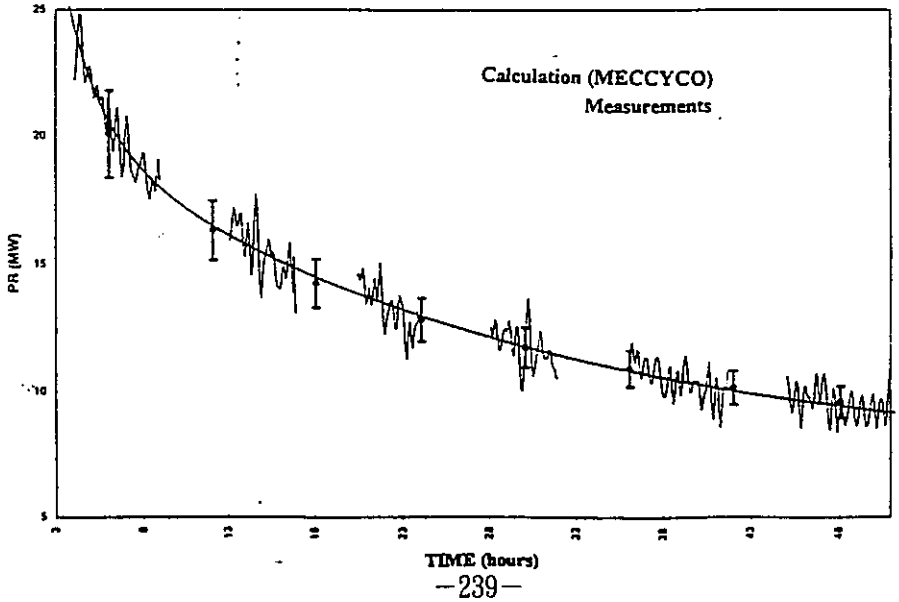


Fig 7.3 - Superphenix residual power test - (25/11/96)



For RB1, a pin pre-irradiated in Phenix to 7 % burnup was prepared in hot lab : a split was machined on the cladding, then sealed by a fusible alloy and coated by a 60μ Ni layer. Then, during the test in Cabri, the rupture was induced by a flow reduction, and then an overpower ramp led to central fuel melting : the target was 10 % molten fuel. the PIE indicates it should rather be around 5 % ; anyway, the important result is that the realisation was quite successful, and the result is clearly that no fuel ejection occurred. (fig 7.4)

7.3.3. Core Disruptive Accident Codes

7.3.3.1. SAS4A (initiation phase)

As indicated last year, a version SAS4A-96.1 was developed, including new models, among which fuel stub motion ; this version was finalised and made available at mid 1997. The version SAS4A96RO.1 was used for calculation on Superphenix cores. A study of the sensitivity of clad failure models was made on the BI6 Cabri experiment. Another achievement was the realisation of an interface between SAS4A and ERANOS.

7.3.3.2. SIMMER 3 (transition phase)

The collaboration with FZK and PNC proved to retrain quite fruitful and substantial achievements were obtained on the development of the code ; the thermalhydraulics part was qualified for corium pools (2D multiphase, diphasic flow and interaction corium / sodium) ; the neutronic park was validated by comparison with ERANOS and various calculations of molten core behaviour were conducted, for the first time.

7.3.3.3. Expansion phase

The first interpretation of a bubble expansion test, CARAVELLE, was made. On PBDOWN was published a final synthesis of the conducted work.

7.3.4. Control rod withdrawal accident

An assessment of the methodology is being conducted by Russian, in the firms of the CEA / MINATOM collaboration.

7.3.5. Sodium fires

For the codes FEUMIX and PULSAR, the work consisted mainly in writing instructions for code uses. An evaluation of the AIRBUS tests was made by specialists from university, an in the same time an internal detailed review of these experiments, which consist in simulating sodium air by water air interaction, a refined models was established, considering convection, gas compression, vaporisation and condensation of droplets on the walls. This will help to reorientate the programme, rather towards hydraulics aspects as jet fragmentation.

7.4. TECHNOLOGY

7.4.1. Sodium technology

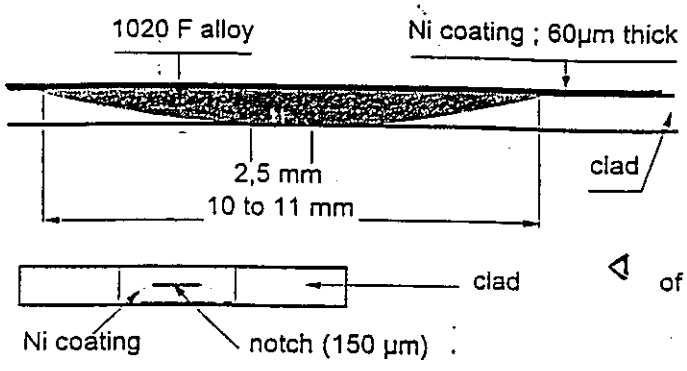
An experimental work on EPINAR loop demonstrated the mechanism of isotopic exchange leading to retention and concentration of tritium (simulated by deuterium) in a cold trap.(fig 7.5).

At the beginning of the year, assistance was given to the replacement of Superphenix secondary cold traps. and reference measurement made of the filling by neutron radiography.

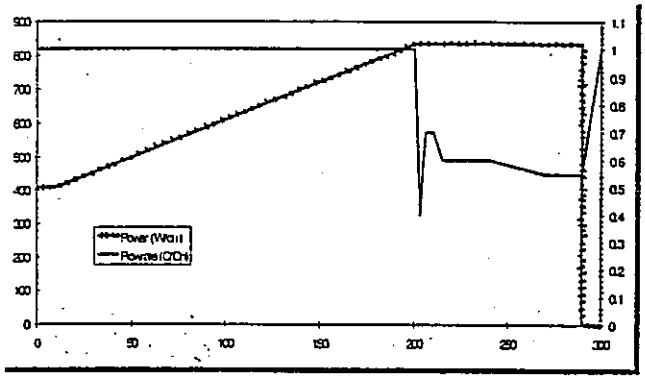
Code qualification was made for ANACCONDA (contamination transfer in sodium) and ELCESNA (caesium trapping).

The knowledge of sodium leaks behaviour and consequences remains a field of great interest ; a new loop. FUTUNA 2. was started in November 1997 ; its main characteristics is to allow work in low rate leaks range (till 0.05 cm^3/mm),which might be difficult to detect and nevertheless lead to significative corrosion ; two mock-ups were prepared for the first tests.

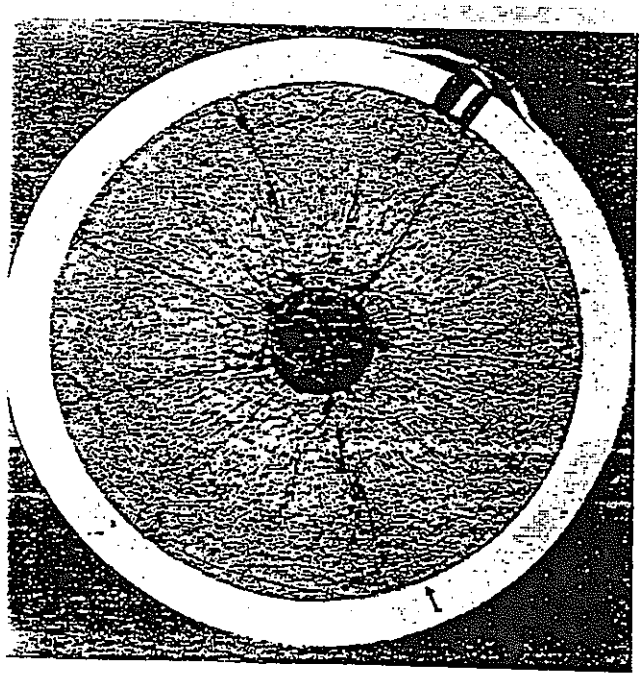
Fig 7.4 - Cabri RB1 test



RB1 notch : at mid-height of the pellet located at PPN



Cabri test procedure



PIE

molten zone ϕ = 6,7 mm

$\Rightarrow$ 45% of the pellet

Fig 7.5 - Isotopic exchange in a cold trap

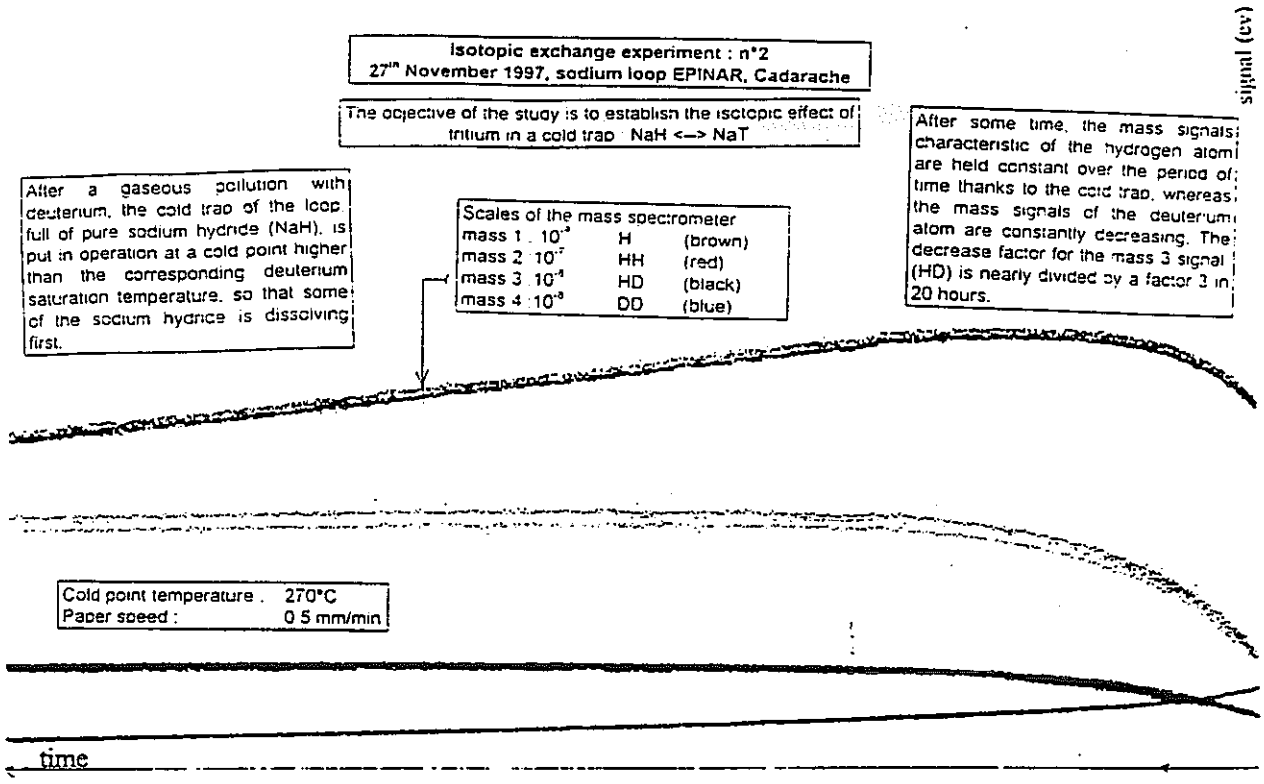
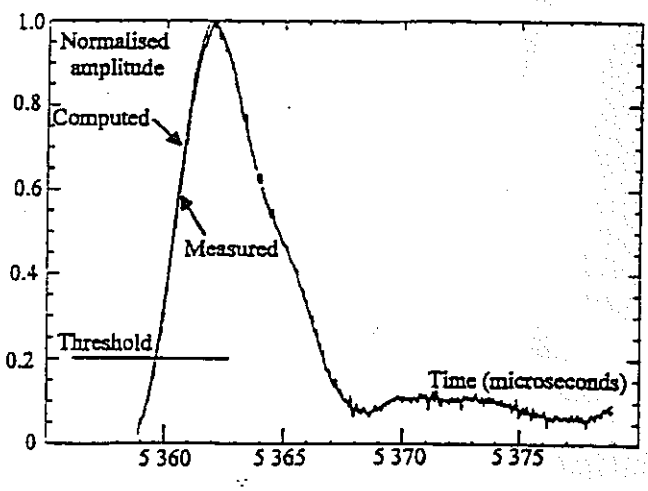
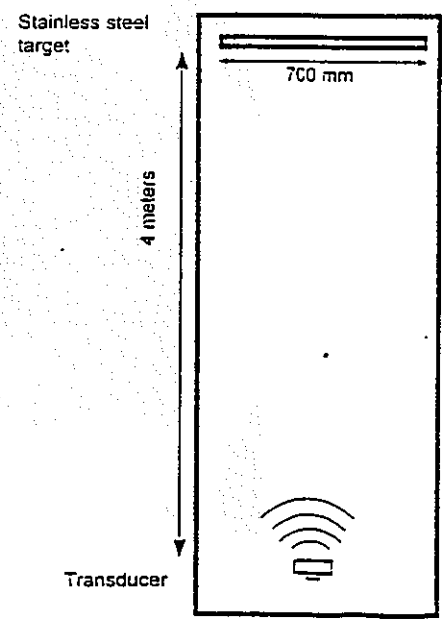


Fig 7.6 - Telemetry under sodium - Performance of SYTRAC.L code





1 : Envelopes of measured and computed echoes, for a 35 mm diameter transducer, at 4.5 MH

7.4.2. Thermalhydraulics

Improvement of TRIO code remains a central work in this field. The model of helicoidal steam generator was qualified on Superphenix tests.

One important issue remains the thermal striping and its consequences ; a key problem is the transfer of sodium temperature fluctuations towards the wall ; comparisons between calculations and test show that the calculations overestimate the fluctuation range, and so margins can be found in mechanical analysis of structures submitted to sodium jets.

7.4.3. In-Service Inspection and Repair (ISIR)

An extensive programme had been initiated, to answer Superphenix requirements (coming from safety Authority demands) ; the main results are the following :

- Confirmation of the possibility of inspection of the core support structure of Superphenix ; a model of propagation of transversal waves in the shell was built.
- For volumetric control under sodium, the availability of materials ensuring a good coupling and compatible with sodium is a key issue ; progress was made in the choice of performant materials.
- Qualification under water of the under sodium telemetry was achieved, and the SYTRAC.L code validated (fig 7.6) ; for in gas telemetry, above sodium, it was shown that the measurements were not disturbed in the case of targets wetted by sodium.
- First tests were successfully conducted for sodium elimination by gas blowing in representative conditions, on the MIRSA facility ; whereas in glove box, tests were made on surface cleaning by laser.

Phenix has also requirements for core support structure inspection ; tests were conducted to assess the feasibility of EMAT transducers, a part of them in the frame in the collaboration with Japanese.

7.5. STRUCTURAL MATERIALS

7.5.1. Alloy 800

The first results on long term creep tests show no difference between samples from tubes and plates used for RCCMR design data.

7.5.2. Ageing behaviour

On this important item, significant results were obtained :

- Austenitic welds : mechanical data (tensile stress, creep, toughness, crack initiation) are available on welds aged during 90000 h at 550 °C, and also creep at 650°C. Studies are in progress on relation between microstructure and behaviour ; heat treated welds are more ductile than raw welds.
- 321 SS : it was shown that ageing effect consist in significant decrease of creep resistance ; some test indicate an increase of about two decades of the secondary rate ; an experimental simulation on a CT sample was able to reproduce the relaxation induced cracking observed in Phenix ; this is a step towards a quantitative evaluation of the risk of cracking.
- 304 M ; some cracks observed in Phenix on sensitised zones were not clearly explained ; mechanical tests show no difference with reference material.

- CF8M pumps cast steel ; characteristics were measured (tensile, resilience, toughness) on samples coming from a Phenix secondary pump, having stayed 70000 h in sodium at 350 °C ; they show an embrittlement resilience divided by 4 - but remain acceptable. Tests at 400°C indicate that these values are stabilised.

7.6. STRUCTURAL MECHANICS

7.6.1. Design rules

RCC-MR work progressed in three directions :

- Progressive deformation : codification was improved, and a document issued, including validations on experimental tests ; simplified methods were compound and a recommendation given ; the Spray tests, allowing improvement on piping calculations, are completed.

- Advanced methods for piping calculations -a generic method was developed for nozzles and Ts ; a simplified method in elastic field was introduced as a modulus in CASTEM ; the introduction of elasticity is under consideration.

- Thermal striping -a synthesis was written, it is not possible to change the ΔT max rule without a refined study of thermalhydraulics and associated mechanical loading in walls.

7.6.2. High temperature fracture mechanics

- Creep fatigue propagation of cracks on CT specimens. The PROFIS programme is completed. It consisted on tests on SS316 L (N) CT25 at 600 °C and 650 °C and allowed progress in modelling : good accordance was found between calculated and observed relation between stress and crack opening.

- Crack propagation in a bended tube - TUBFIS programme allowed to establish a formulation of KI ; the prediction of propagation is good.

- Cracked tube under thermal shock - TERFIS 6 tests were realised since 1990, and a 7th is being prepared ; a cracked tube under traction load is submitted to thermal shocks (Tmax 600 °C) ; good interpretation are obtained with an intergranular damage model ; the evaluation of C* parameter is conservative.

- Propagation under creep-fatigue in plates -PLAQFLU- the method consists in fatigue or creep fatigue solicitation of large plates (590 x 350 x 25) at 650 °C, with or without welded junction (fig 7.7); a fast test and its interpretation were made in 1997.

- Leak before break argumentation - The CHARFIS- tests aim a good representatively : cracked tube under traction, cyclic loading ; the last one with axisymetric cracks were conducted.

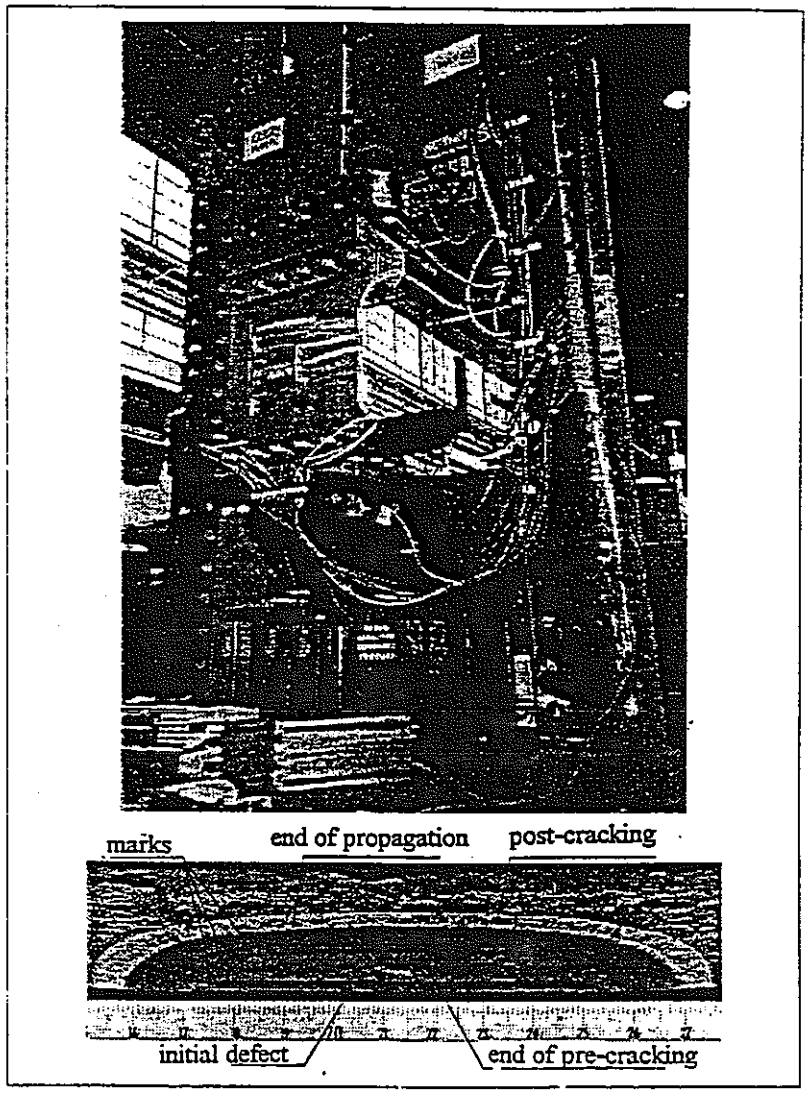
Synthesis were made on the correlation between defect area and flow rate, and between fatigue and tearing.

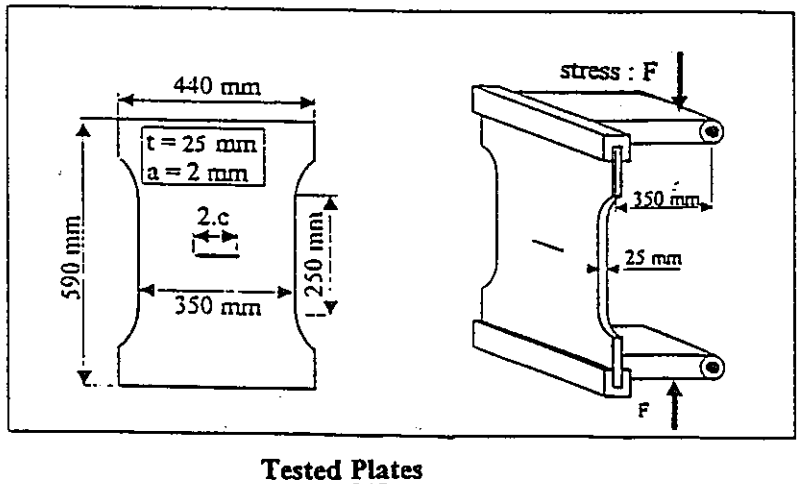
- Welded structures at high temperature SOUFFLE tests (flat plate specimen) and SOFA tests (thin - thick cylinders junctions) were interpreted. A noticeable result is that the efficiency coefficient for a welded junction is not relevant in fatigue and creep fatigue situations.

7.6.3. Seisms

Seismic behaviour of structures - The study was completed in 1997 ; 3D calculation methods were introduced in CASTEM to take into account components and fluid communications ; a good experience calculation agreement is obtained for buckling, not for dynamic buckling.
Seismic behaviour of cores ; tests were made in air on the SYMPHONY mock-up.

Fig 7.7 - PLAQFLU : high temperature propagation tests in plates





Tested Plates

The EFR Associates work programme since 1993 was settled to respond to the EFRUG Board strategy as defined at the end of « phase 2 » studies (1993).

Objective to answer some open issues such as :

- responses to the recommendations expressed by a group of European Safety Experts (Ad Hoc Safety Club),
- the necessity of progress concerning the provisions for In Service Inspection and Repair,
- the need of further R&D validation of some of the options (e.g. Steam Generators),
- and the demonstration that a high load factor can be achieved in combination with ambitious operating and safety standards.

The schedule included two main milestones :

- Innovative Concepts Catalogue in 1998,
- Up dating the Reference EFR (NSSS, economic study, status of validation....) in 2000.

In addition the programme included generic activities :

safety, specifications, codes and standard, international collaboration and feedback from operating experience.

The SPX operation together with other programmes such as EFR formed a consistent scene for the future of the European FR development, scene which was expected to remain valid for several years.

Announcement of a premature closure of SPX operation led EDF and BNFL to propose to their EFR partners :

- **dissolution of EFRUG (effective since January 1998),**
- **an early closure of the EFR programme.**

The considerable amount of work and good progress of the studies makes possible early valid conclusions possibly with some remaining open issues and weak points which should be identified in the EFR dossier.

Objectives for the EFR closing programme

Major part dedicated to the up dating of the Reference EFR documentation (Consistent Design in 1993).

Main expected outcomes for the Reference EFR :

progress in the safety case in response to AHSC recommendations and adaptation to EUR

limited up dating of the Non Site Specific Safety Report

progress in ISIR

evolution of ISI needs, identification of design progress for a better accessibility, possibilities to implement some of the innovative options for a better ISIR approach, interest in advanced technique.

evolution of the economic, plant performance

not a full reassessment of the 1993 economic report but trends resulting of design evolution since 1993 will be given for load factors, capital and production costs (French conditions).

catalogue of options and status of validations

list of the design options and possible alternatives (incl. CAPRA core integration).

identify the promising potential options for further development of the Reference EFR at a later date.

Alternative concepts and coolants

Since 1993 assessments of alternative designs (not compatible with the Reference EFR) have been performed, notably with the benefit of the International Collaboration.

It is expected that the final synthesis will give a clear view of advantages and drawbacks of each of these alternatives, in comparison with Reference EFR.

Concerning the alternative coolants preferred candidates are lead and gas (CO_2).

Future developments will be in the R&D domain : conclusion of EFR studies should identify priorities in the needs for further studies and validation of these alternative designs and coolants.

Final booklet for communication

A booklet presenting the final EFR design and results (same style as the 1993 version), to allow a wide distribution to all those interested.

EFR accounts for the best design options resulting from the current knowledge and operating experience

it will remain for a long time a European Reference FR with no new global design studies of a large FR such as EFR.

Open points, questions and views for the future

The early closure of SPX leaves open a number of questions concerning the FR technology :

ability of the sodium technology to meet the industrial conditions of operation of a large nuclear plant,

degree of validation of a number of options of the Reference EFR,

flexibility regarding the breeding/burning capabilities of a large FR core.

Reference EFR is today the best digest of the European knowledge on Fast reactors, with identified ways of progress thanks to the innovative options which have been considered.

But considering the far future for FRs, as a precaution alternatives to the Reference technology must be considered : incl. lead and gas coolants.

Now interest of EDF, BNFL as other utilities will be mainly through association with the R&D programmes and corresponding R&D agreements.

EFR Reference Design update

Reactor structure studies

Detailed design features have been improved with regard to :

- **The junction of the reactor vessel top flange with the steel ring embedded in the concrete vault,**
- **The peripheral bearings of the reactor top closure** (the roof slab, made of solid steel plate) and the **reactor cover gas sealing,**
- The **vault cooling system organisation.**

Component studies

SG:

- Reassessment of the technical-economical comparison of SGU variants : the **EFR reference SGU with straight tube bundle in modified 9Cr 1Mo ferritic steel remains the lowest cost option**, despite the fact that the A800 alloy (employed in a helical coiled tube bundle) is less sensitive to wastage and hydrogen diffusion ;
- Exploitation of **test results from the ATLAS mock-up**
- Confirmation of **bursting disc reliability** from tests performed on aged discs which have been utilised at Phénix.

Building and layout studies

The **rectangular reactor building is now confirmed as the reference feature**, replacing the previous cylindrical building.

Detailed studies on **roof layout** have been carried out by a joint utility / designer working group.

Generic Studies

Containment performance enhancement

EFR radioactive releases associated with the most severe accident conditions are compliant with the limits set in the **European Utility Requirements** (EUR).

Retention chambers of reasonable size have been shown to be effective

The oxygen availability in the **above roof zone** limits the amount of **sodium that can be burnt to 2,800kg**. The **radiological consequences of such a primary sodium fire remain within EUR limits.** This could remove the need to demonstrate that the sodium spillage hazard through the roof openings is less than the 1,500kg associated with Plant State III.

DHR function optimisation & reactor vault design

An **independent SGU feedwater system** shown potential for improved operational flexibility at plant shutdown.

For the reactor vault, **the floating liner design has no cost advantage over the anchored safety vessel.**

In-service inspection and repair

The design studies and technique development are orientated towards **in-situ ISI&R** after the earlier studies on removability of internal structures and lay down areas above the reactor indicated the difficulty of such solutions.

The promising techniques for **in-gas and under sodium inspection** are agreed and their application to the EFR Reference Design confirms the design and R&D requirements. Various **repair scenarios** have been investigated to identify the needs for equipment and access possibilities.

Innovative Concept Studies

Features potentially integrable in the reference design

The preferred design is the reactor where the core support structure **(strongback) is resting directly on the main vessel bottom head,** instead of being connected to it through a shell. This design has significant advantages in reducing the ISI requirements on the core support load line.

The **cylindrical inner vessel** was also a feature proposed in view of potential ISI advantages due to the improved accessibility to the cold plenum structures. However, several complications, including the connections from the hot plenum to IHX inlets, need to be further investigated and considered against the ISI incentives.

Alternative reactor concepts

Modular Fast Reactor (MFR, proposed by General Electric) : **higher investment cost of the modular reactor,** comprising a large number of smaller sized components, but **potential gains in availability** due to the independent operation of the modules.

However, a meaningful comparison of the economics can only be carried out against the background of identical specifications and regulations, which implies that some features of the modular reactor be altered to be compliant with the requirements imposed on EFR.

FR without intermediate system : the increased cost of the **double wall tube steam generator** compared to the single wall unit significantly reduces the potential economic advantage of the concept. This implies that **few incentives remain for further studies.**

Alternative coolants to sodium :

- A preliminary evaluation of the fast reactor cooled by **lead and lead-bismuth alloy,** which reveals a number of **potential problems** linked to corrosion, high freezing temperature and ISI difficulties, and
- A re-evaluation of the **Existing Technology Gas-Cooled Breeder Reactor (ETGBR)** concept, based on the UK AGR technology, which shows **potential advantages in terms of inspectability and reparability, good flexibility for burning / breeding of Pu and minor actinides and the possibility of competitive economics.** The studies continue on ISI&R issues related to the boilers in the context of a 60-year design life, and on the evaluation of severe accident behaviour.

Specifications, Codes and Standards

Comparative analysis of **EFRUG Technical Specifications** imposed on EFR with the **European Utilities Requirements (EUR)** for future NPPs. **Equivalence can be achieved**, with due consideration of the technical peculiarities of the fast reactor, and to establish compliance of the EFR design with the EUR.

The development and application of **design and construction rules dedicated to the fast reactor is a continuous process** reflecting the experience being gained and improved understanding, with the aim to eliminate penalising conservatism through advanced methods of analysis.

The evolution of the **RCC-MR code** and production of topical reports by the DCRC (experts' committee on design and construction rules) continue. The **DCRC reports** recently issued or updated concern, e.g.:

- Design methods for flat and dished tubeplates.
- Design properties of the modified 9Cr 1Mo steel.
- Leak-before-break procedure for sodium retaining boundaries.
- Interaction diagram for ratchetting assessment.
- Design properties of carbon steels.

Feedback from Operating Experience

This part of the work programme consists of a systematic review of all EFR plant areas and components in the light of operating experience from European fast reactors. Experience from beyond Europe is used where available.

The reviews done so far have concerned :

- the absorber rod drive mechanisms,
- the intermediate heat exchangers,
- the main sodium pumps,
- the primary auxiliary systems (sodium and argon cover gas),
- the main sodium circuits of the intermediate heat transfer system.

A technical synthesis with recommendations for the Reference Design evolution is issued for each area reviewed.

Closure of SPX

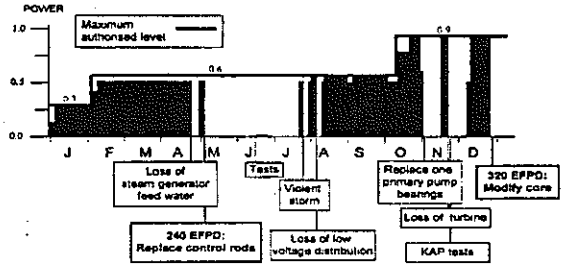
by G. HUBERT (EDF/SEPTEN)

31st IWGFR
12-14 May 1998 Vienna

SUPERPHENIX 1997

- After a very good year in 1996 (246 days coupled to the grid)

SUPERPHENIX 1996



- the reactor did not operate in 1997 = scheduled outage prolonged by politico/ administrative events and beginning 98 : definitive shutdown confirmed

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The SPX closure announcement

- In 1997 a long stop (> 6 months) was planned for :
  - rearranging the core (elimination of the first row of breeders because of the 1994 license requiring progressive evolution toward a burner),
  - hydraulic tests of three of the 4 SGs,
  - several maintenance and inspection operations, e.g. change of the 2nd sodium cold traps, some SG tubes inspection, etc.
- But beginning 1997 a Supreme Administrative Court (Conseil d'Etat) cancelled the 1994 license
- then elections June 97 has led to a new Parliament and a new Government (incl. Green Party):
  - P.M. statement, June 19 : «... not pursue projects whose costs are excessive and successes very chancy.....That's why the reactor called Superphenix will be abandoned. »

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1997 conditions

- After the PM statement in June : long waiting period for the definitive shutdown conditions (despite number of demonstrations in favour of resuming power for burning the 2 available cores) finally confirmed in Feb 98 by a Governmental Statement
- Lack of operating license :
  - did not prevent the planned test and maintenance operations being carried out successfully on the plant (but extended until Feb 1998 instead of lasting 6 months),
  - exceptions were any rearrangement of the core, fuel handling and storage which were not allowed in the administrative situation, e.g., originally planned :
    - replacement of the 3 « SAC » rods (emergency shutdown articulated rods),
    - rearrangement of the core by eliminating the first row of breeders,
    - introduction of CAPRA type sub assemblies and actinide burning test (Np).

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1997 main plant outage activities

- NDT on sodium circuits welds (X-rays), expansion tanks (US) and water-steam piping (X-rays and US)
- decennial hydraulic tests of 3 of the 4 SGs, ·
- replacement of the secondary circuit cold trap filters,
- several maintenance operations :
  - primary cold trap cartridges, clad rupture detection and localisation modules,
  - rod mechanism and rotating plug motors, replacement of one of the primary pump motor,
  - ISI of the reactor block internals (visual inspection and geometrical control of in gas structures, checking of the diagrid geometry using fuel handling devices)
  - check and improvement of the SA handling line and washing installation,
  - condenser, diesel and electrical cabinets, replacement of sections of corroded piping of the reactor pit water cooling system), etc.

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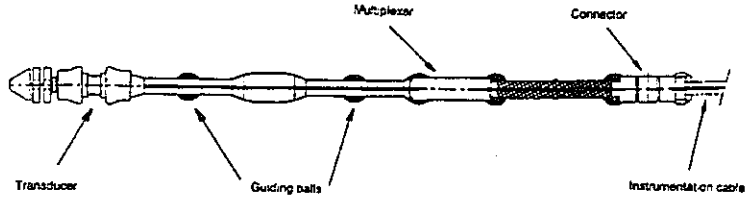
NDE (US) of tubes of one of the SG

- first inspection in industrial environment of 28 (over 36 initially planned) tubes of SG "D" by three US specialised probes (tubes length about 100m)



- allow inspection of helical tube welds, search for possible circumferential cracks and enable measurement of tube thickness.
- despite some difficulties of implementation (e.g. problems of connection) inspection was a success.

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1997/98 main plant outage activities (cont.)

- **NDE (US) of tubes of one of the SG (cont.)**
- Analysis did not reveal any notable defect. The checking covered on average 84 % of the area to be inspected (due to blockage of the probes in some parts of the tubes).
- Work has been stopped after the Governmental Statement of Feb 98, limiting inspection to 28 tubes instead of the 36 originally planned.
- **global R records and security results of the 1997 shutdown**
- dose received by personnel on-site as a whole during this shutdown remained very low and did not exceed 0,046 hSv,
- Regarding work safety, there were 26 accidents for all EDF personnel and contractors as a whole with a total number of 2.200.000 hours worked

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Closure of SPX

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SPX : the steps towards definitive shutdown state and dismantling

- In a letter to NERSA on 20 April 98 the Ministries of Economy, Industry and Environment, specified that due to the cancellation of the 1994 License, the application for restart of ....Oct 92 is now receiving a negative answer.
- The next legal step may be a new authorisation decree by the end of 98, to specify the conditions for core unloading and preparing sodium drainage.
- Then a further decree (around 2000) should specify the definitive shutdown state (level 1 IAEA) before further dismantling (level 2).
- Authorities recommended to avoid any new public enquiry. This requirement (opposite to previous opponents position!) may raise a number of technical and legal problems.
- The Parliament has recently launched a Commission to inquire on SPX and FR development. Outcomes expected end of June.

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SPX post operation : assessment of the possible options

- Preliminary studies since mid97 (should have started 5 years before definitive shutdown) notably to identify the possible options for :
  - the fuel (new or irradiated) and other elements of the core,
  - the primary and secondary sodium and the soda wastes,
  - the handling and storage facilities on the site, etc.,
- no technical difficulties, choices between the different possible options are only a question of economical optimisation.
- For the irradiated fuel the preferred option is to use the storage pond facility (APEC) which is a distinct nuclear installation from the licensing point of view. The new fuels (core 2) could be sent to MOX fabrication units after assemblies and clads dismantling.
- Core unloading (600 SAs) may start mid 99 for 18 months. Use or not of dummy assemblies is an open question in relation also with sodium heating.

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SPX post operation (cont.)

- Draining and storage of the whole sodium (3300t prim. Na, 1550t sec. Na + 670t from the former fuel carrousel storage) will necessitate additional storage capacities on the site
- After treatment (in Na sulphate) release in the Rhone river might raise no environmental concern (but may be political) :
  - neither for the salt : 0.1 mg/l = 3/1000 of the natural content of the river
  - nor for radioactive releases : in 2020 the whole primary sodium activity will be only of about 1 Ci (other sodium can be considered as non radioactive).
- Further studies will consider the future of the nuclear parts of the reactor and auxiliaries circuits and the non nuclear parts (secondary circuits, SG, water steam plant...)
- Interest for end of life inspections and expertises are also being considered in the frame of the R&D studies.

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SPX closure cost

- The total cost of SPX closure will be about FF 16.5 billion:
  - 4.1 billion for the residual debt,
  - 2.7 billion for fuel reprocessing (irradiated and new core),
  - 3.7 billion for post operation activities,
  - 5.8 billion for dismantling.
- Position of NERSA partners will have to be clarified

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Forschungszentrum Karlsruhe
Technik und Umwelt

Nuclear Energy in Germany

Report to IWGFR Meeting

May 1998

Nuclear Energy in Germany

- Controversy between Conservatives and Socialists/Greens still unresolved. No further attempt in 1997 to reach a solution.
- Considerable difficulties on CASTOR Transports
- Existing power plants operated in a most satisfying mode
- Produced electricity: 170 TWh, i.e. about 35% (rest: hard coal/lignite)
- Planning on EPR continued
- Future of Nuclear Energy in Germany will be strongly influenced by outcome of Federal Elections in September 98

Performance of German Nuclear Power Plants in 1997

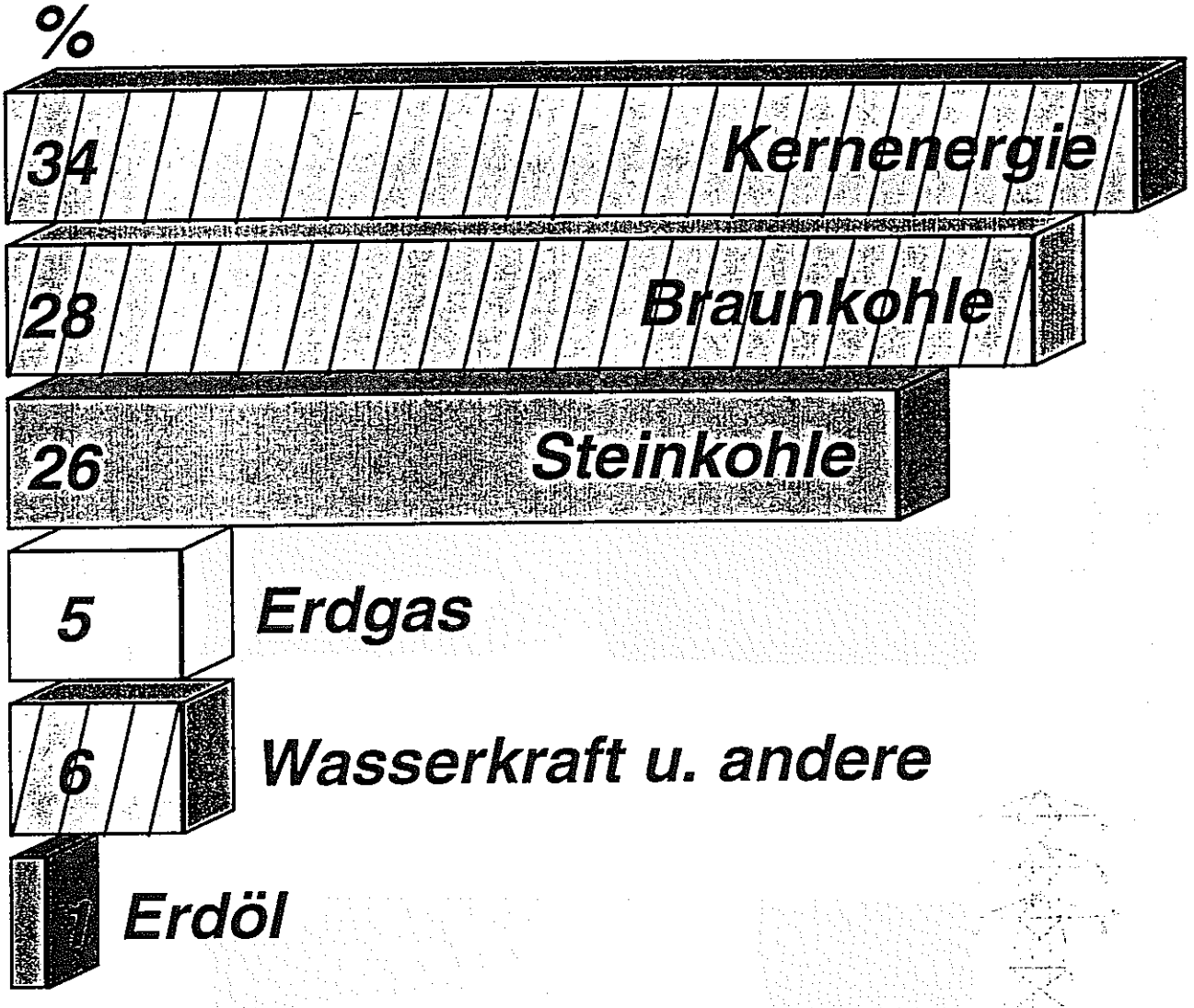
Power Plant	Type	Power MW	Output MWh	Fraction of theoretical output
Biblis A	PWR	1225	8518010	79.4%
Biblis B	PWR	1300	9043800	79.4%
Brokdorf	PWR	1440	11836621	93.8%
Brunsbüttel	BWR	806	5333225	75.5%
Emsland	PWR	1363	11235188	94.1%
Grafenrheinfeld	PWR	1345	10690679	90.7%
Grohnde	PWR	1430	12528660	100.0%
Grundremmingen B	BWR	1344	9710332	82.5%
Grundremmingen C	BWR	1344	9471929	80.5%
Isar 1	PWR	907	6272052	78.9%
Isar 2	PWR	1440	11538964	91.5%
Krümmel	PWR	1302	9671430	84.8%
Neckar 1	PWR	840	6723766	91.4%
Neckar 2	PWR	1365	10805600	90.4%
Obrigheim	PWR	357	2916180	93.2%
Philippsburg 1	BWR	926	6704410	82.7%
Philippsburg 2	PWR	1424	11706785	93.8%
Stade	PWR	672	5218720	88.7%
Unterweser	PWR	1350	10465823	88.5%
Summe		22180	170392174	87.7%

Electricity Production Costs in existing power plants (Source RWE)

Nuclear	4,7 Pf/kWh
Gas	7,9 Pf/kWh
Hard coal	8,0 Pf/kWh

Deutschland

Nettoerzeugung 426 Mrd. KWh

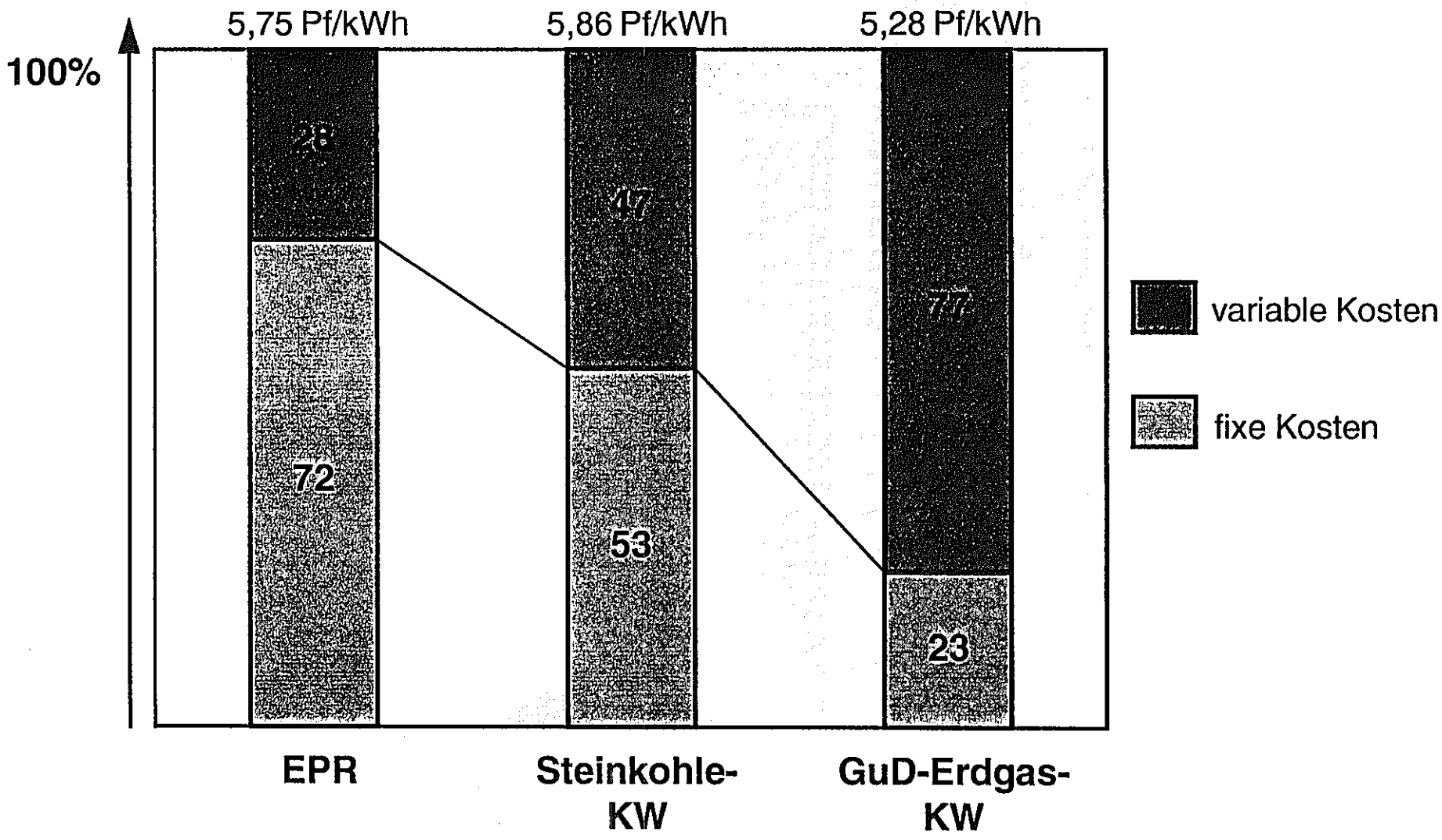


heimische Energieträger

Stand 1995

Zukunftschance: heimische Kernenergie

Stromerzeugungskosten neuer Kraftwerke im Vergleich (Inbetriebnahme 2000, Nutzungsdauer 30 Jahre, 7500 h/a)



Fast Reactor Activities in Germany

- Activities only at FZK. No industrial activities any longer

- Justification: Actinide burning
  Participation in European CAPRA Project

- But: No firm intention to built a facility

- Items:

  - Neutronics
  - Safety analyses
  - Irradiation experiment TRABANT

Neutronics

- Special needs because of unusual geometry of CAPRA cores
  - well established calculational route fails at CAPRA
  - Mesh size has considerable influence on power distribution
  - Nodal variational methods of advantage
  - special effort necessary to calculate sodium-void and control rod worth
- Pu-quality becomes poorer in CAPRA-type reactors

Safety Studies (1)

- SIMMER III

➢ TWODANT implemented

➢ Validation continued on LMR and LWR related experiments

⇒ SIMMER III demonstrates its large and excellent potential to simulate multiphase, multicomponent accident phenomena

- SAS4A

➢ New version released in summer 97 (SAS4A.REF96.Rel1)

➢ Various new modules, in particular modules describing fuel pin behavior

➢ Validation using CABRI experiments

Safety Studies (2)

- SIMMER III application

➢ Prevention and mitigation of recriticalities in CAPRA cores

➢ Special measures can prevent whole core pools with their large potential for energetics

- SAS4A

➢ CAPRA core: Influence of cooling strategies. Slight modification leads to high energy releases

➢ BN 800 (negative void): Only small energy releases in the initiating phase. But no coolable configurations. Transition phase!

GOVERNMENT OF INDIA

DEPARTMENT OF ATOMIC ENERGY

INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH

KALPAKKAM

STATUS OF THE FAST REACTOR DEVELOPMENT IN INDIA

April 1997 - March 1998

S.B.BHOJE

Director, Reactor Group

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP ON FAST REACTORS

31st ANNUAL MEETING, VIENNA. 12-14 May 1998

STATUS OF THE FAST REACTOR DEVELOPMENT IN INDIA
April 1997 - March 1998

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ABSTRACT

India generated 420 TWh of electricity during April 1997 to March 1998. The Central Electricity Authority has estimated a generating capacity addition of 57 GWe in the IX plan (1997-2002). The power and energy requirement by 2020 is estimated to be 325 GWe and 1950 TWh respectively. The installed capacity required to meet these requirements would be about 420 GWe, needing an average of 14 GWe power addition every year.Proposals have been received for private construction of plants for over 80 GWe.

The thermal power reactors performed very well during the year. RAPS -1 was recommissioned in March 1997 after successful remote repair done indigenously. The coolant channel replacement in RAPS-2 by indigenous technology, ahead of schedule, well within the budgeted manrem and cost is a proud achievement by any standards. It has been fitted with Zr - Nb channels and will commence operation shortly.

FBTR power was raised to 11.5 MWt and TG was synchronised to the grid for the first time in July'97. A continuous run of 22 d at this power was achieved. The maximum fuel burnup reached is 39,000 MWd/t without any fuel failure (targeted burnup is 50,000). No reactivity transient occurred during the year. After completion of major shutdown jobs, reactor will be started in May 98 and will be operated upto a power level of 15 MWt.

Detailed design of 500 MWe Prototype Fast Breeder Reactor is in progress. Conceptual design report has been prepared. Internal Safety Committee (ISC) has been constituted for the safety review at IGCAR. Atomic Energy Regulatory Board (AERB) through its PFBR-Project Design Safety Committee (PDSC) continued the safety review. SER has been prepared and is being sent to AERB for approval. Technology development of 45° sector of main vessel has been completed and the sector was received at site in April'98. Construction of PFBR is likely to start in the year 2001. It is included in the IX Plan proposal and Government has approved it. . The important design activities carried out during the year are: finalisation of secondary sodium flow sheet, conceptual design of secondary sodium pump and spend fuel storage bay, establishing temperature limits for the various categories of events, detailed thermal hydraulic and structural mechanics investigations of primary pump discharge pipe, analysis of leakage flow through mechanical seal between inner vessel-IHX, and feasibility study of reactor operation with (n-1) SG modules.

R&D in the domain of engineering development, thermal hydraulics, structural mechanics, metallurgy, non-destructive examinations, chemistry and safety are continued. Important experimental R&D works carried out during the year are testing of heat and mass transfer studies in cover gas, hydraulic tests on 1/3 scale grid plate model in air, creep-fatigue simulation on FBTR bellows as well as on PFBR control plug mock up, flow induced vibration studies, shielding mockup experiments at APSARA, corrosion studies and generation of material data for 9 Cr-1Mo, post -irradiation examination of irradiated fuel pins in FBTR, development work related to fabrication and property evaluation of PFBR fuel, design and developments of equipment for the pilot plant for the fast reactor reprocessing and study of sodium-concrete interactions.

1.0 INTRODUCTION

The year 1997-98 marks the 50th year of India's independence and the following may be said in introspection.

- The economic growth has been slow (~5 % average GDP growth). The country's rural population which is more than 70% is yet to be provided with basic amenities like health, drinking water and shelter. The agricultural production has increased 4 fold, thanks to adoption of high yielding varieties of seeds, increased irrigation, use of fertilisers etc. The industrial production increased

by about ten times. However, the increase in population by about three times, from 350 million to 950 million, has neutralised these gains considerably. Thus India's problem is two fold: to ensure growth and distribute this equitably to all sections of the society.

- In some field of science and technology, India has made very good progress. It has developed capability in design, construction and operation of PHWR based NPP, technology of complete fuel cycle i.e. exploration, mining, fabrication, reprocessing and radwaste management, and production of heavy water. In the field of space technology, the country has developed capability in the design, construction and operation of satellites and design and construction of satellite launch vehicles upto 1000 kg pay load. Development of supercomputer PARAM 10000 with a capacity of 100 gigaflops and with further scope of increasing it to 10 tera flops level is the latest achievement. India is the third country in the world, after USA and Japan to build supercomputers.
- The political stabilities of federal governments have become important issues due to coalition of large number of parties in the formation of the governments. This is the most important concern for the common man.
- The economics of subsidies in the inputs to agriculture, especially in the power sector, has resulted in the bankruptcy of the public sector state electricity boards. Fresh investments in the power sector has to necessarily come from private sector. Hence the government has opened the economy to private sector, both Indian and Foreign from the beginning of this decade. Unfortunately the process of privatisation has not been encouraging because of political, economic and procedural reasons.

The government has recently laid emphasis on strengthening grid network, modernising aged power stations, improving capacity factors and regulating the power tariff to revamp the public sector. Proposals have been received for private construction of plants for over 80 GWe. The government is planning to select, based on competitive bidding, offers for 5 large power stations (4-5 GWe). The Central Electricity Authority has estimated an additional generating capacity needed for the IX plan (1997-2002) as 57 GWe. The power and energy requirement by 2020 is estimated to be 325 GWe and 1950 TWh respectively. The installed capacity needed to meet these requirements would be about 420 GWe, requiring an average of 14 GWe power addition every year.

1.1 Electricity Generation

The electricity generation increased by 6.8% over the previous year because of high capacity factor of operating plants, and addition of capacity.

The total installed capacity in MWe as on 31.03.1998 :

Coal	-	55,969
Hydro	-	21,891
Gas	-	7,805
Nuclear	-	2,225
Wind	-	900
Diesel	-	376
Total	-	89,166

Electricity generated, TWh, in 1997-98:

Thermal (incl gas)	-	336.0
Hydel	-	74.4
Nuclear	-	10.0
Total	-	420.4

Capacity addition, MWe, in 1997-98 :

Thermal (incl gas)	-	3,048
Hydel	-	233
Total	-	3,281

Production of coal, oil and gas during 1997 is about 280 mt, 37 mt and 25 bcum respectively. The oil import has been about 40 mt.

1.2 Nuclear Power

The operating performance of the nuclear power plants for 1997-98 is given below.

Unit	Generation TWh		Availability factor %	Capacity factor %
	Target	Actual		
RAPS-1	-	0.472	48.40	36.50
RAPS-2	Shut down for enmass coolant channel replacement			
TAPS-1	0.960	1.180	90.10	84.20
TAPS-2	0.780	0.955	81.20	68.10
MAPS-1	0.905	0.729	51.40	48.90*
MAPS-2	0.905	1.164	78.30	78.10
NAPS-1	1.260	1.734	91.40	90.00
NAPS-2	1.260	1.715	91.40	89.00
KAPS-1	1.240	0.934	52.70	48.50**
KAPS-2	1.205	1.207	68.05	62.60***
Total	8.515	9.617		71.0

* MAPS-1 is under annual shut down since mid Nov.97 including mandatory inspection of coolant channels.

** KAPS-1 was under annual shut down from April to June 97 and was operating at reduced power due to high grid frequency for about 8 weeks (extended to attend generic failure in the shroud of LP rotor).

*** KAPS-2 was not permitted to synchronise during December 97 due to high grid frequency.

It is seen that the performance has far exceeded the target set and the generation this year has been more than during 1996-97 by about 0.6 TWh which is about 6% higher over previous year. The Nuclear Power Corporation has made a profit of 2.25 b Rs during the year.

A peer review of the Kakrapar Atomic Power Station (2 x 220 PHWR) was conducted in Jan. 1998 by WANO.

MOX fuel bundles loaded in Tarapur BWR have reached a burn-up of 10,000 MW d/t which is roughly 50% of the target. The fuel loaded is about a tonne and has given good experience in design, fabrication and irradiation.

1.3 Nuclear Power Plants under Refurbishing and Construction

- The two PHWR in Rajasthan were shut down since 1994, RAPS-1 for repair of over pressure relief device (OPRD) in the Calandria and the RAPS-2 for enmasse coolant channel replacement. RAPS-1 was recommissioned in March 1997 after successful remote repair done indigenously. The coolant channel replacement by indigenous technology, ahead of schedule, well within the budgeted manrem and cost is a proud achievement by any standards. RAPS 2 has been fitted with Zr - Nb channels and will commence operation shortly.
- Construction of 4 x 220 MWe PHWR is in advanced stages. The two units at Kaiga in Karnataka are expected to go critical in 1999. Similarly, the two units under construction in Rajasthan are also expected to go critical in 1999.
- 2 x 500 MWe PHWR are to be constructed at the site of BWR at Tarapur. Construction activity of Unit-1 is likely to commence soon.
- 2x 1000 MWe VVER are proposed to be constructed under Indo-Russian joint venture at Koodankulam in south India. The contract for detailed project report is to be entered soon after signing the supplement to the original inter-governmental agreement of 1988. The land for the project has already been acquired. The site investigations and establishment of infrastructural facilities have been completed.
- A combined multistage flash-reverse osmosis nuclear desalination demonstration plant of 6300 m^3/d capacity is under construction adjacent to the Madras Atomic Power Station, Kalpakkam at a cost of 8 m$. The plant is expected to be commissioned in the year 2000.

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 operation when turbo generator (TG) is not available. The reactor achieved its first criticality in Oct' 85. Since then it has operated at various power levels upto 11.5 MWt. The small carbide core with 27 fuel subassemblies has been licenced to operate upto 12.5 MWt with 320 W/cm linear heat rating of fuel. TG was synchronised to the grid for the first time, in July 97.

2.1 Reactor Operation

During the year reactor operated for 1594 h (fig 1 - histogram) and the following were achieved,

- All the TG commissioning tests were completed and it was synchronised to the grid for the first time on 11th July '97. TG was synchronised for 12 d in three spells at a maximum power level of 1.2 MWe.

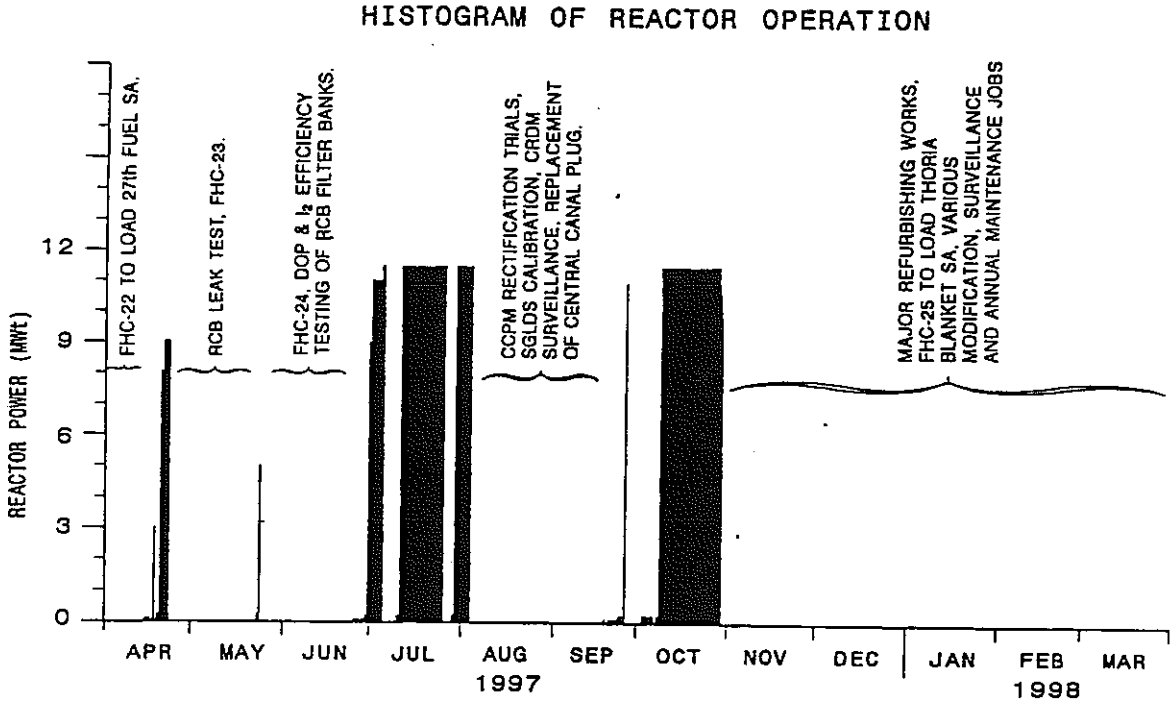

Fig 1

- The power was raised to 11.5 MWt for the first time in July '97 and a continuous run of 22d at 11.5 MWt was achieved.
- Irradiation campaign no.005 with 27 fuel subassembly core was completed.
- Maximum fuel burnup achieved was 39,000 MWd/t without any fuel failure.
- PIE of the central fuel subassembly, discharged after a burnup of 25,000 MWd/t, was completed and based on the results, clearance was obtained from safety authorities to raise the burnup to 50,000 MWd/t and operate the reactor upto a power level of 15 MWt by enhancing the LHR to 400 W/cm.
- No reactivity transient incident occured during the year.
- A 3 D computer code FARCOB was developed which can follow SA burnups and fuel compositions. The operation of 23 SA core during 1989-93 was analysed using the code. The reactivity loss was found to be 1.77 pcm/MW.d, including the estimated loss of 0.35 pcm/MW.d for axial fuel expansion. This compares well with the measured reactivity loss of 1.8 pcm/MW.d.
- During the year reactor tripped 7 times (2 scram + 5 LOR). In addition, there was one forced manual shut down and 4 planned shutdowns.

2.2 System Performance

FBTR has logged 18,712 h of cumulative operating time, out of which 5412 h has been at high power. The performance of reactor systems, sodium systems, CRDM, other safety related systems and auxiliary system were generally satisfactory. The primary and secondary sodium purity was maintained below a plugging temperature of 105° C. A primary sodium sample was taken from the reactor vessel in Feb. 97, 15 d after shut down..

The impurity levels, in ppm, were as follows:

Ag	< 0.02	Fe	3.30
Al	< 0.20	Li	< 0.04
Ba	0.02	Mn	0.03
Bi	0.03	Mo	0.07
Be	< 0.02	Pb	0.10
Co	0.006	Sb	< 0.02
Cu	0.05	Cr	0.60

The radioactivity levels of above sample were as follows

Radionuclide	Activity
Na^{22}	0.0219 μ Ci/g
Hg^{203}	0.0664 μ Ci/g
Ag^{110m}	0.058 n Ci /g
Sb^{124}	0.081 n Ci/g

SG were in operation for 5412 h without any water/steam tube failure. The four sodium pumps are operating very well and have logged 89,239 (P-E), 77,510 (P-W), 93,704 (S-E) & 74,050 (S-W) h without their removal..

The heat transport parameters achieved during 11.5 MWt operation, in July '97, are as follows:

Reactor inlet/outlet temperatures	:	334/420° C
Primary sodium flow through core	:	440 cu.m/h
Central fuel subassembly sodium Outlet temperature	:	470° C
Sodium temperature at SG inlet/outlet	:	418/285° C
Feed water temperature	:	188° C
Feed water flow	:	18.5 t/h
Steam temperature	:	395° C
Steam pressure	:	120 kg/sq.cm

Salient parameters of TG during the synchronisation with the grid, in Oct '97, are as follows:

Turbine speed	:	2880 rpm
Generator load	:	1.2 MWe
Overall expansion	:	4.8 mm
Differential expansion	:	+ 1.6 mm (Rotor expansion wrt casing)
Main condenser vacuum	:	620 mm of Hg
Hot well temperature	:	52° C
Exhaust hood temperature	:	60° C

Eccentricity	:	25 μ
Turbine inlet steam pressure	:	120 kg/sq.cm
Turbine inlet steam temperature	:	383° C
Steam consumption	:	10 t/h
Alternator voltage	:	6.6 kV
Alternator field voltage	:	68 V
Alternator field current	:	205 A
Excitor field voltage	:	11 V
Excitor field current	:	3.7 A
Power factor	:	0.8 lag
Frequency	:	48 Hz

2.3 Shutdown Jobs and Other Works

- A long shutdown was taken in Nov 97 to carry out the following major refurbishing works:
  - Replacement of neutronic channels with state-of-art system to reduce spurious shut downs and ease of maintenance
  - Loopwise segregation of instrumentation and power supplies of secondary sodium circuit to avoid common cause failure.
  - Replacement of uninterrupted power supply system with state-of-art system to reduce failures.
  - Replacement of one main boiler feed pump with indigenously manufactured pump due to non-availability of spares.
  - Augmentation of fire detection system.

The above works are in the commissioning stages.

- Both the secondary sodium loops were drained for executing the work of loopwise segregation of instrumentation and power supplies. The sodium systems were kept hot by recirculation of sodium from the secondary sodium storage tanks through IHX using storage tank surface heaters as the heat source.
- Carbon meter circuit (fig.2) in the secondary sodium east loop was commissioned. The cell output reading was corresponding to about 4 ppm of saturation carbon after 5 h of observation. The initial stabilisation time required is about one week, as expected from the laboratory experiments, to get the saturation carbon activity. The circuit will be put in service during the next reactor operation.
- Hydrogen in Argon Detector (HAD) in the secondary sodium loop (fig.3) was commissioned. Following are the observations: (i) evidence of natural convection flow of cover gas through HAD, (ii) response time and stabilisation time of signal about 5 min each (iii) sensitivity of the detector 0.4 V/100 vpm of hydrogen in cover gas. Hydrogen injection and calibration will be carried out prior to the next start up.

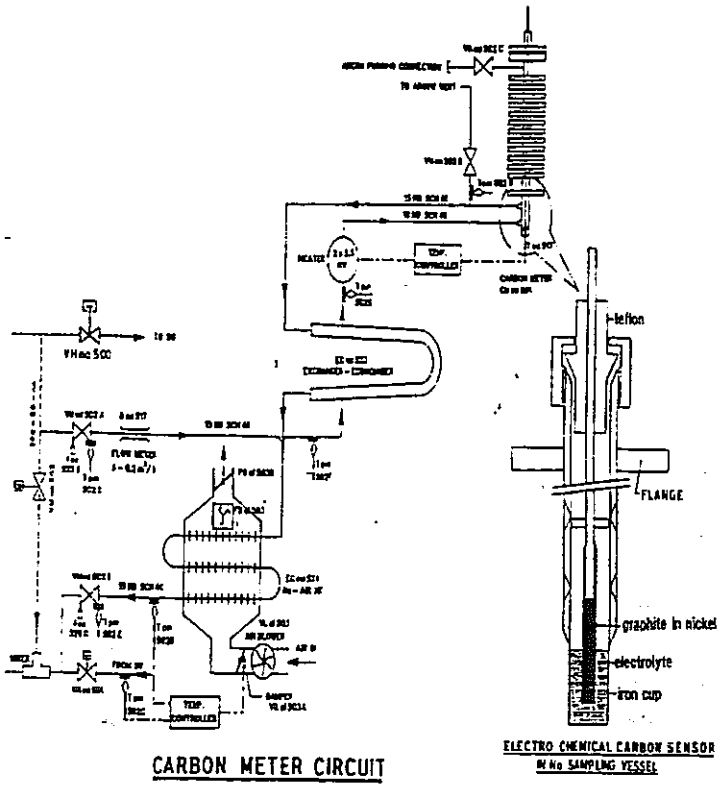

Fig 2

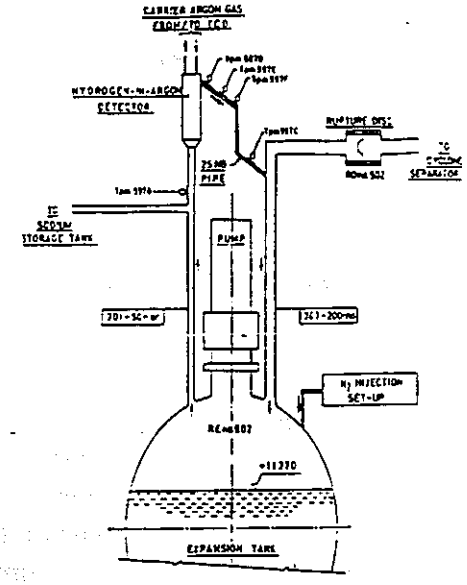

Fig 3 Argon Detector in Sec Sodium Loop

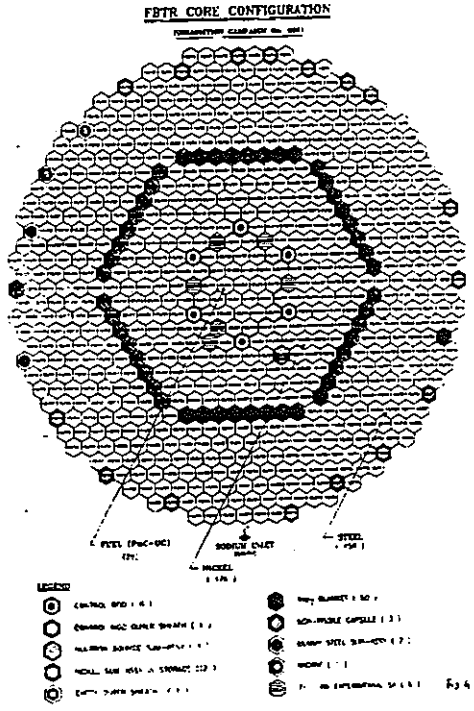

Fig 4

- Various modification works for the improved performance of different systems, major preventive maintenance works and surveillance activities were also carried out during the shutdown. The major among these are viz., online testing of SG safety valves at high power, replacement of central canal plug to make available three thermocouples for measuring the outlet temperatureof central fuel subassembly, biennial RCB leak testing, ultrasonic testing of SG shell welds and secondary sodium main pipeline welds, inspection of primary piping supports and SG hot roller supports, insitu replacement of silicon bellows for one CRDM and outer sheath `O' rings for 4 CRDM, fuel handling to load 48 ThO2 blanket subassemblies (fig 4).

- Number of LOR & SCRAM since 1993 (high power operation) is very high. An exercise was carried out to minimise LOR/SCRAM based on operating experience. As a first step, the number of trips on sodium pump drives was reduced considerably. Safety input parameters to reactor protection system is also planned to be reduced. Two offsite power supplies have been provided.
- The Core Cover Plate (CCP) remains stuck at 80 mm position. Rectification works to restore normalcy of CCP were not successful. This has caused an attenuation of about 7% in the measurement of fuel subassembly outlet temperature. After detailed investigations, it has been attributed to thermal hydraulics in the core outlet plenum at the fuel/nickel reflector boundary due to the present position of CCP. 3D thermohydraulic analysis was carried out to understand the phenomena and presented to safety authorities for clearance for further operation.
- During MBFP operation for readjusting SG safety valve spring setting, water leak was observed in the weld, between one of the boss and thermowells for RTD mounting, in steam subheader of one of the SG module. Rewelding was carried out at the leaked location after removing the thermowell by cutting and plugging with a dummy piece of similar dimensions so as to have the same flow restriction in the path.
- Spare control rod drive mechanism - lower parts are being manufactured & tested in air. One assembly is available as spare after all testing and two are under assembly stage. In one assembly, after testing in air, leak was observed in the translation bellows during helium leak test. The bellows had passed helium leak test, under vacuum, three times during the assembly. Analysis revealed that failure has taken place due to pitting corrosion in storage and absence of leak during assembly was due to blockage of pits by corrosion products which were dislodged during testing.
- A device to measure flow rate in each fuel subassembly in the core using Eddy current Flow Meter (ECFM) is being manufactured. All activities including mockup tests and sodium calibration of sensor will be completed by June 98.
- Ultrasonic under sodium scanner is being improvised for regular use by plant personnel.
- A client-server based computer system, named Maintenance Management Information System (MMIS) is being developed. The MMIS will be utilised for spares management, preventive maintenance and surveillance schedules, preparing equipment performance history, preparing list of all equipment by its function or circuit, listing of maintenance activities that can be done during various reactor states like operation, fuel handling. The MMIS is in the advanced stage of completion.

The radioactivity release to the environment during the year was 1218 GBq (33 Ci) Ar^{41}. The radiation dose to the operating personnel was 1.5 P-mSv.

2.4 Operation Programme

After completion of major shutdown jobs, reactor will be started in May 98 and will be operated upto a power level of 15 MWt to achieve the following:

- Irradiation of Zr-Nb alloy for PHWR programme
- Sustained TG operation with electrical output of about 2.5 Mwe
- Fuel burnup of 50,000 MW.d/t

3.0 PROTOTYPE FAST BREEDER REACTOR (PFBR) - DESIGN

The detailed design of PFBR is continued at IGCAR. The main design parameters are as follows:

• Reactor coolant	Sodium
• Primary circuit concept	Pool
• Thermal power	1250 MWt
• Electrical power	500 MWe
• Fuel	PuO_2-UO_2
• Core height /diameter	1 / 2 m
• Fuel pin dia / No of pins per SA	6.6 mm / 217
• Number of PSP	2
• Number of IHX	4
• Number of sec loops	2
• Number of SG per loop	4
• Number of TG	1
• Na temp at reactor inlet	670 K
• Na temp at reactor outlet	820 K
• Steam condition at SG TSV	763 K at 16.7 MPa
• Invessel fuel handling	2 rotatable plugs + 1 TA
• Spent fuel storage	In water pond
• Reactor shutdown systems	2
• Decay heat removal systems	2
• Containment building	RCC cylindrical shape
• Reactor site	Kalpakkam
• Reactor life	30 y

The important design activities carried out during the year are summarised below.

3.1 Conceptual Design

3.1.1 Conceptual Design Report

Conceptual design report has been prepared giving the design concepts chosen for the various systems and components and the basis for their selection. This report also includes a brief outline of design basis, i.e. safety classification, seismic categorisation, design basis events, applicable design and construction codes, computer codes used for analysis, material data and site data. The document is being discussed in a series of seminars attended by specialists of various related fields.

3.1.2 Core Design

Calculations were done for studying the reactivity management at BOL. 8 blanket diluents were found sufficient and could maintain the power at required value. However, the blanket pin linear rating crossed 450 W/cm during the first operating cycle. Use of SS SA

required 14 diluents but the power output was below full power. Further analysis is in progress.

Two dimensional transport calculations are in progress for evaluating the flux in the bulk shields and in remote locations like the under vessel detectors.

The reactivity resulting from the release of fission gas from the pins in the central subassembly has been calculated. The consequent transient effects and its detectability is being studied.

Design temperature limits for fuel and clad are established under various category of events. The temperature limit for fuel is 2973 K for category I and II events. Fuel melting is limited to 10 % and 50 % of pellet area under category III and category IV events respectively. For the clad, the temperature limits are 973 K, 1073 K, 1173 K and 1473 K for category I, II, III and IV events respectively. Both fuel and clad should maintain coolable geometry under category IV events. Bulk sodium temperature is limited to boiling point under all design basis events.

Acceptability of present thermocouple position (90 mm above the head of SA) in the event of SA blocking has been analysed. Thermal hydraulic analysis, taking into account the dilution of temperature due to the sodium coming from the inner rings, indicates that a blocking which leads to a flow reduction of 8-60 % in peripheral fuel SA can be detected. The hot spot clad temperature is within the limit up to 30 % flow reduction.

Seismic analysis of reactor core under vertical and horizontal excitations corresponding to 0.078 g (S1) and 0.156 g (S2) has been completed and verified that the possible reactivity oscillations is acceptable (<0.5$) during seismic event based on preliminary estimate of reactivity.

3.1.3 Reactor Assembly

I) Main Vessel

For the main vessel cooling, 24 cooling tubes with a single curvature of radius 4.45 m, has been selected based on the optimatisation studies on flow induced vibration and thermal flexibility considerations. The height of the jet at the outlet of the cooling pipe has been predicted for the given flow rate (~0.02 m^3/s) which matched well with the experimental observations. With this, the maximum temperature of the main vessel does not exceed 730 K during 20% power operation (fig 5) and 700 K during full power operation.

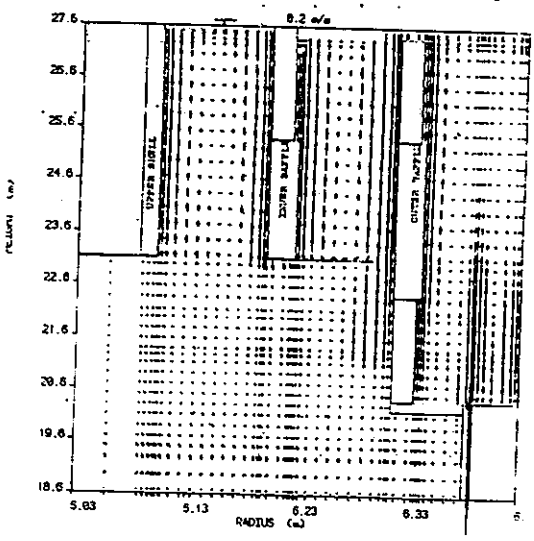

Fig 5a. Flow field in cooling circuit at 20 % power

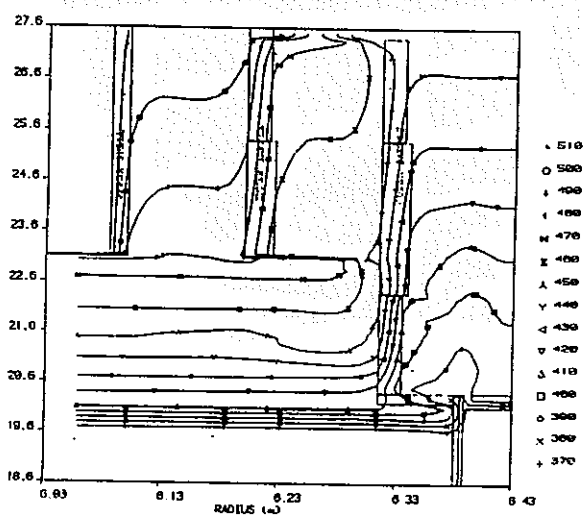

Fig 5b. Temperature field in cooling circuit at 20 % power

ii) Primary pump discharge pipe

In order to see the safety implication of two pump concept in case of a pipe rupture, the component is analysed in detail for various aspects. Detailed analysis has been done to confirm that the stresses under seismic loadings are well below the code limits. With the pessimistic assumption of presence of a crack of 3.75 mm depth (0.25 times the wall thickness) and 90 mm length (6 times the thickness) at the grid plate junction, fracture mechanics analysis has been carried out as per A16 procedure and demonstrated that through-wall crack is not developed to cause any sodium leakage even after imposing load cycles corresponding to 4 plant lives. Based on the analysis, there exists a case that even one pipe rupture can be considered as beyond design basis event. Preliminary thermal hydraulic analysis indicates that the maximum temperature of the sodium in the hottest channel is 1073 K and the hot spot clad temperature is 1254 K which are within the acceptable limits (category IV). As a defence in depth to reduce the possibility of pipe rupture, additional features to ensure the structural integrity such as two independent inspection techniques and independent inspection of the critical weld are under consideration.

iii) Inner vessel

Thermal hydraulic analysis has been carried out to know the temperature distribution on the inner vessel, particularly in the vicinity of Inner vessel -IHX Seal (IIS) due to 1% and 2% leakage flow. Analysis indicates that the leakage in the IIS reduces marginally the temperature drop across the inner vessel wall.

Buckling analysis has been completed for inner vessel using 3D shell elements without considering the leakage through IIS and also considering the thermal loads due to maximum possible leakage (2%). Fig. 6 show the stress distribution and buckling load factors respectively which respect the RCC-MR:RB limits.

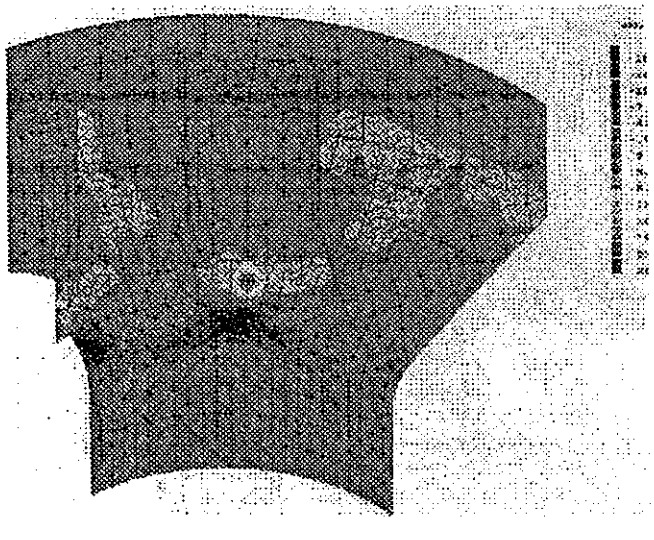

Fig 6a Stress distribution in the inner vessel

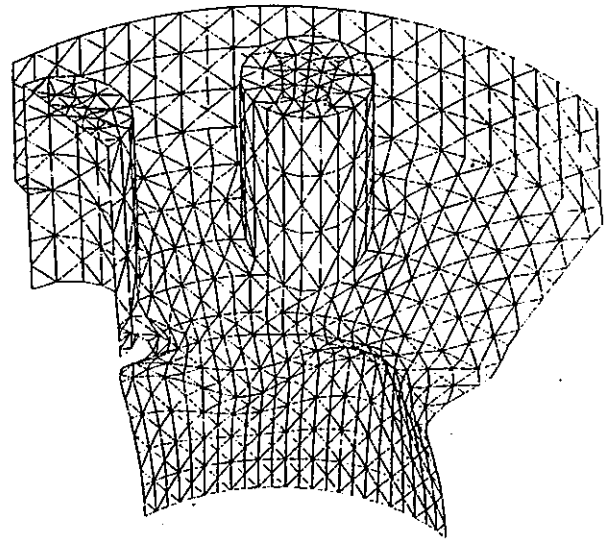

Fig 6b Buckled mode shape

Thermal hydraulic studies of the hot pool have indicated an unacceptable temperature gradient (about 750 K /m) due to sodium stratification with reference to thermal stress fluctuations on the metal wall surface in the redan due to possible oscillations of stratified layer (fig 7). Subsequently a perforated cylindrical skirt has been introduced in the bottom portion of the control plug. The thermal hydraulic studies show that the skirt has reduced the temperature gradient in the stratified layers to an acceptable limit (30 K on the metal wall for the oscillations of ± 150 mm).

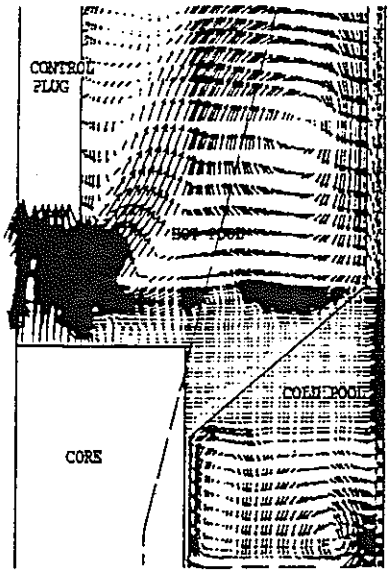

Fig 7a. Flow pattern at full power

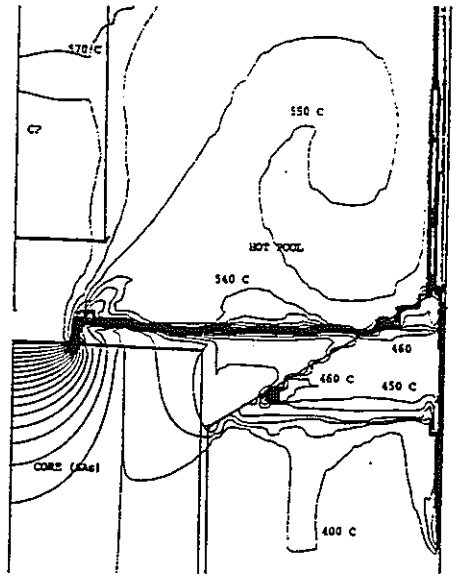

Fig 7b. Isotherms at full power

iv) Absorber Rod Drive Mechanism (ARDM)

ARDM drawings for prototype manufacture were completed. The dashpot details and spring design were modified to comply with RCC-MR:RK rules. The dashpot and spring were designed following normal practices.

v) Thermal hydraulics and thermomechanical studies

Effect of leakage of hot sodium through the affected IHX sleeve valve is studied for the leak rate upto 10% of IHX flow in the case of reactor operation with one secondary loop. The computed temperatures on the reactor assembly components are acceptable from the structural mechanics as well as core temperature monitoring considerations and reactor operation is feasible with one secondary loop for the design leak (2 % maximum) .

Thermal striping limits have been established in compliance with the creep-fatigue design rules of RCC-MR: RB (1993). The temperature limits are 40 K for the locations where the creep-fatigue damage is very high (about 0.99) and on the isolated locations where creep-fatigue damage is negligible (about 0.01) the acceptable limit is 55 K. Accordingly the temperature fluctuations are limited to 55 K for the striping regions of the control plug and 40 K for inner and main vessels.

The temperature asymmetricity in the cold pool components due to reactor operation with (n-1) SG module has been quantified. The associated structural mechanics analysis for the risk of thermal fatigue and local thermal buckling has indicated that the operation with (n-1) SG module is feasible.

vi) Preheating of reactor assembly

Preheating is done by circulating hot nitrogen gas inside the reactor to a temperature of 150^{+30}_{-0} for a duration of about 15 d with heater capacity of 100 kW and nitrogen flow rate of 10 kg/s. For the effective heating, it is recommended that the inlet of the hot gas should be at the hot pool along with heating in annular space between main and safety vessels.

3.1.4 Sodium Circuit

i) DHR

Fault tree diagram has been prepared for safety grade DHR reliability analysis.

ii) Primary sodium pump

The design of primary sodium pump for the revised speed of 680 rpm was carried out. The cavitation criterion chosen is erosion free at the NPSH. Normally both the pumps are started and run together. However, there would be certain instances such as trip/seizure of one primary pump and initial commissioning where only one PSP could run. The safe range of operating speed under different modes (2/2 and 1/2 mode) is worked out. The maximum and minimum speeds under any mode are limited from cavitation and hydrostatic bearing limitations respectively.

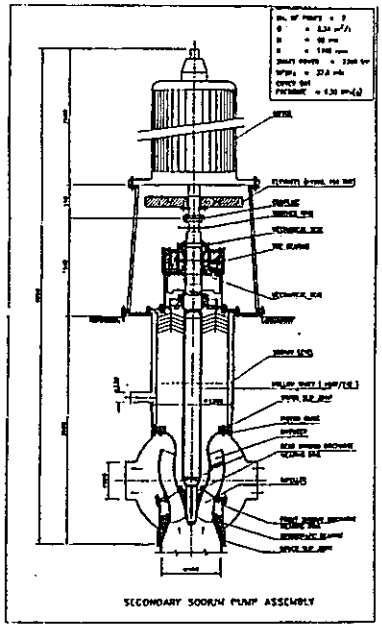

Fig 8 Secondary Sodium Pump

iii) Secondary sodium pump

The conceptual design of secondary pump has been finalised. Each of the two secondary loop is provided with centrifugal pump which delivers a flow of 3.34 m^3/s at a head of 65 mlc. The pump is of single stage, bottom suction design (fig 8) and is located in the cold leg of the secondary circuit, at a lower elevation with respect to the steam generator to reduce the piping length. In order to keep the leakage flow small, and consequently the rating of the electromagnetic pump reasonable, as well as to make the pump compact a high operating speed of 1460 rpm has been selected. This results in a specific speed of 120 rpm/(m^3/s)/ mlc. The pump overall efficiency is 79% and the shaft input power is 2330 kW. A high cover gas pressure of 450 kPa (abs) has been employed to provide NPSH at the high operating speed. The cover gas spaces in the pump, surge tank and secondary sodium tank remain in communication during operation. The pump is provided with a flywheel to ensure a flow halving time of 4 s in case of pump trip.

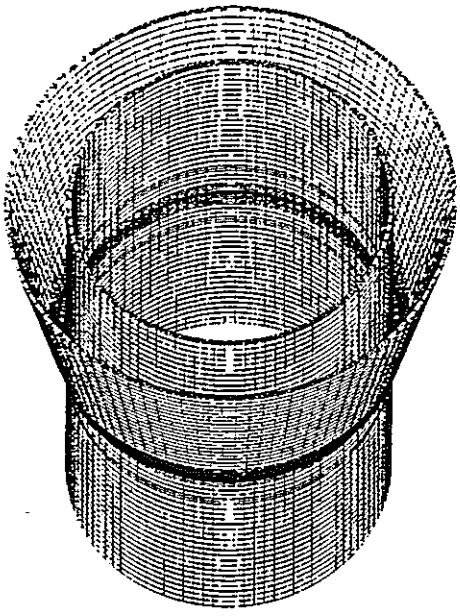

Fig 8 SG support skirt

iv) Steam generator

A preliminary estimate of SG tube wall thinning due to fretting wear at the support locations is 0.5 mm at the end of 30 y operations and rate of thickness reduction is 7.4×10^{-10} mm/s. For this, a computer code TURB-GEN has been developed and used in conjunction with CASTEM 2000.

The support skirt for SG was analysed in detail for seismic and thermomechanical loadings. The junction geometry has been optimised to minimise the overall stresses and creep-fatigue damage (fig 8).

v) IHX

The transient secondary sodium pressure has been reassessed from 1.3 MPa to 1.9 MPa under conditions of large sodium water reaction due to a design base leak in SG. Accordingly the IHX tubesheets are analysed which indicates a minimum tubesheet thickness of 190 mm.

vi) Piping

Secondary sodium flow sheet and piping layout has been finalised (fig 9). Pressure loss in the secondary sodium piping was estimated to be 65 m of sodium, including the margin for uncertainties.

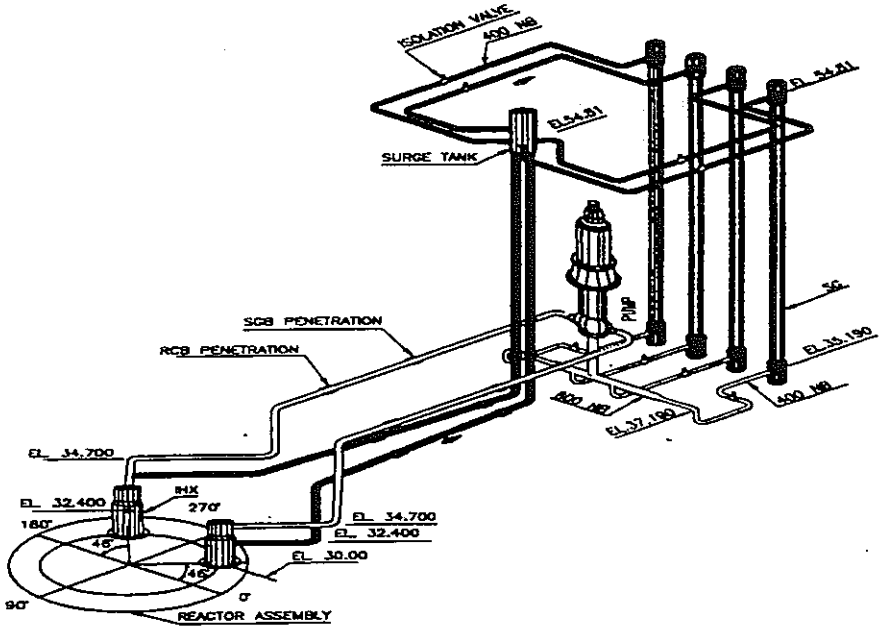

Fig 9 Secondary sodium circuit

3.1.5 Component Handling

I) Inclined Fuel Transfer Machine (IFTM)

Detailed design of IFTM is completed and tender documents have been prepared for manufacture.

ii) Transfer Arm (TA)

Seismic analysis of TA has been carried out for S1 and S2 earthquakes both under horizontal and vertical excitations. Analysis for S2 aims at preventing the sodium leak from reactor assembly through TA support. The structural integrity has been assured as per the RCC-MR:RK. Absence of any mechanical interaction of transfer arm with neighbouring components is also ensured.

iii) Spent Fuel Storage Bay (SFSB)

Conceptual design of SFSB has been completed. Natural convection analysis of water filled pot containing failed fuel stored in SFSB was carried out with THYC-2D and the computed temperatures are within the permissible limits, 318 K, 328 K and 338 K under normal, upset and emergency conditions respectively.

Possibility of using borated steel is being studied to reduce SA interspacing in storage bay. Subcriticality calculations have been performed to optimise the inter subassembly spacing of storage bay under accident scenarios of compact configuration of SA with and without borated steel.

SFSB top water shielding layer thickness was optimized to get the dose rate close to the specified value on the top of the bay. Pool water radioactivity was computed for various reactions. Radiation dose levels in the adjacent cell area of the spent fuel storage was computed.

3.1.7 Reactor Vault

Conceptual design of reactor vault with two layers, inner one for thermal loads and outer one for mechanical loads has been proposed. A preliminary analysis including dead loads, thermal, seismic and shrinkage loads indicates that a minimum wall thickness of 1 m for outer layer and 0.6 m for the inner layer is required.

2D analysis of biological shield cooling circuit was carried out for different tube sizes, tube pitches, concrete cover thickness of the cooling pipes and main vessel temperatures. During steady state with water cooling the concrete temperature is about 343 K.

3.2 Safety Evaluation

Internal Safety Committee (ISC) has been constituted for the safety review at IGCAR. Atomic Energy Regulatory Board (AERB) through its PFBR-Project Design Safety Committee (PDSC) continued the safety review. The committee has discussed the chapters on design bases of structures, equipment and systems, reactor physics, core engineering and component handling system. Site evaluation report has been prepared and is being sent to AERB for approval.

3.3 Technology Development

Technology development of 45° sector of main vessel has been completed and the sector was received at site in April'98. Technology development of inner vessel and SG are in progress. Order has been placed for the development of IHX tube (OD 19 mm & 0.9mm WT) to tubesheet weld joint. Manufacturer for CSRDM has been identified and order will be released shortly. Manufacturing specifications for grid plate, CSRDM, roof slab and transfer arm were finalised.

Large diameter pipe bends and Tees without straight portion have been manufactured with normal SS type 304 and 304L materials for the purpose of technology development as well as for the experimental studies on leak before break justification.

Indigenous manufacture of clad tubes in D9 material has been carried out satisfactorily. Manufacture of SG tube material in T91 grade has been satisfactorily completed for FBTR SG spare module (13 m length).

3.4 Construction of PFBR

Construction of PFBR is likely to start in the year 2001. It is included in the IX Plan proposal and Government has approved it.

4.0 RESEARCH AND DEVELOPMENT

4.1 REACTOR PHYSICS

4.1.1 Nuclear Data

A computer program to calculate cross sections for non-fissile nuclides having Reich-Moore resonance parameters was written and tested. The program was used to investigate the contribution of individual levels to the collision matrix at a particular energy. A new method was developed to compute cross sections with RM formalism which saves computation time considerably.

Isotopes of Pu and Am from ENDF/B-VI were processed to obtain 25 group infinite dilution cross sections and one group cross sections. Revised ENDF/B-VI data received recently from IAEA were used to prepare point data for Fe, Cr, Ni, Au and U-235.

4.1.2 Core Physics

New versions of DORT, CRSRD, MCNP and KENO codes were made operational for production runs. Analysis of the BN-800 core was taken up as an exercise. The results of the analysis show good agreement between the IGCAR values and those predicted by other laboratories, except in the region dependence of sodium void worth.

A standard thermal flux facility was set up for accurate flux measurements and neutron detector calibration and inter-calibrated with BARC facility.

4.1.3 Radiation Shielding

The phase I of bulk shielding mockup experiments at APSARA were carried out. The Mn

equivalent flux at the shielding corner was measured to be 1×10^6 n/cm^2/s as against calculated value of 2.32×10^6 n/cm^2/s. Thermal neutron flux with gold foil and fission equivalent neutron flux values are 1.3×10^6 n/cm^2/s (calculated value is 2.9×10^6 n/cm^2/s) and 7.63×10^6 n/cm^2/s (calculated value is 7.05×10^6 n/cm^2/s). Gamma spectrum and activities of major fission product isotopes in a depleted uranium converter subassembly was calculated to obtain the personnel dose. Activation of some of the important components were studied.

4.1.4 Reactor Safety

The pre-disassembly phase calculation code PINCHTRAN was modified to include better sodium boiling models, better fuel pin temperature distribution calculations including porosity and fuel migration, better temperature calculations in the sodium plenum and upper axial region, refined pressure drop calculations, etc. The modified code has been run for ULOF analysis of BN-800 reactor. A summary report on the "Results of Transient Calculations up to Onset of Boiling of a Comparative Calculations for ULOF in BN-800 with Near Zero Void Reactivity Coefficient" has been prepared. The report includes critical assessment and comparison of the results from various participating organisations of the benchmark.

4.2 ENGINEERING DEVELOPMENT

4.2.1 Heat Transfer Studies in Cover Gas

The LCTR was operated continuously for the heat transfer experimental programme at the sodium temperature above 800 K for more than 3000 h and at 823 K for about 1000 h. Heat flux from sodium free surface to roof slab model was determined for three cover gas heights of 400, 600 and 800 mm at 823 K. The heat flux to the roof slab from the sodium surface is 900 W/m^2 for the cover gas height 400 mm and about 760 W/m^2 for cover gas height of 800 mm. The circulation of air around the vertical surface of the annulus keeps the rotatable plug at a more or less uniform temperature. The circumferential temperature difference is less than 10 K with dip seal below the annular space with the roof slab cooling circuit in operation.

4.2.2 Gas Entrainment Studies in the Hot Pool

A simple slab model of the hot pool involving 1 IHX inlet geometry is being constructed to study the influence of non-dimensional numbers (Reynolds, Weber and Froude) on the incipient gas entrainment. It is proposed to carry out experiments using 1/27, 1/18 and 1/9 scale models in water.

4.2.3 Testing of CSRDM Dash-pot

A dash-pot of CSRDM has been fabricated and lapping of its piston was carried out after which frictional force reduced to < 50 kgf including the resistance offered by sealing rings. The drive assembly, electromagnet with instrumentation have been assembled and dash-pot testing is in progress.

4.2.4 Failed Fuel Location Module

A sodium immersible DC conduction pump has been designed for circulating sodium sampled for failed fuel location. Manufacture of the same is in progress. Thermal hydraulic analysis has been carried out to estimate the temperature distribution in the copper winding and the electromagnet. The maximum magnet temperature is about 890 K (<curie point temperature of 993 K) and winding temperature is about 910 K (to be seen for the acceptability from structural integrity point of view).

4.2.5 Hydraulic Tests on Grid Plate Model

The hydraulic studies on the 1/3 scale grid plate air model have been completed both with and without baffle plate. It is found that the pressure drop due to baffle plate is 48% of the total pressure drop in the grid plate. There is also no difference in the flow distribution with and without baffle plate. These conclusions were found to hold good for 50, 75 and 100% flow rates. The pressure drop in the model without baffle plate is 20 mm of water column which corresponds to 1.77 m for the prototype. With baffle plate this value is increased to 38 mm for the model which corresponds to 3.4 m for the prototype.

4.2.6 Hydraulic and Mechanical Testing of Primary Sodium Pump

The dynamic behaviour of the primary pump is being studied in a full scale test section of the rotor assembly but without the actual pump impeller which is replaced by a steel disk having same mass unbalance. The test section assembly consisting of long shaft, hydrostatic bearing, upper thrust bearing and mechanical seals and the spherical seat machined and assembled to close tolerances was successfully completed and free rotation of the shaft inside the bottom bearing was ensured along with measurement of starting torque of the rotor. The dynamic tests of the rotor section will commence soon.

4.2.7 Thermal hydraulic studies on IHX Model

A 1/2 air model of IHX has been constructed to study the flow distribution on the secondary (tube) side. Different configurations for distribution baffle plates kept below the bottom tubesheet, are being studied. The study is aimed at reducing the radial temperature gradient across the tube bundle as well as in the secondary sodium in the outlet header from the structural integrity considerations.

4.2.8 Steam Generator Test Facility (SGTF)

Financial sanction of 300 mRs has been obtained for 5.5 MW SGTF. Detailed planning for the implementation has been formulated and procurement of materials and equipment has commenced. The detailed civil and electrical layout of the facility building has been completed and tendering actions are in progress. Bulk sodium heating system using oil fired heater has been decided in view of economic advantages and possibility of turnkey supply of the system.

4.2.9 Boron Enrichment Facility

A 5 kg/a capacity B^{10} enrichment facility is under construction at IGCAR. All the fabricated vessels for ion exchange columns and pumps have been received at site. The steel support structures have been erected. Most of the components including instrumentation have been procured. The site installation is expected to be completed by December 1998.

4.2.10 Structural Mechanics Laboratory (SML)

The first phase of SML was completed and the lab was made functional in January 98. 4 test facilities, viz. 15 t high temperature component test facility, thermal ratchetting test facility, IHX/SG tube buckling test rig and an experimental set up to study the FIV of discontinuous level probes in the FBTR reactor vessel are installed. A 2 t slip table using electromagnetic shaker is also under installation. It is planned to further extend the laboratory to install a 20 t shake table, 400 t large component static and buckling test rig, photoelastic stress analysis laboratory . It is expected that the full fledged laboratory will be ready in 2 years.

4.2.11 Identification of Structural Mechanics Failure Modes

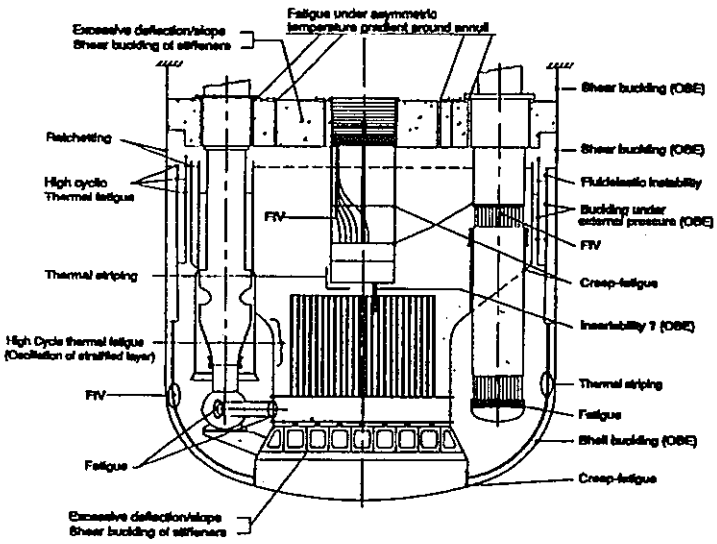

Fig 10 Failure modes under level A & B loadings

All the possible failure modes (fig 10) have been systematically analysed for satisfying the functional and design code limits. For the critical failure modes, a few experimental studies have been completed and continued further.

4.2.12 Creep-Fatigue Damage Simulation Studies

i) Testing of FBTR CRDM Bellows

After successful completion of qualification of the spare gripper bellows of FBTR CRDM by testing for creep/fatigue loading (hold time is about 3 h) which has developed sodium leak after subjecting it to about 1200 deflection controlled load cycles, a new SS347 bellow was subjected to continuous fatigue tests with negligible hold period (about 2 min) at 873 K to understand the effect of creep-fatigue interaction in the bellows. The bellows has withstood more than 40,000 fatigue cycles without developing any sodium leak. These results have shown satisfactory comparison with the theoretical assessment (1040 cycles with hold time and 50,000 cycles with 2 min hold time for crack initiation as per RCC-MR:RB (1993).

ii) Creep-Fatigue Damage Simulation in the PFBR Control plug Mockup

A mockup of upper stay plate-outer shell junction of control plug, kept in a furnace at about 900 K was subjected to 75 thermal shocks (15 K/s) which are the minimum cycles predicted by viscoplastic analysis using 'Chaboche model' in conjunction with RCC-MR:RB criteria for crack initiation. No cracks were observed. This helped to qualify the life prediction methodology adopted for the control plug, which decides the reactor outlet temperature. In order to understand further on crack initiation and propagation, particularly with weld at the junction, tests are to be continued at higher furnace temperature (1000 K).

iii) Creep-Fatigue Tests on ORNL Plates

Two centrally loaded simply supported circular plates are tested for the creep fatigue damage under pure deflection controlled loads to qualify the RCC-MR procedure for estimating the creep damage under the situation of multiaxial stress relaxation at 900 K (±6 K). The state of stress at the inner vessel and stand pipe junction at the hotter face is simulated in this test. Further tests are in progress with welded junction.

4.2.13 Thermal Ratchetting Simulation

Two test facilities have been constructed for simulating the thermal ratchetting in the scaled down models of the reactor assembly vessels. In one of the facilities (fig 11), the hot gas (about 700 K) and water at room temperature are used and the axial movements of thermal gradients are produced by a 'lever-power screw-stepper motor' system. A composite pulley and lever system is used to apply a steady axial stress in the vessel model. Commissioning tests were completed where the simulation of temperature, strain gradients and their movements were verified. Tests are in progress to simulate ratchetting. the another facility, heating coil and water are used as the heating and cooling medium. The movements of axial temperature gradients are produced by the translational movements of the vessel.

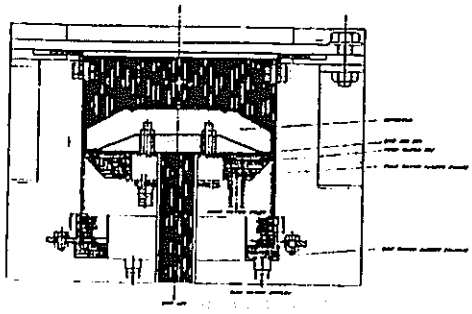

Fig 11 Simulation of ratchetting

Presently 13-Parameter Chaboche Model is used for the theoretical prediction which indicated satisfactory prediction of ratchetting. It is planned to use "Ohno-Wang' constitutive model in future for better prediction.

4.2.14 Thermal Striping

India has participated in the IAEA-Coordinated Research Program on Harmonisation and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes and Relations using Experimental Data' . Thermal striping problem of Phenix Tee Junction was analysed using thermal hydraulic code PHOENICS and thermomechanical code 'CONE'. The prediction of crack initiation life (about 40 h) and propagation life (5 mm growth in 1100 h) for the maximum temperature fluctuation of 90 K on the metal wall , by CONE are comparable to the predictions by France, Japan, Russia and Korea.

A systematic experimental programme is in progress to understand the phenomenon well and propose realistic limits. Water experiments on the scaled models of core cover plate, expansion tank and mixing Tee with maximum ΔT of about 75 K (cold water at 283 K and hot water at 358 K) are in progress. These experiments help to quantify the possible striping frequencies. In parallel, air tests on the flat plate of SS 316 LN and other possible candidate materials with maximum ΔT of about 300 K in the air (colder air at 700 K and hot air at 1000 K) which gives about 200 K on the metal wall are being carried out. These tests are to understand the damage mechanism including crack initiation and propagation. It is planned to further carry out a series of tests in sodium in the 'INSOT' facility which is in commissioning stage now, to confirm the results derived from water and air tests.

4.2.15 Buckling Investigation of Inner vessel

Towards validating the prediction of buckling strength of inner vessel by CASTEM 2000 code, it is planned to carry out tests on 1/15 scale models of inner vessel subjected to mechanical loads. Construction of experimental setup and fabrication of vessel models are in progress. Tests may be over by this year end.

4.2.16 Flow Induced Vibration (FIV) Study

i) Discontinuous level probes of FBTR

Subsequent to the vibration observed in the level probes, detailed theoretical and experimental investigations are made, mainly to ensure the structural integrity and to understand the excitation mechanisms for the PFBR applications. For the experiments, a 1/3 scale model of reactor vessel with level probes has been specially constructed and the vibration of level probes have been simulated in the tests under various flow rates. The investigations indicate that the level of vibration is negligible during normal operation and vibration occurs only at some specified orientation of control plug. Theoretical analysis based on random vibration theory is in progress.

ii) Fuel bundle model

Flow induced vibration measurements were made on a 19 pin dummy fuel bundle in a water loop up to 130% nominal velocity of fuel subassembly. The measured overall strain levels for spacer wire lead of 200 mm are well within the permissible level from fatigue and fretting wear considerations, whereas those for 250 mm lead are acceptable from fatigue point of view but exceeded marginally from fretting wear considerations. A new full scale dummy fuel subassembly (271 pins) has been fabricated with 200 mm spacer wire lead for hydraulic tests.

iii) SG sector model

The flow induced vibration testing of 60° sector model was completed. The excitation mechanisms like fluid elastic instability and vortex shedding are not observed up to a flow of 120% of nominal flow confirming that no FIV problem would exist on the SG. This has been confirmed by the theoretical analysis as per ASME Sec III Appendix-N. The measured pressure drop is 6.84 m of water column which corresponds to 14.67 m for the prototype.

A 3/5 scale model of the inlet plenum of the SG has also been fabricated and studies of velocity distribution at the entry to the tube bundle are in progress.

4.2.17 Leak Before Break (LBB) Investigations

For demonstrating LBB for the secondary sodium pipeline, an experimental programme is in progress. Construction of a loading rig is made ready. 4 large size pipe bends and 2 Tees were fabricated. Benchmark experiments on the plate specimens to qualify the fracture parameters and material data are carried out.

For the fracture assessment of 1/5 model of core support structure as well as primary pump discharge pipe, loading frame has been fabricated. Fabrication of component models are in progress.

4.2.18 Seismic Analysis

India participated in the IAEA Coordinated Research Programme on Intercomparison of analysis methods for seismically isolated nuclear structures. Results of analysis on Italian experimental data for seismic isolators were presented in the 2nd RCM held at Taormina, Italy during 25-29 August 1997. Currently the Japanese and Korean data are being analysed with ABAQUS code for the presentation at the 3rd meeting to be held at Hertford, UK during 25-29 May 1998.

Towards validating the code 'CORESEIS' specific to PFBR conditions, dynamic tests on 7 SA mockup with FBTR dimensions are carried out. The dynamic characteristics of individual SA have been validated based on tests under sinusoidal excitations. The tests for the response under actual simulated base excitations are in progress.

4.2.19 Structural Integrity Assessment under Core Disruptive Accident (CDA)

Analysis capability of FUSTIN code with the control plug in the model has been extended. To validate the code, tests on simple cylindrical shells with rigid cover subjected to low density chemical explosive (approximately equivalent to Pentolite) are in progress. One test with 20 g of explosive has been completed and the prediction of deformation profile in the vessel (maximum 61 mm at the middle location) by FUSTIN is very good compared to the observation (about 60 mm). Further tests are in progress by incorporating a specified cover gas space above the water level as well as increasing the quantity of explosive. Fabrication of 1/10 scale models of main vessel with important internals is in progress to start the next series.

4.3 METALLURGY

4.3.1 Tensile, Creep, Low Cycle Fatigue and Creep-Fatigue Interaction Studies

Detailed investigations have been performed to understand the effects of temperature and Ti/C ratio on low cycle fatigue (LCF) behaviour of Alloy D9. The carbon content of the alloy is fixed at 0.05% and the Ti content is varied to obtain Ti/C ratios of 4 and 8. LCF tests are performed at various temperatures in the range 300-923K. The following are the important observations:

- Fatigue life is high at the intermediate temperature range at around 573K.
- The alloy exhibits initial hardening followed by nearly saturation stress response. At 573K, the cyclic deformation at the saturation stage took place at lower stress amplitudes compares to the saturation stress amplitudes at other temperatures. The degree of initial hardening substantially increases with temperature and exhibits a peak at 823K. The anomalous increase in saturation stress amplitude with increasing temperature between 573 and 823K results from the dynamic strain ageing (DSA) effects.
- DSA also causes a faster reduction in life over the range of its operation between 573 and 823K.
- Oxidation effects plays a significant role in reducing life at 923K. The lower life at 300 K has been attributed to the higher cyclic stress response exhibited by the alloy at that temperature. The Ti/C ratio had negligible effect on stress response at 823K, while Ti/C ratio of 8 showed higher

response stress compares to Ti/C ratio of 4 at 923K. There is no pronounced effect of Ti/C ratio on fatigue life at 823K. At 923K, the alloy with Ti/C ratio of 8 displays poor fatigue resistance.

Low cycle fatigue tests are performed on SS 316L(N) at various temperatures (298-923K), strain amplitudes (±0.25% to ±1.0%) and at various strain rates (3×10^{-5} s^{-1} to 3×10^{-2} s^{-1}). The maximum stress amplitude developed during a LCF test displays an anomalous increase with increasing temperature between 573-873 K. This negative temperature dependence of maximum stress has been considered to result from the operation of DSA process. general, fatigue life decreases with increase in temperature and decrease in strain rate. The decrease in fatigue resistance with increasing temperature between 773-873 K is more significant at lower strain amplitudes where the crack initiation phase consumes larger fraction of fatigue life. The quantitative measurement of surface cracks clearly reveals an increase in intergranular crack density and crack coalescence with increasing temperature due to deleterious effects associated with DSA.

The creep rupture behaviour of 2.25Cr-1Mo and 9Cr-1Mo dissimilar weld joint has been assessed at 823K over a stress range of 80-250 MPa. Specimens for creep testing were taken from single-V weld pads, fabricated by shielded metal arc welding process using basic coated 9Cr-1Mo electrodes. Creep results of 2.25Cr-1Mo/9Cr-1Mo dissimilar weld joints are compared with the data obtained on both the base metals and respective similar joints. Creep rupture lives of similar weld joint specimens were inferior to those of respective weld metals. The 2.25Cr-1Mo/9Cr-1Mo dissimilar weld joint revealed much lower values of rupture life.

Hot tensile properties of an indigenously produced forged 9Cr-1Mo steel are determined at different nominal strain rates in the temperature range of 298-873 K. Both the yield and ultimate tensile strength values of this steel are well within the ASME scatter band and are slightly above the ASME average value. But the steel exhibits %elongation values well below the ASME scatter band. The lower values of ductility has been attributed to the combined effects of larger prior austenitic grain size, and to the slightly higher values of sulphur and phosphorous in the steel compared to the ASTM specification.

4.3.2. Weldability

SS 316L, 316L(N) and Alloy D9 are tested for their susceptibility to hot cracking using the Varestraint Hot Cracking device. The susceptibility of fusion zones to hot cracking as represented by the total crack length (TCL) decreases in the order 316LN>D9>316L. The influence of solidification mode on cracking is quite apparent when hot cracking susceptibilities of 316L and 316LN are compared. Despite a low (P+S) content, 316LN shows higher TCL than 316L because the former solidified in the primary austenitic mode as against the primary ferritic mode in 316L. Alloy D9 with fully austenitic microstructure shows cracking tendency in between those of 316L and 316LN even though the (P+S) content is very low (0.013 wt.-%). The presence of Ti in D9 enhances susceptibility to cracking because of low melting austenite/TiC eutectic. The influence of Ti/C ratio (4, 6 and 8) on hot cracking of D9 is also investigated. general the TCL at a given strain level increases with increase in Ti/C ratio. The composition with Ti/C ratio of 4 is the least susceptible. A study on the influence of nitrogen on hot cracking susceptibility of 316L shows that the TCL first increases and then decreases beyond a particular nitrogen level. Microstructural analysis indicates that the initial increase in susceptibility is because of change in the solidification mode from primary ferritic to primary austenitic. Beyond this, the addition of nitrogen refines the weld metal microstructure and

decreases its cracking tendency.

SS 316L(N) /Alloy 800 dissimilar metal weld, welded with 16-8-2 filler metal, is subjected to accelerated ageing at 873K up to 5000h to simulate long-term service exposures, and is characterised for elevated temperature service at 773K. The analysis includes, apart from conventional structure-property correlation, characterisation in terms of the empirical parameters Γ_f and η_f at 773K. The parameters Γ_f and η_f characterise the resistance to ductile fracture (for the section with least resistance to micro-void growth and coalescence in an inhomogeneous specimen like a weld joint), and are determined from tensile tests of smooth cylindrical transverse-weld specimens. The results show that this weldment has adequate stability against elevated temperature ageing, and should therefore show stable behaviour on exposure at the service temperatures.

4.3.3 Corrosion Studies in Ferritic 9 Cr - 1 Mo Steel

Systematic investigations were carried out to study the influence of various possible microstructures on the corrosion and hydrogen embrittlement. The investigations reveal that microstructure resulting due to insufficient tempering after normalising is more susceptible compared to untempered as well as tempered microstructures. Such a microstructure is likely to occur at the HAZ during repair-welding where post weld heat treatment (PWHT) is either not performed or not possible. The need to perform sufficient tempering after normalising or proper PWHT is emphasised in this work. Though due recognition is given in the welding codes to the importance of PWHT still there might occur intricate structures wherein PWHT might not be possible after repair-welding. To such cases the results of these studies do serve as a caution in calling for a continuous monitoring for possible failures during service either due to corrosion or hydrogen embrittlement.

4.3.4 Thermodynamic Activities of Major Elements in ASS

The thermodynamic activities of major metallic elements at the locations of abnormal compositions (viz., zones of the Radiation Induced Segregation, RIS and some of the Radiation Induced Precipitates, RIP) that are produced due to high levels of radiation were computed using a thermodynamic model developed recently. Computations have revealed that the Cr and Mo activities in G-phase (one of the RIP phases) is abnormally high. Perhaps, such phase might not be observed in reactors under in-pile conditions as these could become centres of localized corrosion.

A thermodynamic model on the lines of the one developed for the FCC phase during the previous year for predicting the activities of the major metallic elements in BCC type iron alloys (Ferritic single phase alloys) is formulated. The model incorporates the magnetic contributions to the thermodynamic functions. The results of this model show a steep variation of thermodynamic activities of all the constituent elements around the Curie temperature, T_C (as in the case of 9 Cr - 1 Mo steel at a T_C of 986 K).

4.3.5 Physical Characterization of D9

Characterization of microstructure and defect evolution has been carried out using Positron Annihilation Spectroscopy (PAS), through annealing experiments in D-9 alloy, addressed to the effect of cold work and Ti/C ratio. This study is important, because the production of cold work induced microstructures that are stable at the service temperature in the reactor, is one of the controlling factors for irradiation induced void swelling. The main results of PAS studies can be summarised as follows :

- TiC precipitates are found to be more stable in 20% cold worked alloy that in 17.5% cold worked sample.
- From the isothermal annealing studies on the kinetics of the process, it is found that TiC precipitation is controlled by dislocations.
- The limited temperature dependence of TiC precipitation is governed by an apparent activation energy of 1.6 eV. 20% cold worked alloy, TiC precipitates are found stable against growth even after 1000 h of annealing at 923 K.
- For higher annealing temperatures, TiC precipitate coarsening occurs due to recrystallisation.

4.3.6 Fuel Fabrication

For FBTR, fabrication of Mark II core of $(U_{0.45}Pu_{0.55})C$ fuel is continued. Compatibility of hyper stoichiometric mixed $(U_{0.45}Pu_{0.55})C$ fuel with sodium and SS 316 (20 % cold worked) are being established experimentally by heating miniature capsules containing the fuel, clad and coolant at 873 K and 973 K for 1000 and 2000 h. Estimation of clad carburisation by photomicrography and microhardness measurement of the cladding material is in progress.

As a part of the development work related to fabrication and property evaluation of mixed oxide fuel for PFBR, fuel pellets of compositions: UO_2-21wt.%PuO_2, UO_2-28wt.%PuO_2 and UO_2-40wt.%PuO_2, were fabricated via mixing, cold compaction and sintering route. Samples of appropriate dimensions were fabricated for the measurement of physical properties and fuel-coolant-clad compatibility. Pellet samples (~5.55 mm diameter) were also prepared for chemical dissolution studies. Thermal diffusion and thermal conductivity have been measured by laser flash method over the temperature range 973 - 1873 K.

4.4 POST IRRADIATION EXAMINATION (PIE) AND NDT DEVELOPMENT

4.4.1 PIE of FBTR Fuel

PIE was carried out on the central fuel SA which had seen more than 25,000 MWd/te burn-up, to assess it's performance and to evaluate the possibility of exploiting this unique fuel for higher linear heat rating and burn-up. The details of examinations carried out upto metallography has been reported last year. Further studies were done on the cut sections of fuel. The etched structure of the fuel cross sections did not reveal any evidence of restructuring of fuel. The clad microstructure also did not reveal any carburisation, indicating that gaseous phase carburisation is insignificant under the relevant conditions.

The micrographs of cut sections of fuel pins were further subjected to image analysis to quantitatively estimate the fuel-clad gaps. For fuel pin number 0211 which has the initial fuel-clad gap of 88 μ (g_i) estimated from the pre-irradiation data, the apparent fuel-clad gap after irradiation, with cracks in the fuel cross section. is 45 μ (g_a) and effective fuel-clad gap available for further swelling, taking into account the crack gaps 58 μ (g_a). The corresponding values of g_i, g_a, g_a for pin number 0141 are 98 μ, 39 μ and 48 μ respectively. The accuracy of measurements is $\pm$12μ.

Considering the effective fuel-clad gap g_e which takes into account the combination of apparent fuel -clad gap and the crack gaps, it can be seen that space is still available to accommodate further swelling of fuel. The fuel which had initially occupied about 91.9% of the cross sectional area inside the clad tube, has now swelled to fill up an effective area of 96.4%, after a burn-up of 25,000 MWd/t. This indicates that the average swelling rate of the fuel is around 1.2% per atom percent burn-up, which correlates well with the value obtained from stack length increase measured from radiographs. Based on a conservative assumption that further swelling of the fuel takes place uniformly at the current rate, it can be seen that an additional burn-up of around 20,000 MWd/t is achievable for the fuel, before the fuel-clad gap is fully closed and Fuel-Clad- Mechanical-Interaction (FCMI) is initiated.

4.4.2 Requalification of FBTR Fuel Pins from a deformed SA

An FBTR fuel sub-assembly deformed during the fuel handling incident in the reactor was earlier dismantled, cleaned in ethanol and stored in a bird cage. These fuel pins were taken up for a comprehensive inspection programme, both non-destructive as well as destructive, to evaluate their suitability for reuse in the reactor. These tests were carried out with the objective of examining whether these fuel pins meet the specifications prescribed for new fuel pins and to ascertain that no caustic embrittlement of the clad material has taken place due to exposure to sodium oxide. After thorough investigations mentioned, 58 out of 61 fuel pins are qualified for reuse in the reactor.

4.4.3 ISI of Weld Joint in FBTR SG Shell and Secondary Sodium Pipelines

As part of the surveillance, as per the technical specifications of FBTR, ultrasonic examination was carried out on six numbers of SG welds and forty four numbers of secondary sodium weld joints, as per ASME Section-XI procedure. In one of the steam generator shell weld joints, a defect equal to a reference level of 10% wall thickness was observed at the ID region. This was recorded and will be monitored during future ISI. The ultrasonic examination of the secondary sodium weld joints did not reveal any defect indication equal to or more than the reference level.

4.4.4 Ultrasonic Examination of PFBR Main vessel Sector Joint

The integrity of the weld joints was checked for the sector of main vessel fabricated for the purpose of technology development by ultrasonic test method using both A-scan and P-scan systems. The reference standard notches were made as per PFBR specifications. The details of the observed flaw indications have been recorded giving information about its size, orientation etc.

4.4.5 Eddy Current Inspection of FBTR Cladding Tubes

The quality control procedures for the indigenously manufactured FBTR fuel cladding tubes include the ultrasonic and eddy current inspection of these cladding tubes. Analysis of the eddy current results combined with the destructive examination of a few of the tubes revealed a periodic variation in wall thickness throughout the tube length, leading to a kind of banded structure. This problem had been overcome by the use of a novel technique called the Phased Array Eddy Current Testing (PAECT). The PAECT technique was optimised using a calibration tube (a similar banded tube containing through holes of 0.4 mm and 1.0 mm). Testing was carried out on all the 205 tubes by inserting each tube through the encircling probe and pulling it manually at a uniform speed. Only thirty tubes were found to give indications. A total of 175 tubes (full length) have been accepted. Since TiC precipitates are known to be efficient traps for vacancies and helium, these results may be significant in the context of evaluation of void swelling and helium embrittlement in D-9 alloy.

4.5 KAMINI REACTOR

The U233 fueled, pool type, KAMINI experimental reactor achieved its first criticality in October 1996. After carrying out the commissioning tests, rectification of deficiencies in radiation shielding around the beam hole, repositioning of neutron detector for uniform signals and providing a cage for the FSA to avoid the reactivity changes due to FIV, the rated 30 kW power was achieved in September 1997. The reactor is being used for activation analysis and neutron radiography.

4.6 CHEMISTRY

4.6.1 Sodium Removal from the Sodium Wetted Components

Ether containing alcohols such as ethyl carbitol and butyl cellosolve are used as solvents for cleaning of sodium from sodium wetted components. These alcohols have a tendency to form peroxide on long time exposure to air, light and heat. In order to evaluate their tendency to form peroxide, experiments were carried out. The preliminary results indicate their decreasing tendency to form peroxides. From the results it is seen that dissolved sodium in used solvents tends to inhibit the production of peroxide.

4.6.2 Carbon Release Behaviour from Reticulated Vitreous Carbon

Reticulated vitreous carbon (RVC) used as cesium trap material can release carbon into sodium and increase the carbon activity during operation of radionuclide trap. The increase in carbon activity will lead to carburisation of structural material. In order to verify this possibility of carbon release into sodium during operation of cesium trap containing RVC, experiments were carried in radioactive sodium chemistry loop (RASCL) at temperature of 473 K. with sodium flow rate of 250 l/h for 500 h. During operation of the trap, the change in the carbon activity in sodium was continuously monitored using an electrochemical carbon meter. From the results it is found that there is negligible change in the carbon activity in sodium.

4.6.3 Determination of Cadmium in Sodium by Cold Vapour AAS

A sensitive analytical method with ultra trace detection capabilities has been developed for the determination of cadmium in sodium. In the CV-AAS, the elemental atomic vapour is directly produced from the solution by using a suitable reducing agent. Subsequently, the elemental vapours are efficiently carried to the optical cell of AAS by a carrier argon. The technique enjoys high sensitivity as the transport efficiency is more than 95% as compared to the poor 2% efficiency obtained with the conventional pneumatic nebuliser used with flame atomiser. Samples of secondary sodium from FBTR were analysed and found to have cadmium concentration below detection limits viz., 5ng/g. The same method can be employed to analyse water, the detection limit being 50 pg/ml.

4.6.4 Studies on Vapour Pressures at Very High Temperatures

The Laser Induced Vaporisation Mass Spectrometry designed and assembled at IGCAR has been used to study Vapour pressures of : Uranium Oxide, Thorium Oxide and Uranium Carbide systems.

4.6.5 Thermochemistry

Thermodynamic studies on alloys of interest in pyrochemical processes were continued. The enthalpies of formation of $CaGa_4$ and Ca_3Ga_8 at 298.15 K were determined by high temperature gallium solution calorimetry. The enthalpies of formation of these compounds at high temperatures were also determined by precipitation calorimetry. The integral enthalpies of mixing of liquid Ca-Ga alloys at 1309 K were also calorimetrically measured.

Drop calorimetric measurements on thoria rich urania - thoria solid solutions were carried out. Enthalpy increments of $(U_{0.1}\ Th_{0.9})\ O_2$ were determined by drop calorimetry in the temperature range of 1000-1700 K from which other thermodynamic functions have been derived. Oxygen potential studies on urania-thoria solid solutions were continued and the oxygen potentials of $(U_{0.9}Th_{0.1})\ O_{2+x}$, $(U_{0.77}\ Th_{0.23})\ O_{2+x}$ and $(U_{0.54}Th_{0.46})\ O_{2+x}$ were measured at 1023 K by using the gas equilibration method.

In order to measure the solid-liquid phase transformation temperatures in nuclear fuel systems, an experimental assembly has been designed and installed. The solidus and liquidus are determined by using the "spot-technique", in the temperature range 1273 to 2273 K in a vacuum atmosphere to prevent the oxidation of the sample. The experimental conditions were standardised by measuring the melting points of pure elemental gold, silver and copper. The technique is validated by the experimental determination of the solidus and liquidus in the binary system Cu-Ni.

4.6.6 Pyrochemical Processing

Electrorefining experiments on uranium were carried out at 200 g level. The effect of cathode surface area, the design of the collector crucible for cathode deposit and the current density on the morphology of the deposit and the collection efficiency are being studied.

4.6.7 Preparation of Nuclear Materials

Studies on the microwave synthesis and sintering of nuclear materials were continued. Uranium oxide powders were prepared by chemically aided denitration of uranyl nitrate by heating uranyl nitrate - PVA mixtures in a microwave oven. UO_2 pellets were found to be rapidly sintered up to 94% theoretical density by heating them in a microwave oven for 20-30 min. Studies are being continued on the effects of heating power, heating time, etc. on the densification of the pellets.

Combustion synthesis of thoria was studied using urea and citric acid as fuels. A microwave oven was used to provide the initial heat to carry out the synthesis. The powders obtained were characterised for surface area, morphology and crystallite size. The powders were compacted into pellets and sintered at 1573, 1673 and 1773 K in air to study the ceramic nature of the powders and also to study the effect of temperature on the sintered density of the compacts. Using citric acid as fuel, thoria pellets with densities as high as 93% and 95% of theoretical density could be achieved with hot plate heated and microwave oven heated high citric acid containing compositions respectively. Urea was found to be a poor fuel that could give only 85-88% of theoretical density pellets. The synthesis of solid solutions of uranium and thorium dioxides were also carried out using the polyvinyl alcohol as fuel. The solid solutions were characterised as above.

4.6.8 Post-Irradiation Examination

For the radiochemical measurements of the burn-up of the irradiated fuel discharged from FBTR, an apparatus for dissolving the fuel pin samples was commissioned inside the hot cells. Dissolution of the cut sections of the fuel pins in this set up has been initiated. In addition, the electrochemical dissolver set up has also been fabricated and installed in the hot cells. Trial runs were carried out initially employing a dissolver made of stainless steel, on un-irradiated mixed carbide fuel pellet, and complete dissolution of the carbide in nitric acid was achieved. However, corrosion of the dissolver by nitric acid in the presence of the oxidant was also observed. The dissolver proposed to be used in the hot cells has therefore, been fabricated out of titanium material.

4.6.9 Actinide Solution Chemistry

Recovery of U and Pu from irradiated fast reactor fuels by solvent extraction process (PUREX), triamyl phosphates possess advantages over the tri-n-butyl phosphate (TBP) presently used as the extractant. Studies were conducted on the effect of temperature on the extraction of plutonium {Pu (IV) and Pu (VI)} and other actinide ions by Tri-n-butyl phosphate as well as Tri-n-amyl phosphate (TAP). The enthalpy changes for the extraction of Pu(IV) and Pu (VI) ions by tri-n-amyl phosphate from nitric acid media were determined by the temperature dependence on the distribution coefficients, for the first time. The extraction of nitric acid by tri-n-amyl phosphate of 303 K was also studied systematically.

4.7 FAST REACTOR FUEL REPROCESSING

Erection of equipment and piping works of the pilot plant is in progress. Cold commissioning of this plant is slated for December, 1998. The design of the demonstration plant for FBTR fuel

reprocessing is nearing completion. Fabrication orders for major equipment like centrifugal extractors, process vessels, evaporators are being placed. The first phase of this plant will be commissioned in 2001.

A new non-dimensional form of equation has been proposed to correlate density of organic solutions containing TBP, n-dodecane, uranyl nitrate and nitic acid for any TBP concentration (0-100%). This form of equation could correlate any liquid system with multiple solutes. The equation is tested thoroughly also for other cases of organic-organic solutions (methanol- PGDME250), solutions of ultra high density (Zn amalgam-mercury binary system, Cd amalgam-mercury binary system) and third phase densities (Th(IV) third phase in TBP/SST). In all cases, the performance of the developed equation has been very good.

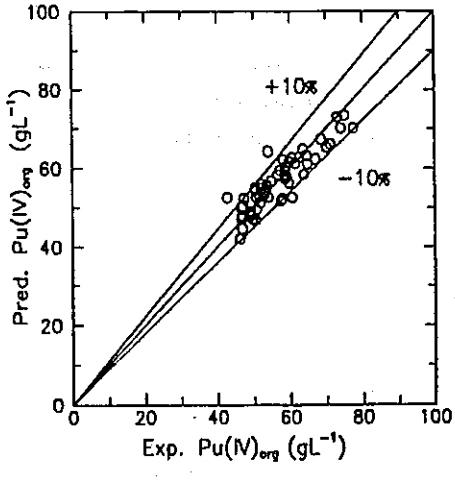

Fig. 12 Parity plot at 303.2 K

An empirical modelling of third phase formation for U(IV) and Pu(IV) nitrates in 30% TBP/n-dodecane solvent has been established. Pu(IV) LOC in 30% TBP/n-dodecane has been correlated as an empirical function of aqueous molar concentrations of uranium and plutonium and total nitrate salting strength of aqueous phase. Parity plot in fig 12 represents the comparison between experimental and predicted values. The mean error and standard deviation were 3.03% and 8.54%.

4.8 SAFETY

4.8.1 Sodium-Concrete Interaction

2 kg of sodium at 723 K was brought in contact with limestone concrete block of 190 sq cm area, kept in an argon filled chamber of 220 l capacity. Soon after dumping sodium, an immersion heater was switched on to maintain sodium temperature at 823 K and the experiment was carried out for about five hours. The heater was switched off when the pool temperature started rising above 823 K due to on set of energetic reactions after 103 min (time reckoned after sodium dumping). The pool temperature reached a peak value of 956 K and the interface temperature between concrete and sodium attained a maximum value of 1056 K in 122 min. With on-line hydrogen measurement, it was observed that hydrogen release started within 10 seconds of initiation of the experiment and reached a peak value of 29.7 %V in 142 minutes. Chromatographic analysis of gas samples indicated the peak concentration of 164 vpm for carbon monoxide, 330 for methane, 325 for carbondioxide, 100 for acetylene and 60 for ethane.

4.8.2 Air Cleaning System for Sodium Fire :

For evaluating the air cleaning system for sodium fire containment an experimental run was carried out using submerged gravel bed scrubber (material size 6 mm to 13 mm, mass 1 kg) with water volume of 6 l. The amount of sodium burned was about 100 g and the duration of fire was 50 min. Throughout the experiment air flow was maintained at 20 l/min. Aerosol samples in scrubbing solution, sodium burn pot residue, filter paper samples and sodium deposited on inner surfaces of fire chamber were collected and analysed for mass balance. Sodium aerosol removal efficiency of the system was found to be better than 90%.

4.8.3 Simulated Fuel Coolant Interaction Studies

Analysis of results obtained from the experimental investigations carried out on tin-water, lead-water and zinc-water systems in which the melt temperature was maintained at 873 K and melt to water ratio was 0.35, indicates the extent of finer fragments generated was in decreasing order for the above three systems which could be explained by the relative explosion index defined as $\Delta T_s^2 / q^2_{FB}\, \sigma\mu$ where ΔT_s is the $(T_{hot}-T_{M.P})$, q_{FB} is the film boiling heat flux calculated at the interface temperature, σ and μ are the surface tension and viscosity of the melt.

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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

April 1997 - March 1998

April 1998

Power Reactor and Nuclear Fuel Development Corporation

A REVIEW OF FAST REACTOR PROGRAM IN JAPAN
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Abstract

This report describes the development and activities on fast reactor in Japan for the period of April 1997 - March 1998.

During this period, two important results were drawn by the Special Committee on Fast Breeder Reactors (FBRs) and the Power Reactor and Nuclear Fuel Development Corporation (PNC) Reform Committee, respectively. The Special Committee on FBRs discussed on the future FBR development in Japan including the Prototype FBR "Monju" operation, and proposed its conclusion as the final report to the Japan Atomic Energy Commission (JAEC) on December 1, 1997. The PNC Reform Committee reviewed PNC's management and safety assurance system, and recommended to reform PNC to a new organization. Each committee result is outlined in this report.

The Experimental Fast Reactor "Joyo" operated 30th - 32nd cycle. In parallel with the operation, the Joyo Upgrading Program (MK-III program) is in progress. Five MK-III driver fuel subassemblies were loaded to the core in the 32nd cycle.

Monju comprehensive safety review, which was started in December 1996, was continued through 1997, and was completed in March 1998.

The DFBR Plant Optimization (phase 2) design study was launched by the Japan Atomic Power Company(JAPC) with goal of constructing FBR plant that achieves both reliability and economy from FY 1997 for three years.

Research and development works are underway under the discussion and coordination of the Japanese FBR R&D Steering Committee, which is composed of PNC, JAPC, Japan Atomic Energy Research Institute(JAERI) and Central Research Institute of Electric Power Industry(CRIEPI).

I. General Review

1. After the sodium leak accident at Monju, it was required to discuss FBR research and development including Monju operation to obtain the public understanding of its significance and process. Thus the Special Committee on FBRs was formed by the JAEC on January 31, 1997. Committee members heard various views from a local government representative, FBR specialists from overseas, an FBR critic, and many others. The deliberations of the committee and the materials presented in the discussions were open to the public. Conclusions of the Committee are that pursuing FBR research and development - with commercialization as a goal - is appropriate given the technology's place as a key alternative to fossil fuels; as part of a flexible development plan, regular reviews reflecting public opinion must be undertaken; Monju is a part of this research and development plan, which must be undertaken with full public understanding of its significance and implications, and they were proposed to the JAEC as the final report.

2. By the lessons of PNC's series of troubles which are the Monju sodium leakage on 8 December 1995 and the fire and explosion in Bituminization Demonstration Facility in Tokai Reprocessing Plant on 11 March 1997, the necessity of reforming PNC heightened. Therefore, in order to construct management reform, system of safety assurance and structures open to the society, it was decided to reform PNC as a new organization. Core functions of the new organization were also defined as FBR and related fuel cycle technology development, and as high-level nuclear waste treatment and disposal research and development.

3. Three cycles of 100MWt rated power operation were carried out in Joyo by the end of March 1998. The Joyo upgrading program named MK-III program is planned to improve its irradiation capability. The main objects of this program are the increase of neutron flux, the modification of the cooling system related to the power increase, extension of irradiation period and upgrading of irradiation technology. The 32nd cycle was the first transition core in which five MK-III fuels were loaded.

4. Monju comprehensive safety review, which was started in December 1996, was continued through 1997, and was completed in March 1998. The review was performed in the fields of (1) structural integrity against flow induced vibration, (2) structural integrity of sodium boundaries, (3) sodium leak detection, prevention of accident expansion and mitigation of sodium leak accidents, (4) the safety design basis of Monju, (5) the operation manuals, (6) reflections of R&D products and technical information, and (7) quality assurance(QA) system and QA activities. Based on the review results, PNC will make an licensing application to STA for modified design.

5. Based on recent situation on both FBR and LWR field, the DFBR Plant Optimization (phase 2) design study was launched with goal of constructing FBR plant that achieves both reliability and economy from FY 1997 for three years. The top priority of design subject in FY 1997 was given to strengthening sodium leakage and fire measures in the sodium cooling systems. It is accomplished by integrating and concentrating sodium equipment, and enclosing whole sodium boundary of both the primary and secondary circuits that the risks of sodium leaks and fire is virtually eliminated.

II. Special Committee on FBRs of the JAEC

The sodium leak accident at the FBR Monju had developed strong sentiments of uneasiness and distrust of the local residents and of the nation on the safety and the future FBR research and development in Japan. In order to obtain the public understanding of its significance and implications, it was required to discuss FBR research and development, including Monju operation.

Special Committee on FBRs was formed at 31 January 1997 by the JAEC. The committee members heard various views from a local government representative, FBR specialists from overseas, an FBR critic, and many others. The deliberations of the committee and the materials presented in the discussions were open to the public.

Draft deport of the Special Committee was proposed on 14 October 1997, and then opened to the public to collect the public opinions on the report. Following to the collection of the public opinions from 14 October to 14 November 1997, a meeting was held in Tokyo on 7 November 1997 to hear public opinions. Reflecting the public opinions, the final report of the committee was submitted to the JAEC. The JAEC declared on 5 December 1997 that the JAEC have a respect for the report of the subcommittee on FBRs and FBR research and development should be carried out under its policy.

The Final Report of the Special Committee, "Future FBR Research and Development in Japan", dated on December 1, 1997 is outlined as follows:

1. Introduction

(a) Background

The "Special Committee on FBRs" was formed by the Atomic Energy Commission (JAEC) at the end of January 1997 as a result of deliberations of the "Roundtable Conference on Nuclear Energy Policy" held in 1996 following the sodium leak incident at the FBR Monju.

(b) The Special Committee on FBRs

The Special Committee consists of specialists from various different fields spanning Japanese society. As part of their meetings, the committee members heard from a local government representative, FBR specialists from overseas, an FBR critic, and many others. The deliberations of the committee and the materials presented in the discussions are open to the public.

2. Energy and Nuclear Power in Japan

Given Japan's paucity of energy resources, the committee feels that it is prudent for Japan to maintain nuclear energy as a supply source for the future. There are quantitative limits to achieving possible gains through increased energy efficiency or the use of the new energy sources. Nevertheless a variety of energy supply options must be maintained for future generations. Therefore, based on a foundation of safety and of peaceful use, nuclear energy is a wise energy supply choice for continued maintenance and development into the next century. In addition. increased energy efficiency and the use of new energies should be actively promoted.

3. The Significance of FBR Research and Development

Although opinions vary as to the levels of future energy demand or the quantities of remaining uranium reserves, most of the committee members agree that FBRs can realize a role as a long-term energy supply resource, as well as the fact that their development represents a duty to humanity. Some members expressed concerns about the economic feasibility of FBR technology, but a majority agree that it is necessary to pursue the possibility of commercialization through concrete efforts. Safety and nuclear proliferation were also mentioned as risk factors, but it is felt that these problems will have technological solutions, especially in a 21st century society which places great emphasis on technologies for recycling. Hence, the committee feels that In the future FBRs will play a significant role as an alternative to the use of fossil fuels, and that it is therefore prudent to promote the research and development of FBRs in order to determine the feasibility of their commercialization.

4. Matters for Further Consideration

(a) Insuring Safety

The organization responsible for FBR research and development must be capable of guaranteeing the highest level of safety. Every possible accident prevention measure must be taken, and when an incident does occur, all efforts must be made to prevent there being any effect on humans or the environment.

(b) Promoting the Understanding and Agreement of Local Residents and the Nation

Due to a series of accidents and mishaps at facilities operated by the Power Reactor and Nuclear Fuel Development Corporation (PNC), residents of the areas surrounding these facilities have developed strong sentiments of uneasiness and distrust. It is therefore essential to reform PNC and put into action an effective plan of safety assurance that will convince the local community.

(c) Insuring Cost Effectiveness and Flexibility

Periodic review of the research and development plan is necessary in the areas of R&D economic efficiency (investments versus results), economic efficiency improvements of the FBR, and the relationship between the reactor and the fuel cycle as a whole. In addition, periodic reviews to assess the viability of the plan are essential to promote flexibility.

(d) Promoting Nuclear Non-Proliferation

Every effort must be made to prevent the proliferation of nuclear weapons.

5. Monju's Research and Development Role

While PNC must be subjected to reform from the ground up, a complete abandonment of Monju at this stage followed by a future resumption of FBR research and development would be an immense waste of human and fiscal resources. With the understanding of the local community, and realizing that Monju is a reactor in a research and development phase, operation must be undertaken with great care in order to reach the research and development goals for the facility.

6. Demonstration FBR Development and Beyond

A concrete plan for developing a demonstration FBR as a successor to Monju can only be formulated following thorough examination of the research results from Monju and from the design studies for such a successor. Ample efforts must be made to insure that safety and economic efficiency are maintained while adopting a flexible approach in view of future energy circumstances on a date for commercialization.

7. Conclusions

Pursuing FBR research and development with commercialization as a goal is appropriate given the technology's place as a key alternative to fossil fuels use. As part of a flexible development plan, regular reviews reflecting public opinion must be undertaken. The prototype reactor Monju is a part of this research and development plan, which must be undertaken with full public understanding of its significance and implications.

III PNC Reformation

By the lessons of PNC's series of troubles which are the Monju sodium leakage on 8 December 1995 and the fire and explosion in the Bituminization Demonstration Facility in Tokai Reprocessing Plant on 11 March 1997, the necessity of drastic reformation of PNC heightened.

PNC Reform Committee was formed in April 1997 by the Science and Technology Agency(STA) to review PNC's management and system of safety assurance, etc., and to design the reformation. The Committee proposed recommendations for the reformation on August 1, 1997.

According to the recommendations, it was decided to establish a new organization in order to construct management reform, system of safety assurance and structures open to the society.

With the PNC Reform Committee monitoring its progress, a working group, formed with the industry's cooperation and as part of the Science and Technology Agency, drafted concrete measures based on these recommendations.

The bill related to PNC Reformation is under deliberation at the Diet scheduling to establish the new organization on October 1, 1998.

The outline of PNC reformation is shown in figure 3.1, and the PNC Reform Committee Report is summarized as follows:

1. Summary of the PNC Reform Committee Report

(1) Part One: The Conceptual Framework for PNC Reform

Japan's Nuclear Energy Policy and PNC

Developing a nuclear fuel cycle to insure a future energy source is connected with ideas of both national energy security as well as of contributing to the international community. PNC, being at the center of this effort, was entrusted by the people of Japan with the responsibility to develop the fuel cycle.

The Structure of the PNC Problem

- PNC has been simultaneously faced with a variety of conflicting circumstances requiring differing analytical and practical tools, such as: the fact that it must conduct unprecedented research and development; the need to supply competitive technology; and the need for particularly high safety standards stipulated by the fact of nuclear energy use.
- In spite of this, PNC found itself in a state of "management lapse" as a result of its inability to formulate precise responses to the changing circumstances surrounding the organization.

(2) Part Two: Basic Concepts for PNC Reform

Examples of Management Lapse

- Overemphasis on research and development resulting in insufficient safety and crisis management capabilities.
- Neglect of seeking outside feedback leading to a closed corporate culture.
- Increases in project size resulting in problems controlling the daily business and structure of the organization.

The Basic Design of the Reform

- Required elements for strong management are: a clear definition of the new corporation's realm of discretion; clearly defined project goals and their precise evaluation; and continuing internal reform as well as external appraisal.
- Regarding the new corporation's projects, the sphere of operations must be clearly delineated, clear criteria for reevaluation must be set, and programs to be undertaken must be distinctly specified.

(3) Part Three: Working Toward Realizing Reform

Concrete Aims of the Reform

- Projects must be fundamentally reevaluated and PNC reorganized prior to establishing the new corporation.
- The new corporation must carry out practical, project-oriented research and development with the goal of achieving the nuclear fuel cycle.

Management Reform

- The new corporation will implement clear project goals based on instructions from the Atomic Energy Commission in the form of the Long Term Nuclear Development and Utilization Plan.
- Management will be subject to systematic external evaluation.
- Personnel affairs will be thoroughly reformed through training and exchange efforts.
- The new corporation will be responsible for conducting its own business, with the Science and Technology Agency acting as its auditor and evaluator.

Core Functions to be Undertaken by the New Corporation

- Fast breeder reactor research and related fuel cycle technology development.
- High-level nuclear waste treatment and disposal research and development.

Functions to be Discontinued

- Overseas uranium exploration, uranium enrichment, and development of the advanced thermal reactor will all be discontinued following periods of transition and consultation with affected local governments and other bodies.

Strengthening Safety Assurance

- Eliminating the overemphasis on research and development, the operations and maintenance sections will be separated from research and development sections.
- The operations and maintenance sections will be suffused with the experience and know-how of private sector industries such as the electric utilities.
- Views on general disaster prevention will be enhanced.
- Perfection will be the goal in formulating a structure for crisis management.

An Organizational Structure Open to Society

- The new corporation's main office will be located close to the site of its actual operations in order to demonstrate the importance of the local community.
- Openness of communications and information to the public will have as a basis a thorough explanation of self-administered reform.
- Openness will be sought in forming domestic and international research networks.
- The understanding and support of area communities will be pursued through efforts to enhance harmonious coexistence.

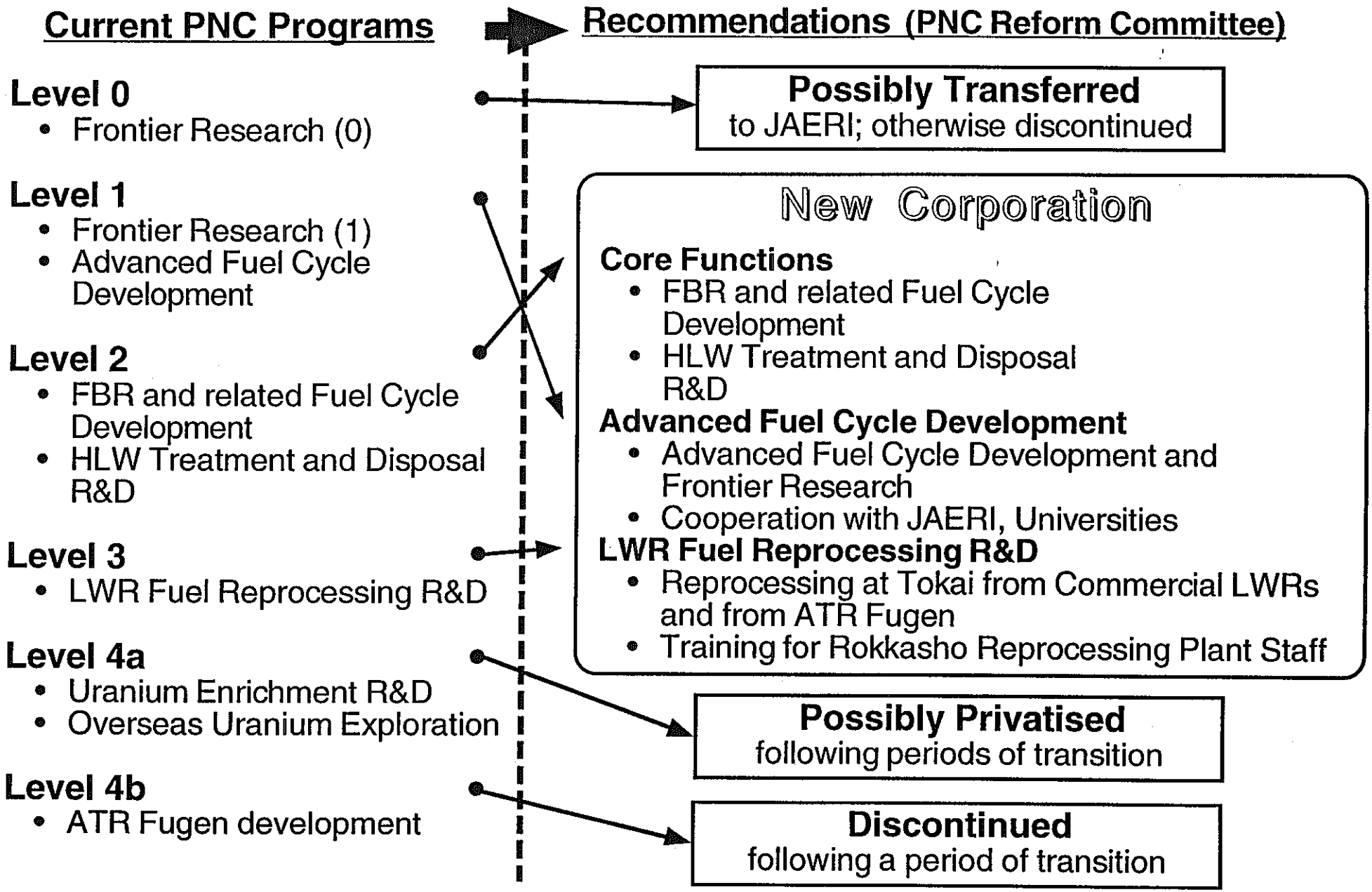


Figure 3.1 Outline of PNC Reformation

IV. EXPERIMENTAL FAST REACTOR JOYO

1. General Status

This report covers the activities of JOYO from April 1997 through March 1998. The operating history of JOYO is shown in figure 4.1.

After finishing the 11th periodical inspection in March 1997, JOYO started the 30th cycle operation with the MK-II core. Three cycles of 100MWt rated power operation were carried out by the end of March 1998.

The 32nd cycle was the first transition core. The transition core is the transformation process from MK-II core to MK-III core, which the MK-III fuels are gradually loaded into the MK-II core. Five MK-III fuels were loaded into the MK-II core for the 32nd cycle.

The irradiation tests carried out in these cycles are summarized as follows.

(1) The creep test of fuel cladding materials under irradiation with MARICO (Material Testing Rig with Temperature Control)

(2) Advanced fuel test (mixed carbide fuel, nitride fuel)

(3) Sodium bond type control rod test

The 12th periodical inspection was started on February 24 in 1998 after completion of the 32nd cycle operation. This inspection will finish in March 1999.

Main items of the 12th periodical inspection are as follows.

(1) ISI of the primary piping boundary

(2) Visual inspection of thermocouple well in the secondary loop

(3) Leak rate test of the containment vessel

(4) Overhaul of the control rod driving mechanism

(5) Inspection of power supply system

2. Upgrading program of JOYO (MK-III program)

The JOYO upgrading program named MK-III program is planned to improve its irradiation capability. The main objects of this program are the increase of neutron flux, the modification of the cooling system related to the power increase, extension of irradiation period and upgrading of irradiation technology.

An outline of this program is shown in figure 4.2.

The license of the MK-III program for the modification of nuclear plant was issued in September 1995. Then, the design approvals on the driver fuel and cooling components such as IHX and DHX were submitted. After obtaining the design permit, the fabrication of fuels and cooling system components was started. The fabrication of outer driver fuels of 25 subassemblies were already finished, and five MK-III driver fuels were loaded for the MK-II core in the 32nd cycle. The manufacturing of IHXs' parts were completed, and the fabrication of the IHXs have started. The manufacturing of DHXs' parts are in progress.

The modification of the MK-III core and the cooling system are planned to start in the year of 2000 with the 13th periodical inspection after completion of the 35th cycle operation. The initial criticality of the MK-III core is now scheduled for the year of 2002. The schedule of the MK-III program is shown in figure 4.3.

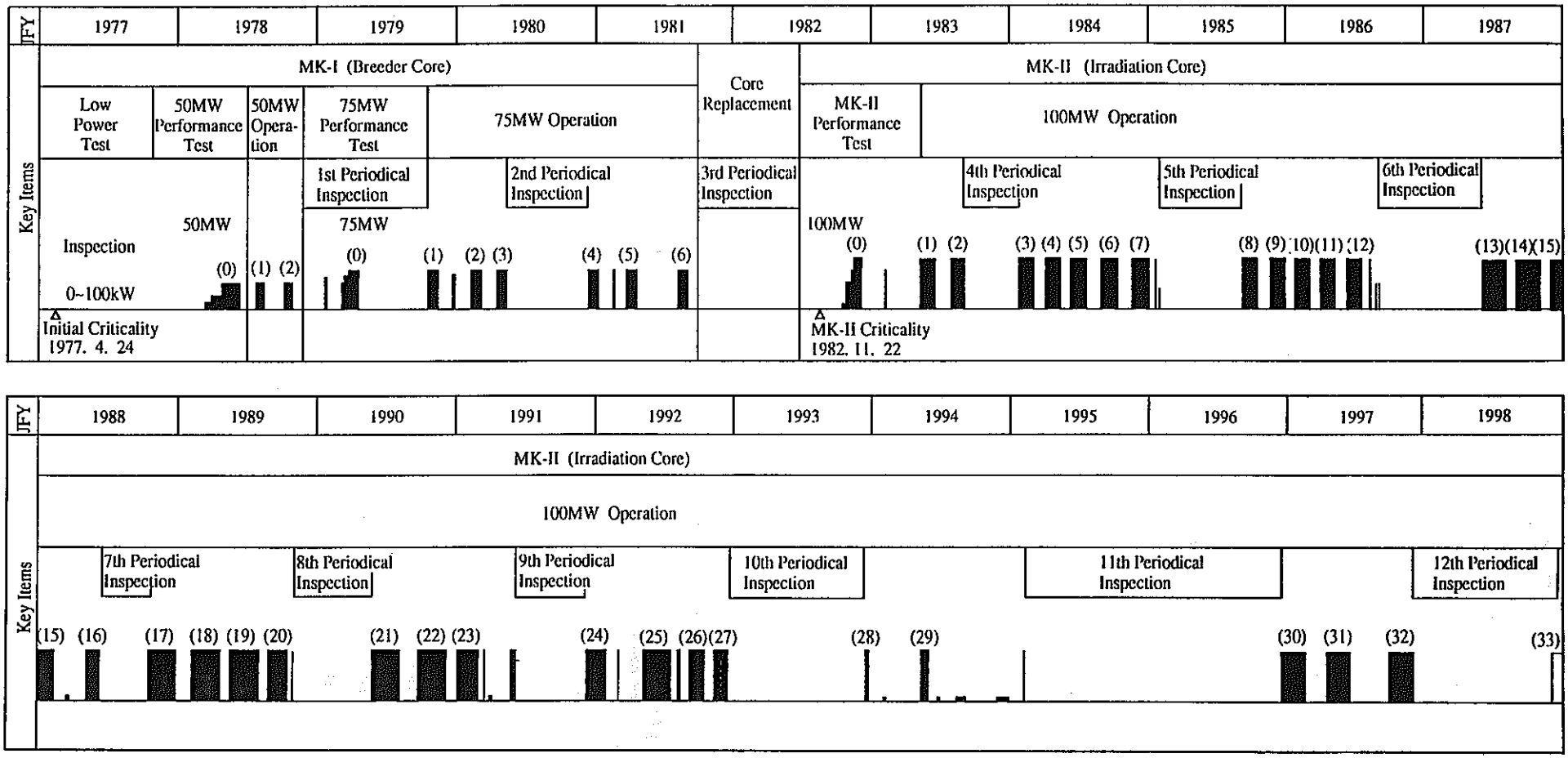


Figure 4.1 Operating History of Experimental Fast Reactor JOYO

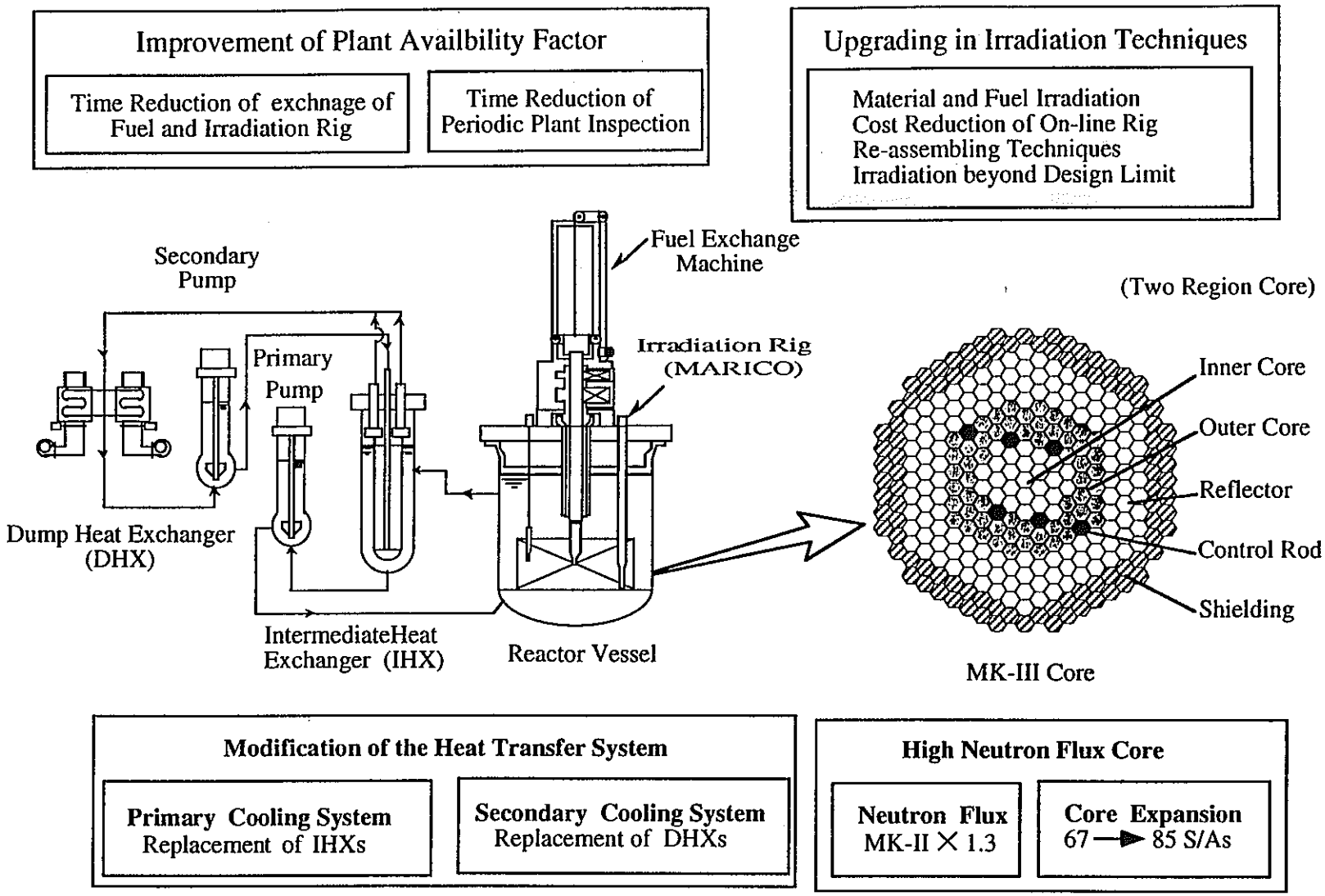


Figure 4.2 Whole Outline of MK-III Program

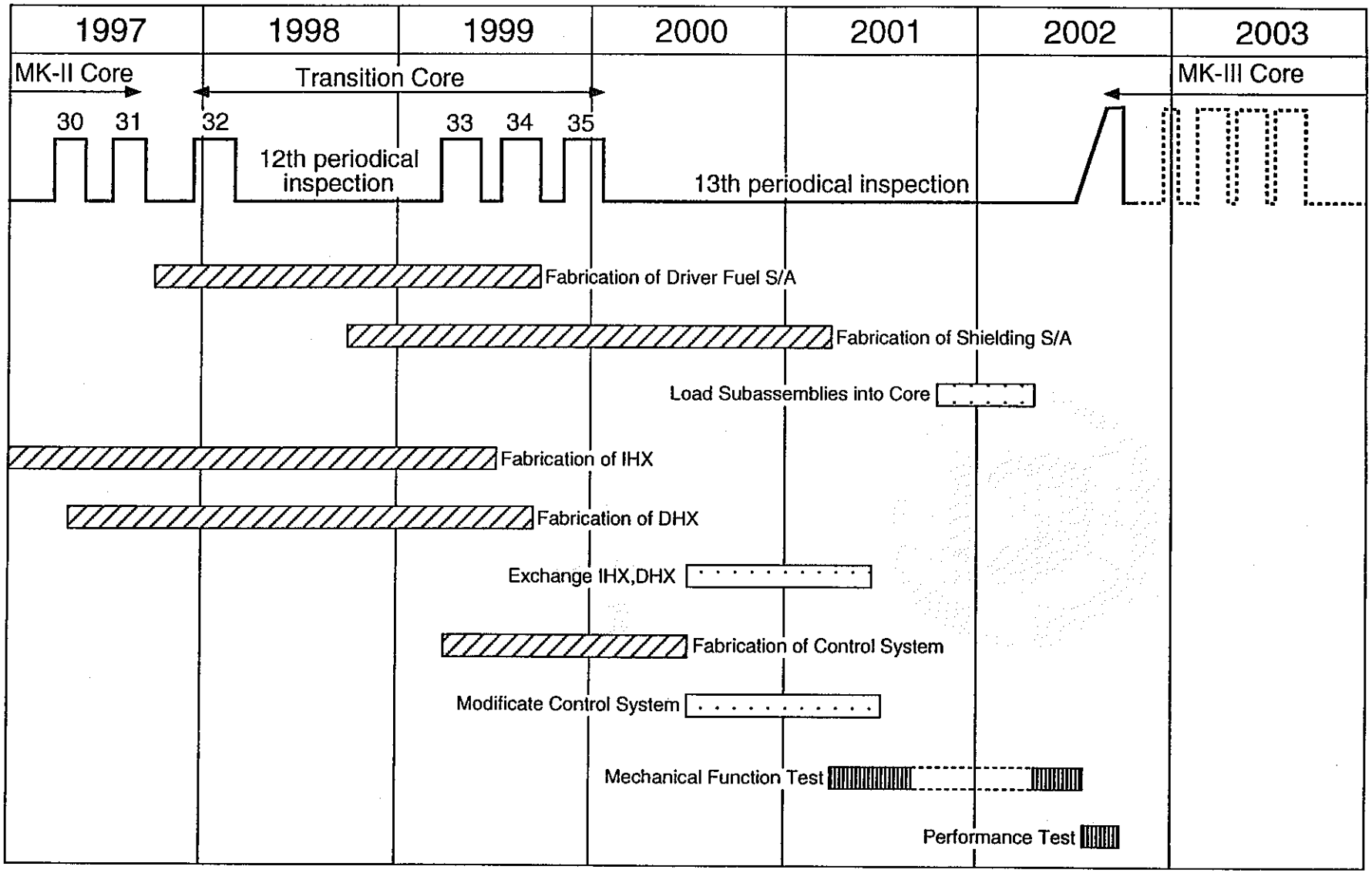


Figure 4.3 Schedule of JOYO MK-III Project

V. Prototype FBR, Monju

1. Introduction

Monju is the Japanese prototype fast breeder reactor, which is located on the north side of the Tsuruga Peninsula in the central Japan, facing the Sea of Japan. Monju is fueled with mixed oxide of plutonium and uranium, and cooled by liquid sodium. It's principal design is shown in Table 5.1.

The construction of Monju began in 1985. Monju achieved the initial criticality in April 1994 and started generating electricity in July 1995.

On December 8, 1995, a sodium leak occurred on the secondary heat transport system (SHTS) while the reactor power was being increased to 40% for a plant trip test as part of the performance tests. About 0.7 ton of sodium leaked out by a break-off of the thermometer well tube installed on the intermediate heat exchanger (IHX) outlet pipe of the SHTS. There was no effect on the general public or the employees due to radioactive materials. However, small thinning of the floor liner was observed where sodium leaked .

In 1996, the cause investigation of the accident was done and it was found that the break-off of the thermometer well tube had been caused by high frequency fatigue by the flow-induced vibration parallel to the direction of the sodium flow. On the other hand, experiments simulating the accident were carried out to examine the sodium leak and combustion behavior. In the test, the floor liner of the experimental facility was perforated; this had not occurred in the Monju accident. By these experiments, it was found that high temperature Na-Fe compound oxide type corrosion had resulted in the small thinning of the floor liner of Monju and molten-salt type corrosion by peroxide ion (O_2^{2-}) had occurred on the floor liner in the experimental facility.

In December 1996, Monju comprehensive safety review was started and continued through 1997.

2. Out line of Monju comprehensive safety review

The aim of Monju comprehensive safety review is to improve the safety and reliability of Monju by considering the problems revealed by the cause investigations. In this work, PNC performed comprehensive review of the Monju design and the operational safety. Special attention paid to the characteristics of Monju such as utilizing sodium as coolant and being in a prototype R&D-stage .

3. Organization for Monju comprehensive safety review

Headquarters for the comprehensive review is composed of the personnel of Monju construction office, PNC head office, O-arai engineering center, and Tokai works. PNC set up an advisory group of learned men in order to ask extensive recommendation for the comprehensive safety review.

Science and Technology Agency (STA) set up a safety review team in Oct. 1996. The STA team consists of many kinds of experts. PNC has been reporting the progress and the results of the comprehensive safety review to the STA team . The STA team plays a leading part in the comprehensive safety review. PNC respects comments of the local administration as well.

4. Review Fields

The comprehensive safety review was performed in seven major fields below.

(1) structural integrity against flow induced vibration

(2) structural integrity of sodium boundaries

(3) sodium leak detection, prevention of accident expansion, and mitigation of sodium leak accidents

(4) safety design basis of Monju

(5) operation manuals

(6) reflections of R&D products and technical information

(7) quality assurance (QA) system and activities

In order to extract safety problems easily through the confirmation of the plant quality and reliability, these fields are set.

5. Review of the structural integrity against flow induced vibration

Integrity of the equipment making sodium boundaries or water/steam boundary is reviewed form a perspective of prevention of a leak accident. Objects of the review are as follows.

(1) thermometer well tubes
(2) structures that penetrate or make sodium boundary or water/steam boundary
(3) heat transfer tubes of heat exchangers such as IHX, SG, and air cooler
(4) equipment submerged in sodium.

Vibration modes and effects of each equipment are listed first. Integrity of the equipment is evaluated from viewpoints of avoiding synchronization with flow-induced vibration and preventing high cycle fatigue in comparison with the "design criteria for preventing flow-induced vibration of thermometers" established by PNC.

As a result of the review, it turned out that 22 thermometer well tubes of the water/steam system was unsuitable to the design criteria. These well tubes will be replaced or removed if unnecessarily. Additional 15 well tubes under similar condition will be replaced by way of precaution.

6. Review of the structural integrity of the sodium boundaries

Structural integrity of the whole sodium boundaries is reviewed from a perspective of prevention of a sodium leak accident. The systems that have sodium boundaries are as follows.

(1) reactor vessel
(2) primary cooling system
(3) secondary cooling system
(4) auxiliary cooling system
(5) ex-vessel fuel storage system
(6) maintenance cooling system

The design requirements are checked to be reflected in the detailed design, the manufacturing, the installation, the testing, and the inspection. As a result, there is no problem that damages the integrity of the boundaries but a few issues to be improved.

Five flow junctions (outlets of the plugging instrument) where temperature difference exists will be taken measures such as heating up by way of precaution. Thermal displacement and magnitude of vibration of the small piping will be measured under the actual plant operating condition after re-starting of Monju.

7. Review of the systems relating sodium leak detection, prevention of the accident expansion, and mitigation of a sodium leak accident

The measures against a sodium leak accident have been reviewed from the viewpoint of early detection , prevention of expansion, and mitigation of the consequence of the accident.

Objects of the review are as follows.

(1) sodium leak detectors
(2) fire extinguishers
(3) HVACs
(4) charge/drain system
(5) heat-retaining structure surrounding the sodium piping
(6) liners on the floor concrete

(7) sodium transfering or storing equipment after a leak accident

(8) control system such as safety protection system, level indicators, and interlocks

(9) central monitoring and control system

The design philosophy of the equipment is inspected in the light of lessons from the accident, the experience and the knowledge of the preceding reactors, up-to-date R&D products, analyses of the accident.

As a result of the review, following improvements are decided:

(1) Increasing monitoring capability of leak detection.

Cell monitors that detect aerosol existence and temperature rise in the room where sodium leak occurs accidentally are going to be installed to enhance the capability of leak detection.

A unified monitoring system that supports the operators will be installed in the central control room. This system will gather the information about sodium leak, fire, visual scene, and process data.

(2) Early sodium drainage following immediate shutdown of the HVACs and the reactor.

When the operators verify sodium leak, the reactor should be scrammed regardless of the leak scale. HVACs will be shutdown automatically by the signal from the cell monitors. The operator will drain the sodium of the broken system then. To prevent the floor liner from failing due to corrosion, time for drainage will be decreased from 50 minutes to 20 minutes. As a result, the drainage will be completed in 40 minutes after initiation of the accident as to the secondary cooling system.

(3) Discharging nitrogen gas to extinguish sodium fire.

After evacuation and safety of the plant personnel is verified, the operator injects nitrogen gas into the concerned area for firm termination of an incident. The nitrogen-gas-injection-system is able to extinguish sodium fire of the leakage of medium scale in 15 minutes.

(4) Preventing spread of sodium aerosol.

The rooms where the secondary cooling systems are installed are separated independently to secure the reactor cooling in a sodium leak accident. However, the volume of the room is too large for the effectiveness of the nitrogen injection and for the suppression of the spread of the sodium aerosol. Each room will be compartmentalized into four smaller area.

(5) Suppressing the temperature rise of the wall and ceiling concrete to prevent discharge of moisture.

If moisture comes out too much in a sodium leak accident, the uncertainty of production of hydrogen and chemical products became large. To prevent the discharge of moisture from concrete wall, thermal insulation structure will be paneled on the wall. Temperature of the wall will be suppressed less than 100 °C.

8. Review of the safety design basis of Monju plant

Objective of this review is to confirm the succession of the plant design basis through the detailed design, the manufacturing, the installation, the testing, and the inspection.

As a result of the review, there is no safety problem generally. However , establishment of the impurity (oxygen) control in the primary cooling system is needed to ensure the fuel cladding integrity during power operation.

9. Review of the operation manuals

Suitability of the following items are reviewed.

(1) notification and communication system and structure at emergency.

(2) operation manuals for transient and accident.

(3) education and training of the operators.

As a result of the review, following improvements are decided.

(1) It is necessary to provide the manuals against fire with all the supposed situations.
(2) The manuals will be modified to check sheet type ones.
(3) Introduction of the symptom-base manuals in the not-too-distant future is recommended

10. Review of the reflections of R&D products and technical information

Objects of the review are as follows.

(1) R&D products regarding FBR (about 16,000 reports)
(2) operational experience from experimental reactor Joyo
(3) operational experience from domestic and foreign nuclear power plants
(4) operational experience from Monju itself
(5) data using in the safety analyses

As a result of the review, following items for further investigation will be taken out.

(1) Evaluation of the fuel maximum temperature.
(2) Life extension of the control rods.
(3) Confirmation of no SG tube failure due to the mode of high temperature rupture.

11. Review of the system and the activities of quality assurance

Objects of the review are as follows.

(1) QA activities for the design and the manufacturing of Monju
(2) examination and approval process of the operation manuals
(3) organization of the education and the training of the operators
(4) implementation system of reflecting the most up-to-date technical information.

As a result of the review, following plans are decided.

(1) A manual for design review will be prepared.
(2) Definition of the new design and modified design will be cleared.
(3) WANO peer review will be invited.

12. Concluding Remarks

The comprehensive safety review was completed in March 1998. A licensing application is to be made to STA for modified design. Major modifications and improvements recommended through the safety review are scheduled by restart.

Table 5.1 Principal Design of Monju

Reactor type	Sodium-cooled / loop-type
Number of loops	3
Thermal output	714 MWt
Electrical output	280 MWe
Fuel material	PuO_2-UO_2
Core dimensions	
Equivalent diameter	1,790 mm
Height	930 mm
Plutonium enrichment (inner core / outer core)	(Pu fissile %)
Initial core	15 / 20
Equilibrium core	16 / 21
Fuel inventory	
Core (U+Pu metal)	5.9 t
Blanket (U metal)	17.5 t
Average burnup	80,000 MWD/ t
Cladding material	SUS316
Cladding outer diameter/thickness	6.5 / 0.47 mm
Blanket thickness	
Upper / lower / radial	30 / 35 / 30 cm
Breeding ratio	1.2

Reactor vessel	
height / diameter	18 / 7 m
Primary coolant systems	
Coolant sodium mass	760 ton
Inlet / outlet reactor temperature	397 / 529 ℃
Coolant flow rate	5.1×10^6kg /h / loop×3loops
Coolant flow velocity	6m/s(inlet), 4m/s(outlet)
Secondary coolant systems	
Coolant sodium mass	760 ton
Inlet / outlet IHX temperature	325 / 505 ℃
Coolant flow rate	3.7×10^6kg /h / loop×3loops
Coolant flow velocity	5 m/s
Water - steam systems	
Feed water flow rate	113.7×10^4kg/h
Steam temperature (turbine inlet)	483 ℃
Steam pressure (turbine inlet)	12.7 MPa
Type of steam generator	Helical coil
Refueling system	Single rotating plug with fixed arm FHM
Refueling interval	6 months

VI. DESIGN STUDY OF DFBR

Based on the results of previous design studies and the major specifications determined by the Federation of Electric Power Companies(FEPC) , the design study for the DFBR plant optimization was carried out during 3 years from FY 1994 to establish the overall concepts of the DFBR plant by designing a consistent overall plant and evaluating the safety property and economical advantages.

The plant concepts was optimized. The optimization was executed upon the reactor structure, the cooling system, BOP and building layout, incorporating the results of the study on the core safety enhancement, pressure capability of the containment vessel and the technical subjects to be solved in case of applying the horizontal seismic isolation design. The technical reports for application of license of the plant construction were drafted based on the results of the evaluation of the plant safety and the structural integrity of major components. The economical evaluation showed that the construction cost of the DFBR was lower than the target cost, which is, 1.5 times that of LWR on 1000MWe basis.

In December 1995, the sodium leakage incident occurred in the prototype reactor MONJU and after that, the measures for the sodium leakage incident became the most important design issue in the DFBR. And, the reduction of the construction cost also became more important because a great effort on further cost reduction of the LWR has being carried out.

Based on such recent situation on both FBR and LWR field, the DFBR Plant Optimization (phase 2) design study was launched with goal of constructing FBR plant that achieves both reliability and economy from FY 1997 for three years.

The top priority of design subject in FY 1997 was given to strengthening sodium leakage and fire measures in the sodium cooling systems. It is accomplished by integrating and concentrating sodium equipment, and enclosing whole sodium boundary of both the primary and secondary circuits that the risks of sodium leaks and fire is virtually eliminated. Further, the basic concept of the main cooling system in line with reducing construction costs has been established. From the point of view of simplifying the structure and improving the reliability of plant, a concept of a well balanced, reduced cost, high reliability structure has been achieved. Concepts of rationalizing the fuel handling equipment and BOP have been clarified related to achievement of sodium leak and fire.

VII. RESEARCH AND DEVELOPMENT

Research and development works are underway under the discussion and coordination of the Japanese FBR R&D Steering Committee, which is composed of PNC, JAPC, JAERI and CRIEPI.

R&D items for DFBR among four organizations in Japanese fiscal year 1997 are shown in Table 7.1.

Table 7.1 R&D items for DFBR among four organizations in JFY 1997

1. Plant System Design

Classification	JAPC	PNC	JAERI	CRIEPI
1. Plant System Design	• Design Study of DFBR (phase II)	• Core & Fuel Design Study for DFBR		

2. Safety

Classification	JAPC	PNC	JAERI	CRIEPI
2.1 Design Principle & related Standard	• Study on DFBR Safety Design Principle	• Study on FBR Safety Evaluation Principle		
2.2 Safety Evaluation	• Study on Mechanical Consequences of CDA : Evaluation & Countermeasures • Study on permission Scenario • Study on Local Core Faults Events : Event Sequences & Countermeasures • Study on Sodium-Water Reaction : Consequences & Countermeasures • Study on Sodium Leak : Consequences & Countermeasures • Study on Protection against Sodium Spray Mitigation of Sodium Leakage	• Development of CDA Evaluation Method : SIMMER-III/SAS4A • Study on CDA Scenario for DFBR • Development of Local Core Faults Events Evaluation Method • Study on Fuel Pin Behavior under Accidental Conditions • Development of Evaluation Method for FP Behavior • Development of Evaluation Method for Sodium-Water Reaction • Development of Evaluation Method for Sodium Fire	• NSRR In-pile Transient Fuel Pin Behavior Tests	
2.3 PSA		• Development of PSA Method & Data Base		

Table 7.1 continued

3. Core & Fuel

Classification	JAPC	PNC	JAERI	CRIEPI
3.1 Enhancement of Core Safety	• Development of Gas Expansion Module(GEM)		• Reactivity Evaluation Test of Gas Expansion Module(GEM)	
3.2 Core Design	• Application Study of Improved Core Physics Method • Design Study of B4C Shielding Assembly	• Improvement of Core Physics Method : Codes & Nuclear Data Development • Study on B4C Shielding Assembly : Irradiation Behavior and B4C Material Properties	• Improvement of Core Physics Method : Nuclear Data Development	
3.3 Fuel Design	• Evaluation of Fuel Design Related to Safety Design Principle • Development of Evaluation Method for Core Mechanics	• Study on Fuel Design Principle, Design Criteria & Related Material Properties • Study on Deformation Behavior of Fuel Subassembly		
3.4 Core Components	• Investigation of PFR and European Large Scale FBR Long Life Control	• Development of Long Life and High Performance Control Rod		

Table 7.1 continued

4. Components

Classification	JAPC	PNC	JAERI	CRIEPI
4.1 Advanced Reactor Shutdown System	• Development of CRDM for Simplified Reactor Upper Structures • Development of Self-Actuated Shutdown System	• In-pile Reliability Tests for SASS in JOYO		
4.2 Fuel Handling System	• Development of Manipulator Type In-vessel Fuel Handling Machine • Development of Seal Structure & Sodium Vapor Deposition Protection of Rotating Plug • Development of Spent Fuel Dry Cleaning System			
4.3 Instrumentation	• Development of Flow Meter for Primary Sodium Circuit, CR Position Detection System & Leak Detection System in SG • Study on Reliability Improvement of Reactor Shutdown System • Development and Improvement of Sodium Leak Detection System • Development of Fuel Failure Detection and Location System	• Development of Leak Detection System in SG : MONJU Test Data Analysis • Study on Operating Procedure in case of Fuel Failure		• Development of Leak Detection System in SG : Active System
4.4 Inspection & Repair	• Development of Inspection Technology for Sodium Components : Development of Under Sodium Viewer and under Sodium Volumetic Inspection • Development of Repair Technology for Sodium Components			
4.5 Others	• Development on Sodium-immersed Self-cooled Electromagnetic Pump • Development on Slide Joint of In-vessel Piping • Application Study of Rapture Disk under Sodium for DFBR			

Table 7.1 continued

5. Structure, Materials, Thermal-hydraulics and Seismic Design

Classification	JAPC	PNC	JAERI	CRIEPI
5.1 Structure & Materials	• Development of Elevated Temperature Structural Design Guide for DFBR • Development of Material Strength Standard for Mod. 9Cr Steel, SUS316 • Application Study of LBB for DFBR Primary & Secondary Pipes • Development of Design Guide of Rectangular RCCV (Reinforced Concrete Containment Vessel)	• Under Sodium Experiment for Data Base • Development of Material Strength Data Base • Study on LBB Evaluation Method : Crack Propagation Test under Sodium Condition		• Under Air Experiment for Data Base • Development of Material Strength Data Base for Applicability of Inelastic Analysis • Study on LBB Evaluation Method : Crack Propagation Test under Long Term Load Condition
5.2 Thermal-hydraulics	• Design Study of Cover Gas Entrainment Protection Structure, Reactor Wall Protection Structure • Evaluation of Coolant-Cover Gas Interface Stability • Evaluation of Natural Circulation Capability • Large Scale Water Test for Design Confirmation	• Development of Evaluation Method for Natural Circulation Capability : Sodium & Water Test, Code Development		• Development of Evaluation Method for Cover Gas Entrainment & Coolant-Cover Gas Interface Stability • Development of Evaluation Method for Natural Circulation Capability : Water Test
5.3 Seismic Design	• Development of Seismic Isolation Design Method			

Table 7.1 continued

6. Large Scale Tests

Classification		
6. Large Scale Tests	PNC will take responsibility • Overall In-sodium Test for Reactor Cooling System • Performance Test for Reactor Wall Protection Structure • Heat Removal Characteristic Test for DHRS • Performance Test for Neutron Detector • Overall Test for SG Tube Rupture • Structural Reliability Test for SG	Not yet decided which organization will take resonsibility • Overall Seismic Test for Reactor Cooling System (Overall Seismic Isolation Test for Reactor Cooling System) • Reliability Test for Main Reactor Shutdown System • Reliability Test for Backup Reactor Shutdown System • Reliability Test for Main Loop Sodium Pump • Reliability Test for Fuel Handling Machine • Reliability Test for Fuel Transfer Machine • Structural Reliability Test for Heat Resistant Concrete Liner • Overall Test for Flow Meter

7. Reflection of Experience in MONJU

Classification	JAPC	PNC	JAERI	CRIEPI
7. Reflection of Experience in MONJU	• Reflection of Pre-operational Test and Operational Experience of MONJU to DFBR	• Evaluation of Pre-operational Test and Operational Experience of MONJU : Design Technology, Radioactive Material Behavior, Breeding Capability, Core Performance, Inspection & Maintenance Technology		

A REVIEW OF FBR DEVELOPMENT IN JAPAN

Presented to the 31st Annual Meeting of the IAEA/IWGFR
Vienna, Austria, 12-14 May 1998

POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
JAPAN

1. GENERAL REVIEW(1)

- The Special Committee on FBRs was formed by the Atomic Energy Commission(AEC) on January 31st,1997.
- The Special Committee submitted a Final Report to the AEC on December 1st,1997
- The AEC had a respect for the Report, and concluded that FBRs R&D should be carried out under its policy on December 5th,1997.
- PNC Reform Committee was formed in April 1997 to review PNC's management & system of safety assurance and to design the reformation.
- PNC Reform Committee submitted a Report to the Sience and Technology Agency on August 1st,1997.

OUTLINE OF THE FINAL REPORT OF THE SPECIAL COMMITTEE ON FBRs

STANCE OF THE FBRs R&D

- FBRs will play a significant role as an alternatives to the use of fossil energy resources.
- It is therefore prudent to promote R&D of FBRs in order to determine the feasibility of their commercialization.

MONJU

- With the understanding of the local community,and realizing that "Monju" is a reactor in a R&D phase,operation must be undertaken with great care in order to reach the R&D goals for the facility.

OUTLINE OF THE FINAL REPORT OF THE SPECIAL COMMITTEE ON FBRs(Continued)

DEMONSTRATION FBR

- A concrete plan for developing a DFBR as a successor to "Monju" can only be formulated following thorough examination of the research results from "Monju" and from the design studies for such a successor.

TOWARD COMMERCIALIZATION

- As FBR is considered to be one of the best alternatives to the future fossil energy resources so far,it is appropriate to promote the FBR R&D in order to determine its practicality.
- The development program including the target year of the FBRs commercialization should be set flexibly considering the future energy circumstances in Japan,while pursuing improvement of the safety and the economical efficiency of FBRs.

1. GENERAL REVIEW(3)

MAIN POINTS OF THE PNC REFORMATION

- By the lessons of PNC's series of troubles,which are the "Monju" sodium leakage on 8 Dec. 1995 and the fire and explosion in the Bituminization Facility on 11 March 1997,the necessity of drastic reformation of PNC heightened.

- Examples of the PNC's Problem

- Overemphasis on R&D resulting in insufficient safety and crisis management capabilities.
- Neglect of seeking outside feedback leading to a closed corporate culture.
- Increases in project size resulting in problems controlling the daily business and structure of the organization.

MAIN POINTS OF THE PNC REFORMATION(Continued)

- The successor organization to PNC aims for concrete realization of the technology of the nuclear fuel cycle through a long term and steady process of R&D.

- Core Functions to be undertaken by the New Corporation
  - FBR research and related Fuel Cycle Technology Development
  - High-level Radioactive Waste Treatment and Disposal R&D

- Functions to be discontinued
  Overseas Uranium Exploration,Uranium Enrichment, and Advanced Thermal Reactor(ATR) R&D will all be discontinued following period of transition (within 5 years).

- The bill related to PNC Reformation is now under deliberation at the National Diet scheduling to establish the New Corporation on October 1st,1998 .

Frontier Research

Advanced Fuel Cycle Development

"Monju"

"Joyo"

Pu-Fuel Manufacturing

FBR Fuel Reprocessing R&D

LWR Fuel Reprocessing

Waste Disposal R&D

HLW Treatment & Disposal R&D

Uranium Enrichment R&D

Overseas Uranium Exploration

ATR"Fugen"

Works of Present "PNC"

FBR & related Fuel Cycle Development
- "Monju"
- "Joyo"
- Pu-Fuel Manufacturing
- FBR Fuel Reprocessing R&D
- Advanced Fuel Cycle Development

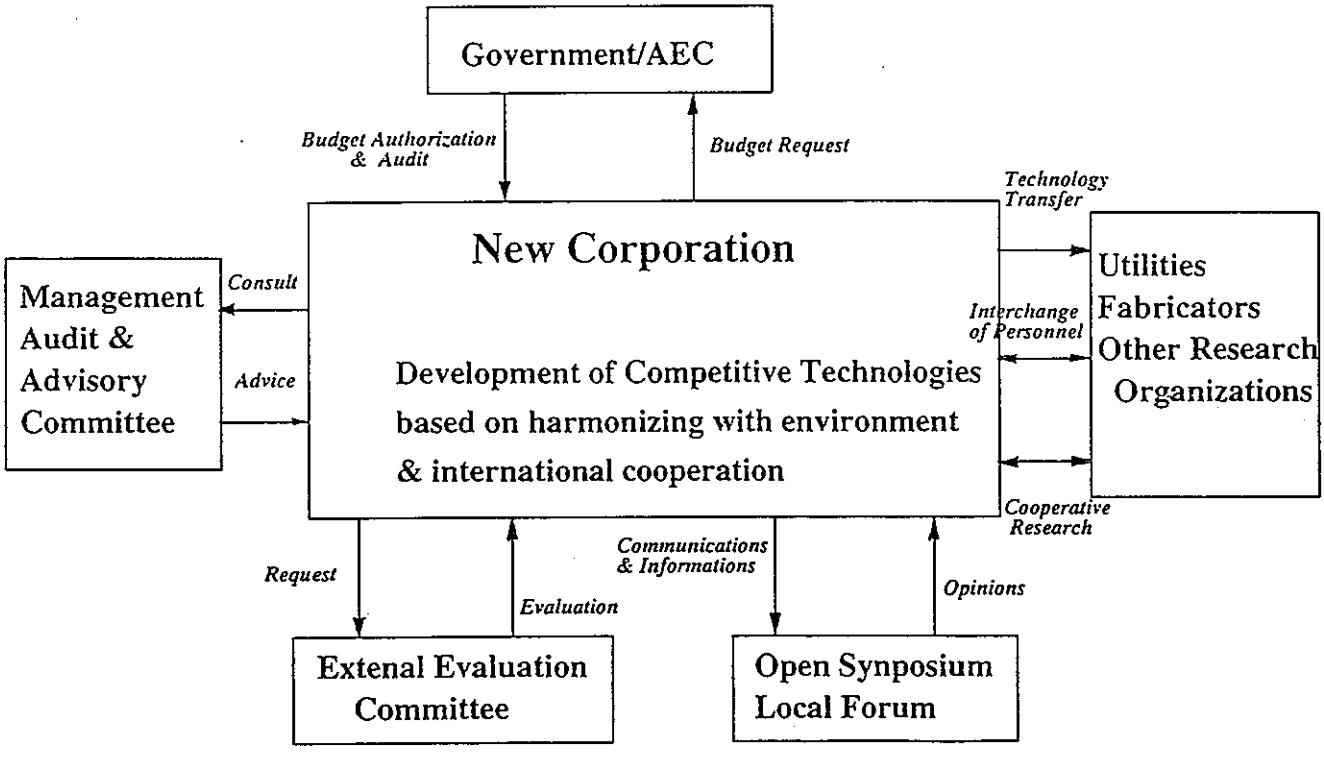
HLW Treatment & Disposal R&D

LWR Fuel Reprocessing

Waste Disposal & Decommisioning R&D

Works to be discontinued
- Uranium Enrichment R&D
- Overseas Uranium Exploration
- ATR"Fugen"

Works of the New Corporation



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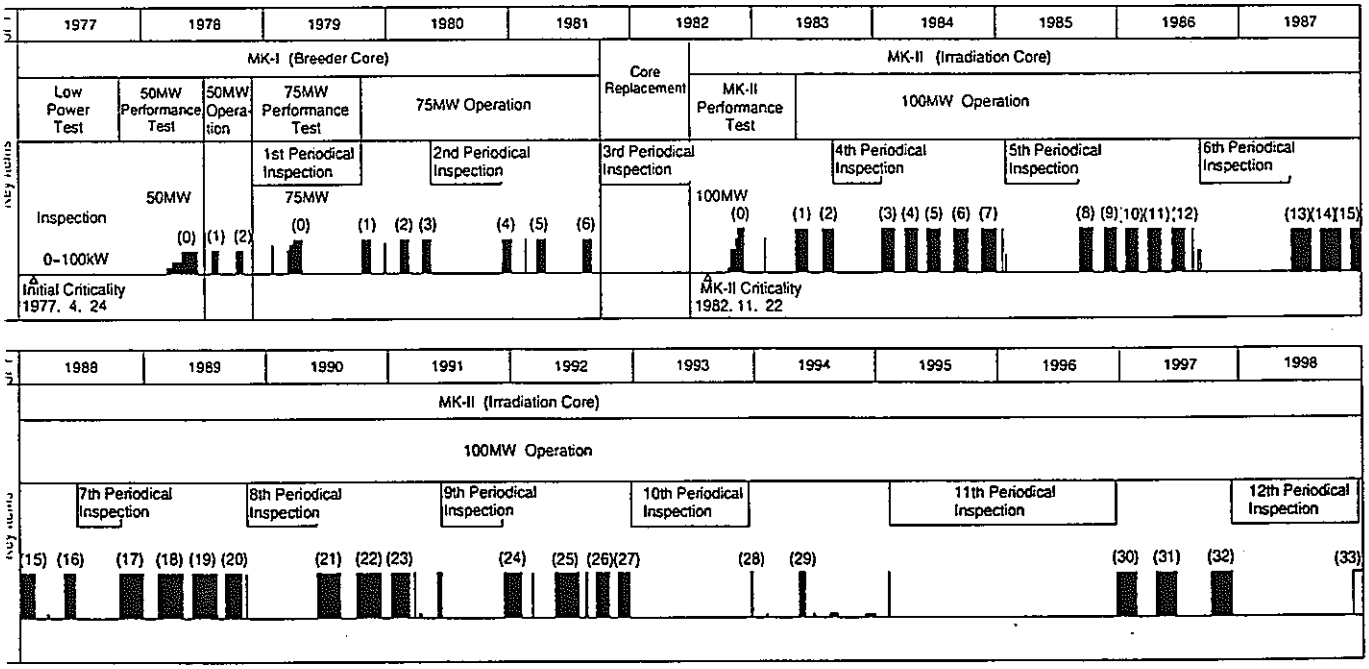
2. EXPERIMENTAL FAST REACTOR, "JOYO"(1)

- Experimental Fast Reactor "Joyo": 55,945 hours operation since 1977.(as of April 1998)

- 11th Periodical Inspection and Improvement of Countermeasure for Sodium Leak was completed on March, 1997.

- 30th Duty Cycle Operation : March 3 ~ May 8,1997
  31st Duty Cycle Operation : July 14 ~ Sept. 12,1997
  32nd Duty Cycle Operation : Dec. 9,1997 ~ Feb.24,1998
  12th Periodical Inspection : Feb.24,1998 ~ (March,1999)



Operating History of Experimental Fast Reactor JOYO

2. EXPERIMENTAL FAST REACTOR, "JOYO"(2)

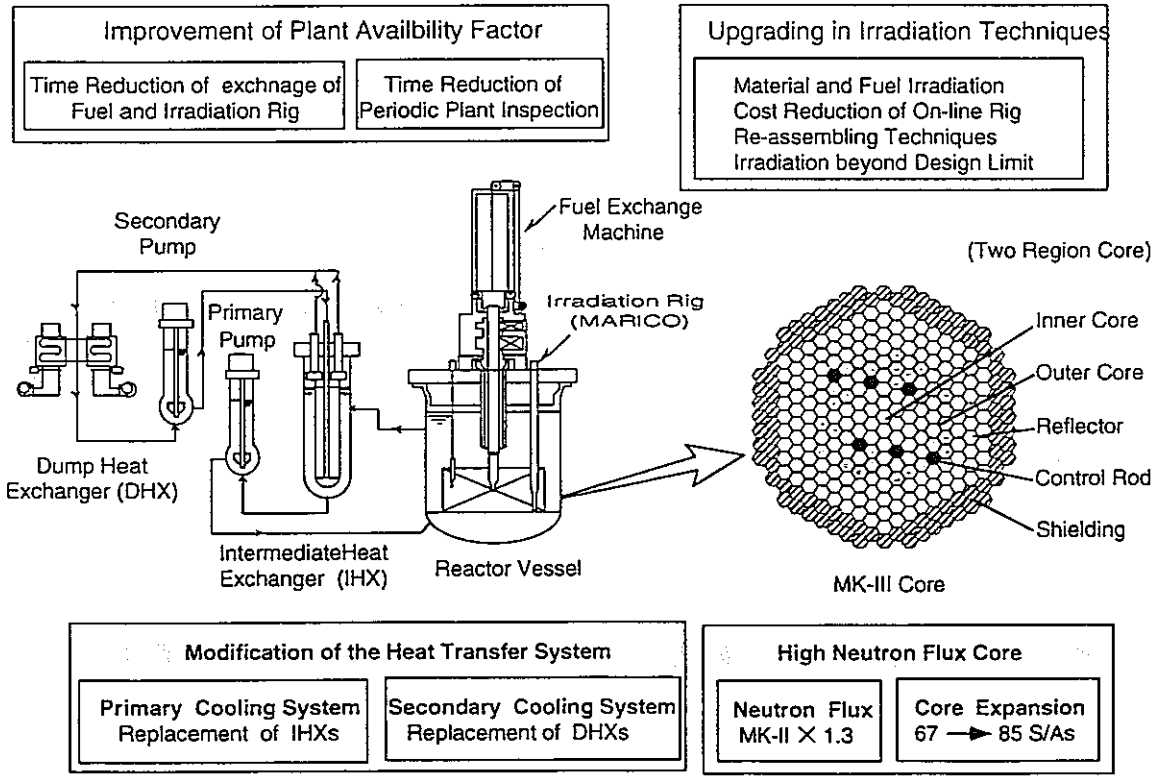
- "Joyo" Upgrading Program, "MARK-III Program"

  - 5 MARK-III Driver Fuels were loaded for the MARK-II Core from 32nd Duty Cycle.

  - The IHXs and DHXs are under manufacturing for the increasing Plant Cooling Capability 100MWt to 140MWt.

  - The Modification of the MARK-III Core & Cooling System are planed to start in the year of 2000 with the 13th Periodical Inspection.

  - The initial criticality of the MARK-III Core is now scheduled for the year of 2002.



Whole Outline of MK-III Program

6

3. PROTOTYPE FBR "MONJU"(1)

● Construction Completed	:	Apr. 1991
● Function Tests Start	:	May 1991
● Function Tests Completed	:	Dec. 1992
● System Start-up Test Start	:	Dec. 1992
● Core Loading Start	:	Aug. 1993
● Initial Criticality	:	Apr. 1994
● First Connection to Grid	:	Aug. 1995
● Sodium Leak Occurred	:	Dec. 1995
● Cause Investigation Completed	:	Apr. 1997

3. PROTOTYPE FBR "MONJU"(2)

● Total Plant Review completed : March 1998
(including the improvement planning)

● For the Future Steps
- Safety Review Approval
- Improvement Works
- Commissioning Test Restart
- Plant Operation

It is essential to reform PNC and to promote the understanding and agreement of Local Residents and the Nation.

3.PROTOTYPE FBR "MONJU"(3)

IMPROVEMENT ON THE THERMOCOUPLE POCKET

Breakage Prevention due to Hydraulic Vibration

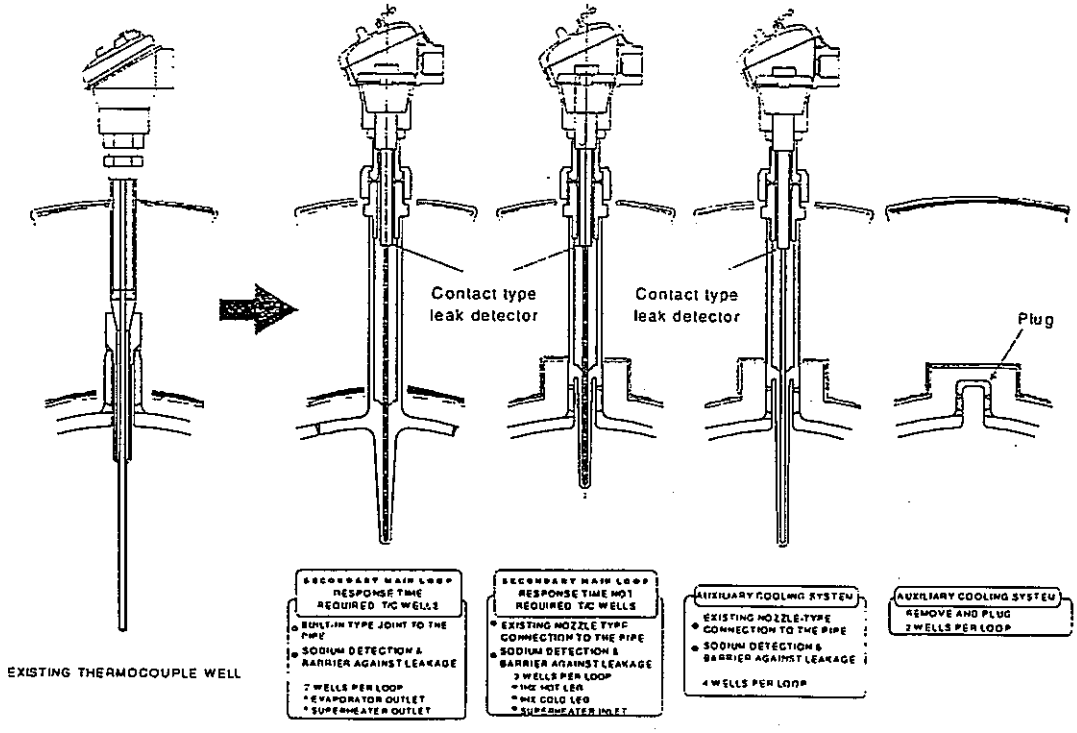
- 2ry Cooling System

  Structure Improvement : 42pieces
  Removal : 6pieces

- Water-Steam System

  Exchange or Removal : 37pieces

Modified Thermocouple Design Concepts



3.PROTOTYPE FBR "MONJU"(4)

MEASURES OF 2RY SYSTEM AGAINST SODIUM LEAKAGE

- Improvement on the Sodium Leakage Monitoring System

  -Installation of *the Air Atmosphere Cell Monitor*
  -Installation of *the Remote Supervision TV Camera*
  -Installation of *the Sodium Leakage Unified Monitoring System*

- Improvement on the Preventing and Mitigating Countermeasure of the Leakage

  -*Shortening of the Sodium Drain Time* required(20min.)
  -*Quick and automatic shutdown of the Ventilating Installation*
  -Sure extinguishment and preventing recombustion by *the Nitrogen Gas Injection System*
  -Installation of *the Insulation to wall and ceiling concrete*

4. DFBR DESIGN STUDY

- Design Study for Plant Optimization(Phase-II) by the JAPC is underway.

- Focused Issues of the Study in 1997:
  - Strengthening Sodium Leak & Fire Measures
  - Improving the Reliability of Plant
  - Simplifying the Structure
  - Improving the Construction Cost(DFBR<1.5*LWR)

- Fuel Design & CDA Analysis are carried by PNC as taking partial charge of the Design Study.

- Concrete Program of the DFBR Construction will be decided after that Operating Experience of the "Monju" will be sufficiently accumulated.

5. R&Ds(1)

R&D Items for DFBR among 4 Organizations in FY 1997

- Design Study on Plant System(J:Total Plant Design
  P:Fuel Design & CDA Analysis)
- Safety(P, J, A)
- Core & Fuel(P, J, A)
- Plant Components & Equipments(J, P, C)
- Structure/Material/Thermal-Hydraulics/Seismic(J, P, C)
- Large-Scale Tests (Partial Tests:P, Others:Not yet decided)
- Reflection of Experiences in the "Monju"(P, J)

* J : JAPC, P : PNC, A : JAERI, C : CRIEPI

5. R&Ds(2)

R&D Items for the Commercialization among 4 Organizations

- Safety(P, A, J)
  Core Safety/PlantSafety/PSA etc.
- Core & Fuels(P, A, C, J)
  Core Physics/Advanced Mox Fuel/MA Burning etc.
- High-Temperature Structural Systems(P, J, C)
  Structural Design Method/Sodium Technology etc.
- Plant Technologies(P, J, C)
  Operating Management/Maintenance & Repair Technology
  Large Scale EMP/Double Wall SG/3D Seismic Isolation etc.
- Design Study(P, J, C)
  Commercialization Plant/2ry System Elimination Plant etc.

International Cooperations

CDA

- CABRI-RAFT Program,Improvement & Verification on *SAS-4A,SIMMER-III* (FZK,CEA/IPSN - P)

Fuel & Cores

- Bundle Irradiation in Phenix(CEA - P)
- PIE of the Irradiated Cylindrical Pellet Fuel in PFR (BNFL/AEA,CEA - P)
- Fuel Material Development(CEA - P)
- Metal Fuel Development(UKAEA - C)
- MA Incineration(ORNL - A,CEA - P)

System & Components

- EMP Development(GE/ETEC - J) etc.

NUCLEAR POWER IN KAZAKHSTAN and CURRENT STATUS OF THE BN-350 FAST REACTOR

Presented at the 31th Annual Meeting of the
International Working Group on Fast Reactors
IAEA, Vienna, May 12-14, 1998

by S.Krechetov
Atomic Energy Agency

Atomic scientific-industrial complex of Republic of Kazakstan consist of:
Number of natural uranium mines [using the mining and underground leaching techniques], two plants of U3O8 production at Aktau and Stepnogorsk towns, metallurgical plant producing fuel pellets for RBMK and VVER fuel assemblies. Fast breeder reactor with sodium coolant BN – 350 at Aktau.
Three different type of the research reactors on the territory of the former Semipalatinsk Nuclear Test Site and one research reactor nearly Almaty are exploiting for the investigation in field of reactors nuclear safety and other type of investigations.
The average share of BN-350 in total electricity production is 0.7%.
Taking into account common condition industrial in Kazakhstan have no significant improvement the total electricity production on goal and oil station stayed on the same level as in 1996.
According to government decision in 1998 the following structure of atomic complex have been established (Fig.1).
Several rather serious events from last meeting should be mentioned.

- In January 1998 the Provision of licensing in nuclear field was signed by Prime Ministry and now Kazakhstan have all necessary acts for starting this process.

- In April 1998 the General Program of development atomic scientific and industrial complex of Kazakhstan had been reported to Government and got approval in whole. In particular this program are including the design and construction NPP for electricity production on the lake Balhash, and two NPP for heating Almaty and new capital Akmola.

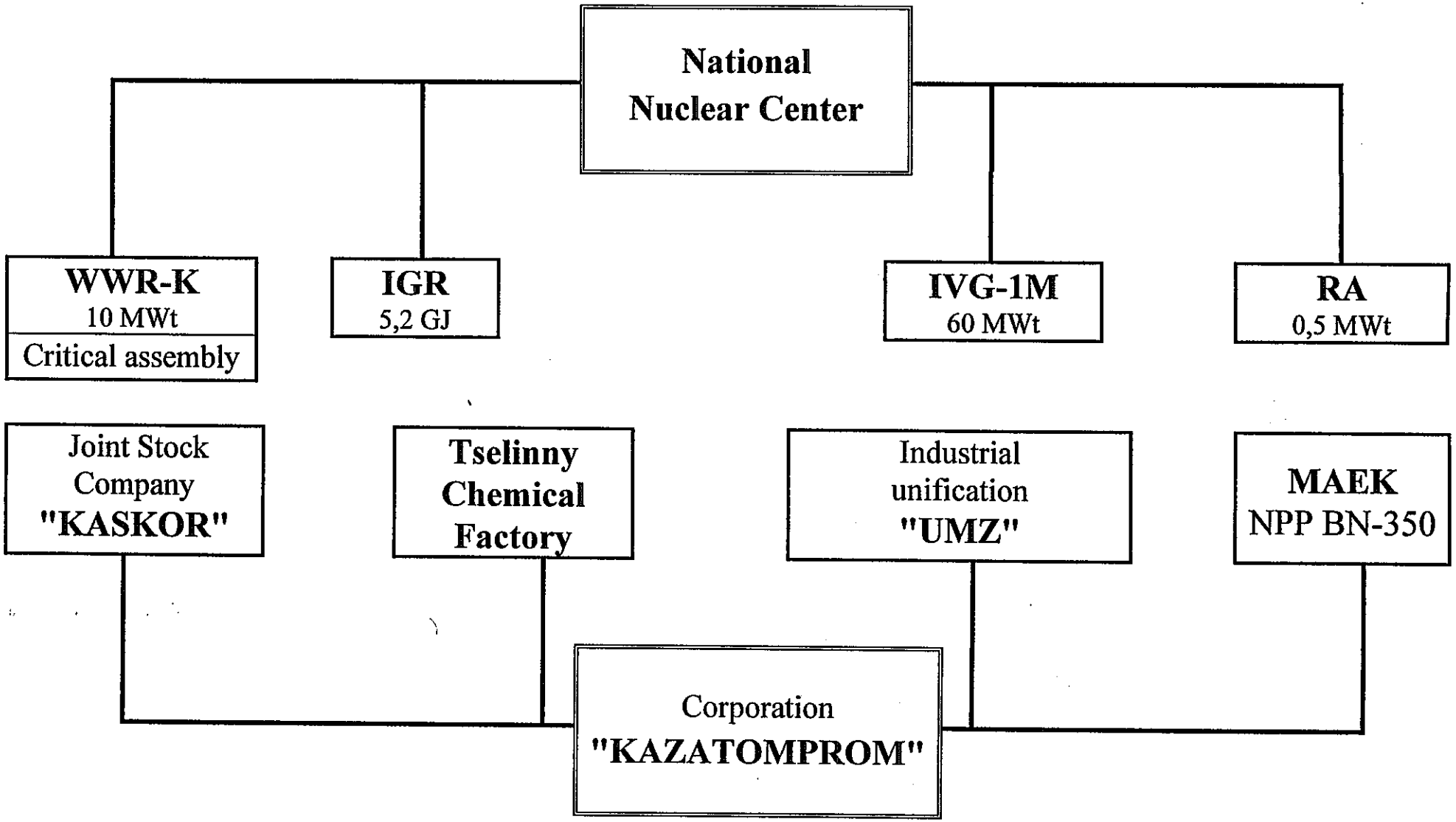
- In April 1998 the law on Radiation protection had got approval of Parliament and now President should sign it.

- In January the Nuclear Technologies Safety Center (NTSC) had been established by group of organizations such as KAEA, NNC, University, Nuclear Society of Kazakhstan, Center of standardization and Almaty local administration.
  NTSC have established as a society independent experts in the field nuclear safety (projects expertise, safety investigations and etc.)

Under Kazakhstan-USA bilateral agreement Argonne National Laboratory is providing to us technical and methodological support. As a first step it will be expertise of BN-350 nuclear safety aspects for long-term spent fuel storage and decommissioning program. We hope that in near future NTSC will collaborate with other research centers.

BN-350 was in operating since February 1997 up to March 1998 with one outage for fulfillment regular maintenance and works on safety justification program for annual permission.
A lot of works were carrying out on characterization of spent fuel cladding conditions.
According to minutes from November 1996 "On harmonization of International assistance for BN-350" in February 1998 have been conducted International workshop on BN-350 decommissioning program.
In July 1998 will be 25 years as BN-350 had produced first electricity and for such event the short article (enclosed) were published in ATW magazine.

Fig. 1 Structure of Atomic Scientific-industrial Complex of Kazakhstan.



"THE 25 YEARS IN SUCCESSFUL OPERATION."

BN-350 is the first in the world industrial fast breeder reactor with liquid sodium cooling system. NPP BN-350 is located nearest to Aktau on a peninsula Mangyshlak on coast of the Caspian sea in Kazakhstan. NPP BN-350 has 2-target purposes: production of the electric power and steam for desalination of seawater. Its enters in to the structure of MAEC (Mangyshlak Atomic Energy Complex).

NPP BN-350 was entered into operation in July 1973. Operational design term by capacity 1000 MW (thermal) is 20 years. Taking into account, the reactor worked on capacity 750 MW (thermal), and the average capacity was about 600 MW (thermal), the technological systems, pipelines of I and 2 circuits, the reactor's vessel and in-vessel systems have not spent their resource. Therefore, since July 1993, is carried out the procedure of annual prolongation of BN-350. On base of the analysis operation of installation for previous period, was made a conclusion of the developers of the equipment and reactor systems about an opportunity of prolongation of a resource of these systems, realization of their auditing and tests. It is very important research of materials science - control of metal is more 20 years old, study of its mechanical properties, corrosion and erosion of damages, the condition of welds.

BN-350 formed in those years, when did not existed the modern norms of safety. The developers of the project generated and used such level of the requirements, which allows to confirm high safety of BN-350 at present.

Nevertheless, fulfilled by the scientific chief (IPPE), Main designer (OCBM) and General designer (VNIPIET) the analysis revealed a number of deviations from requirements of the contemporary normative documents of safety NPP. The "Summary program of maintenance of safety, the exception and indemnification of deviations " was developed. The fulfillment of this program implemented by MAEC at participation IPPE, OCBM, VNIPIET and is supervised by Agency atomic energy of Kazakhstan. This work is conducting within the framework of the agreement between Ministry atomic energy of Russian Federation and Ministry of science and new technologies of Kazakhstan. The agreement between Regulatory Bodies RF and RK. on co-ordination of actions and co-operation in the field of maintenance of nuclear safety work was signed. The fulfillment this program discussed periodically on meetings of working groups relevant to safety BN-350.

Since May 1997 MAEC has become a member of WANO. BN-350 participates in number of programs WANO, namely: " Good practice ", information interchange of operational events, participation in seminars spent in frameworks WANO.

Today BN-350 has good international connections and collaboration.

In this " Summary program... " was determined five directions:

- Reconstruction and creation of new safety systems;
- Confirmation a resource of equipment, which don't commute;
- Manufacture and delivery of the equipment instead the used equipment after capital repair;
- Development of the additional operational documentation according to new requirements;
- Development of the additional documentation on a substantiation of safe operation BN-350 according to the requirements of new rules.

The calculation results of design and over-design failures confirm the resource prolongation of reactor systems. BN-350 has sufficient self- protection level, even at such over-design failures as:

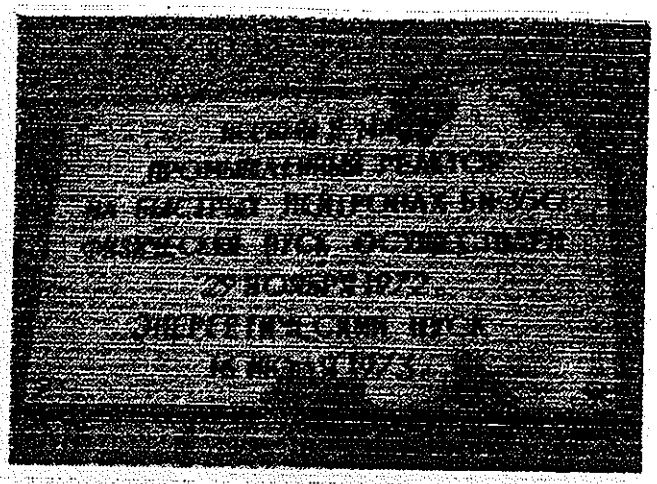
- loss of all power of main circulating pumps and all of the control rod failing;
- the absence provision of nutritious water in all steam generator and failing of emergency reactor protection ;
- break of circulation in basic loops of first circuit and operator's errors with delay inclusion of emergency heat removal system.

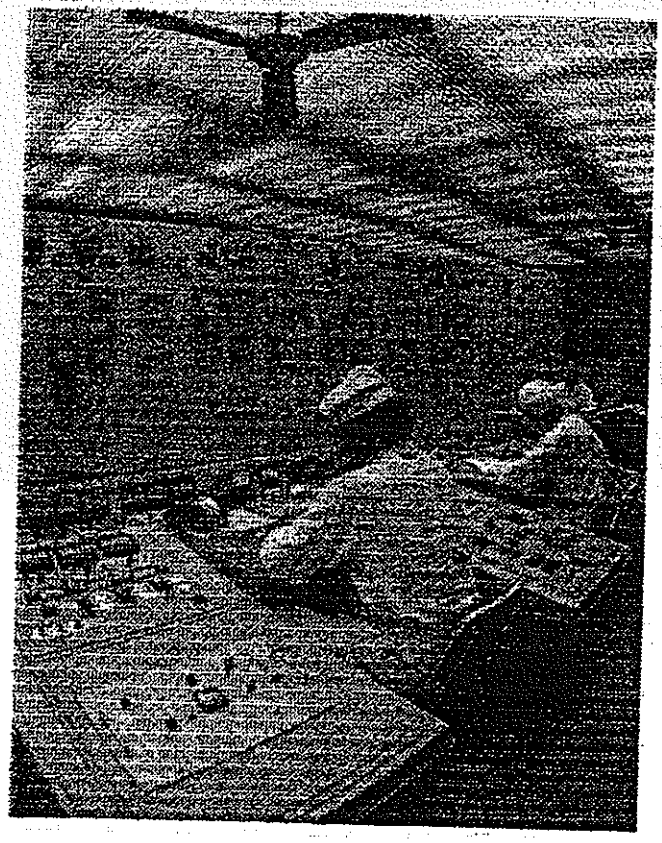
This self-protection of reactor is explained by general advantages of FBR with liquid sodium cooling system, neutron-physical characteristics of core, properties sodium as the heat-carrier, on the one hand, and by lowered capacity levels of BN-350, on the other hand.

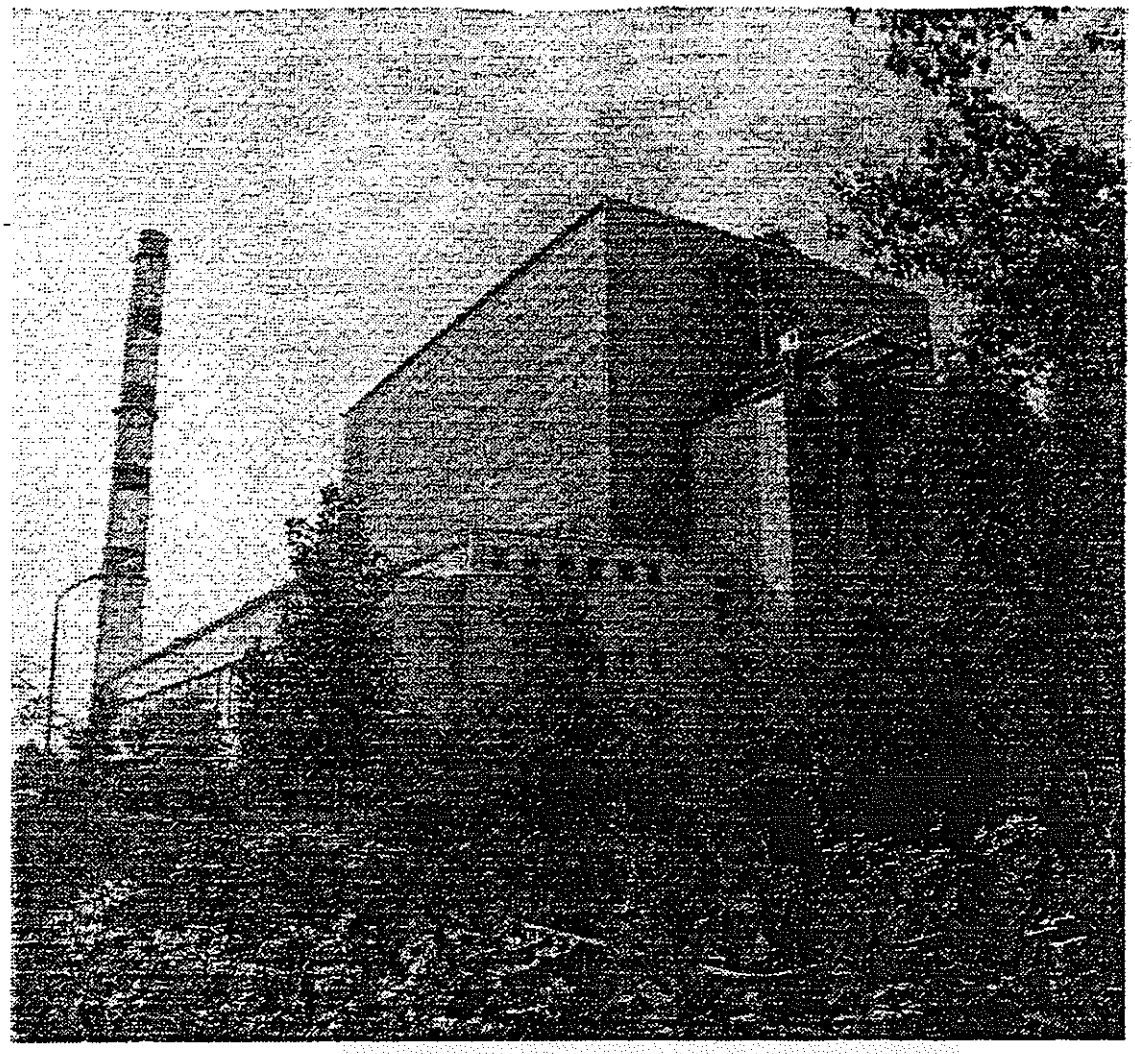
BN-350 to work steadily in the last years. Its playing the important role in water- and power supply of region: already 7 years we haven't reactor shut down by scram system. The current releases eject of radioactivity is much lower, than on thermal reactor. Vessel of reactor has neutron flux is much lower in compassion with design one, consequently materials of reactors vessel and other equipment under irradiation degraded less than expected.

Agreeably to the data of control a radiation conditions in region of arrangement plant is safe and is defined, in basic, by natural radionuclides.

Along of annual procedure prolongation the term of operation plant the work of preparation for decommissioning BN-350 was begun. This work includes creation the concept and project with attraction of international organizations, possessing experience in this work, methodical co-ordination by IAEA.







DEVELOPMENT OF KOREA ADVANCED LIQUID METAL REACTOR

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ABSTRACT

Future nuclear power plants should not only have the features of improved safety and economic competitiveness but also provide a means to resolve spent fuel storage problems by minimizing volume of high level wastes. It is widely believed that liquid metal reactors (LMRs) have the highest potential of meeting these requirements. In this context, the LMR development program was launched as a national long-term R&D program in 1992, with a target to introduce a commercial LMR around 2030. Korea Advanced Liquid Metal Reactor(KALIMER), a 150 MWe pool-type sodium cooled prototype reactor, is currently under the conceptual design study with the target schedule to complete its construction by the mid-2010s. This paper summarizes the KALIMER development program and major technical features of the reactor system.

1. Introduction

Korea has continued its quest to develop nuclear power since the introduction of its first nuclear unit Kori-1,a 600 MWe PWR, in 1978. In the mid-1980s, the country started to develop Korea Standard Nuclear Power Plant (KSNPP), a 1000MWe PWR plant. The two lead units of KSNPP are presently in operation and another six units are under construction. Along with the construction of KSNPP, Korea Next Generation Reactor (KNGR), an evolutionary 1300MWe PWR, is being developed to deliver a safer, more economical and more environmentally friendly energy source for the early 21st century. As of the end of 1997, Korea's total nuclear capacity was more than 10 GWe, with 12 units in operation. In addition, 8 units are currently under construction.

It is expected that the country's present nuclear capacity will be more than doubled by the year 2010, by which time nuclear generation will account for 40 % of total electric power production. Nuclear generation currently stands at 35 % of the total. The heavy dependence on nuclear energy raises the issue of spent nuclear fuel storage or disposal as well as that of utilization of uranium resources. To date, more than 3,000 MTU of spent fuels have been stored in At-Reactor(AR)pools of the 12 operating nuclear power plants. Taking only nuclear power plants currently in operation or under construction into account, the cumulative amount of spent fuels is estimated to reach up to about 26,000 MTU by 2030.

From the viewpoint that liquid metal reactors(LMRs) have the potential of enhanced safety utilizing inherent safety characteristics and of resolving spent fuel storage problems through proliferation-resistant actinide recycling, LMRs appear to be the most promising nuclear power option of the future. In this context, the KALIMER development program was launched as a national long-term R&D program in 1992 and has been carried out by Korea Atomic Energy Research Institute(KAERI) since then.

The objective of the KALIMER Program is to develop an inherently and ultimately safe, environmentally friendly, proliferation-resistant and economically viable fast reactor concept. The KALIMER core is initially designed with 20% enriched uranium metallic fuels, which generates a net negative reactivity with an inherent safety characteristics. KALIMER also has inherent passive means of negative reactivity insertion and decay heat removal, sufficient to place the reactor system in a safe stable state for bounding ATWS events without significant damage to the core or reactor system structure. The reactivity control and shutdown systems result in extremely high shutdown reliability. High seismic margins are achieved by a simple horizontal seismic isolation system. Recycling of transuranic elements by the pyroprocessing process would avoid the expense and potential long-term risk of their disposal in a geological repository, and would provide increased proliferation resistance.

2. KALIMER Development Schedule

Up until July 1997, efforts had been concentrated on the development of basic sodium technologies and design methodologies unique to the LMR design and operating characteristics. An initial design concept also was proposed through the feasibility study of various innovative design features. Based on the insight and results from the previous work, the KALIMER program plan was updated to call for the completion of the basic design and supporting R&D work by 2006. The program is to be carried out in three distinctive phases as follows :

- Phase 1(July 1997–March 2000): Conceptual design and design&analysis methods development
- Phase 2 (April 2000 – March 2002) : Optimized conceptual design and key feature evaluation
- Phase 3(April 2002-March 2006) : Basic design and validation of the design and methods

In Phase 1, an effort is being made to establish a self-consistent conceptual design of system configuration, arrangement and key features satisfying design requirements. In parallel, computer codes and methods for engineering design and analyses are being developed or updated all along. An investment is also being made on the key design features testing, such as electromagnetic pump and self-actuated shutdown system. Effort continues to be made on the development of sodium technologies, such as measurement or detection technique as well as the investigation on thermal-hydraulic and chemical behavior.

Phase 2 is a two-year program to evaluate alternative design features to optimize the reference design formulated in Phase 1 in terms of plant performance, safety and economics. A comprehensive set of computer codes and design methodologies should be developed in Phase 2.

Phase 3 covers a five-year period from 2002 to 2006. The focus of the KALIMER basic design is to produce the specific information required in the preliminary safety analysis report(PSAR) and preliminary cost estimate. Engineering and design tasks to be included are system performance analysis under normal, transient, and faulted conditions, specifications of major component and equipments, safety margin analysis, and probabilistic safety assessment(PSA), among others. Supporting R&D will be carried out to validate design codes and methods along with the performance testing of the major components or equipments.

3. Major Design Features of KALIMER

Table 1 summarizes some of the major design parameters of KALIMER, which is currently under the conceptual design phase. A salient feature of its key system designs is briefly described in the following.

Table 1. KALIMER Key Design Parameters

OVERALL		**PHTS**	
Net plant Power, MWe	150	Reactor Core I/O Temp., °C	386.2 / 530.0
Core Power, MWt	392	Total PHTS Flow Rate, kg/s	2143.1
Gross Plant Efficiency, %	41.5	Primary Pump Type	Electromagnetic
Net Plant Efficiency, %	38.2	Number of Primary Pumps	4
Reactor	Pool Type		
Number of IHTS Loops	2		
Safety Shutdown Heat Removal	PSDRS		
Seismic Design	Seismic Isolation Bearing		
		IHTS	
CORE		IHX I/O temp., °C	339.7 / 511.0
Core Configuration	Radially Homogeneous	IHTS Total Flow Rate, kg/s	1803.6
Core Height, mm	1000	IHTS Pump Type	Electromagnetic
Axial Blanket Thickness, mm	0	Number of IHXs	4
Maximum Core Diameter, mm	3447	Number of SGs	2
Fuel Form	U-10% Zr Alloy		
Enrichments (IC/OC) for Equilibrium Core, %	14.4 / 20.0		
		Steam System	
Assembly Pitch, mm	161.2	Steam Flow Rate, kg/s	175.5
Fuel/Blanket Pins per Assembly	271 / 127	Steam Temperature., °C	483.2
Cladding Material	HT9	Steam Pressure, MPa	15.50
Refueling Interval, months	12		

Core and Fuel Assembly

The KALIMER core system is designed to generate 392 MWt of power. The reference core utilizes a homogeneous core configuration in radial direction with two driver fuel enrichment zones, surrounded by a layer of blanket assemblies. The core layout, shown in Figure 1, consists of 96 driver fuel assemblies, 42 radial blanket assemblies, 6 control rods, 1 ultimate shutdown system(USS) assembly self-actuated by a Curie point electromagnet,6 gas expansion modules(GEMs), 48 reflector assemblies, 54 B_4C shield assemblies, 72 shield assemblies, and 54 in-vessel storages(IVSs) in an annular configuration. The in-vessel storages(IVSs) are located between the stainless steel shielding zones. There are no upper or lower axial blankets surrounding the core. The reference core has an active core height of 100 cm and a radial equivalent diameter (including control rods) of 172 cm; the height-to-diameter ratio (H/D) for the active core becomes 0.581. The physically outermost core diameter of all assemblies is 344.7 cm. Table 2 shows major nuclear design parameters for an equilibrium core with a cycle length of 12 months.

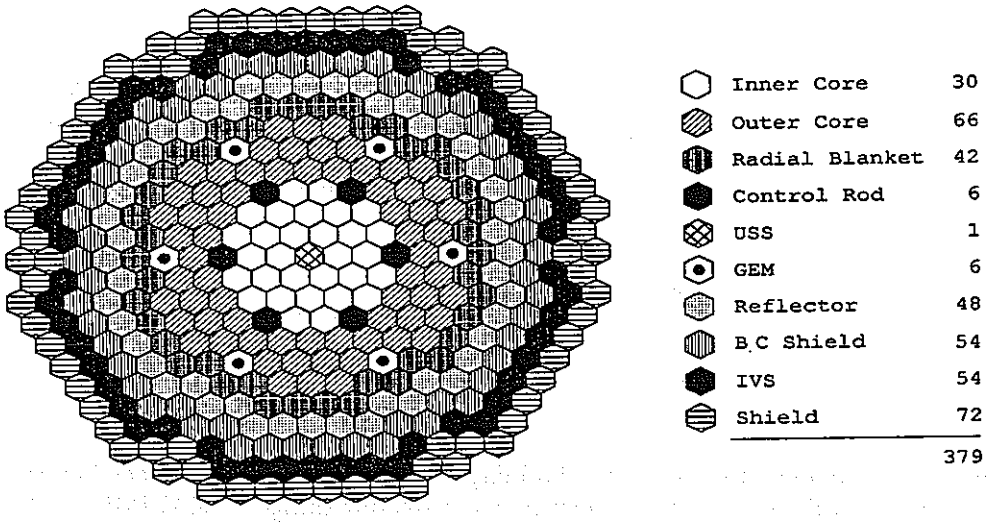


Fig 1. KALIMER Core Layout

The core structural material is HT9. It's low irradiation swelling characteristics permits adequate nuclear performance in a physically small core. The fuel pin is made of sealed HT-9 tubing containing metal fuel slug in columns. The fuel is immersed in sodium for thermal bonding with the cladding. A fission gas plenum is located above the fuel slug and sodium bond. The bottom of each fuel pin is a solid rod end plug for axial shielding. The driver fuel, blanket fuel, reflector, and shield assemblies use identical structural components with only the bundle and its mounting grid changing from one assembly type to the other. The control assemblies use outer hardware (nosepiece, duct and handling socket) that is identical to that in the other assemblies. Reflector assemblies contain solid HT9 rods. The absorber assemblies use a sliding bundle and a dashpot assembly within the same outer assembly structure as the other assembly types. In all assemblies, the pins are in a triangular pitch array. The bottom end of each assembly is formed by the nosepiece which provides the lower restraint function and the coolant inlet.

Reactivity and power are controlled by means of the control rod system in the driver fuel region of the core. The control rod design satisfies both the one rod stuck condition and the unit control rod worth condition against the unprotected transient over-power(UTOP) event. The gas expansion modules(GEMs) are passive reactivity feedback assemblies that insert negative reactivity into the core during a loss of flow. The Self-Actuated Shutdown System(SASS) located at the center of the core is designed as an ultimate shutdown system by using a Curie point electromagnet which loses its magnetic force holding the shutoff rod when the temperature of the primary sodium reaches the curie point, hence a passive shutdown can be achieved.

Table 2. Major Nuclear Design Parameters for an Equilibrium Core

Average Breeding Ratio	0.669
Burnup Reactivity Swing (delta k/k_{boc})	1.150
Average Fuel Burnup (MWD/kg)	
Driver Fuel(Inner/Outer)	25.35/22.82
Radial Blanket	2.31
Peak Fuel Discharge Burnup (MWD/kg)	
Driver Fuel(Inner/Outer)	39.67/42.67
Radial Blanket	6.27
Feed Enrichment (Inner/Outer) (%)	14.41/20.00
Fissile Inventory at BOC (kg/cycle)	1715.15
Total Fissile Gain (kg/cycle)	-63.99
Average Power Density (W/cc) For Driver Fuel at BOC	160.2
Peak Linear Power (W/cm)	208.5
Power Peaking Factor for Driver Fuel(BOC/EOC)	1.603/1.569
Peak Neutron Flux (x 10^{15} n/cm^2 s)	2.304
Peak Fast Fluence, MOEX (x 10^{23} n/cm^2)	1.399

Reactor Structure

The reactor vessel has overall dimensions of 17.6m height, 7.02m diameter, and 5cm thickness in preliminary concept design and is composed of a cylindrical shell with an integral hemispherical shell bottom head. The structural integrity and safety of the reactor vessel has been achieved by providing no penetration nozzle and no attachments other than the core support structure. The shape of the core support structure is skirt-type. All equipment like IHX, EM Pump, IVTM, and UIS are supported by a reactor head and a rotating plug is adopted for the refueling operation. The support barrel, which is a major component of reactor internal structures, serves as a redan to separate the hot sodium pool and cold sodium pool and as a support of internal structures including the reactor core. The containment vessel, which encloses the reactor vessel, is easy to access from the reactor vault so that the inspection and maintenance of the vessel can be easily accomplished. General arrangements for NSSS and reactor building are tentatively developed as shown in Fig.2&3.

The seismic base isolation for the rector building using high damping rubber bearings has been adopted to achieve sufficient structural integrity and economic design of KALIMER, when subjected to the design basis earthquake such as a horizontal Safe Shutdown Earthquake of 0.3g. The development of a design concept adopting 3 dimensional seismic base isolation is under consideration to reduce both horizontal and vertical seismic responses.

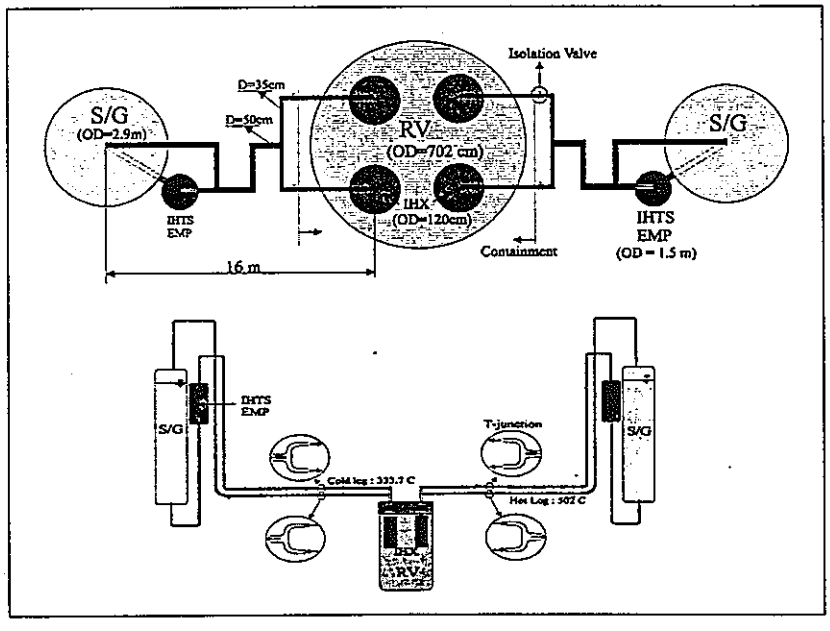


Fig 2. General Arrangements of NSSS

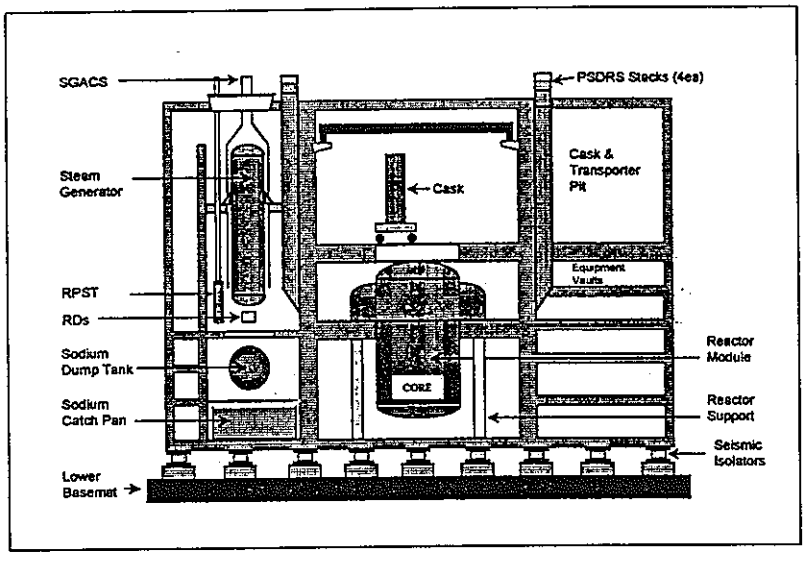


Fig 3. General Arrangement of Reactor Building

Heat Transport System

A superheat steam cycle is implemented to have a high plant efficiency noting that high thermal efficiency reduces the heat discharge from the plant, resulting in less impact to the environment. IHTS consists of two loops and each loop is equipped with one steam generator unit to simplify the system design and increase the plant operation flexibility. For safety, large system thermal inertia is achieved by using a pool based primary system. Strong emphasis has been given to the prevention and mitigation of possible sodium-water reaction events for the IHTS piping routing. Valves for isolation of IHX from the sodium-water reaction products are installed at each IHTS piping penetrating the containment. The system reliability is improved by using electromagnetic (EM) pumps, which do not have moving parts, for both of the primary and intermediate coolant pumping. The low momentum inertia of the EM pump is compensated for by using an auxiliary device which keeps a certain amount of rotating kinetic energy when the EM pump runs normally but supplies electricity from the rotating kinetic energy to the EM pumps when the electricity supply to the pumps is interrupted. The operating temperature and component size were determined to make the net plant thermal efficiency higher than 38%. Preliminary analysis on economic effects was made in setting up the plant heat balance, as shown in Figure 4, for system design optimization.

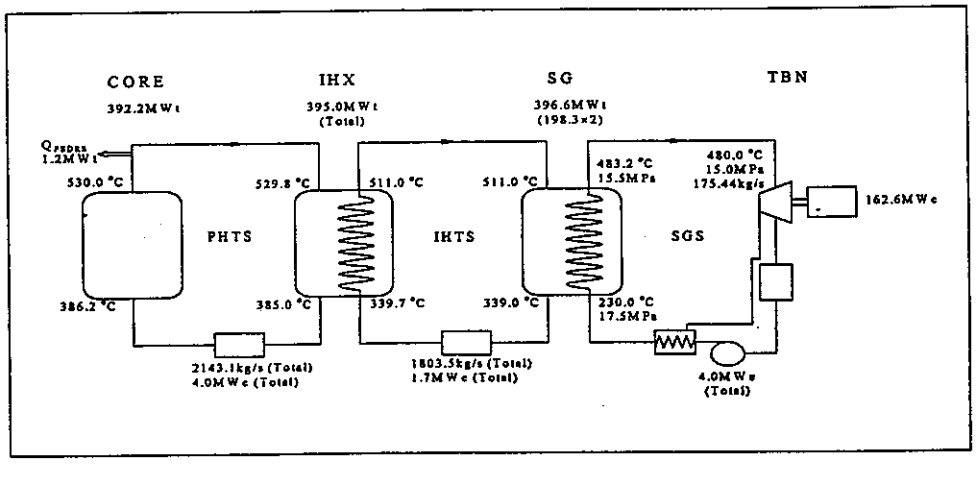


Fig 4. KALIMER Plant Heat Balance

Residual Heat Removal System
In KALIMER, the shutdown heat removal system is designed with the emphasis on system reliability to achieve a higher level of plant safety. Safety grade heat removal is achieved by the Passive Safety Decay Heat Removal System(PSDRS), which consists of the air path around the containment vessel and takes the decay heat from the reactor pool and discharges the heat to the atmosphere. Normally the decay heat is removed by steam generators and the condenser. During the maintenance of any IHTS, heat is removed by the remaining IHTS loop. Also there is the Steam Generator Auxiliary Cooling System(SGACS) to aid the decay heat removal. SGACS induces natural or forced circulation of atmospheric air past the shell side of steam generator. Intensive analysis on the system performance and design parameter changes is under progress for system level design optimization.

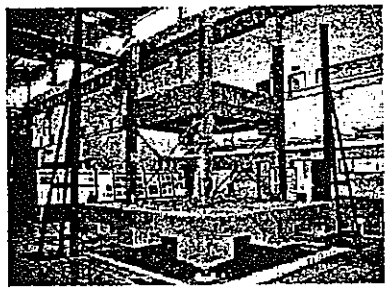
4. R&D Program Status and Plan

Core Design & Safety Analysis
The K-CORE system, an integrated system of KALIMER physics calculation modules, has been under development to provide major data links among the principal core design modules. The present works in progress include the developments of cross-section generation methodologies, a perturbation theory module and a reactivity feedback modeling in the kinetics module for coupled transient calculations. The benchmark experiment for uranium metallic fueled core having similar KALIMER core characteristics were performed jointly with Russia in 1997 and an extensive analysis on the experimental data is now being carried out to produce backup data for the verification and validation of the K-CORE system. Some of modelings employed in flow grouping and core temperature distribution modules have been improved, and core thermal-hydraulic subchannel analysis code is being developed

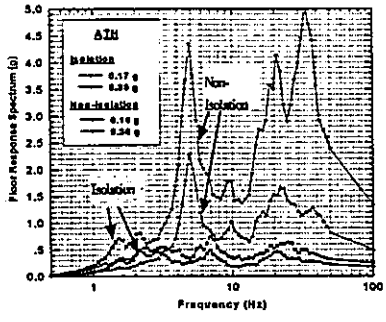
A plant-wide transient analysis code is being developed for the analysis of KALIMER's inherent safety and for the assistance in the development of design, where new design features will frequently demand not just new data but new models. A simple model for the scoping analysis of HCDA energetics is also being developed for the analysis of ultimate safety.

Fluid System Design and Analysis
Various computer codes have been under development for thermal hydraulic design of the KALIMER system. The code development has been so far focused on the support of the initial stage of the system design. The codes for equipment sizing, HSGSA, 2DHX, and ASTEEPL, and the code for preliminary system analysis, LSYS and PARS have been developed. An analysis code system is also being setup for detailed three dimensional analysis of the thermal hydraulic behaviour of the reactor pool using the codes CFX-4 and COMMIX-1AR/P which are supplied by foreign organizations. The code system development efforts are now being shifted to verification of the developed codes and setting of the additional code systems for the support of the main phase work of the conceptual system design. For the verification, experimental data are planned to be generated within the KALIMER development program but international cooperation for the code verification data is also sought. The additional code systems for the development are for the analyses of steam generator stability, plant operation and performance, and free surface behaviour of the pool sodium.

Structural Analysis & Component Development
Structural R&Ds for KALIMER are mainly seismic base isolation and high temperature structural design technology. Structural dynamic test for base isolated structure equipped with 2 dimensional 4-1/8 scale high damping rubber bearings were performed using 30 ton-6 dof shaker. Fig-5 shows test model structure and seismic response results at upper slab of the test model to artificial time history input of SSE 0.3g. Development and tests of 3D isolators to identify the performance by using shear compression tests and shaker tests are in progress. Methodology development for high temperature structures such as thermal striping and ratcheting is focused to evaluate thermomechanical fatigue, creep, and crack damages. In the area of key component development, the submersible-in-pool type electromagnetic pump of operating temperature of 600 °C and 200 ℓ / min maximum flow rate is developed using the theory of magneto-hydraulics and the equivalent circuit analysis. Its prototype was manufactured and its operation tests were performed. SASS and IVTM were developed and their mock-ups are manufactured and their theoretical validation tests were performed.



Test Model



Test Results of Isolated and Non-isolated Test Model to Artificial Time History 0.3g & 0.15g

Fig 5. KALIMER Seismic Isolation Test Program

Sodium Technology

Small-scale sodium experiments have been performed to investigate the coolant thermal-hydraulic behavior, such as turbulent mixing in compact reactor space, flow reversal by natural circulation with an electromagnetic pump operation, and decay heat removal by wall cooling, among others. The effect of the magnetic field on pressure drop is currently being investigated. Sodium experiments continue to be performed to develop the technologies to measure such parameters as differential pressure, local flow rate, and void fraction.

R&D work on sodium - water reaction has been carried out in the areas such as small water leak experiments for the determination of a design base leak rate, safety evaluation of the heat transport system in case of large leaks, development of reliable and real time detection system of water leaks using the acoustic signal as well as hydrogen detection. Additionally, sodium fire characteristics and phenomena are investigated with about $50m^3$ of rectangular type fire cell. Analyses of various types of sodium fire phenomena, development of sodium leak detection system, and fire extinguishment, prevention and mitigation, aerosol filter and scrubbing devices will be carried out.

5. International Cooperation

KAERI has been in international collaboration with a number of organizations having LMR experiences, such as General Electric(GE) in US, AEA Technology in UK, IPPE in Russia, among others. Briefly described in the following are some of the recent international cooperation works.

As part of the partnership agreement between GE and KAERI(1995-1997), a number of KAERI staff had been assigned to GE to work together for the evaluation of design concepts of KALIMER over the two years in 1995 and 1996. GE also evaluated design options of the KALIMER in-vessel transfer machine(IVTM) and supported KAERI on the baseline schedule of the KALIMER program in 1997.

More recently, AEA Technology and KAERI put in place the 'Agreement relating to Joint R&D for the KALIMER Project' early this year and ,as a first-year cooperation part of it, collaborated on the computer progams for the KALIMER core design and safety analysis and discussed various topics to collaborate in the following years.

KAERI has been collaborating with IPPE in Russia since 1995 in carrying out critical experiments

with uranium metallic fuels at the BFS facility, to secure core physics experiment data representing KALIMER core concept so as to verify and validate, to greatest extent possible, the design concepts as well as core physics design methodologies. Early this year, both organizations reached an agreement on collaborating in data acquisition, computer code development and validation for sodium-water reaction.

KAERI has been participating in multinational cooperation programs; OECD/NEA program for the measurement of effective delayed neutron fraction at the FCA in Japan, OECD Halden Reactor Project in Norway, and the IWGLMFR sponsored by IAEA.

The KALIMER development program requires that the conceptual design of KALIMER be done by March 2000. KAERI plans to collaborate with one of the major organizations having LMR experiences and expertise for engineering reviews and consultations on the evolving details of the conceptual design of KALIMER ,and for the support on developing and validating the design methods, computer codes and database, as well.

6. Conclusion

The national long-term R&D program updated in 1997 requires Korea Atomic Energy Research Institute(KAERI) to complete by the year 2006 the basic design of KALIMER, along with supporting R&D work, with the capability of resolving the issue of spent fuel storage as well as with significantly enhanced safety. We realize that it is quite a task to achieve the requirements all by ourselves, with a limited program budget and LMR experiences. We believe, however, that we can make it through international cooperation with the countries having LMR experiences and their moral support on the program.

STATUS OF FAST REACTOR ACTIVITIES IN RUSSIA

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ABSTRACT

This paper outlines state-of-the-art of the Russian nuclear power as of 1997 and its prospects for the nearest future. Results of the BR-10, BOR-60 and BN-600 reactors operation are described, as well as activity of the Russian institutions on scientific and technological support of the BN-350 reactor.

Analysis of current status of the BN-800 reactor South-Urals NPP and Beloyarskaya NPP designs is given in brief, as well as prospects of their construction and possible ways of fast reactor technology improvement.

Studies on fast reactors now under way in Russia are described.

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2. Fast reactors operating experience
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1. NUCLEAR POWER STATE-OF-THE-ART AS OF 1997

In 1997 in Russia there were 29 nuclear power units in 9 NPPs having total installed capacity of 21.242 GW, namely:

- 13 power units with vessel type VVER reactors (including 6 VVER-440 power units and 7 VVER-1000 power units);
- 15 power units with uranium-graphite channel type reactors (including 11 power units with the RBMK reactors and 4 power units with EGP-6 reactors);
- 1 power unit with the fast reactor (BN-600).

27 power units having total capacity of 19.242 GW were in operation in 1997, while 2 RBMK-1000 reactor power units were modified (one power unit in Leningradskaya NPP, and the other in Kurskaya NPP[1]).

The total amount of energy produced by the NPPs in 1997 was $108.286 \cdot 10^6$ MW· hours, this value being 99.5% of the previous year production.

Average load factor value for all NPPs was 58.2% in 1997 as compared to 58.32% value of 1996. Values of scheduled and unscheduled losses of electricity production in 1997 were respectively 29% and 13%, total losses amount being 42%.

The total number of incidents occurred on NPPs during 1997 was 79, only 3 of this number being safety related events. These three incidents are attributed to the first level of INES international scale.

Diagrams presented below are showing NPP incidents, including those safety related, over the previous 6 year period (Fig.1) and their correlation to the INES scale (Fig.2). These diagrams demonstrate stable tendency of NPP incidents number decrease.

During 1997 there were 18 unscheduled disconnections of the power units from the grid, 9 out of them being caused by the reactor scrams.

Operating reliability of Russian NPPs is third following Japan and Germany.

"Program of Nuclear Power Development in the Russian Federation for 1998-2005 Period and Prospects up to 2010" is now considered by the Russian Government.

According to this Program the following activities are planned to be completed until 2000:

- modernisation of existing NPPs in order to continue their safe operation;
- resumption of construction of the frozen NPPs.

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[1]Kurskaya NPP power unit was put into operation in December 1997.

Fig. 1. Trends of operational events at NPPs in Russia during 1992-1997

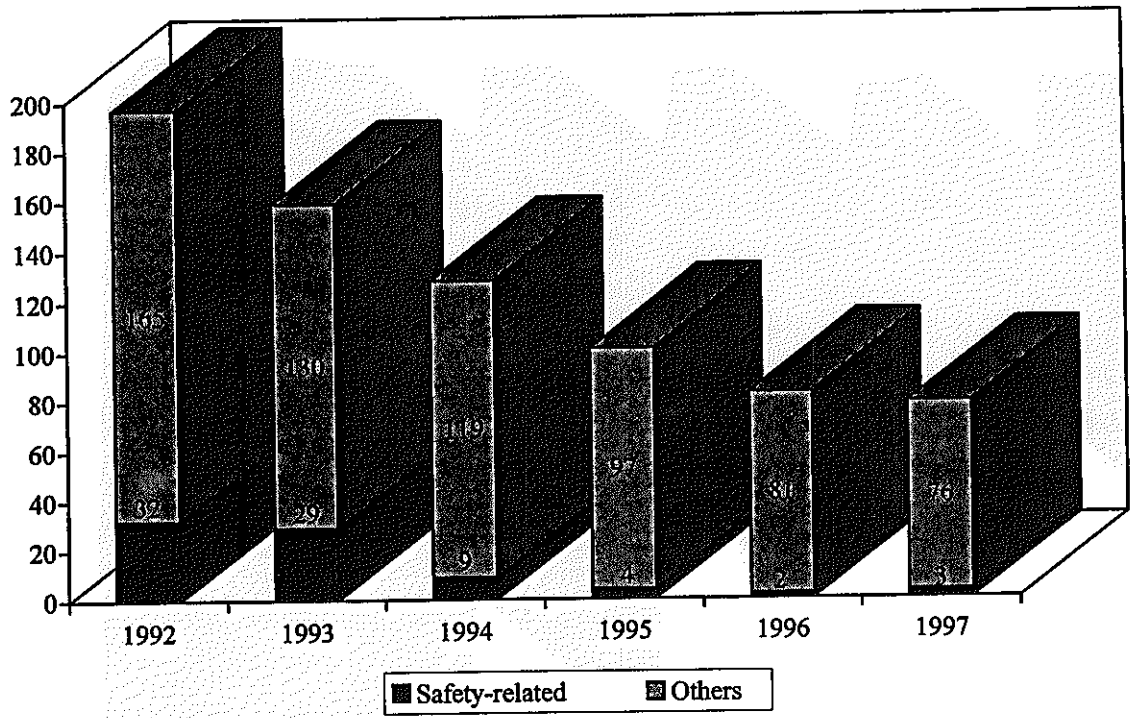


Fig. 1. Trends of operational events at NPPs in Russia during 1992-1997

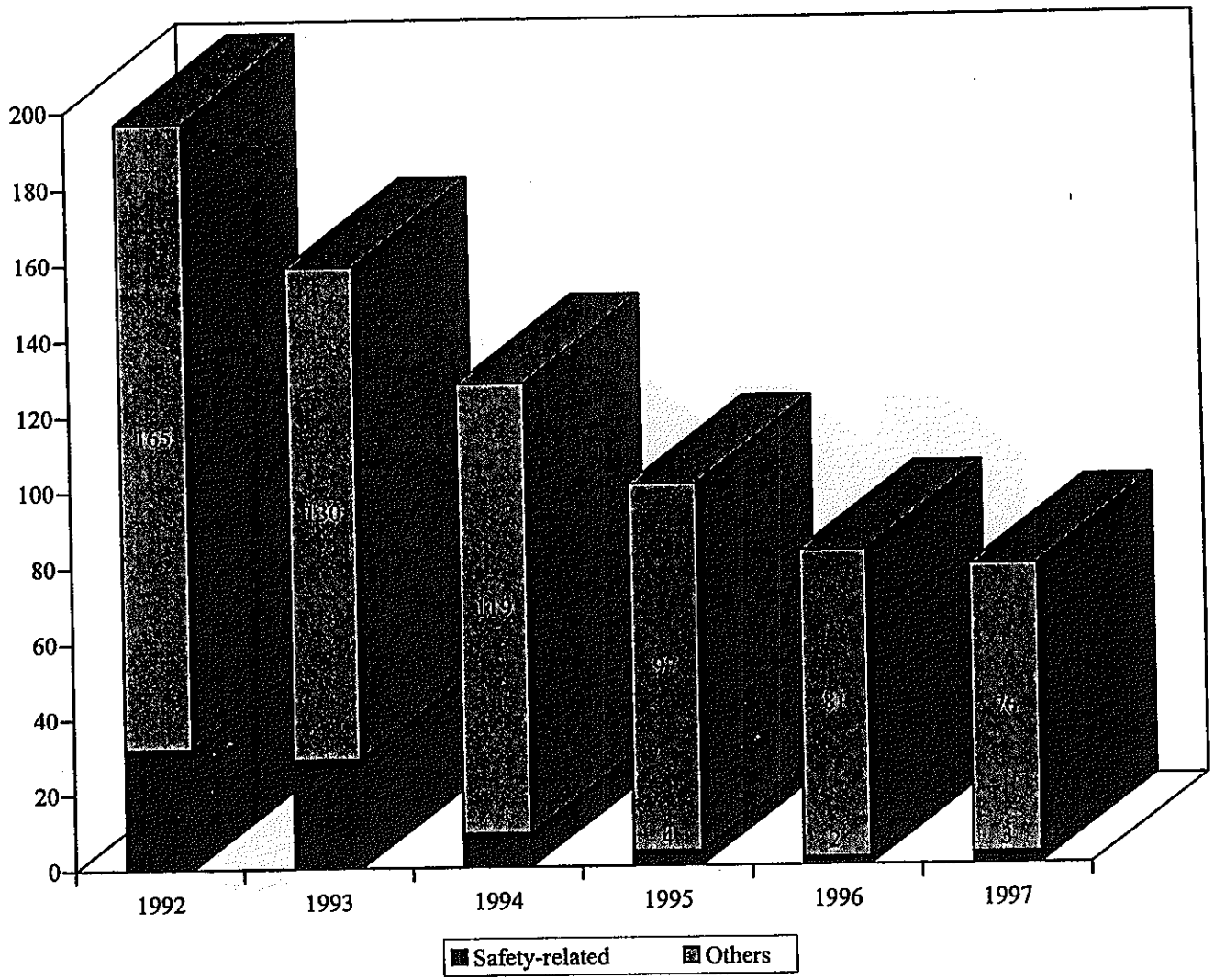
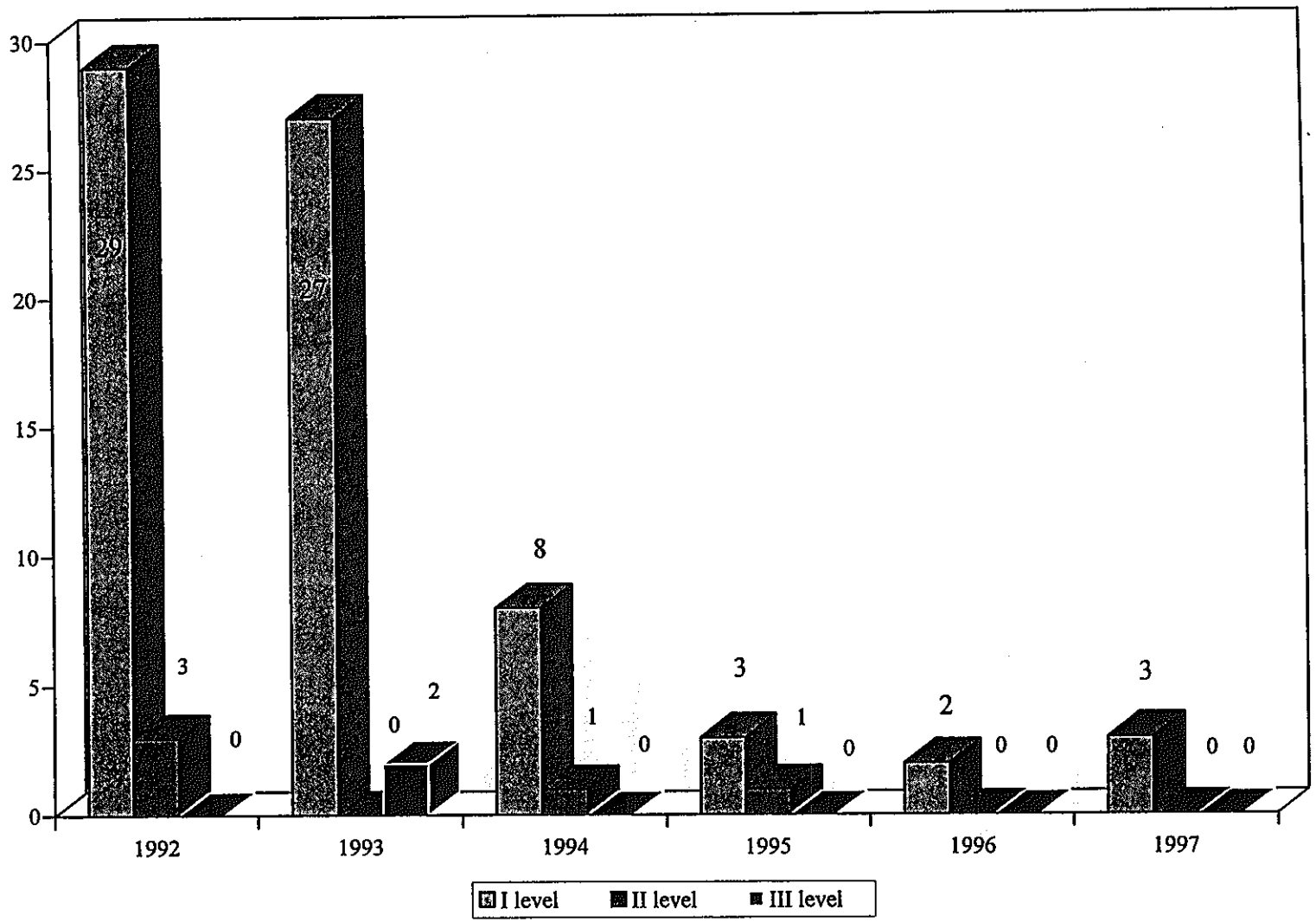


Fig. 2. Trends of safety-related operational events at NPPs during 1992 -1997



Construction of the front-end new generation power units with VVER-640 reactor (Sosnovy Bor) and VVER-1000 reactor (Novovoronezhskaya NPP-2) is to be completed by 2005.

Also, the license has been issued for the construction of the BN-800 reactor power units on Beloyarskaya NPP and South-Urals NPP.

2. FAST REACTOR OPERATING EXPERIENCE

Three fast reactors are currently in operation in Russia, namely BR-10, BOR-60 and BN-600 reactors. The BN-350 fast reactor NPP is operated in Kazakhstan under the scientific and technological support of the Russian specialists.

2.1. BN-600 reactor NPP

Histogram of the BN-600 reactor operation in 1997 is shown in Fig. 3.

There were two reactor shutdowns during 1997, caused by the core refuelling and scheduled maintenance works.

During 1997 no incidents occurred resulting in the abnormal operating conditions of the NPP, its load factor (73.0%) being the highest of those of all NPPs. Some reduction of the load factor value as compared to that of 1996 (76.3%) was caused by the increased scheduled maintenance period (90 days in 1997 as compared to 74 days in 1996).

The main characteristics of the BN-600 reactor over the previous 5-year period are presented in Table 1.

Table 1

Characteristic	Units	1993	1994	1995	1996	1997	From the start of operation up to 01.01.98
Electricity production	10^6 kW·hr	4220	4110	3695	4022	3835	65278
Heat supply	10^3 Gcal	–	–	258.7	312.8	313.5	2674.0
Load factor	%	80.3	78.2	70.31	76.3	73.00	70.00
Number of power unit shutdowns		2	2	3	2	2	79
Number of loop outages		0	0	6	3	0	66

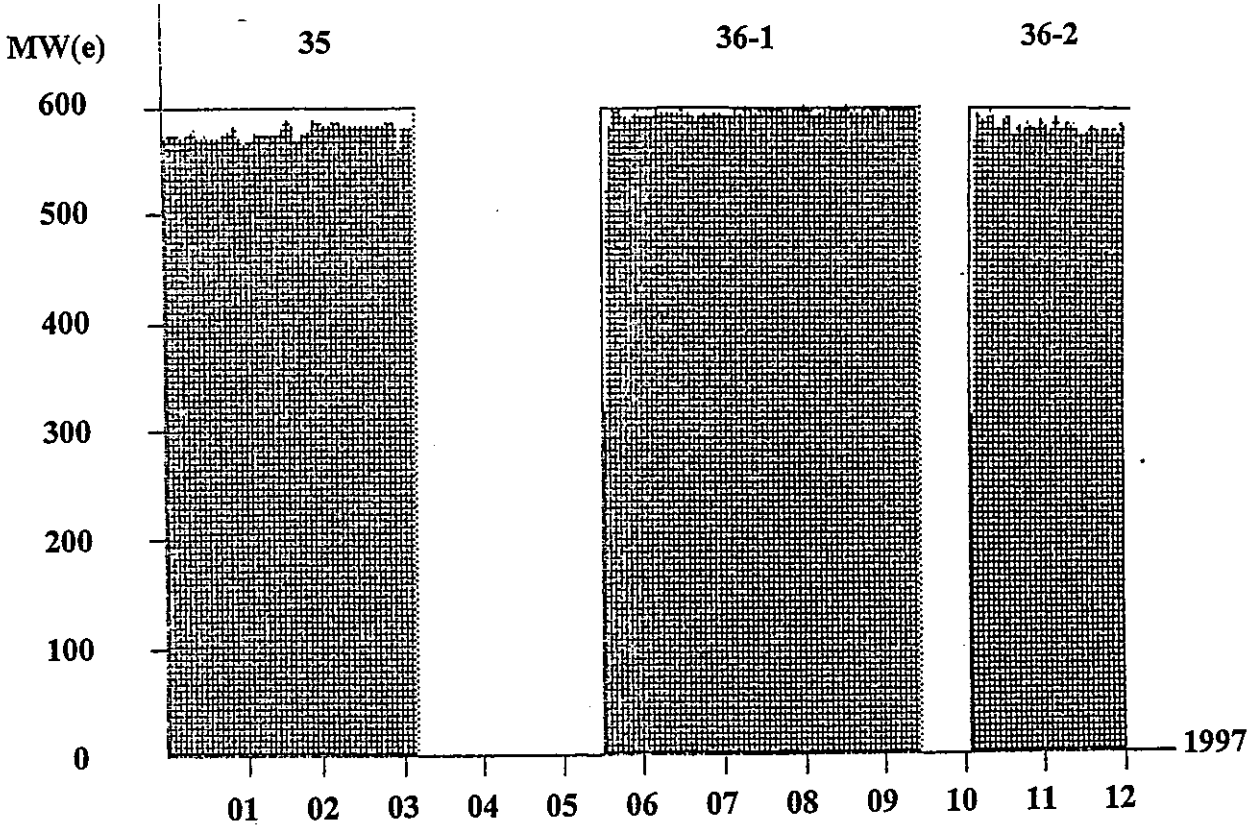


Fig. 3. BN-600 Operating Histogram

Crisis Centre has been established by "Rosenergoatom" Concern in order to render special support to the nuclear power plants of the Concern in the radiological dangerous situations. Local crisis centres are established at the NPP, and special engineering centres are established at some large research institutions for the support of the Crisis Centre. In particular, local crisis centre has been created at the Beloyarskaya NPP, and special engineering centres have been set up at the IPPE and OKBM to render timely support to the BN-600 reactor NPP in case of emergency.

In the framework of the establishment of these centres the following organising activity was fulfilled:

- expert groups were formed;
- on-line communication was organised providing transmission of ~200 parameters related to the NPP safety;
- total archives of the BN-600 reactor design and operating documents is being set up, some part of it being transformed into the electronic version.

The establishment of the engineering support centres is supposed to improve scientific maintenance of the BN-600 reactor operation.

2.2. BOR-60 experimental reactor

During 1997 the power of the BOR-60 reactor varied within the range from 48 MW to 55 MW.

The main operating parameters of the reactor for 1997 and for its whole life are presented in Table 2.

Table 2

1	2	3	4	5	6	7	8
Parameters	Units	1 quarter	2 quarter	3 quarter	4 quarter	1997	From the start of operation up to 01.01.98
Time of reactor operation on the power over minimum detectable level	hours	2146	1607	1632	1386	6771	153731
Reactor availability	–	0.99	0.735	0.743	0.627	0.774	

1	2	3	4	5	6	7	8
Maximum reactor power	MW	55	50	48	55	55	
Production of: heat electricity	MW·hr	112475 15098.4	68480 11932.8	60331 12556.8	68909 10437.6	310195 50025.6	6359219 1042012.4
Time of steam generator operation: SG-1 SG-2	hours	2141 2141	1595 1595	1597 1597	1374 1374	6707 6707	86343 39311
Heat supply of the consumers	Gcal	41481	10855	1189	19392	72917	321724

During 1997 tests of fuel, absorber and structural materials were in progress under conditions of high coolant temperatures and neutron flux.

Standard subassemblies with the vibropacked oxide fuel (UO_2, $UPuO_2$) were installed in the core.

Maximum fuel burn-up in the standard and experimental subassemblies was respectively 15.5 % h.a. and 26 % h.a. In some experimental fuel elements (3 fuel elements installed in a special demountable device ~32 % h.a. maximum fuel burn-up was achieved.

In one subassembly containing UPu-10 % Zr sodium bonded fuel elements ~8 % h.a. burn-up was achieved.

2.3. BR-10 experimental reactor

In 1997 the BR-10 reactor was in operation during 4674 hours on different power levels, the time of its operation on the rated power (7.6 - 8.0 MW) being equal to 2830 hours.

The following works were done on the reactor:

- target irradiation for the production of the isotopes required for pharmaceutical agents manufacture;
- irradiation of specimens of four types of structural materials used for the fuel element cladding;
- irradiation of the lavsan film by the fission products in the thermal neutron beam in order to produce hyperfine membranes;

- irradiation of oncological patients by the fast neutron beam.

During reactor operation on power permanent monitoring of the fuel element cladding integrity was carried out by measuring activity of delay neutron predecessors in the primary coolant and gaseous fission products in the cover gas of the primary pump vessels.

Four fuel element failures accompanied by the gas release were detected.

Fast neutron fluence (E_n>>0.1 MeV) on the reactor vessel was as high as $5.95 \cdot 10^{22}$ n/cm^2, design value being equal to $7.0 \cdot 10^{22}$ n/cm^2.

Development of basic design of the BR-10 test reactor decommissioning is under way.

2.4. Co-operation with the Republic of Kazakhstan on the BN-350 reactor NPP

Since the design lifetime of the BN-350 reactor was expired in 1993, permissions for continuation of the reactor operation are obtained annually.

Experts from Russian institutions participate in the activities on the improvement of safety systems and the NPP as a whole to meet the requirements of current regulatory documents, in the issuing of annual safety report, proving NPP safe operation in the next time period.

In 1997 maximum permissible power level of the reactor was 420 MW on condition that five loops are available for the reactor decay heat removal.

3. FAST REACTOR PROSPECTS

Fast reactor prospects can be divided into the near-term and long-term categories. The former includes plans for the BN-800 reactor NPP construction on Beloyarskaya and South-Urals sites (2005 - 2010), while the latter implies development of design of advanced NPP with the new generation fast neutron reactor having safety characteristics inherent to LMFRs and competitive engineering and cost parameters (2020 - 2030).

3.1. BN-800 reactor NPP

As it was mentioned above, a license for continuation of South-Urals NPP and fourth power unit of Beloyarskaya NPP with the BN-800 reactor was issued by the Russian regulatory authorities in December 1997.

The South-Urals NPP construction is oriented mainly to the budget financing, the Beloyarsk NPP 4 unit construction is planned to continue on joint-stock basis. The BNPP-2 joint-stock company has been created, the founders of which are Sverdlovsk region government,

"Sverdlovskenergo" power concern, Rosenergoatom concern, machine-building, construction and installation enterprises of Sverdlovsk region, and BNPP.

In November 1997 agreement was signed based on the results of the meeting of the Minister of Atomic Energy and the Governor of Sverdlovskaya region on corrections to 'The Program of Development of Nuclear Power in Russia", and further it was corrected so that the construction of Beloyarskaya NPP fourth power unit with the BN-800 reactor should be completed by 2005.

3.2. Studies on advanced fast reactors

In spite of the current decrease in the rate of commercialisation process of fast reactors (FRs), there is a common opinion that their large-scale introduction would be required after 2030-2050 in order to provide fuel for the nuclear power [1]-[2].

Construction of single commercial NPPs with the fast reactors is possible in some countries during the previous period.

Analysis of the process and nature of FR development and their prospects would lead to the conclusion given below.

Now the first stage of sodium cooled FR technology development has been completed, the basic result of which is the demonstration of their satisfactory reliability and safety. This is a principal result showing the validity of the principal engineering solutions.

The tasks of the second stage which is previous to the large scale FR introduction into the nuclear power structure can be considered as follows:

- assimilation of the experience gained, improvements to be made on its basis and development of the new technologies in the FR area;
- development of FR designs having higher safety, reliability and technical and economical parameters;
- expansion of FR application area.

Among the innovative advanced FR technologies lead and lead-bismuth alloy coolants are considered.

In the process of analysis of different options some issues should be taken into account.

Experience has been gained on the use of lead-bismuth alloy in the submarines nuclear power plants. The possibility of this coolant application for the large scale nuclear power will be finally determined after the development of related technology taking into account the requirements to the commercial power engineering.

As regards the reactors cooled with lead, design studies have been made in this area, requiring related experimental approval.

As far as the sodium cooled fast reactors are concerned, it can be stated that their potential is far from its exhaustion. As it has been shown by the studies, technical and economical parameters of the sodium cooled fast reactors can be significantly improved on condition that the required reliability and safety are provided.

The development of the concept of advanced NPP design with the sodium cooled fast reactor has been initiated recently by the Russian specialists using positive experience gained on the development, construction and operation of the BR-10, BOR-60, BN-350 and BN-600 reactors as well as on the BN-800 and BN-1600 reactors design development.

In connection with the complexity of tasks lying ahead in scientific, engineering and economical aspects it could be rather useful to combine efforts of the countries-participants of nuclear community.

There is a need to create under the aegis of IAEA a long-term (7-10 years) work program, the objective of which will be the elaboration of agreed viewpoint on extension of application area, main directions of improvement of future NPP with fast reactors and the methods of their realisation.

This program including technical committee meetings, specialist meetings, conferences, fulfilment of joint R and D studies under agreed international programs etc will benefit the increase in international activity of the countries possessing fast reactor technology, as well as the involvement of countries interested in FR introduction into their future nuclear power.

The basis for the program mentioned realisation might be the scientific and engineering potential of France, Japan, Russia as well as United Kingdom, Germany and USA with participation of India, China, Korea and other countries.

It is believed that a gradual realisation of R and D studies in the framework of unified international program will allow to eliminate the major at present disadvantage of sodium cooled FR – higher cost as compared to thermal reactors. In doing so the scientific and engineering experience having been gained during 40 years in many countries will be used to the maximum extent which will noticeably decrease total expenditures for future NPP construction.

The FR prospects are as well rather promising if the expansion of their application is considered. Below the main possible FR applications are given.

1. Industry:

- production of high specific activity isotopes and unique isotopes obtained in fast neutron flux;

- fast reactor high-temperature heat utilisation for technological purposes (for example, production of motor fuels from coal and petroleum residue).

2. Energetics:

- fast reactor utilisation with extra long lifetime for autonomous and unattended small and medium power units.

3. Ecology:

- efficient plutonium utilisation (without breeding blankets, with MOX-fuel on the basis of increased enrichment (by plutonium) fuel or fuel without uranium-238 based on an inert matrix);
- efficient minor actinide burning (homogeneous and heterogeneous MA location in core and blankets when using different fuel types);
- use of FR for long-lived fission product burning.

4. R&D ACTIVITIES

In 1997 scientific research and design development works on many issues of the FR area were in progress. Below the brief description of some of these works is presented.

4.1. BN-600 reactor hybrid core design

Within the framework of utilisation of weapons plutonium stocks in Russia it is supposed to use the BN-600 reactor as plutonium burner.

For this purpose hybrid core design of the BN-600 reactor is under development.

This work was initiated in 1996 in the co-operation with the US Department of Energy and CEA of France.

The main directions of this work are as follows:

- development of the hybrid plutonium containing core design and comparative analysis of its characteristics with those of existing uranium fuel core;
- safety approval of the BN-600 reactor hybrid core, taking into consideration beyond design accidents.

The following requirements are to be observed in the BN-600 reactor hybrid core design:

- initial design of the core SA should be kept;
- zero value of the sodium void reactivity effect (SVRE) should be provided in accordance with the requirements of the Russian regulatory authorities.

The studies have shown that the latter requirement can only be met if some part of the uranium SAs are replaced by those containing MOX fuel with 21% of plutonium. High enrichment zone SAs are replaced in this case.

The studies were performed for two hybrid core designs, namely:

- core with fertile radial blanket containing depleted uranium dioxide, similarly to the existing design;
- core with non-fertile radial blanket made of steel and natural boron carbide.

Core design with the non-fertile blanket was chosen for the further development.

Both Russian and American computer codes are supposed to be used for the safety approval of the BN-600 reactor hybrid core. In this some American codes on fast reactor safety approval were handed over to the Russian specialists. Two training courses on SAS4A/SASSYS code for the Russian specialists were held at the ANL. By now the input data base on the BN-600 reactor hybrid core has been prepared for the SAS4A code calculation.

4.2. Training simulator centre for the BN-600 reactor

According to the TACIS program (1992 budget), works on the training simulator for the third power unit of Beloyarskaya NPP were initiated in September of 1996 by the Russian specialists in co-operation with the Siemens/Corys/Belgoatom consortium.

This simulator is intended for the modelling of all three circuits of the power unit, turbo-generator, electric systems and technological parameters automatic control system (TPACS). Neither auxiliary systems of the power unit, nor reactor refuelling system are assumed to be incorporated into the simulator. The simulator concept is based on the display interface supplemented with the mimic panel and the limited number of control keys.

By December 1997 the following works had been performed (and accepted by the NERSA supervising organisation):

1. Package of technological parameters on all simulated technological systems.
2. Models of all technological systems, namely:

- <u>reactor</u> - three-dimensional synthetical analysis of the core neutron profile, calculation of the core thermal characteristics using channels averaged over 9 radial flow profiling zones, 5 vertical zones and 3 angular zones;
- <u>primary circuit</u> - one-dimensional thermohydraulic calculation;
- <u>secondary circuit</u> - one-dimensional thermohydraulic calculation taking into account transfer of both dissolved and gaseous hydrogen, with partial modelling of he argon gas system and sodium tanks;

- steam generator - 4 sections of one half of one steam generator (SG) are simulated, while the second half of this SG as well as the halves of all other SGs are simulated as the single sections; modelling of all operating modes, from SG filling with water to the rated power operation, is provided in two-phase approach;
- third circuit - one-phase one-dimensional thermal hydraulics analysis of heat exchangers, turbo-generator and some auxiliary systems required for the start-up and shut-down operations;
- electrical systems - modelling of turbo-generator, auxiliary power supply systems and relaying devices connecting power unit with the grid;
- TPACS - modelling of the main system logic and main types of abnormal operation in all systems.

3. Approval procedures with the description of power unit operation during ~ 100 hours of the main transients.

The assembly of the simulator was planned to start on Siemens site in January 1998. However the work has been suspended because of the lack of the financing.

4.3. ULOF accident benchmark for the BN-800 type reactor

Comparative analysis of severe accidents in the BN-800 type reactor went on in 1997 using input data prepared at the IPPE.

Specialists from Germany, France, Italy, Japan, India and Russia are involved in this work, which was supported by the IAEA and CEC. Mr. G. Van Goethem and Mr. A. Rinejski have contributed much to the organisation and fulfilment of this work.

In the previous year reports were prepared on the steady state core pre-accident condition (FZK, Karlsruhe is responsible), input data on the core neutronics for the accident analysis (IPPE, Obninsk is responsible) and on the initial stage of the accident (IGCAR, Kalpakkam is responsible).

It should be noted that the work carried out in the framework of the group mentioned above, has made considerable influence on the software development for the fast reactor safety analysis in the Russian Federation. The experience gained by the West European, Japanese and Indian specialists, and their critical approach to the results obtained turned out to be useful and valuable for the Russian specialists.

It is very important to keep the fruitful co-operation of the large group of scientists formed during six years. The Russian party applies to the IAEA and CEC to continue support of this group.

4.4. Studies on the unauthorised withdrawal of absorber rod

In the framework of co-operation between Minatom (RF) and CEA (France) works are carried out on the improvement of the existing approach to the approval of safety in case of the absorber rod unauthorised withdrawal (ARUW) accident in the LMFR type reactor.

Modification of the approach to this design basis accident is made in two directions, namely:

- development of methods and related tools for identification of suspicious SAs and related values of maximum linear power as a function of time duration of the ARUW accident;
- analysis of conservatism extent in the assumptions used for the development of algorithm of calculation of the probability to avoid fuel melting in the end of the accident.

On the basis of RBR-3D three-dimensional code for the reactor analysis calculation tool has been developed at the IPPE in order to form the maps of potentially dangerous SAs in the ARUW accident, changes of maximum linear power in time being registered.

Evolution of the power profile during the reactor run is evaluated taking into account both movement of the absorber rod lattice and change of the fuel isotope composition. This approach allows decreasing uncertainty of the maximum linear power value.

In order to verify developed method of analysis of the accident under consideration, experiments on its modelling were carried out on the BFS facility.

The work on this method improvement is continued.

4.5. BFS critical facility studies

During the first half year of 1997 works went on at the BFS-2 critical facility on the preparation of experimental studies on the simplified model of the BN-800 reactor core with the increased plutonium content and non-fertile blankets.

In the second half-year critical assembly creation was started, its completion being planned for the first six months of 1998, when the experimental studies are to be initiated.

In the beginning of the last year, the standard part of the experimental program was completed on the BFS-71-2 critical assembly of the BFS-1 critical facility. The insert containing about 14% of neptunium dioxide replacing partially removed depleted uranium dioxide was places in the central part of the assembly. The results of measurements are being processed and analysed.

Plutonium content in the BFS-71-1 and BFS-71-2 critical assemblies MOX fuel is about 55%.

Starting from March studies were in progress on the BFS-1 critical facility using preliminary (conceptual) model of the KALIMER Korean fast reactor under the contract with the KAERI. Single-zone model of the reactor was assembled on the basis of 20% enrichment metal uranium fuel, and the following characteristics were measured:

- criticality;
- central spectral indices (using different methods and detectors);
- central reactivity coefficients of samples of different reactor materials;
- radial and axial distribution of fission numbers;
- Doppler effect on the samples; etc.

Results of measurements have been handed over to the customer.

Tests were completed on the FCA critical facility in Japan carried out within the framework of the international Program of studies on the efficient delay neutron fraction. After the results are summarised they will be presented at the special meeting.

In the framework of the program of control, accountancy and physical protection of nuclear fissile materials operation and modification of some elements of the system, created at the BFS facility continued.

4.6. Sodium fire studies

Studies of sprayed sodium fires were under way. These tests were carried out either in the closed chamber or in the ventilated room, their volumes being respectively 8 m^3 and 220 m^3. Ventilation ratio was about 5 hr^{-1}.

The sodium outflow parameters were as follows:

- sodium outflow velocity 2 m/s;
- sodium temperature 500°C;
- amount of outflowing sodium 10 kg;
- outflow hole diameter15 mm.

The results obtained are of preliminary nature. The experiments are in progress.

In addition, works continued on the development of analytical methods of the sodium fire studies, in particular, on modification of BOX and AERO computer codes.

4.7. Analytical and experimental studies of the coolant boiling

Several tests were carried out to study the boiling process of liquid metal coolant (sodium-potassium eutectic) and its stability using fast reactor SA model under natural flow conditions.

Experimental model is shown schematically on Fig.4.

The methodology of the experiments was based on the increase of power rating of the SA with stationary coolant which would result in appearance of the coolant natural flow within the closed circuit. As the SA power increased the coolant temperature also increased reaching saturation temperature in the heated area and coolant boiling started.

Tests were carried out for mass flow velocities varying within the range from 20 $kg/m^2 \cdot s$ to 700 $kg/m^2 \cdot s$, heat flux density reaching 260 kW/m^2 value.

It was determined that three boiling modes existed, namely bubble, slug and annular-dispersion modes. The latter is the ultimate boiling mode, providing SA cooling.

As it has been shown by the experiments, the slug boiling mode is characterised by the fluctuations of the coolant flow rate within the range limited by values typical for the bubble mode and the annular-dispersion mode. When changing from the bubble mode to the annular-dispersion mode about three time increase of the coolant flow rate occurred in the circuit. When the heat flux exceeded 250 kW/m^2 value, the coolant flow rate decrease was detected in the circuit accompanied by transition from the annular-dispersion boiling mode to the dispersion mode (heat transfer in the beyond-crisis area).

The analysis of the boiling stability of the coolant natural flow in the circuit has shown that there are two most typical kinds of dynamic thermohydraulic flow instability, namely:

- instability, observed in the region of low outlet vapour content values, when the variations of the gravitation related component of the pressure drop value are of great importance;
- instability, observed in cases of high and intermediate vapour content values, i.e. when the key component of the pressure drop is that caused by friction.

On the stage of tests preparation and planning and their methodology development computer code was worked out for the analysis of the liquid metal boiling dynamics. It was based on the one-dimensional mathematical model of two-phase flow in the closed circuit in one-dimension approach taking into account phase sliding.

Both qualitative and quantitative similarity of sodium-potassium eutectic and sodium thermophysical characteristics makes it possible to transfer experimental results, obtained for the eutectic, to the sodium.

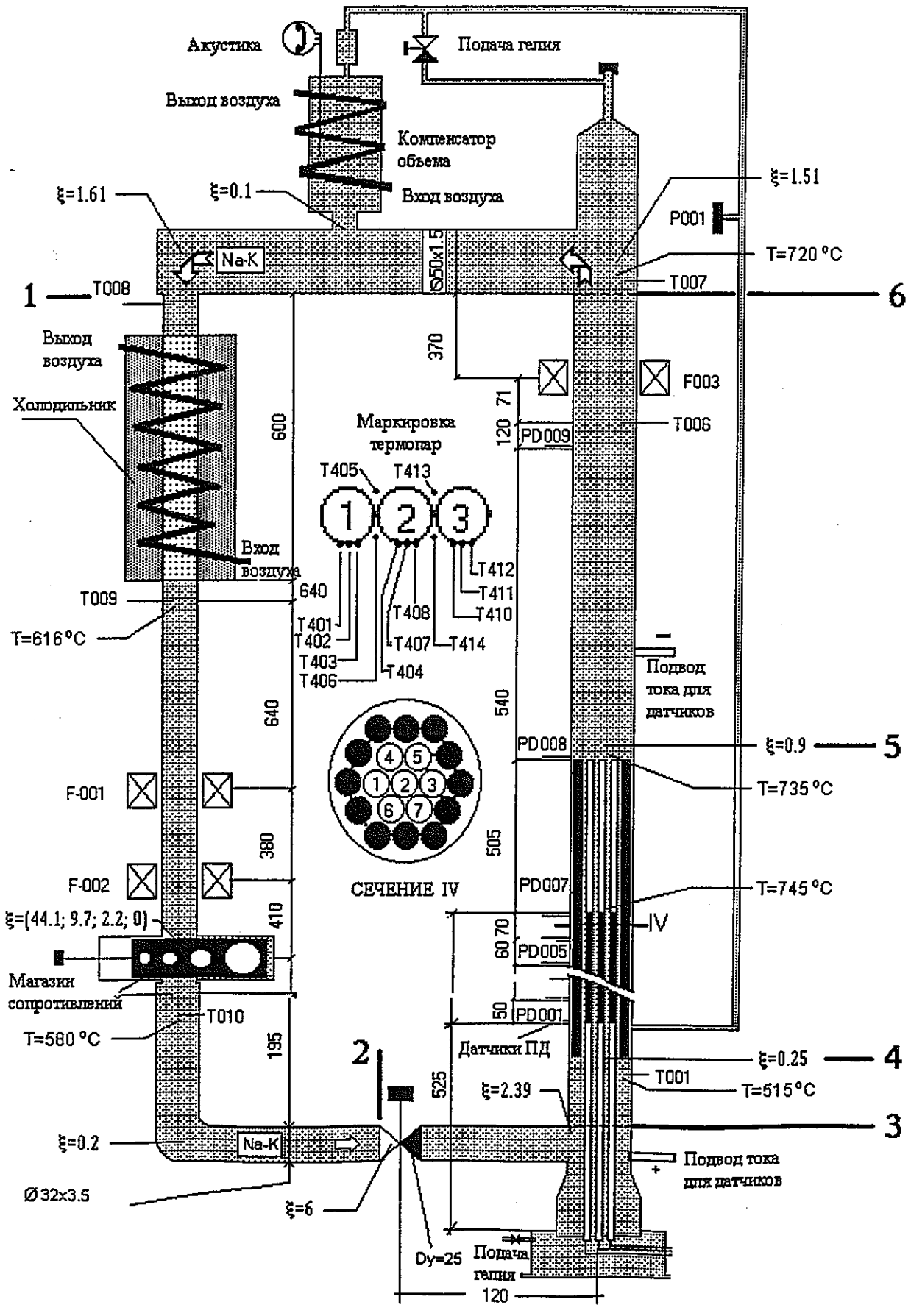


Fig. 4. Scheme of test facility for coolant boiling

Continuation of both experimental and analytical studies is planned in order to provide detailed and comprehensive approval of the possibility of stable heat removal from the fast reactor core under conditions of the natural convection flow of the boiling coolant that can take place in the reactor decay heat removal circuit in case of beyond design accident.

4.8. Studies on modelling of decay heat removal processes in fast reactor vessel

Studies are carried out at the IPPE on the development of theoretical foundations of using water for the modelling of the liquid metal coolant natural convection flow in the upper plenum of fast reactor in the decay heat removal mode. In particular, work is under way on the modification of the approximate modelling method proposed in [3], and development of the new approach to its basic principles interpretation.

In order to verify some principles of water models application for thermohydraulics studies of the sodium natural flow, rather simple experimental rig made of the acrylic plastic was constructed (Fig.5), on which the following studies are carried out:

- testing the possibility of method [3] application for studying transients in the stable natural convection flow;
- checking influence of the Reynolds number change (accompanied by the similar change of the Peclet number) on the modelling results and determining causes of the self-similarity appearance when certain values of these numbers are reached [4];
- determining relationship between the combined convection and the stable natural convection flows in various designs of immersed heat exchanger, etc.

Experimental studies carried out at the IPPE have confirmed the possibility of modelling process of transition to the developed natural convection for sufficiently high Reynolds number values. Analytical studies have shown the possibility of rather significant errors caused by the application of experimental data for the evaluation of the real reactor characteristics, if the hydraulic resistance coefficient is not taken into account as a function of Reynolds number according to the method [3].

It has been shown that the difference between resistance laws for the reactor and the model is caused by some modelling problems.

The works on this issue are in progress at the IPPE.

Water model tests are planned to study the reactor decay heat removal processes occurring in the reactor vessel as a whole, including those in the core, taking into account inter SA gap coolant flow, upper mixing plenum, intermediate heat exchangers and decay heat removal heat

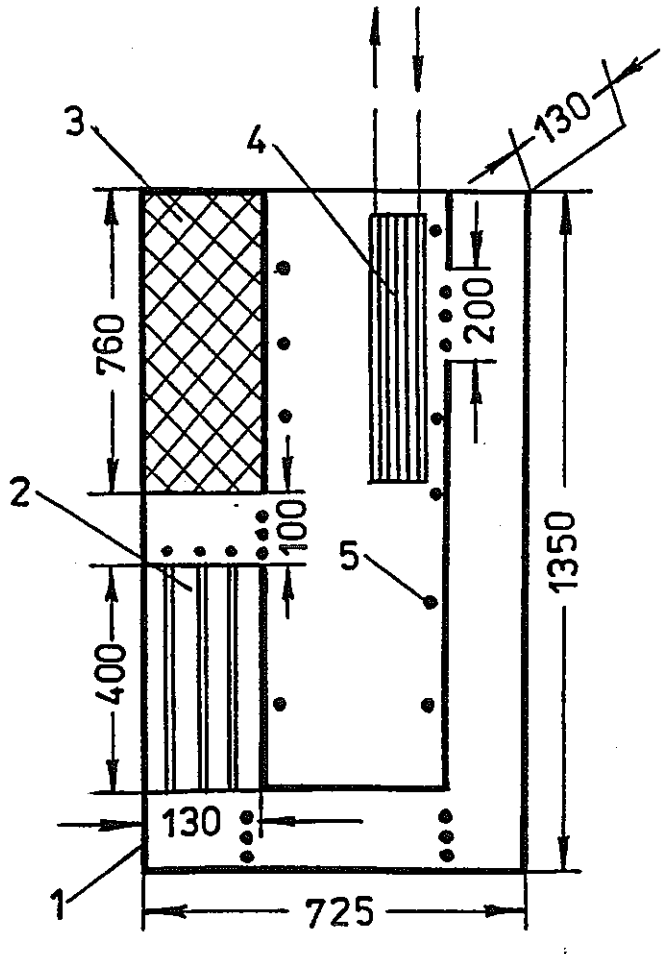


Fig. 5. Scheme of water experimental rig

1 - vessel, 2 - 16 heating rods, 3 - "displacers",

4 - cooler, 5 - thermocouples

exchangers, coolant path from the intermediate heat exchanger to the core diagrid, and the reactor vessel cooling circuit.

In addition to the criterion-based approach to the modelling of decay heat removal process in the reactor vessel, three-dimensional computer code is supposed to be used for the coolant temperature and flow patterns evaluation.

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A REVIEW OF THE UK FAST REACTOR PROGRAMME 1997/98

For Presentation to the 31st Annual Meeting of the IAEA IWGFR
Vienna; May 1998

1. The UK Nuclear Industry

In the UK, the position on nuclear generation remains very much the same as reported to the IWGFR last year. The Sizewell B PWR and the 20 Magnox units continue to perform well. The 14 AGRs continue to operate with greatly improved load factors and reliabilities. In 1996, the last year for which full statistics are available, the 35 nuclear power plants supplied 83 TW(e)h, ie. about 26% of the total electricity production.

The UK electricity market is now de-regulated and highly competitive. Generation by gas has risen to over 20% of the total (largely at the expense of coal) and seems likely to increase further.

Last year's report provided details of the restructuring of the UK Nuclear Industry, in particular, the formation and privatisation of the new utility company 'British Energy plc' and also 'AEA Technology plc'. The last phase of this restructuring has recently taken a significant step closer with the agreement (at Government level) of the financial package which will underpin the process by which Magnox Electric will become a wholly-owned subsidiary of BNFL.

The anticipation is that the combining of the financial and operational responsibility for the discharge of UK Magnox liabilities will provide clear incentives to maximise income from generation and to minimise liabilities costs.

Full integration is expected to be complete in about 12 months. Meanwhile BNFL are to apply to take over the responsibility for the Magnox site licences and discharge authorisations. This will necessitate their relicensing by the NII and the granting of new discharge authorisations by the Environmental Agency.

2. UK Activities on Fast Reactors 1997/98

The UK continues to support the international development of fast reactor technology, mainly through participation in the European Fast Reactor collaboration. Work has been carried out by the appropriate organisations: BNFL, AEA Technology and NNC, with funding provided by BNFL. As in previous years, the main activities have been centred on the continued development of the EFR design and on the CAPRA programme, led by the CEA, which are reported in detail by our French colleagues. The notes below summarise the main UK contributions.

EFR

BNFL as a member of EFRUG and NNC as a member of EFR Associates continue to participate in the EFR work programme. An important area of the EFR programme is the feedback into the EFR design of the operating experience from fast reactors in Europe and, where possible, around the world. NNC contributed to this task with feedback of information from the design and operation of PFR, with the assistance of AEA Technology. Reviews undertaken in the past year have included the main and auxiliary sodium circuits, the DHR sodium-air heat exchanger and the gas and water auxiliary systems.

The study of alternative concepts to EFR is another major area of the EFR work programme. NNC contributed to this work with an investigation of the potential of the gas cooled fast reactor based on existing UK AGR technology. Reviews of the work performed in the 1970s and 80s have revealed advantages in inspectability and maintainability, favourable breeding characteristics, in regard to present energy use scenarios, and potential for competitive economics. Studies performed over the last year have progressed the earlier design in terms of an updated core design, the specification for a suitable containment and the measures needed to make the boilers easier to replace, in order to improve the prospects of a 60 year plant design lifetime. The major remaining uncertainty, with respect to the viability of the gas cooled fast reactor as an alternative to sodium, is the behaviour of the reactor in case of severe accidents. Studies on this topic are planned for the coming year.

NNC also participate in the ongoing work of the Design and Construction Rules Committee for EFR, which is responsible for the development of design rules, and their continual improvement, aimed at eliminating penalising conservatism through the application of advanced methods of analysis. A related activity is the continuing international collaboration with Japan and Russia to compare fast reactor design codes.

NNC have also contributed to the analyses performed for the validation of the EFR rectangular containment building.

CAPRA

The UK continues to make a major contribution to the CEA-led CAPRA programme in the areas of advanced fuel, core physics and safety, and fuel cycle. BNFL participates fully in the CAPRA Project committee with technical contributions being provided by BNFL, AEA Technology and NNC.

A major topic of study in the current year has been the minimisation of the sodium void worth of the CAPRA core. A review of options to minimise the sodium void worth has been carried out, including various reduction measures, smaller cores and alternative coolants. The data and methods applicable to the assessment of the sodium void worth have been investigated by AEA Technology and detailed physics studies of specific

measures have been carried out. This has led to the recommendation of a reference core with a reduced sodium void coefficient, achieved by adopting B_4C axial shields and moderator (these measures are specifically applicable to burner cores).

A specific objective of the CAPRA programme is to investigate the burning of Minor Actinides (MA) in CAPRA cores. The UK has investigated a CAPRA core with MA target subassemblies around the periphery of the core and Np237 homogeneously mixed throughout the core. Fuel cycle scenario studies have investigated the contribution of MA burning to the fuel cycle.

The CAPRA programme presents particular challenges for fuel design and development as a consequence of the very high plutonium enrichments. AEA Technology has carried out extensive modelling studies of nitride fuel (PuN (Pn,Zr)N), both to assist development of the knowledge-base in this area and to optimise the characteristics of a nitride fuel pin design particularly for a Pu-without-U core.

IAEA IWGFR

The UK attaches great importance to the activities of the IWGFR and participates in meetings as appropriate. In 1997, this included attendance at the Annual IWGFR meeting in Beijing, and at a Consultant's meeting on LMFR Operating Experience in Vienna in December. It has also provided assistance with preparation of a Status Report on fast reactors and on the fast reactor database.

Through participation in other IAEA-led activities, the UK, and in particular BNFL, have contributed significantly to reviews, etc of other advanced concepts, including Accelerator-driven hybrid systems and Thorium-based fuel cycles.

3. Position on Dounreay Plants

The Stage 1 decommissioning programme has continued to make satisfactory progress. The major thrust is currently the disposal of the ~ 1500 Te of sodium/NaK that was used in the UK Fast Reactor development programme, of which 1150 Te is the PFR primary and secondary sodium coolant. The installation of the Liquid Metal Disposal Plant (LMDP), sited in the Turbine Hall, is well advanced and is on schedule to be completed by end of September 1998, when inactive commissioning is due to start.

A contract has been placed to supply a caesium removal plant through which the neutralised effluent from the LMDP will be passed before it is transferred to the Dounreay low level liquid discharge system.

Good progress has been made with the installation of the sodium transfer system to link the reactor primary circuit with the LMDP. Preliminary work for the installation of the pump system, which will be installed in the primary circuit over the next few months, is well advanced. Work to install back-up heating systems in the primary circuit is also

well advanced. There will be two independent systems, one using a NaK filled coil and the other using an electrical heater. Primary circuit heating is presently achieved by running a primary pump.

No transfers of fuel from PFR to the reprocessing plant at Dounreay have taken place since November 1996, because of the leak in the dissolver. However, the transfer of three sub-assembly remnants that contain a few stuck and/or broken fuel pins is now planned later in 1998, to allow recovery of these pins.

The route for the recovery of the reprocessing plant from the dissolver leak has not yet been decided. This decision is likely to take at least a further 2 years and it seems probable that it will be 3 to 4 years before irradiated fuel can again be taken from PFR. In the meantime, the Residues Recovery Plant is to be used to reprocess non-irradiated fuel from PFR, as well as residues, beginning later in 1998. Small amounts of residues have been recovered by this route before.

Consideration is now being given to the contractual arrangements that will be necessary for the continuation of operation and maintenance work on PFR in support of the decommissioning programme. The present contract for this work is due to expire on 31 March 1999. In line with Government policy, which is to allow open competition for as much as possible of the decommissioning work, it is likely that the maintenance work will be put to open tender.

The justification for reducing the category of the PFR facility from CAT 1 to CAT 2 is being progressed through the necessary safety committee stages. The CAT 2 argument is based on toxic hazard consideration and not on nuclear hazard.

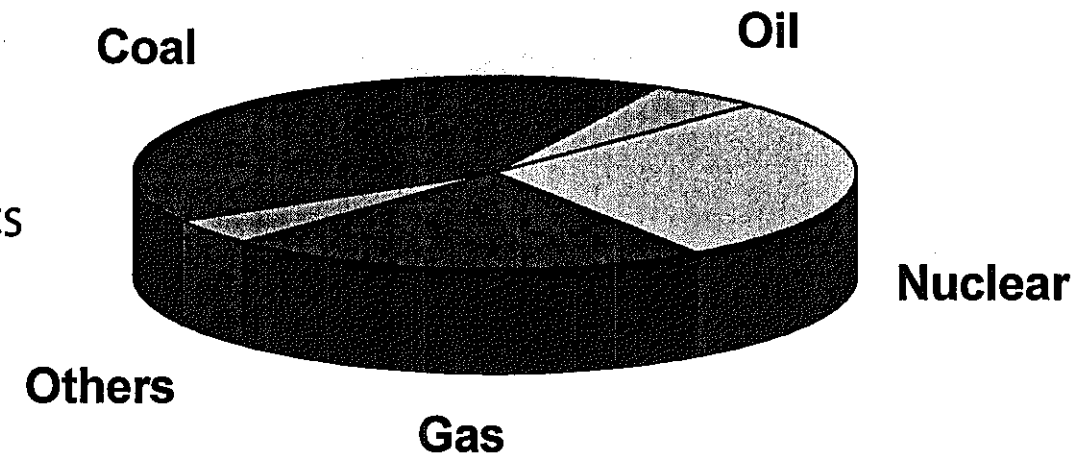
A REVIEW OF THE UK FAST REACTOR PROGRAMME 1997

Mr K F Ainsworth
British Nuclear Fuels plc

Dr C Picker
AEA Technology plc

Electricity Supplied 1996

Total NPP = 35
20 Magnox Units
14 AGR's
1 PWR



- AGR and PWR stations held by privatised British Energy
- Magnox stations held by public-owned Magnox Electric (ME) and by BNFL
- ME to become wholly-owned subsidiary of BNFL

UK Fast Reactor Activities 1997
(1 of 2)

Focus on sustaining the UK participation in the European Fast Reactor (EFR) collaboration and maintaining the many international links built-up over the previous decades

- Membership of the 'European Fast Reactor Utilities Group' (BNFL)
- Engineering Support to EFR Associates design studies including a UK secondee to Framatome/Novatome, Lyon (NNC)
- Technical support to CEA CAPRA Project and UK secondee to MASURCA, Cadarache (AEA-T, NNC, BNFL)

UK Fast Reactor Activities 1997
(2 of 2)

- Advanced Reactor System and Fuel Cycle Assessments (BNFL, NNC, AEA-T)
- Maintaining International links, IAEA meetings, CEC activities
- Prototype Fast Reactor (PFR), fuel reprocessing and reactor decommissioning (AEA-T, UKAEA funded by DTI)

UK Contribution to European Studies 1997
(1 of 4)

EFR

- Feedback of PFR design and operation information
  - main and auxiliary sodium circuits
  - DHR sodium-air heat exchanger
  - gas water auxiliary systems

- Study of Alternative Concepts
  - potential of gas cooled reactor

- Participation in Design and Construction Rules Committee

UK Contribution to European Studies 1997
(2 of 4)

CAPRA

- Modelling of nitride fuel performance
  - (Pu,U)N and (Pu,Zr)N
- Establishment of a calculation route to predict the performance of CAPRA cores
  - ERANOS code scheme
  - JEF 2.2 nuclear data
- Preliminary investigation of the use of the CAPRA core to burn Minor Actinides
  - Adaptation of the oxide-fuelled Pu-burning core appears to be feasible

UK Contribution to European Studies 1997
(3 of 4)

- Review of options to reduce Na void reactivity
  - Poisoned blanket
  - Moderated core
  - Reduced core height
  - Axially heterogeneous core

- Detailed analysis of a MA-burning core
  - Poisoned axial blanket and moderated core adopted

- Radical options to reduce Na void reactivity (NNC)
  Small core
  Metal fuel
  Gas coolant
  Sodium "plenum"

- Gas coolant selected for further study

UK Contribution to European Studies 1997
(4 of 4)

Support to IAEA Activities

- International Working Group Fast Reactor
  - Annual meeting of IWGFR, Beijing
  - Consultants Meeting on LMFR Operating experience
  - Assistance with preparation of Status Report and database for FR
- Thorium Fuel Cycle Advisory Committee

Support to CEC Activities

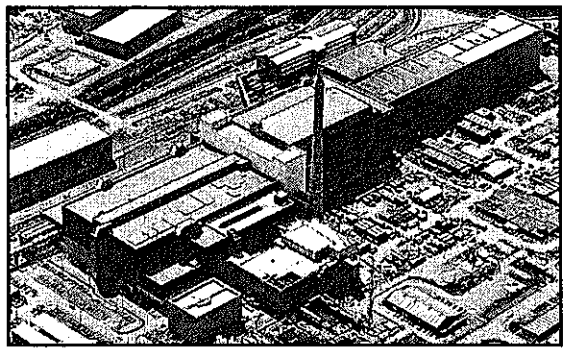
- STC review of Accelerator-driven hybrid system

Position on New UK Facilities

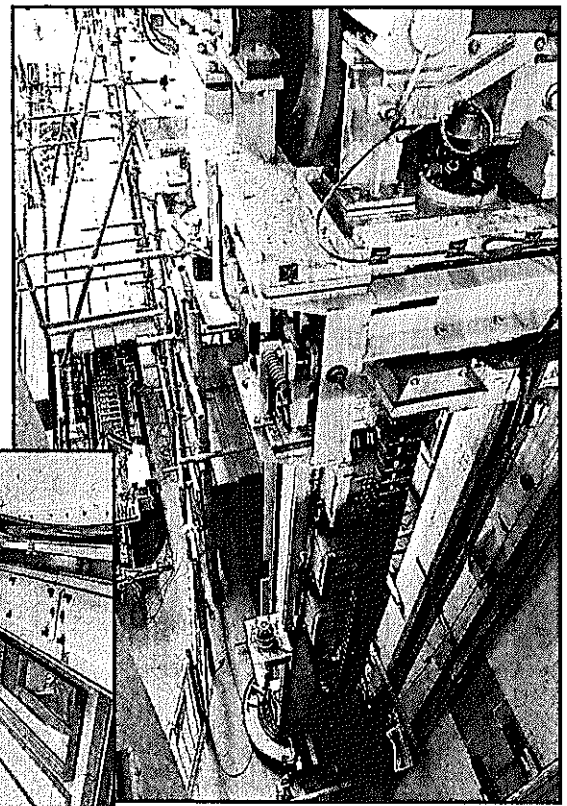
- BNFL Thermal Oxide Reprocessing Plant (THORP)
  - operating currently at 900 t/a
- BNFL Sellafield MOX Plant (120 t/a)
  - awaiting Environmental Agency approval to commence Uranium commissioning
- BNFL Technology Centre (advanced fuel cycle studies)
  - civil construction 60% complete
  - first occupancy anticipated late 1999

RU and Pu Route

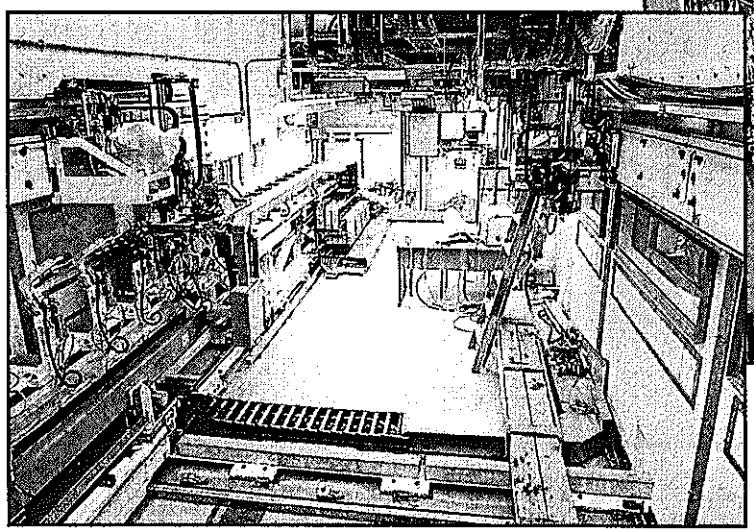
BNFL



Thorp & SMP



SMP:
PWR Build
Cavern Fuel
Handler



SMP: PWR Build Cavern

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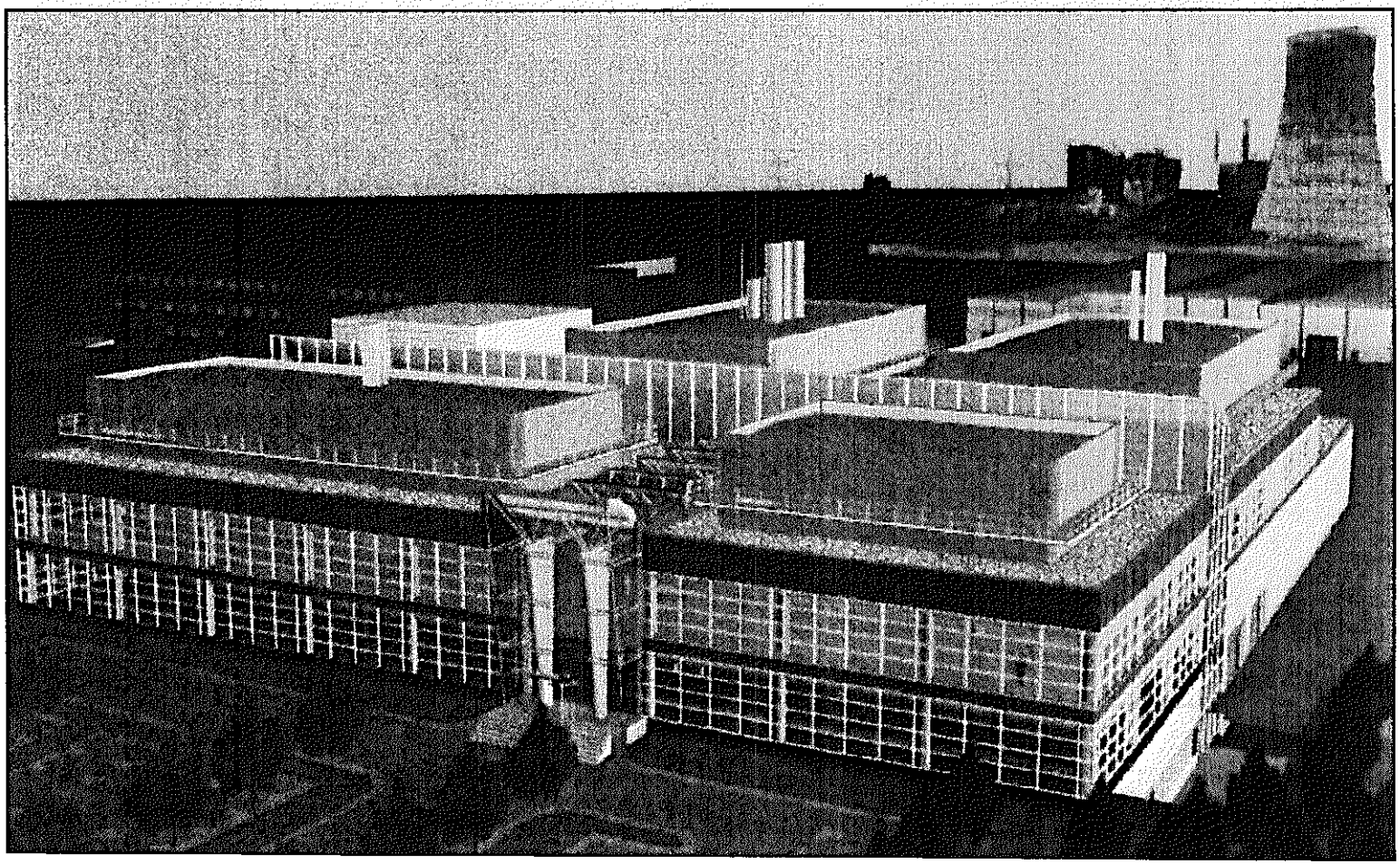
BNFL Technology Centre

BNFL



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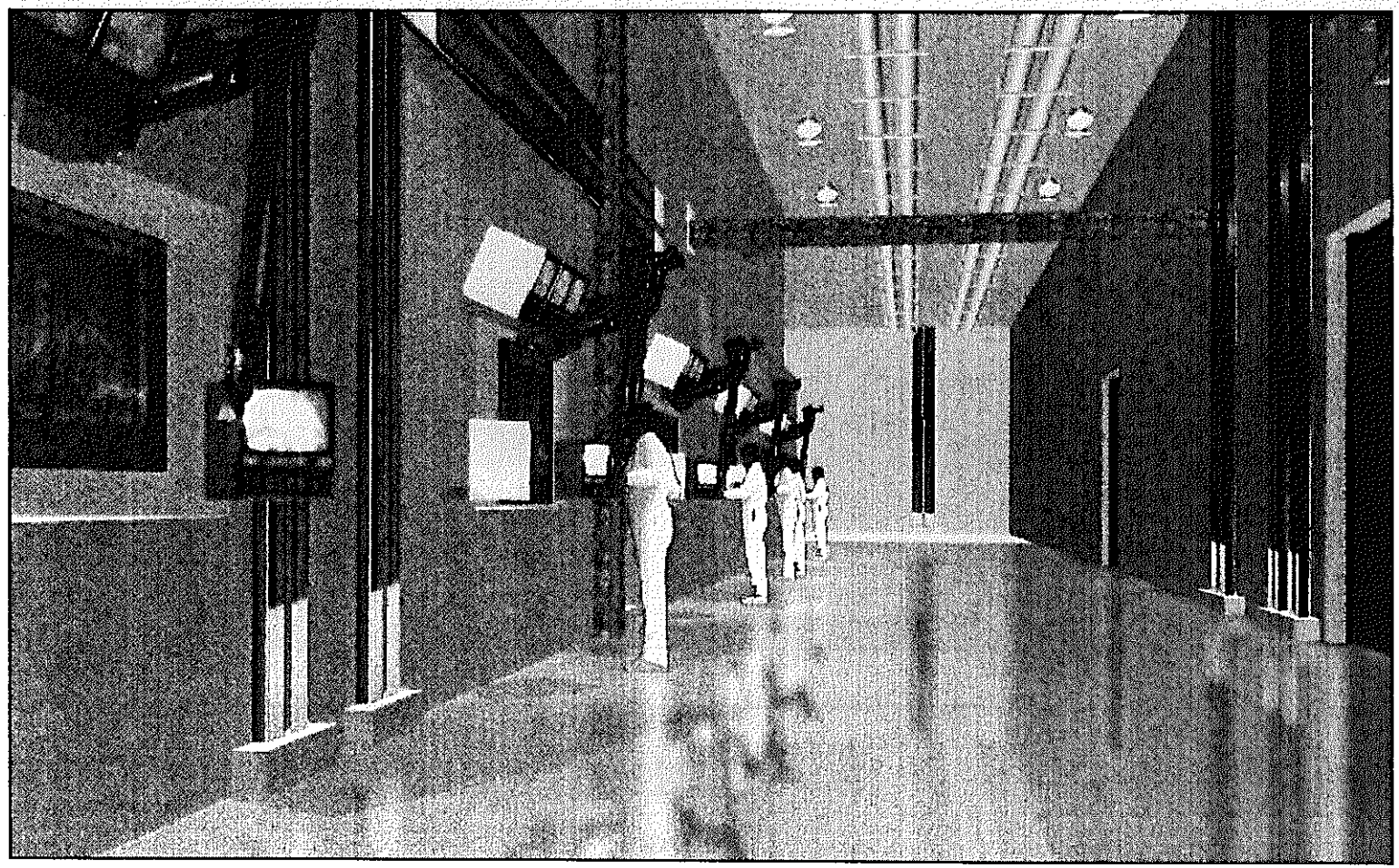

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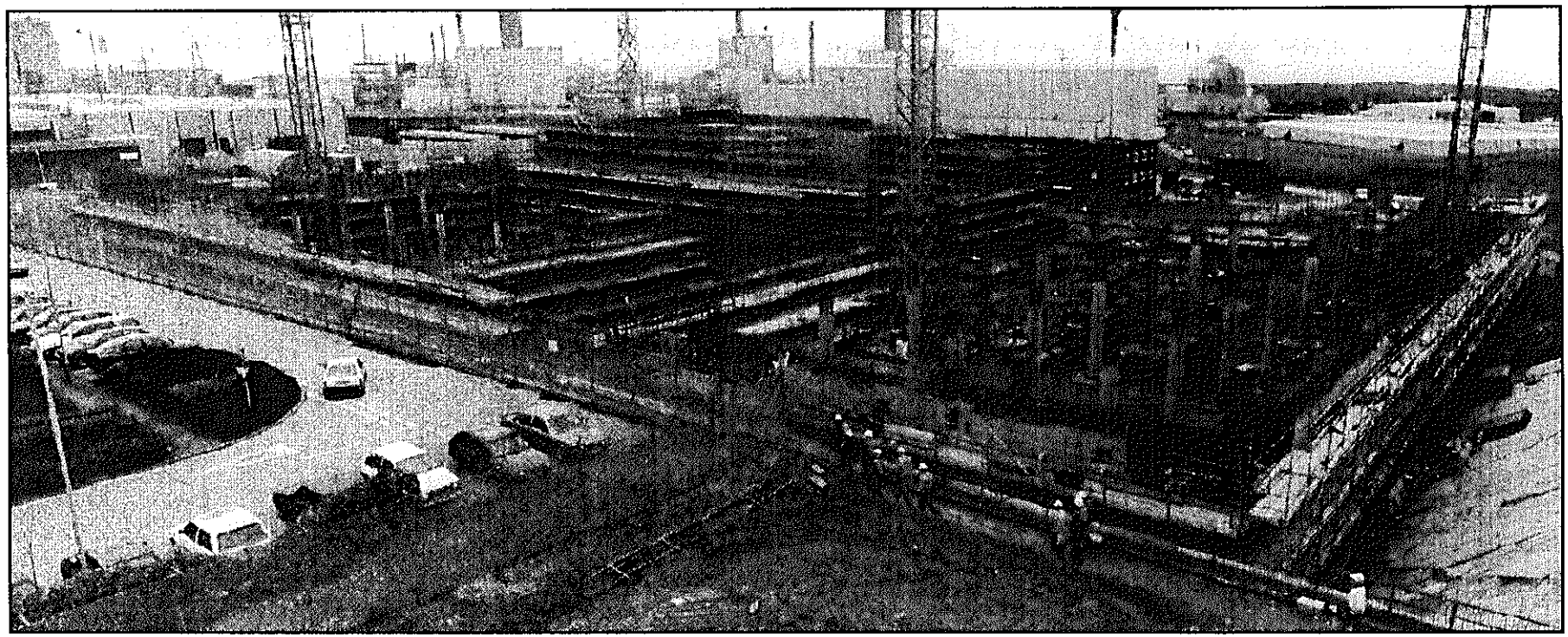

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PFR Decommissioning
(1 of 2)

- Satisfactory progress on Stage 1
  - installation of Liquid Metal Disposal Plant on schedule
  - contract placed to supply a cesium removal plant
  - sodium transfer system to link the reactor primary circuit with the LMDP progressing well
  - work to install back-up heating systems in primary circuit well advanced
  - consideration presently being given to contractual arrangements necessary for the continuation of operation and maintenance work

PFR Decommissioning
(2 of 2)

- Justification for reducing category of the PFR facility from CAT 1 to CAT 2 being progressed through the necessary safety committee stages

  - CAT 2 argument based on toxic hazard consideration, not on nuclear hazard

PFR Fuel Reprocessing

- No transfers of fuel from PFR since November 1996, because of dissolver leak
  - route for recovery of the reprocessing plant from the dissolver leak not yet decided
  - Residues Recovery Plant to be used to reprocess non-irradiated fuel from PFR, beginning late in 1998
    - small amounts of residues recovered by this route before
- Transfer of 3 sub-assembly remnants planned later in 1998 to allow recovery of some stuck or broken fuel pins