高速炉資料 43―0028

高速実験炉フランスチエック
才2回派遣団および駐在員出張報告書

昭和44年3月

動力炉・核燃料開発事業団

高速炉資料 43―0028　　FBR-MEMO-0028

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要　　旨

高速実験炉のオ2次概念設計に関し，その設計および安全性の検討評価をフランス原子力公社ＣＥＡに依頼した。オ1回派遣団出張報告書はすでに発表されている。この報告書は昭和43年10月から昭和44年1月下旬までサクレー研究所に長期駐在していた2名の報告と，昭和44年1月下旬にサクレー研究所で開かれたオ2回会議の綜合報告である。

ＣＥＡからのコメントおよび日本側から提出した質問に対するＣＥＡの回答を項目別に整理してまとめ，実験炉の設計に十分反映されるよう考慮した。

ＣＥＡの最終報告書は，この報告書より後に提出される予定である。

実験炉設計全般に対するＣＥＡの意見は下記の3点である。

Ⅰ　実験炉はできるだけ単純な設計にして建設すること。

Ⅱ　つぎの原型炉（あるいはさらに大型炉）に必要な技術を開発することを十分考慮すること。

Ⅲ　大型炉における燃料の高燃焼度達成に必要な照射施設として役立つこと。

オ2回派遣団団員は下記の通りである。

（団長）	高速増殖炉開発本部・（実験炉準備室長）	主任研究員	石川　寛
同	上	主任研究員	川口　修
同	上	副主任研究員	阿部康弘
同	上（日本原子力研究所・研究員）	嘱託	井上晃次*
同	上	研究員	尾尻洋介*

上記のうち＊印2名は，昭和43年10月から昭和44年1月下旬まで，サックレー研究所に長期駐在した。

高速増殖炉開発本部

目　　次

概　　要

フランス原子力公社ＣＥＡに高速実験炉第２次概念設計の設計および安全性検討評価を依頼した。このために原研の全面的協力を得て、英文の設計説明書および図面集を作製し、これをＣＥＡの高速炉専門家に説明し、また多くの技術的質門に対し回答および意見を得た。まだ中間報告の段階で、最終報告は昭和４４年３月末に動燃事業団へ送付されてくる予定である。

この検討評価を効率的に実施するため昭和４３年１０月７日から１０月１８日まで２週間にわたつて、ＣＥＡと日本側５名によつて第１回会議がサクレー研究所で開かれた。第１回会議後、２名は長期駐在者としてサクレー研究所に滞在しＣＥＡとの討議および動燃事業団との連絡に当つた。ＣＥＡの中間回答が大部分提出された段階で、第２回会議が同じくサクレー研究所で昭和４４年１月２０日から２７日まで行なわれた。この会議には長期駐在者２名と、日本からの出張者３名の合計５名が参加した。この会議終了後、長期駐在者もふくめて全員帰国した。３月末に最終報告書受領後、東京において第３回会議を開く予定で、ＣＥＡから専門家２名が来日し、約１０日間にわたつて討議を行なうことになつている。

この報告書は第２回会議終了の時点でまとめたものであるので、英文の部分は最終報告書でさらに整理、調整されたものが提出されるであろう。しかし日本文のＣＥＡ口頭説明の部分などはこの報告書が最も詳細であると考える。

本報告書は、関係者が利用しやすい形式とするために、設計の内容に主眼をおき、各部門別にＣＥＡから入手した情報を整理した。

まず、設計上指摘された主要な問題点のみを項目のみ列記して、重要な問題点が何であるかを明瞭にするよう努めた。

次に、第Ｉ部では、設計について得た情報を内容別に整理し、各部門における　1）一般的な comments、2）日本側より提出した質問およびＣＥＡ側からの回答、3）Rapsodie および Phenix などに関して入手できた情報などを別にまとめた。日本文で記述してある部分はＣＥＡ側の口頭説明によるものである。

第Ⅱ部では、ＣＥＡ関係の研究所、電力庁（ＥＤＦ）および民間会社を見学したときの印象記を訪門日別にまとめた。

第Ⅲ部および第Ⅳ部では、入手した資料の一覧表、派遣団および駐在員の活動した日程、折衝したＣＥＡ、ＥＤＦおよび民間会社の人名、所属などを列記した。

できるだけ早く関係者に利用して頂くために、時間の関係から充分な報告書内容の検討ができなかつたが、不明の点は直接派遣団員に問い合わされるように希望します。

設計上指適された主要点

本項は設計について得た数多くの情報のうち、特に重要と思われる主要事項のみを簡単にまとめたものである。

（設計の部）

1. 実験炉概説

1.1 インパイル・ループは、炉が複雑になるため、やめた方がよい。

1.2 FMFの Philosophy は、技術的に困難な、または材料物性上未知な問題点があり、考え方を変えた方がよい。

1.3 運転中の燃料出入れは、トラブルを起す原因となるので、やめた方がよい。

1.4 燃料照射を主目的と考えれば、最大中性子束はもつと高くできる。

1.5 機械設計および修理の点から、制御棒は dual purpose にして、本数を減らした方がよい。

1.6 原子炉容器下部はもつと Smooth で simple なものにした方がよい。

1.7 燃料取替機および出入機はもつと簡単な方式に設計し直すこと。

1.8 回転ラツクは、ナトリウム中で使用するベアリングに関して技術的に未知な問題を含んでいるので、やめた方がよい。

1.9 冷却系および Ar 系はもつと単純化した設計にやり直した方がよい。

1.10 計算機制御、自動制御の必要性は特にない。非常に安定であり制御しやすい。

1.11 原子炉容器内でカバーガスを通つて Na 中に挿入されている管類は、Ar 圧が異常に上昇した場合、Na が管内を通つて上昇する恐れがあるので Na レベルより少し高い位置に穴をあけておく必要がある。

2. 核 設 計

2.1 燃料照射を主目的とするならば、最大中性子束密度は炉心の体系を変えて現在の 4.3×10^{15} n／cm² sec から $8\sim9\times10^{15}$ n／cm² sec 迄上昇させることが可能である。

2.2 制御棒チヤンネルの吸収材の体積比が小さすぎるため B^{10} の濃縮度が大きくなつている。

2.3 negatiue な feed back（Doppler effect）がある限り、制御棒1本当り1ドル以下にすべきである。という基準の必要性はなく、dual purpose を採用すれば制御棒の本数を減少できる。

2.4 ABN set は Doppler 係数、Critical mass に対して error が大きい。

3. 炉心熱設計

3.1 燃料ペレツトと被覆管との間の最小ギヤツプは、unground pellet では 0.02㎜ では製作上困難な問題を生じるので、0.05㎜ とる必要があるだろう。

3.2　ギャップ熱伝達率0.85 W／㎠・℃は too optimistic である。0.6 W／㎠・℃を使うようおすすめする（ギャップの値は Rapsodie が基準）

3.3　hot spot factor の取扱いは cumulative と statistical とを組合わせた方がよい。

3.4　mixing factor は spacer wire を使用する燃料集合体の場合は無視出来ないので、将来実験で確かめること。

4.　炉心および炉体設計

4.1　被覆管強度に関する設計基準は $\sigma_t + \sigma_p < \sigma_y$ を使つた方がよい。plastic deformation が現われるのは long life の後にである。

4.2　Creep rate 1％は大きすぎる。

4.3　制御棒と駆動機構は相対的変位が可能なように rigid にしてはならない。

4.4　回転ラックをやめることにより、Vessel 直径は小さくできる。

4.5　Vessel からのノズルの数はできるだけ少なくした方がよい。

4.6　緊急系ノズルの位置を上げた方がよい。

4.7　球形プレナムの製作は困難であるのでやめた方がよい。

4.8　ブランケット冷却材流量制御は、これによる利益が小さく系統が複雑になり信頼性が下るので、やめた方がよい。

4.9　再臨界防止板は不要である。燃料溶融事故時はこの板ももたない。

4.10　Labyrinth structure は、製作が困難であり、キャビテーションやエロージョンを起す恐れがあるので、充分考慮せよ。

4.11　Core cover plate は固定した方がよい。可動の場合は trouble が多い。

4.12　回転プラグの冷却方式は温度分布を考慮して再検討する必要がある。parallel gas flow がよい。

5.　燃料交換系統

5.1　internal storage は回転方式をやめた方がよい。もし回転させるなら回転する部分および Na に浸つている部分は取替可能にしておくこと。

5.2　燃料取替機はもつと simple なものにできる。

1）多数の同軸 type 駆動は製作が困難である。

2）冷却系は不要であり、加熱は必要である。

3）hold down system の周囲6集合体を押し拡げる方式はやめた方がよい。

4）core element orientation はやめた方がよい。

5.3　燃料出入機はもつと simple なものにできる。

1）冷却系を空気冷却にできる。

2) Coffinは現在2つあるが、1つで十分である。

3) 運転中の燃料出入れはやめた方がよい。高温のためAr中でのNa depositの問題がある。

5.4 Transfer rotorはclad failureを考えて、Arパージを可能にしておくこと。また1次シールと2次シールの間にさらに高い圧力のArを送り込みactive Arのリークがないようにしておくこと。

6. 冷却系統設計

6.1 主冷却系管長はRapsodieに比較して長いので、チエックし直してみること。

6.2 2次系の全配管を2重管にする必要はない。

6.3 ポンプNaレベル調整用連結管はIHXに戻すよりも主冷却機ポンプ入口管に連結した方がよい。

6.4 中間熱交換器は全面的に変更しなおす必要がある。

1) バッフルをなくせば数分の1になる。

2) Naは水に比較して熱伝達がよいので並向流を採用してはどうか。

6.5 storage tankへの配管系の貫通部は熱膨脹を避けるように再考慮せよ。

6.6 純化系は単純化した方がよい。プラギング・インデイケーターは1次、2次系各1つづつで十分である。Cold trapのtopにAr管をつなぐ必要はない。

6.7 Ar管はNaによる洗浄が可能なようにしておくこと。Ar管に傾斜をつけておくとよい。

6.8 各機器へのAr管はタンクからとらずにpipesからとればもつと系統を単純化できる。

6.9 PumpsおよびControl rod mechanismへのAr供給量を少なくした方がよい。その解決策として、Control rodにベローを使いAr供給をやめると同時に、Pumpのmechanical sealにoil sealを使う。

6.10 Naポンプは保修を容易に行なえるようにseal部分を考えておくこと。

6.11 EMPは保温などの関係から予備をもつのはよくない。現在tankの上に置いてあるが、電源喪失時空になる恐れがある。

6.12 空気冷却器は5unitsで十分である。

6.13 逆止弁は修理が容易な設計にしておく必要がある。

7. 計測制御設計

7.1 Controlについてはautomatizeされすぎており、Operationに関してmanual法を採用した方がよい。

7.2 中性子安全系として次の2つを設置することが望ましい。

1) 炉心爆発時に通常位置のneutron detectorsが破損した場合、より離れた位置にあるCICによりneutron level測定が可能なようにしておくこと。

2）反応度の急速な変化を検出するための reactivity meter を設置しておくこと。

7.3　冷却系に Pressure measurement 用の計器をとりつける必要はない。Pump 特性がわかつている限り normal operation 中は必要ない。

7.4　熱電対の数が多すぎる。特に Vessel 周辺については多過ぎる。

7.5　50℃／h の運転基準は非常に小さすぎる。

7.6　安全系が alarm, set back ……など6種類のものを採用しているが多すぎる。

7.7　Computor function で処理すべき情報が多すぎる。

8. 遮蔽設計

8.1　2次系 Na が activity をもたないように design すること。

8.2　遮蔽設計全体が too pessimistic な design である。

9. 材料

9.1　spring あるいは grid への Inconel X の使用はよくない。

9.2　irradiated graphite と Na との compatibility はよくわかつていないので、フランスでは目下実験中である。

10. 燃料使用中検査施設

10.1　JEFRのFMFの計画では膨大な設備費と運転費を必要とする。ホットラボを2つに分けて、1つは非破壊試験用とし、他の1つは冶金学的試験用のラボとする方がそれぞれの機能が分離できてよりよいであろう。

10.2　irradiated fuel を取り出して steam cleaning を行なうと clad 材の mechanical property を悪化させ、炉心で再使用することに問題がある。

10.3　過期的に全ピンを検査しなくても、同一照射条件下の燃料集合体を sampling によつて check することで十分だと考えられる。

10.4　形式が異なるので比較しにくいが、非破壊試験用ホットセル内での作業能力が、10 subassemblies／week は大きすぎる。Rapsodie では2 subassemblies／week を目標としている。

11. 放射性廃棄物処理系および炉サービス系

（特になし）

12. 建屋施設

12.1　格納容器外コンクリートは要らないと思う。仮想事故時の民衆への許容放射能レベルを維持するためには、距離を十分にとることにより考慮されうる。

12.2　もし12.1が可能ならcontainerの冷却施設の必要性はない。

12.3　Container内のoperating floorは、Na ejection事器時にNaを回収出来る機構とすること。

（安全解析の部）

1. maximum hypothetical accidentの考え方は、explosionの過程、使用数値などでフランスと日本との間に相違がある。フランスの考え方はrealisticであると思われる。
2. Vessel上部機構の耐爆設計についてはscrew jacketでflangeをfixするとか、諸機構をボルト締めする前にbaionet systemでholdするなど・・・・・の対策が必要である。
3. Start up accidentの解析上の仮定はpessimisticすぎる。
4. Threshold scram 108MWとnominal flowの1%とを同時に事故解析で仮定するのは現実的ではない。
5. 集合体上部熱電対のTime ConstantはJEFRの解析で仮定しているものよりも大きい。
6. Channel blockageの場合、時間遅れが短いcaseだと熱電対では間に合わず、Reactivity meterによつてScram作動を起こさせること。
7. 緊急系の必要性は特にない。つけるとしても1系統で十分である。それよりも自然循環による除熱またはガス冷却・放射熱を利用しての除熱か炉内コイル方式などもつと信頼度の高い設計をすべきである。

第Ⅰ部　設計について得た情報

第 1 章　実験炉設計の部

1. 実験炉概説

In our opinion, three main objectives could be taken for JEFR.:

1) To design and build this facility as simple as possible.

2) To prepare following larger fast breeder reactors as far as technology (and especially sodium cooled reactors technology) is concerned.

3) To have a test bed for development of fuel elements capable of high burn up in large fast reactors.

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1) Simplicity is very important, because it generally means reliability. And it is obvious that the first goal to be achieved is to get the reactor as reliable as possible.

Nothing can be obtained from the reactor if the reactor itself cannot sustain an important load factor.

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2) Concerning sodium technology, the only question is to design the main components, keeping in mind what could be larger reactors, to get real experience easily useful for the future.

After construction, this experience would be a normal result of reactor operation, and does not generally need any special program.

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3) Very important is the use of such a reactor for development of fuel elements for future reactors.

On that field, very interesting results can be obtained with the own fuel elements of the reactor (driver fuel), especially if this fuel is sufficiently close of future reactors one.

But, in parallel, irradiations of fuel elements and cladding materials typical of larger reactors need special consideration. That would mean appropriate neutron flux, sufficient space for irradiation inside the core, and proper examination facilities.

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Along these main lines, the more important comments concerning the JEFR design, are the following:

(i) We are not in favour of an irradiation loop. The experiments intended with this loop can be performed easier and safer inside the reactor. Moreover, this loop would certainly complicate in a large extend the detail design of the reactor. As regards the operation, nothing good can be expected.

(ii) the philosophy about periodic watching of the fuel element is very different from ours. We think that scheduled destructive inspection of one or two subassemblies after say each 10,000 MWd/t gives a good understanding of the general behaviour of the fuel. It would be of course very interesting to examine one subassembly from time to time and put it again into the reactor, but we think that the foreseen procedure is too ambitious and anyway not necessary. Furthermore, that would mean only non destructive examination, and consequently only limited results.

Moreover, we never put fuel pins into the core after dismantling, cleaning, and washing. Such a operation could be possible, but would need special care and attention.

(iii) In our philosophy, we think that the necessary decay time of a fuel subassembly inside the reactor is not too harmful. Not neglecting the value of a prompt fuel examination, we realize that the necessary inoperation-unloading means a large number of complications and possible sources of trouble: difficult design and operation (rotating storage system), high level of decay heat during unloading, and consequently important argon cooling, etc...

(iv) The fuel monitoring facility (FMF) designed for the JEFR could usefully be divided into two different facilities: one in close connection with the reactor (one or two hot cells) for preliminary and non destructive examination, and the other quite independent of the reactor operation for deeper metallurgical study of irradiated fuel.

(v) As a fuel and cladding irradiation facility, the reactor would take advantage of sufficiently high neutron flux and needed space inside the core. The maximum flux of $4.2,10^{15}$ n/cm^2. sec. is pretty high. But, due to the size of the core and the power of the JEFR, higher flux could probably be obtained, in good comparison with larger future reactors (8 to 9.10^{15}).

(vi) Structure and cladding materials irradiation is an important point, and enough space and reactivity margin is to be provided for, taking into account that results can rapidly be obtained, only in high flux region.

(vii) Furthermore, the two values of fuel density (94% and 94% of theoretical density) are not very clear, as for the reasons of not retaining only one value.

(viii) Going further with the fuel, it appears that the design of the pins are perhaps not the best: the gap between fuel and cladding seems too small and the criteria for clad probably not very adequate. Gap conductance seems too optimistic.

(ix) Concerning the engineering design of JEFR, other main comments are:

- o the lower part of reactor vessel could be smoother and more simple (and so probably safer).

- if any valve could be avoided on main primary loops, it would be more reliable
- separate cooling pipes for blanket could be dropped
- emergency cooling system is perhaps not very safe in its proper design
- transfer machine (from core to internal storage) and loading-unloading machine could be made more rustic
- control and safety rods system could be reduced with dual purpose rods (control and safety)

(x) It is probably ambitious and not suited to such a reactor to envisage any automatic operation. Manual control is very easy and safer.

(xi) Some hypotheses of accidents considered for safety seem pessimistic. However, a better clamping of top shielding plugs could be envisaged and lead to softer criteria as far as the reactor building is concerned.

(xii) A great attention has to be paid to the subassemblies outlet thermocouples which can give an actually good feeling of the behaviour of the fuel elements.

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2. 核設計

2.1 General Comments

1) Comments

(i) We do not very well understand why two types of fuel are selected (94% TD and 84% TD). It would probably be of interest to turn to account the results already obtained in different countries in order to fix the characteristics of one type of fuel.

Concerning the neutron flux, it would be of interest to be able to raise up the value until 8 or 9.10^{15} n/cm^2.s, which is the flux available in large fast reactors. This increase could be obtained by the same type of modifications we have foreseen for Rapsodie (Project Fortissimo).

The specific power of 430 W/cm seems to be somewhat conservative and could be increased.

(ii) Control rods: we think that the rate of absorbing material in the rods is rather small. This material (B_4C) fills only about 19% of the available space, versus 60% in Rapsodie, and 40% in Phenix. For this reason, the enrichment is necessarily high (95%) which could be difficult to obtain by manufacturers.

- o control and safety rods system could be reduced with dual purpose rods (control and safety)

(iii) Heat production by gamma radiations has not been calculated in details for all media. The corresponding power is not negligible at all and should be carefully evaluated (In case of Phenix for instance, the gamma rays absorption in the blanket produces 20% of total power of blanket).

(iv) The ANL-Report 7320 gives the results of calculations made by many laboratories, on the assembly ZPR N^o 48. As you can see, the Russian set did not give a correct answer. At Cadarache, we use now a selfmade cross section set.

(v) We agree with you that for small fast reactors, heterogeneity effects are negligible. However, we think that diffusion theory over-estimates the critical mass up to 5 % as compared to transport theory.

(vi) Precise comments on the neutronic characteristics, as published in your report, cannot be done without Cadarache calculations on JFER. Suggestions for a precise calculation program were made to PNC representatives.

2) Questions and Answers

(i) Q. When the plutonium fuel is imported from abroad, how much is the distribution of the component of the plutonium fuel? What degrees are permissible for the design of a fast reactor?

A. We do not understand this question. In any case, fast reactors can accommodate any type of isotopic plutonium composition. We have already considered in our projects studies, plutonium compositions from 10 up to 25 % of Pu-240 content, and also, different types of plutonium compositions in the same core.

(ii) Q. Kindly show us your comments on the physical constants used in the nuclear design of JEFR except the group constants. Especially how much is the accuracy of the physical constants concerning the fuel?

A. Our physical constants are generally consistent with the data given in "the Handbook of Chemistry and Physics" (Chemicals Rubber Publishing Co.) Your physical constants agree well with ours; however, JFER B_4C density (72,8 w/o T.D.) seems to be rather low (see 94 % T.D. for Rapsodie). Accuracy on fuel physical constants must be asked to fuel managing people.

2.2 Cross section

1) Comment

No comment.

2) Questions and Answers

(i) Q. Nuclear design of our experimental reactor has been performed using Abagyan (ABN) set. If you have any experience on the reliability of ABN set, kindly show us the results (in comparison with experiments or detail calculations) on the following items,

① Doppler effect calculations
② sodium void reactivity effect
③ critical mass
④ danger coefficients
⑤ application to burn up calculations
⑥ fission products

A. On the reliability of ABN set, one can say that:

① Doppler effect calculations are not correctly calculated, especially for small fast reactors with hard neutron spectra (very poor accuracy on the self-shielding factors).

特に hard core に対しては more higher な Doppler effect の計算値となる。
Phenix 位の炉心の大きさについては ABN set でよい。

② Neutron spectrum calculations seem to agree not too badly with experiments. So, sodium void reactivity effects may be correctly calculated by ABN set (See report SETR 017 by J. MARTIN and P. CAUMETTE).

③ Critical mass calculated with ABN set may be too small by about 8 to 10 % (5 % in $\Delta k/k$ for Rapsodie).

④ Fission ratios calculated with ABN set are in good agreement for U 233/Pu 239 - U 235/Pu 239 - U 238/Pu 239 - Pu 240/Pu 239 but danger coefficients for diffusing materials disagree generally B-10 danger coefficient is quite well calculated, thanks to the good neutron spectrum calculation.

⑤ + ⑥ Burn-up calculation and fission products seemed to give correct answers for both the $\Delta k/k$ / day of reactor operation and the fission product effects.

(ii) Q. Do you improve partially the ready-made cross section set to the neutron cross section used in nuclear design, or do you make design data for each particular reactor type from a nuclear data library from the point of view of your unique phylosophy? If the former, is it considered to be actual and effective? If the latter, is it considered not to be sufficiently exact that the nuclear design is performed by the ready-made cross section set improved partially? Kindly show us your unique phylosophy for the cross sections used in nuclear design or for these developments, if any.

A. - for cross sections, our philosophy is the following:

- from the Windfrith nuclear data file, reviewed with our own evaluations, we prepare a 25 g set (infinite dilution), valid for a given range of neutron spectra.
- for each region of a given fast reactor and in order to take into account the actual volume fraction of each element, we prepare a "corrected" cross section set from the original "infinite dilution set", by using self-shielding factors and elastic corrections.

(iii) Q. In the data of fission products, how are they dealt with in the nuclear calculation?

A. The data of fission products seem to be in good agreement with the data used at Cadarache (see report by M. OHTA on long term behavior of Rapsodie - to be published -)

(iv) Q. In the evaluation of Doppler effect, kindly show us your opinions about how to deal with the unaccuracy of the nuclear data of ^{239}Pu.

A. Doppler effect is of prime interest in large fast reactors. In that case, due to the actual Pu/U-8 ratio, the rather large unaccuracy of the nuclear data of Pu-239 is not very important, since the major contribution in the Doppler effect is negative and du to U-238. Of course, this involves a very good mixture of PuO_2 and UO_2 powders (cf. safety). Recent German experiments were in good agreement with a Doppler calculation made with the following assumption: increase of 50 % of the α value in the range of 10 Kev. (See Schomberg measurements). In that case, the Pu-239 Doppler effect was negative, due to the higher value of α Pu-239.

3) Informations

① Cadarache set については１９６６年の ANL meeting (ANL 7390) でその基本的考え方について説明した。 現在 ^{239}Pu の α value, ^{238}U, ^{235}U などの cross section を改良して新しい set に開発中であり，後１ケ月位で整備される。 ^{239}Pu の new α value については Note CEA-N-989 (internal report) に従つて改良を行なう。新しく整備される cross section set (Cadarache new set) の σ infinite value および self-shielding value などはテープにて保存される。 この Cadarache set から，与えられた体系に対する群定数を作成する方法は，"BARRAKA" 法と "BARBAK" 法の２つがある。

② "BARBAK"法：この方法は 1966 年 Washington 会議に提出された Baker の方法 (ANL 7320) からヒントを得たもので，CEA において，ZPRIII，ZEBRA，VERA，... など世界の臨界実験装置から入手しうる全部のデータを集大成して，K_{eff} value を spherical，homogenious 体系にて単純に評価出来るようにしたもの (ANL-7320 を参照せよ)

③ "BARRAKA" 法： 世界の臨界実験データと Cadarache set とを調整出来るようにしたもの。 各実験の spectrum の影響は perturbation にて補正してある。

2.3 Temperature and power coefficients

1) Comment

No comment.

2) Questions and Answers

(i) Q. Did you calculate the power coefficient considering the special distribution of temperature in France? If it is so, kindly show us the outline of its calculation method.

A. We did calculate the power coefficient considering the exact temperature distribution in the core and the blankets.

Core and blankets may be divided into many regions (up to 200) for the danger coefficients calculation. Then, in each region, one computes the average Δ T for each material. For the fuel, two cases are considered:

a - transient calculations: the fuel pin is divided into two annular zones (calculation involving the whole reactor, core and sodium loops and steam generator)

b - power coefficient calculation (stability calculation) - exact distribution of the fuel temperature is calculated (only core calculations).

In any case, these schemes allow us to take a special care of the Doppler effect.

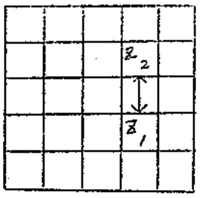
3) Informations

(i) RAPSODIE, Coefficients calculés de puissance et de température (*)

	coefficients de puissance pcm/MW		coefficient de temperature pcm/^{o}C	
	comb. libre de dilatation	comb. lié à la gaine	comb. libre de dilatation	comb. lié à la gain
Dilatation du sodium	-3.6	-3.6	-1.4	-1.4
Dilatation de l'acior	-2.0	-2.0	-0.7	-0.7
Dilatation du combustible	-11.6	-1.7	-0.2	-0.4
Effet des plaquettes	-4.7	-4.7	-1.1	-1.1
Effet du système de contrôle	-3.6	-1.8	-0.0	-0.0
Effet d'arcage	0.1	0.1	-	-
Effet du sommier	-	-	-0.4	-0.4
T o t a l	-25.4	-13.7	-3.8	-4.0

(*) Le calcul est fait en supposant le coeur compact.

(ii) CEA の計算方法

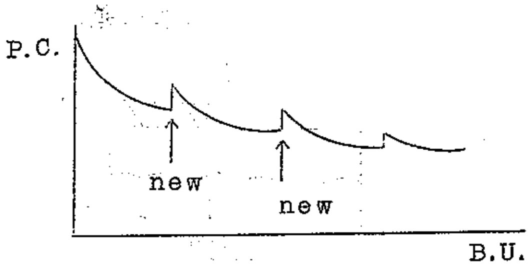


まず2次元計算にてdanger coefficientを求める。そして最大60領域，炉心約30領域をとり，各領域のNa平均温度$\Delta\overline{T}_{Na}$，構造材平均温度$\Delta\overline{T}_{ss}$，燃料平均温度$\Delta\overline{T}_{f}$，clad 以外の構造材，制御棒，vessel などの平均温度$\Delta\overline{T}$ をpower coefficient cal に考慮する。例えばある領域の$\Delta\overline{T}_{Na}$ は $\int_{Z_1}^{Z_2} T(Z)dZ/\Delta Z$ として求める。power coefficient Kpは$K_p = \Sigma K_T \times \Delta T$ となる。K_T は isothermal temperature coefficient。特にDoppler Coefficient のみは燃料pin中の積分値をとつて $\int_{Z_1}^{Z_2}\int_0^R f_D(T<r,z)dr,dz$ とする。f_D は Doppler effect である。

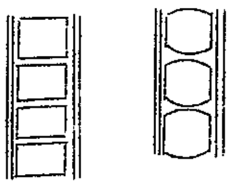
(特にcladのみ)

（しかしこの方法は余り複雑すぎるとのCEA内部の人の意見もある。）しかしこの方法を使つている人の話では，stability については余り問題がないので，単純に扱う方がよいだろうが，しかし温度係数の方は温度分布があるので出来るだけ詳細に扱う方がよい，とのこと。要するに細かく計算体系の温度分布を考慮しているということである。

EBR－IIではpower coeff が現在殆んど0で(+)の値が生ずることを心配しているとのこと。



Rapsodie の場合もnew fuel を中央領域に入れたときのみ多少power coeff が回復する。JFERでは潜在的にある約125 pcm の反応度（Doppler 効果）の上に （ΔT） があるのでRapsodie より多少楽であろうが，Bowing によるある power level での(+)の効果はpads の gap 間隔を変えるなどして(+)を生じないようにしておく方がよい。



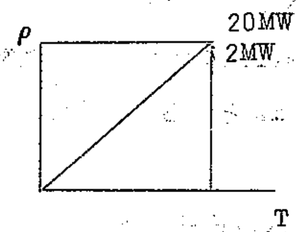
Rapsodie の温度係数； 0～20MWまでの間の温度係数の計算値は約350 pcm，実験値は600 pcm であつた。この差が生じた理由の第1は，右図の如くpellet の膨脹は中央領域で最大となるが，これを計算では考慮していなかつた。第2は，Pellet にクラックが入ると膨脹がよくなるらしい。

Burn up によるpower coeff の減少効果を評価するため次のような簡単な物理的推察が行なわれた。つまりpower coeff を次のように3つの効果に分離する。

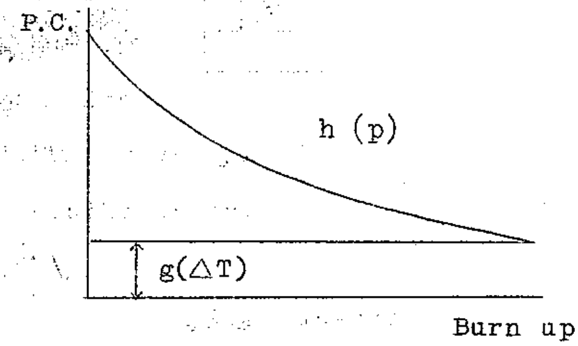
$$P = f(T_{inlet}) + g(\Delta T_{in \to out}) + h(P)$$

f は inlet temp. に依存する部分で最初はこれを無視していた。g は入口，出口温度差による power coeff，h は燃料温度の power level に係数。g は Na と SS の expansion に依存する。Rapsodie で実際に上の効果を separate して実験が行なわれた。same power で different flow を流して g と h を分離する。結果は次のような結論となつた。

1967年10月以来 h(P) = 0 である。
(右図)

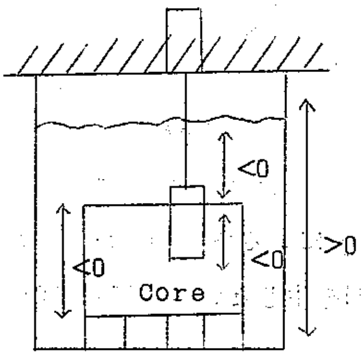


つまり左図の通り power level によつてPは変化せず。



h(P) = 0 以来，constant power coeff であつたが，その後 g(△T) が少しづつ減少してきた。これは pads の部分が前ほど膨張しなくなつたためなのか，SS の expansion coeff が変つてきたことによるのか，現在誰も説明できない状態である。

要するに Rapsodie の場合は計算値が −4 pcm/°C の isothermal temp. coeff で，実験値と10%の差のみであつたことはすばらしい。



Rapsodie で見落していたもので制御棒膨張と他の構造材との relative motion による温度係数がある。これは（左図参照）

i）制御棒膨張　㈠の効果
ii）Vessel 〃　㈩ 〃
iii）Core 〃　㈠ 〃（制御棒との相互作用として）

これらの効果の Time constant の大小関係は
　（iii）<（i）<（ii）となる。

2.4 Control Rod

1) Comment

No comment.

2) Questions and Answers

(i) Q. On the array of the control rods of JEFR, kindly point out the problems from your experience of the operation of Rapsodie, if any. Particularly on the position and the number of the control rods of JEFR.

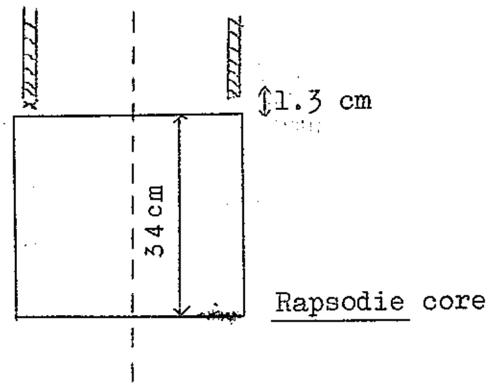
A. We have no special comment on the array of control rods. However, we can say that in Rapsodie

- fine control rods were replaced by coarse control rods
- coarse control rods (6) are also safety rods.

In normal operation, the 6 rod lower ends are approximately in the same horizontal plane in order to minimize the flux perturbation. One among them is used as the fine control rod, and every week, one adjusts the position of the six control rods.

(ii) Q. In our design of JEFR, the maximum position of shim rods and safety rods withdrawn is the upper surface of the core. Kindly show us the problems from your experience of the operation of Rapsodie. And kindly give us your comments whether the position should be upper than the present one designed.

A. In Rapsodie, the upper surface of the core is not taken into special consideration. For a core height of 34 cm, the total stroke of the rod is 44,5 cm and the control rod may be withdrawn 1,3 cm out of the core.



We think in fact, that it is more a question of operational philosophy, than a question of safety, within of course given limits.

(iii) Q. We are planning to change the present design as the following,

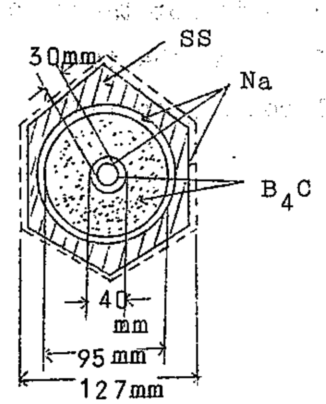
1) The B_4C rods (pins) of the shim rods and the regulating rods have no inner space.

2) The gas plenums are set up at the upper part of a rod.
Kindly show us the problems of this design change, if any.

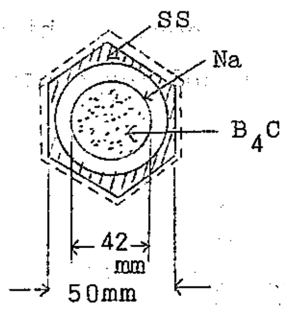
A. Questions 1) and 2) seem to be more on the technologic field than on the physics field. From Rapsodie experience, we are trying to develop vented control rods in order to avoid the problem of Helium release (quite important in Rapsodie). Price of B-10 in France is about 6 S/g (enrichment can go up to 90 % B-10).

Rapsodie では ^{10}B 90%濃縮のものを使い, B_4C の実効密度は2.34～2.364（94%TD）となつている。Phenix は natural B で1本当りの worth は1140 pcm, 全部で6本あり, 合計6840 pcm となつている。その内訳は下表の通りである。1$ = 304 pcm

Burn up 用	2000pcm（3ケ月相当）	regulating 用にはわずか約0.2% $\frac{\Delta K}{K}$ のみ。
(△T,△P)用	1500 〃	
Safety 用	3340 〃	



（Phenix 用制御棒）



（Rapsodie 用制御棒）

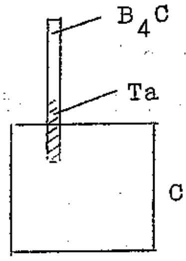
Rapsodie と Phenix の制御棒の構造を左図に示す。Phenix の場合 B_4C：SS：Na の比は39：18：43となる。
Rapsodie では B_4C：SS：Na の大体の比は60：16.5：20となる。
これら2つの B_4C 体積比に対して JFER の B_4C 体積比は16，19% など実に低い。

これが原因で ^{10}B の濃縮度が９５％迄と高くなつているのであろう。Phenix, Rapsodieの場合どちらもこれらの体積比で熱的な問題点を生じてはいないので, JEFR の場合も，もつと単純かつ ^{10}B の低濃縮使用の構造に変更されては如何ですか。

参考迄にフランスにおける B_4C の製造は９０％ ^{10}B のものをまず作り，後はもし低濃縮のＢを必要とする場合９０％ ^{10}B と natural B を混合する。９０％ ^{10}B の１ｇ当りの価額は３０Ｆ（≃約６＄）。濃縮度が多少変つても値段は余り変らない。

PHENIX の６本の制御棒構造は下記の通りとなつている。

Double purpose であり上部に B_4C，下部に Ta を使用している。下部は Dose rite が高くなるので Ta に L,He の発生を少なくしている。



(iv) Q. Kindly show us your opinions whether the maximum reactivity insertion ratio of each control rod and its maximum power increasing ratio are suitable for the reactor operation and safety.

A. We think that, as far as the regulating rod and the shim rod are concerned, the maximum rates of reactivity insertion are suitable for the reactor operation and safety. In the case of safety rod, the maximum speed might be considered to be a little high, but should be suitable if, by means of operational procedures, one is sure that the reactor is subcritical when regulating rods and shim rods are in the reactor.

(v) Q. On the production of the B_4C containing highly enriched ^{10}B,

1) How do you perform the regulation of the enrichment of ^{10}B in France?

2) What degrees of the pellet density are actual?

A. In France, enriched Boron is furnished on the basis of 90 % B-10 enrichment. This is the enrichment value needed for the Rapsodie control rods. Lower enrichment of Boron is obtained by mixing the 90 % B-10 boron with natural boron. The pellet density of Rapsodie B_4C is about 94 % of theoretical density.

(vi) Q. On the methods of the control rod calculations, what kind of approximations are used in France? Then, the outline of the calculation code? And, the calculation model? Besides, kindly show us the results compared with the experiments.

A. Control rods are calculated by the use of transport theory (However, diffusion theory is applicable, when using corrected diffusion coefficients in the control region, see paper by T. LACAPELLE, based on AMOUYAL-BENOIST-HOROWITZ Method). As a general rule, the control rod is calculated in one-dimension transport code, at the centre of the core. Then, the use of danger coefficients and diffusion theory enables us to give the control rod value at every position in the core.

Agreement between calculation (1.6 % $\Delta k/k \pm 0.1$ %) and experiment was very good for coarse control rods in Rapsodie (within a few percent, but with the Hanson-Roach cross section set). Of course, the relative error in the calculation of the initial fine control rod (natural boron) was worse (0.2 % $\Delta k/k$ $\pm$ 0.1 %) but again in good agreement with experiment. New calculations with our new cross section set have to be done.

See CEA Reports n° 3354 and 3416.

(vii) Q. What plenum volume/B_4C volume ratio did you adopt in the Rapsodie and the Phenix control rod design?

A. Plenum volume/B_4C volume in Rapsodie control rod

Where we designed Rapsodie control rod, we had only very poor informations about B_4C irradiation behaviour. Plenum volume was calculated for 10% the release, what was too optimistic. New design is being made for the next control rods, using the data we give you above.

(viii) Q. You are using 42mm O.D. solid pellets in the Rapsodie control rod and 95mm O.D. and 40mm I.D. hollow pellets in the Phenix.

Did you make these pellets by hot pressing or by press and sintering?

Did you polish these pellet?

In the case of JEFR, we are thinking of hot pressing 19.3mm x $30mm^h$ pellets at 2100°C for 1 hr. without polishing.

A. B_4C pellet fabrication

B_4C pellets are pressed and sintered by "Partiot" directly with good tolerances. The detailed process of barication must be asked to the manufacturer. Partiot made also fabrication for USA reactors.

(ix) Q. What amount of gaps are you considering necessary for B_4C pellet and the sheath?

A. Gaps between B_4C pellets and sheath

Clearance between B_4C pellets and sheath is pretty large (some 1/10 of mm) because pellets are not ground. This is permissible because heat flux is low at the pellet surface.

(x) Q. How did you estimate the release fraction of He gas and the irradiation swelling rate of B_4C in the control rod design, and how was those in the actual control rod of the Rapsodie?

A. (a) the release fraction

This question was already answered: For about one year of operation, 20 liters were calculated to be produced. About one half seems effectively released from B_4C.

(b) B_4C swelling rate

The undamaged pellets showed no swelling but underwent only a very; small burn-up. The most irradiated pellets were broken due to thermal gradients.

No swelling of the stainless steel can was measured.

(xi) Q. We expect that the mechanical disintegration of the B_4C pellet would not take place even after 10 $^a/_o$ ^{10}B burnup. Or, would the mechanical disintegration of the pellet effect the design of control rod?

Would you please give us your comment on this point?

A. Desintegration of the B_4C pellet above 10% burn-up

Our experience is limited to 1% burn-up. As we wrote above, at the burn-up pellets are broken into pieces and pieces number from a pellet is increasing with burn-up in a same control rod.

(xii) Q. We think the analysis of the thermo-mechanical shock and the countermeasure for the shock should never be excluded in designing the control rod.

Would you please give us some information on this point?

A. Analysis of the control rod thermo-mechanical shocks

We made no analysis. As we guessed during design and as we told you above, pellets are broken into pieces, if thermal gradients are important.

2.5 Burn up

1) Comment

No comment.

2) Questions and Answers

(i) Q. ① The method of the fuel exchange described in the JEFR design is one sample. Kindly show us your comments on this formula from your experience of fuel irradiation in Rapsodie.

② Kindly show us your criteria of the fuel exchange in Rapsodie and your formula of it.

A. Rapsodie is an experimental reactor; so at the beginning we had no special criteria on the fuel exchange scheme because of the leak of knowledge on the long-term behavior of the fuel, the type of irradiations to be made, etc...

If one wants to have the same flux value during all the irradiation, it is necessary to have a constant core volume in order to not increase the power. So in Rapsodie, for the special fuel irradiation (Phénix, for example) we try to get fuel test assemblies of about the same reactivity value as given by the nominal Rapsodie fuel subassemblies.

(ii) Q. The compensation of the reactivity decrease accompanied with the burn up of JEFR is performed by control rods until 1 % $\Delta k/k$. For the higher burn up than 1 % $\Delta k/k$, in addition to that, the radial blanket assemblies are exchanged by the core fuel assemblies so that a control rod may have the worth of 1 % $\Delta k/k$. Therefore, the mixed region composed of the blankets and the fuels may appear in the certain stage of burn up. Then, our r-z two-dimensional burn up calculation code can not give us the correct results of the respective burn up of the blankets and of the fuel assemblies. Therefore, it is necessary to consider the methods of three dimensional estimation from two dimensional calculations. Kindly show us your opinions about such methods.

A. We did not consider this kind of problems in our studies. In fact, if one tries to obtain a homogeneous loading and unloading at the core boundary, burn-up calculation, based on two dimensions (r, z) code, must give correct answers to this question. In any case, if one is able to define a "core and blanket" equilibrium configuration in which constant flux hypothesis is valid, it is therefore possible to calculate the special burn-up behavior of any kind of material, located at any place in the core and the blanket. We are planning to develop a special code for this purpose.

(iii) Q. The calculation of burn up for the reactor system which include several different nuclear elements of the fuel is made by using the values averaged over one region in our burn up calculation code. This method is considered to be sufficient to know the burn up reactivity, but it has no mean when reaction rates are calculated traversing specially. When the limited regions are 20 in a burn up code and it is intended for the axial distribution of reaction rates to be much more exact in addition to our fuel exchange method, what kind of method is considered to be the best? Is it necessary to calculate the cell burn up in fast reactors? Kindly show us your opinions on the problems above mentioned.

A. The special code, described in section (13) is of the cell-burnup type. But, in fact, in our code "REVE", the number of regions can go up to 50 (2 D - (r,z) diffusion theory burn-up code). Moreover, a new code (2 D, r, z diffusion theory burn-up code, based on a synthesis method) is being debugged and will be ready by spring 1969. This new code will alow up to 200 regions.

(iv) Q. In a burn up analysis, it is ideal to do this by combination with the control rods compensating the decrease of reactivity by burn up. But it was not performed in our design. In order to analysis a long burn up at a comparatively short time, what kind of methods in considered to be proper to deal with the control rods?

A. Our code REVE allows us to compensate the decrease of reactivity due to burn-up by moving the control rods. In fact, we did never use this possibility for easier computation. Indeed, the burn-up decrease (10 to 20 $10^{-5} \Delta k/k$ / day) is small enough to enable us to take this simplified method. See Report M. OHTA.

(v) Q. The reactivity decrease by the accumulation of fission products is so large as to cancel the positive values by various effects. Therefore, is it necessary to consider specially the creation and the transfer of fission products? Is such consideration paid in general for a burn up analysis?

A. We do not think that the reactivity decrease due to the accumulation of fission products is the largest negative effect. In fact, Plutonium burn-up in the fuel is the most important effect especially in small fast reactors where the internal breeding ratio is very low. So, we did not pay any special consideration on the transfer of fission products. Gaseous fission products in any case represent a small fraction of the total fission products.

(vi) Q. In the transient state of burn up, the heavy nucleus having a small decay constant brings often problems to us. But it is not dealt with in our design. If it should be dealt with, what kind of data can be used? How is that result? For example, ^{239}Np.

A. Our burn-up code allows us to deal with the Np 239, in the Pu-Uranium cycle. Quite often we take the simplified hypothesis to forget about this heavy nucleus. As a matter of fact, when the equilibrium state is reached (after a few periods) the Np-239 content gives a small constant decrease for the reactivity versus time, almost negligible for a small reactor as Rapsodie, and of the order of 0,15 $10^{-2} \Delta k/k$ for a typical 1000 MWe with large U-238 content inside the core.

N.B. - This problem would be very different in the case of Th-232-U-233 cycle, because of the quite different decay constant of Pa-233.

(vii) Q. It is necessary to spend a long time for calculation in general in the present burn up calculation code. This reason is considered to be owning to resolving the nuclear diffusion equation at each time step of burn up. If it is true that in future a burn up analysis will be developed toward an engineering side rather than a nuclear side, it is necessary to simplify the method of the nuclear analysis of burn up (for example, to use perturbation method or variational method). How is this problem considered in your country?

A. In the internal report by C. GIACOMETTI, we compared results obtained with direct burn-up calculations and results obtained by perturbation calculations. The agreement seems to be very good.

In fact, the code REVE we are using and the new synthesis code we are planning to use in the next future are rather little time-machine consuming, and with the new computers, being available soon, we think that this problem of long time machine consuming will disappear.

(viii) Q. In general, are you considering the effects of the control rods in a burn up calculation? And, in Rapsodie?

A. In general, we did not consider the effects of control rods in a burn-up calculation, although our code REVE allows it. For Rapsodie, we did not consider this effect (see report by M. OHTA).

(ix) Q. What is the principal factors to be considered on the scheme of the fuel exchange in the experimental reactor which aims to irradiate fuels?

A. We think that the main factors to be considered on the scheme of the fuel exchange in the experimental reactor which aims to irradiate fuels are the following:

- constant maximum flux value versus time (by keeping unchanged the core volume and of course the total power)
- unvariable power distribution in the core.

In order to meet these objectives as well as possible, we have to be very careful on reactivity problems and to favor an exchange scheme of the "homogeneous" type as described in the report SETR 023 on Phénix (core equilibrium and blanket equilibrium).

(x) Q. What stage of burn up should be selected to the reactor system which gives us the best nuclear information for the heat- or hydromechanics design? And what kind of factors should be considered to determine the reactor system? (especially, in the system of the experimental reactor which aims to irradiate fuels)

A. For Phénix, as we already have written, we define a "core and blanket equilibrium". All thermal and hydraulical calculations were made at this stage. In the case of Rapsodie, we just consider the fresh fueled core. We think, that it is valuable to consider a Phénix type scheme, even for an experimental reactor.

2.6 Code

1) Comment

No comment.

2) Questions and Answers

(i) Q. Kindly show us the outline of the nuclear calculation codes for fast reactors in France, and are these codes open or not?

A. The outline of the nuclear calculation code is the following:

- Cross section set evaluation
- Nuclear codes calculation
  . Diffusion theory 1-D and 2-D
  . Transport theory 1-D and 2-D
- Burn-up calculation
  . Diffusion theory 1-D and 2-D
  . Synthesis 2-D (being debugged)
- Perturbation
  . 1-D diffusion theory and transport
  - 2-D diffusion theory

- Gamma heating calculation (transport theory)
  (code being debugged)

- Special code for heterogeneity calculations
- Doppler coefficient calculations.

Our general philosophy is to prepare a set of nuclear codes used as subroutines based on about the same philosophy that the Argonne Laboratory has adopted (see ANL Report 7332).

On your special request we can ask our computation center, if the codes your physicist want to dispose are open or not.

口頭説明による核設計に関係した code list は下記の通りである。

① Cross section set に関係するもの

a) Cadarache set (reference)

b) Cadarache set (new one)

c) "BARRAKA" (前に説明した)

d) "BARBAK" (〃)

② "MUDE" 1次元拡散理論

③ "ALCi" 2次元拡散 〃

④ "DTF" 1次元輸送 〃

⑤ "DDT" 2次元輸送 〃

⑥ "REVE" 2次元燃焼 (R−Z) 現在は Test 中

最大 region 200, mesh point 2000 (但し rad. max. 80, axial max, group 12, isotop element 200, one region 中への element 数 15, control rod motion は考えない。

⑦ "new MUDE" 1次元燃焼

⑧ γ ray による heat の estimation code (γ source, neutron capture γ)

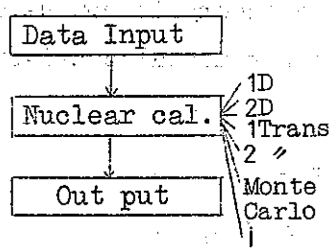
まず DTF_4 にて γ sources (fission, capture, inelastic によるもの)を算出し，次に σ_γ ……… γ spectrum を出して，DTF_4 にて τ_γ (gamma source) を求める。この code は heating および shielding の計算に可能。

⑨ heterogenity code

⑩ Doppler code（2つある）by Nevier and by Martin et. al.

⑪ "new MUDE" の問題は ANL 7332 と同である。

サブルーチンの問題で 1D., 2D, Sn, ... など計算手法が異なるごとに



Input data を用意していたのが現在迄の nuclear cal でとられてきた方法であるが，これらをサブルーチンでうまく結合して一完した方法をとろうとするもの。

⑫ hexagonal code （2次元拡散）

2.7 Comparisons between claculation and experiment

1) Comment

No comment.

2) Questions and Answers

(i) Q. From your experience in Rapsodie, kindly compare the data obtained in your nuclear design with the experimental values concerning the following items, and kindly show us your opinions about the problems of the calculation methods, if any.

① critical
② temperature coefficient
③ power coefficient
④ reactivity worth of a control rod
⑤ power regulation
⑥ spacial distribution of fission ratio
⑦ transfer function
⑧ neutron intensity at a sub-critical state
⑨ peaking factor
⑩ reactivity change by burn up

A. 1° critical -

3 to 4 subassemblies on the difference (^{57}calculated - ^{52}experimental) = 5 were explained by various errors in the previous calculations. (We had a critical experiment in ZPR III). Diffusion theory is now used and the results adjusted on the first experimental configuration of Rapsodie.

2° temperature coefficient -

good results within about 10 % of error.

3° - Power coefficient -

For the fresh core : $\frac{\text{experim.}}{\text{calculated}} = 1{,}5$

For the used core : $\frac{\text{experim.}}{\text{calculated}} = 0{,}6$

We notice a very progressive effect due to bowing; we think that, in the initial stage, the assumption of an average fuel temperature for the axial fuel expansion was wrong, and we have to use the central fuel temperature for the axial fuel expansion. The interpretation of the behavior of the power coefficient values are still under study.

4° - Reactivity worth of a control rod -

very good agreement, but with Hansen & Roach constant Calculations will be done again with our new set of cross sections.

5° - Power regulation - what do you mean by this question?

6° - Spatial distribution of fission ratio -

Very good agreement between neutronic measurements and neutronic calculations. However, we notice a small disagreement on the thermal point of view; this point is under study (higher radial peaking factor?)

7° - transfer function - measurements were not accurate enough, so we still encountered some difficulties for intercomparison.

8° - Neutron density at a subcritical state - The neutron source value effect was quite well estimated; no problem for the neutron intensity at a subcritical state.

9° - peaking factor - good agreement from the neutronic point of view; however, you have to check up more carefully from the thermal point of view, see point 6°.

10° - Reactivity change by burn-up - very good agreement - See report on Rapsodie by M. OHTA, already mentioned.

口頭説明による内容は下記の通りである。

① Critical mass

予測値(計算)57本，実験値52本であつた。その後Cylindrization, Snとの比較，BARBAK法との対比(Baker法)などで次のような評価を行なつた。

Hausen Roach set (HR set)使用によりHexagonal Code でRapsodieのCylindrization について評価を行なつている。

$$52本 \longrightarrow K_{exp}^{non\ cyln} = 1 \begin{cases} K_{exp}^{cyln} > 1 \text{と予測} \\ \text{従つて} K_{exp}^{cyln} = 1 \text{のときの} V_e < 52 \end{cases}$$

control rod up のときのnon cyl. とcyl. との差は800 pcm と計算されている。従つて $52 - \frac{0.8}{0.4} \simeq 50$ (0.4は周辺worth)と予測され $V_c < 52$ の傾向と一致。これより以後の解析はSETR(cadarache)set により行なわれた。52本体系で2D.diff.，と2D.Transport(S_4)の計算，56本の体系でreaction ratio を同じく2D.diff.，と2D.Trans. で計算した。

experimental critical mass は heterogeneous, non cly. の条件で51.8であり，non cyl., homogeneousの条件で52.16である。一方52本に対する計算値はdiff = 997, trans.(S_4) = 1.040±0.001である。従つて

diff. → S_4 …… 4.3% △k ← 2D.Caderache set
diff. → S_4 …… 4.14% 〃 ← HR, 1D
diff. → S_4 …… 3.81% 〃 ← Cadarache, 1D
diff. → S_∞ …… 3.58% 〃 ← Baker の 5/6 法による推定

の関係をうる。結論は以下の通り。52.16 ass. に対する計算値は S_4 で 1.0435，従つて S_∞ ≒ 1.0364， X ass. の cyl. された実際の実験値は < 1.0364である。つまり cyl. 効果を例に 0.006 とすると（現在計算中），cyl. された実際の実験値は 1.03 となる。別の観点から Cadarache setを応用した Baker 法（BARBAK法）によると $K_c - 1 = 0.0309$ となり，非常によく一致する。

② temp. coeff. SETR set(cadarache set のこと）で danger coeff. を求めて今，評価中

③ power coeff. （前にすでに説明した。）

④ fission ratio, core center fission ratio は下記の通り

	HR set cal.	Cadarache set cal.	Rapsodie exp.	ZPRⅢ44 Rapsodie mock up exp.
σ_f^8/σ_f^5	0.084	0.0765	0.0789	0.806 and 0.09
σ_f^9/σ_f^5	1.210	1.191	1.284	1.179
σ_c^8/σ_f^5	0.112	0.114		
σ_c^8/σ_f^8			1.44	

238U capture の exp. はまわりの材料，control rod の影響（Spectrun）など気を付ける必要がある。

2.8 Check calculation by CEA

Nuclear design に関して CEA 側から Comments を与えるためには check calculation が必要である，との提案があり，下記の通り実施することゝなつた。

JEFR 核設計の check 計算に必要な時間は下記の通り

①	ー 核断面積の準備	Band SETR"Barraca 法" （これはCadarache set の１つの手法で詳しくは後で説明）	5分
②	ー Code"MUDE"（拡散１次元コード）２５群 DTF$_4$（S$_4$）２５群 （球形計算） （拡散理論と輸送理論との比較を行なう）		10分 15分
③	ー Code ５２４５（別名"REVE"）１２群 （２次元拡散コード）	SETR"Barraca"の準備 必要な核特性計算 （Kept，臨時量，breeding rates，など Burn up 特性も一緒に出てくる）	5分 40分
④	ー 2DTF$_4$ １２群 （２次元輸送理論）	safety rod の計算 （もしsafety rod 以外の shim rod 又はregulating rod を計算に加えると）	30分 （＋15分）

（註）i）以上の計算はsafety margin を含んでいない。　（合計）1時間４５分

ii）Rapsodie の実験結果を考慮に入れて実験結果よりKeff をスペクトルを考慮した上で"Barraca 法"によつて外挿する。　（制御棒2 case を行なえば）→（2時間）

但し，計算開始までに以下３点についてＰＮＣにて検討することとした。

(i) B$_4$C の体積比（構造材，冷却材に対する）がRapsodie，Phenix と比較すると余りにも小さすぎる。

(ii) 動く制御棒が１＄以下であるという基準は必要ない。（Storrer）

(iii) 制御棒の本数が多すぎる。（ＣＥＡの意見では６本でよい）

2.9 Physical experiments

1) Comments

(i) 必要最小限度の試験項目は

① core outlet temp test, ② rod drive mechanism test, ③ reaction ratio (power calibration at low power condition), ④ Control rod calibration, ⑤ behavior of reactor incidents operation: 例えば, primary pump stop など, ⑥ Oscilation test (Transfer function & Stability の解析のため) ... 解析が非常に難しい, ⑦ Rod Drop Test, ⑧ Temp. Coeff. measurement, ⑨ Power Coeff. measurement ⑩ pressure による reactivity change, ⑪ pomp flow 変化による reactivity change at zero power, ⑫ thermal output power calibration, (thermal leakage は Temp. measurement により推定, Double wall vessel を流れる N_2 gas 温度と flow より推定する。これは rough な estimation である。

以上, いろいろな Test のうち, 興味のある Test も多々あるが, 全部が必ずしも必要だとは考えていない。

2) Questions and answers

(i) Q. Power calibration method adopted in Rapsodie operation.

1-1 Do you make power calibration each time after refueling?
1-2 If not, when and how often do you make calibration?
1-3 The method and the procedure of power calibration.

(ii) Q. We are facing to the problem that there is the flux level difference of about one (1) decade when the spent fuels are loaded in the spent fuel rack.

We would like to know the necessary precaution when you determine the neutron detector position in order to avoid the effect of spent fuel.

For example, do you not load the spent fuel between the core and neutron detector?

3) Informations

(i) Rapsodie の臨界実験，制御棒校正，反応度係数についてはReport CEA −R3354（報告第6号付12）を，Fission Ratio の測定についてはReport CEA−R−3416（報告第6号付20）を参照せよ。又CEA−R3406 にも必要なTest，Test を通じて起きた問題点などがまとめてある。

(ii) Rapsodie の燃料交換手順につき現在迄交換した燃料の位置，月日，MWD／T は以下の通り

年 月 日	MWD／T	位 置
67. 4. 20	150	01−01
67. 7. 20	2,300	01−06
68. 3. 16	12,500	04−19
68. 3. 16	20,000	01−04
68. 6. 19	30,000	01−01
68. 8. 26	35,000	02−01
68. 10. 31	40,000	00−00

この位置はCEA−R−3416 Report P.14 を参照せよ。
1968年12月4日現在Rapsodie は24MWe運転を継続中，42,000MWD/T に達している。非常に順調である。
1969年1月下旬には50,000MWD/T に達する。

3. 炉心熱設計

3.1 General comments

1) Over power conditions (110%)

We do not use any "over power condition" in our calculations. We just compute temperature for nominal conditions at 100% power. We call them "nominal temperatures".

Every excess of temperature coming from general or local over power is included into hot spot factors.

2) Mixing

(i) We are running an extensive program of sodium mixing tests right now we have only preliminary results which show us that some mixing results from helical spacer wires.

In the case of JEFR.

a) If helical spacer wires are not used, mixing is only due to diffusivity and heat conduction. The real mixing is narrowly dependent from subassembly internal geometry and especially from grids form and distance. We do not think that mixing can be calculated but it is certainly not very large.

b) If helical spacer wire is used with a pitch not far from Rapsodie one, we guess that it can be estimated that 50% of the flow-rate influenced by helical wire left one channel every pitch and is replaced by sodium coming from other channels, but this figure has to be prudently used.

(ii) SKOK 氏からの私信

a) Rapsodie のMixing効果は最初は0としていたが，その後10/1のプラスティックス製燃料集合体模型によりTestを行なつて，妥当な値を得た。

b) 新しく入れかわる炉心に対してはこのMixing効果を考慮し，被覆管最高温度を限界値(650℃)以下にあわすように，各集合体のオリフィスを調節であろう。

c) JEFR のdesignでMixing 効果0とするのはおかしい。しかし，もともと計算値は現状では非常に信頼しにくいので実験によつて確かめることをおすすめする。

d) フランスではCEA−EDFの共同で今日迄5ヶ年間Mixing に関する実験を水中およびNa中の雰囲気で行なつてきた。近いうちにReportにまとめることが出来ると思うが，計算の方は誠に複雑で評価を正しく出来なくて困つている。

3) Rapsodieの power up について
(SKOK 氏からの私信)

(i) Rapsodieの power up に対して熱設計基準の何を変えたか

a) 流量を増加させたのみで，原則的には何も変えていない。

b) 設計基準として被覆管のホット・スポット，温度650℃は同じだし，ホット・スポット係数の処理のしかたも「累積的処理」と4γ法による「系統的処理」との両法を使つている。

c) 燃料中心温度に対してover power の基準はとり入れていない。理由はRapsodie の場合・燃料中心温度が融点よりはるかに低いからだ。

d) ホット・スポット・ファクターそのものの数字は実際に物を製作し，運転した後に縮少しうる。例えば臨界質量，tolerance，流量分布などは実際に運転経験を重ねることにより，factor をさげうる。Rapsodie の最初の炉心に適用したfactor は非常にconservative であつた。

e) Phenix design の場合は燃料中心温度はRapsodie の場合よりも大きく，考え方を変えるであろう。(詳細について説明はなかつた)

3.2 Temperature limit of the fuel

1) Comment

No comment.

2) Questions and answers

(i) Q. The critical limitation on the JEFR design is maximum temperature of the fuel center or of the clad. What criteria do you adopt in the CEA? How about on the Rapsodie design? What centigrade do you use for each?

The melting point of the fuel on the JEFR design is estimated to be 2760 ± 30°C at initial state.

The maximum center temperature of the fuel by calculation is 2650°C at the hot spot with 110 % over power condition. Please show us the design examples in the CEA corresponding to the problems above mentioned.

A. We dont calculate the maximum temperature of the fuel. We appreciate hot spot conditions as follows:

for high density fuel (95 %) we appreciate the integrated conductivity $\int_{900^\circ C}^{Tc} kdt$ and we compare the value obtained to the integrated conductivity $\int_{900^\circ C}^{Tc} kdt$

TF: melting temperature of fuel

Tc: maximum temperature of fuel

(at the point of maximum neutron flux).

We evaluated $\int_{900^\circ C}^{TF} kdt$ at 47 watts/cm, after experimental results.

$\int_{900^\circ C}^{Tc} kdt$ is calculated taking into account the influence of every hot spot factor.

(ii) Q. The maximum temperature at the middle point of the clad should not exceed 650°C at the hot spot with 100 % nominal power condition on the JEFR design (This temperature limitation is adopted on the JEFR as there are reliable back data around this temperature.).

How do you take account of critical temperature limitations for the clad in the CEA? What is the reason?

A. For Rapsodie the maximum temperature of the cladding, calculated in the hot spot conditions should not exceed 650°C.

Main reasons of this limitation is the big decrease of the mechanical properties above 700°C for 316 type stainless steel, loss of ductility of such materials with increasing in pile temperatures.

For change of power in Rapsodie (I suppose what is mentioned here is the change of from 20 MWth to 24 MWth) we maintained the maximum temperature calculated in hot spot conditions, at 650°C. We adjusted the sodium flow across the core to fulfill this condition. We adopted for Fortissimo the value of 700°C.

(iii) Q. Give us your comments on the hot spot factors in the JEFR. Which factors have you changed for raising the Rapsodie power?

A. The method of calculation of hot spot temperatures used in C.E.A. is entirely different from the Japanese method. So we cannot compare with great precision the two methods, and can give only impressions. As a whole, it seems to us that the different temperatures you calculated, sodium outlet of the reactor, mean temperatures at the channel outlet (with complete mixing) maximum coolant temperatures, nominal and hot spot, maximum cladding temperatures, nominal and hot spot, are coherent between themselves and are in the good range. They approach the corresponding temperatures we now obtain in Rapsodie with a nominal of 24 MW. About your hot spot factors, perhaps the uncertainties of power and neutron flux are a little optimistic and the uncertainty of coolant flow rate pessimistic. But these are points of minor importance.

About hot spot temperatures of fuel, we think you are also in a good range. Nevertheless the value you have choosen for the gap conductance (0,85 W/cm^2) is too much optimistic and so is the value of the corresponding hot spot factor as we have already tell you, we choose 0,6 W/cm^2 $^\circ$C for nominal conditions, and 0,33 W/cm^2 $^\circ$C in hot spot conditions. As for the second part of your question, we never changed any hot spot factor for raising the Rapsodie power.

3.3 Constants

1) Comment

No comment

2) Questions and Answers

(i) Q. Material constants of PuO_2-UO_2 pellet being reached at high burn up (melting point, thermal conductivity, expansion coefficient, etc.,). Material constants of mixed oxide as a function of burn-up.

A. Material constants: We do not take account of any variation of meling point, thermal conductivity and expansion coefficient of UO_2-PuO_2 with burn-up in our calculations. Our calculations are made with the out of pile values.

(ii) Q. The effects of material constants changes of fuel pellets by irradiation are not fully considered on the design of the JEFR. Are there any appreciable differences on the design by those effects?

A. The maximum fuel linear heat rate which is forecast for JEFR is reasonable we think that, for the burn-up which is forecast in JEFR (5 a/o) it is not necessary to provide with the changes of thermodynamical material constant with burn-up.

3.4 Hot spot factor

1) Comments

(i) Statistical and semi-statistical method

In the JEFR project, you use a completely statistical method and every hot spot factor is assumed to be a random factor.

For Rapsodie, we have separated hot channel factors into two groups:

- True random factors are in first group and they are statistically treated.
- The factors issued from variables which give a systematic error are in second group, and are treated by a cumulative method.

This procedure is called "semi-statistical method" and it is more conservative than a simple statistical method.

(ii) Calculation formulas

In JEFR calculations, formula used for random variables computer the square of sum of every random ΔT (f-1). In Rapsodie calculations, formula computer sum of the square of ΔT (f-1)

Rapsodie $$\sqrt{\Sigma_{i=1}^{n} \left[\Sigma_{ij}^{n} \Delta T_j^2 (fij-1)^2 \right]}$$

JEFR $$\sqrt{\Sigma_{i=1}^{n} \left[\Delta T_1 (fi_1-1) + T_2 (fi_2-1) + \cdots\cdots \right]^2}$$

(iii) Numerical value of hot channel factors

It is difficult to compare the JEFR and Rapsodie hot channel factors because the JEFR factors given in the final conceptual design report are not elementary factors.

For example, F_{81} = 15%, and is given as uncertainty on cooling flow rate. It comes from factors of different origins, as:

- total flow rate random variations
- uncertainly on subassembly flow rate adjustment
- dimensions tolerances, etc...

(iv) Numerical results

a) Fuel clad hot point

The fuel clad hot point is not very different from Rapsodie

b) Fuel hot point

Comparison is difficult because:

- heat transfer coefficients are different

- we do not compute fuel temperatures in Rapsodie. We prefer to compare nominal and maximum conductivity integral to allowable conductivity integral.

We think JEFR fuel hot point is in a good range, but a little less conservative than in Rapsodie. We do not have changed hot channel factors from Rapsodie to Fortissimo, but we have increased maximum clad temperature (700°C instead of 650°C).

3.5 Gap conductance

1) Comment

No comment.

2) Questions and answers

(i) Q. Give us your comments on the gap conductance value of 0.85 w/cm^2 °C (He bond; the gap is 0.05 mm at 20°) in the JEFR.

A. We think that 0.85 $watts/cm^2$ is a too optimistic value for the gap conductance. We recommend, in calculation of nominal thermal conditions to use the value of 0.6 $watts/cm^2$ °C. In hot spot conditions for appreciate the probability of central melting of the pellet we recommend to use the value of 0.33 $watts/cm^2$ °C.

These values are recommended after estimations and measurements made on irradiated pellets. They take into account for the accumulation of gaseous fission products in the pin. As for the problem of the initial stage, we think that there is a real risk to center - melt some pellets, if the smear density of the oxide is low and the gap is large. To avoid this central melting one can increase by small steps and slowly the power of the reactor during the first start following the input of the fuel pins in the reactor. We can assert now that if the increase is too rapid the fuel will center melt. If it is sufficiently slow, oxide can density and adopts itself a good configuration for thermal exchange inside the pellet.

(ii) Q. What value of gap conductance did you adopt in Rapsodie?

A. We think that 0.85 watts/cm^2 is a too optimistic value for the gap conductance. We recommend, in calculation of nominal thermal conditions to use the value of 0.6 watts/cm^2 $^\circ$C. In hot spot conditions for appreciate the probability of central melting of the pellet we recommend to use the value of 0.33 watts/cm^2 $^\circ$C.

These values are recommended after estimations and measurements made on irradiated pellets. They take into account for the accumulation of gaseous fission products in the pin. As for the problem of the initial stage, we think that there is a real risk to center - melt some pellets, if the smear density of the oxide is low and the gap is large. To avoid this central melting one can increase by small steps and slowly the power of the reactor during the first start following the input of the fuel pins in the reactor. We can assert now that if the increase is too rapid the fuel will center melt. If it is sufficiently slow, oxide can density and adopts itself a good configuration for thermal exchange inside the pellet.

(iii) Q. What value of gap conductance between the fuel pellets and clad tubing shall be adapted to the design?

For manufacturing of fuel pins and lowering the smear density of fuel pins, the gap between the pellets and tubing should be large to minimize the swelling on burn up. On that case, the center of fuel pellets may melt or the temperature of the pellet center shall rise in the initial stage of burn up since the gap conductance shall be low. But the gap shall be closer after one or two days burn up, and the conductance shall be rise, then the center temperature shall become lower. What do you recommend about the possibility of center melting caused by high temperature in the initial stage?

A. We think that 0.85 watts/cm^2 is a too optimistic value for the gap conductance. We recommend, in calculation of nominal thermal conditions to use the value of 0.6 watts/cm^2 ^{o}C. In hot spot conditions for appreciate the probability of central melting of the pellet we recommend to use the value of 0.33 watts/cm^2 ^{o}C.

These values are recommended after estimations and measurements made on irradiated pellets. They take into account for the accumulation of gaseous fission products in the pin. As for the problem of the initial stage, we think that there is a real risk to center - melt some pellets, if the smear density of the oxide is low and the gap is large. To avoid this central melting one can increase by small steps and slowly the power of the reactor during the first start following the input of the fuel pins in the reactor. We can assert now that if the increase is too rapid the fuel will center melt. If it is sufficiently slow, oxide can density and adopts itself a good configuration for thermal exchange inside the pellet.

3.6 Decay heat curve

1) Comment

No comment

2) Questions and Answers

(i) Q. We have received your F.P. decay heat curve for your design which were based on the thermal fission.

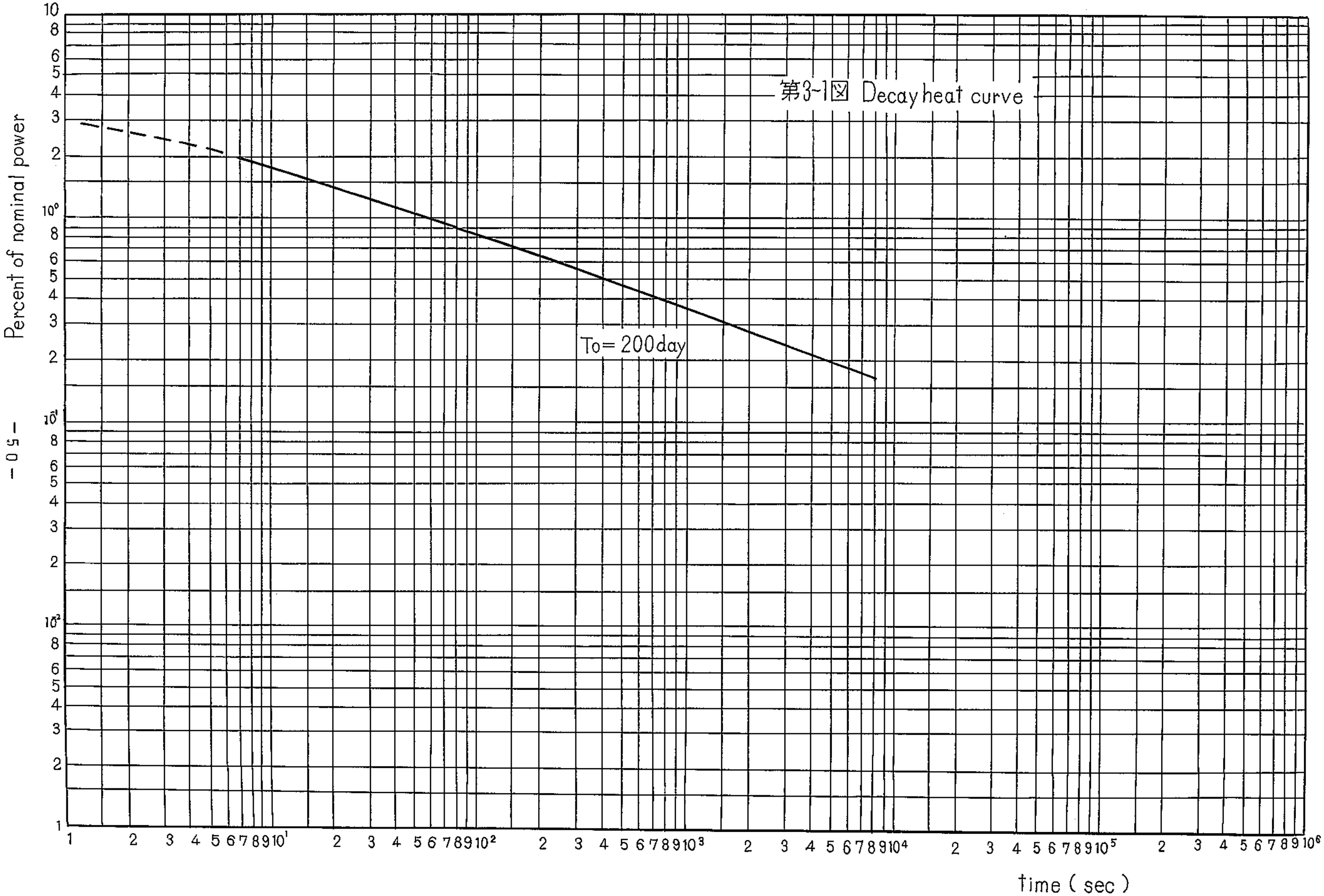
Would you suggest us again the basis that this curve is sufficiently conservative for fast reactor?

A.

3) Information

(i) Courbe de décroissance de la puissance résiduelle due aux produits de fission -

La courbe est le résultat d'un calcul effectué en utilisant les rendements directs dus à une fission thermique dans le Pu 239. La partie de la courbe correspondant à des temps de refroidissement inférieurs à 1 heure est sous-estimée, car les calculs no tiennent pas compte des émetteurs à vie courte d'une manière assez précise.



第3-1図 Decay heat curve

4. 炉心および炉体設計

4.1 General comments

1) Comments

(i) We are not in favour of an irradiation loop. The experiments intended with this loop can be performed easier and safer inside the reactor. Moreover, this loop would certainly complicate in a large extend the detail design of the reactor. As regards the operation, nothing good can be expected.

The JEFR is designed so that an irradiation loop can be used. Such a loop presents very large difficulties in its own design and includes many important consequences for the general design of the reactor:

- core support plates and upper shielding must permit the loop location
- the loop has to be cooled by a special circuit where active sodium is flowing
- the irradiation loop must be loaded and unloaded and it is active
- the active part of the loop has to be dismantled in a cell.

In regards with these large difficulties, we do not see a real advantage of an irradiation loop. Indeed, we think that is is possible to irradiate, in any subassembly, some fuel pins with harder characteristics, since the reactor must be able to operate with ruptured fuel clads.

(ii) Furthermore, the two values of fuel density (95% and 84% of theoretical density) are not very clear, as for the reasons of not retaining only one value.

(iii) Structure and cladding materials irradiation is an important point, and enough space and reactivity margin is to be provided for, taking into account that results can rapidly be obtained, only in high flux region.

(iv) You propose to use inconel X in subassemblies for springs or grids. We made no irradiation of inconel X and we think you have to find information on the behavior of this material under neutron flux in literature. It is possible that it becames brittle. An irradiation test is probably to be made.

(v) You bind outside reflector subassemblies at their upper parts. We do not have such a device in Rapsodie.

This bundling may be good for JEFR because you have very hard earthquake specifications and it forms a material boundary against a large oscillation of subassembly bundle.

But you have no check whether it changers reactivity coefficient due to subassembly bundle expansion.

(vi) Maintenance of control rod mechanisms

Room must be provided to allow a easy access for maintenance of control rod mechanisms, even if the mechanisms are located in a pit as it is proposed for JEFR.

(vii) CEA 側の経験から次の３つの comments を与えられた。

a) B_4C の体積比をもつとあげることは充分可能である。

b) ^{10}B 濃縮度は９０％以上のものをつくることは可能である。

c) 上の i) および ii) の結果より制御棒の本数を減らすことは可能である。

(viii) JEFR の制御棒は非常に dirigity だがflexibility をもたせた方がよい。設計変更が必要であろうがflexibility をもたせれば cover plate を固定させることができる。 Rapsodie は駆動機構を持ち上げる方式である。（図面入手）

2) Questions and answers

(i) Q. How long is the lifetime of safety rod?

(% burn up)

A. Rapsodie では full power condition で計算値は 220 JEPP である。 JEPP は Jours Equivalents à Pleine Puissance (Equivalent days at full power の意味 ）を意味する。 この限界は control rod 内の He pressure に依存する。実際の実験結果では計算値よりも大きい圧力となつた。

Lifetime is limited by technological conditions, for example by gas pressure rising, and not by burn-up.

(ii) Q. How much is the volume of He gas generated in Safety Rod at the lifetime?

A. 20°C で 20 ℓ の volume である。これは real temperature で充分な volume である。圧力の機械的限界は 40 bars, gas release rate は 40 ~ 50% と考えておくのがよい。この値は後で check する。前に 15 ~ 20% release rate で計算したが，よくなかつた。制御棒部分の実際の温度を知ることは難かしいので時々圧力を check した方がよい。例えば Rapsodie の場合 80 JEPP で diameter を check し，100, 160, 220 JEPP でそれぞれ pressure の check をした。

(iii) Q. What non destructive testing methods except dye penetrant testing are available and applied for bellows of bellows sealed valves?

A. Bellows were used for the pipe double wall in Rapsodie. Bellows and double wall tightness were checked by the same test. Over pressure was inserted into the double wall and it was checked there was no pressure decrease.

Bellows had been tested previously by manufacturer during fabrication.

(iv) Q. Please let us know your practice on the model and type of motors used for sodium equipment in and out of the containment vessel.

① Model

a) standard model or explosion proof model?

b) If it is standard, is it enclosed, totally enclosed, drip proof or splash proof model?

c) If it is explosion proof, is it flame explosion proof or increased safety explosion proof model?

d) If it is increased safety explosion proof, is it drip proof or totally enclosed model?

② Type of coil

Is it winding or squirrel cage type?

A.

3) Informations

(i) Rapsodie の Bellows

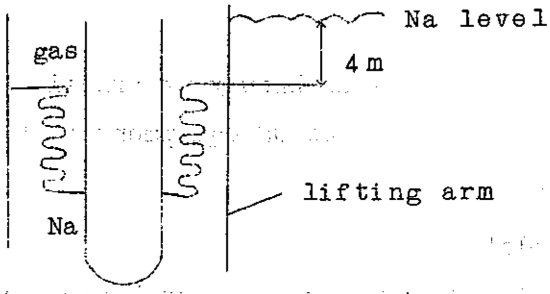
フランスではround corner の溶接ベローを開発している。これはsharp edge のベローに比べてたわみ率を大きくとれない。

PHENIXの燃料交換機にはround corner のベローを使用する考えである。

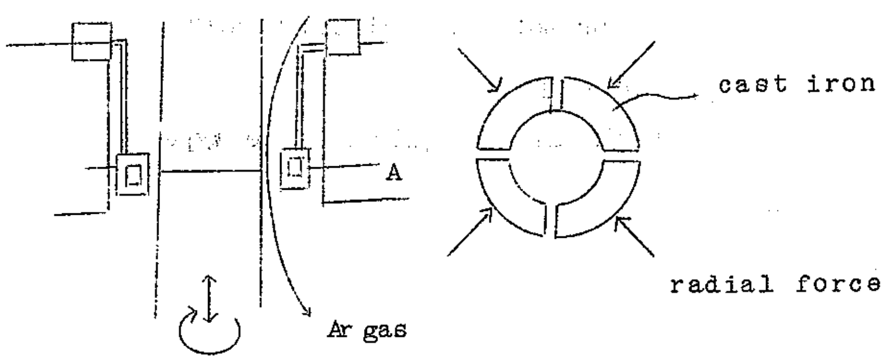
control rod は大きいたわみ率を必要とするのでsharp edge のベローを使用する。これはRapsodie で十分試験済であり問題ない。

round corner　　　　sharp edge

PHENIXの　交換機のベローの使い方



(ii) Rapsodie のシール方式(lifting arm の seal)



Hispano Suizn が製作

4.2 Core design

4.2.1 Fuel pellet

1) Comment

No comment

2) Questions and answers

(i) Q. How do you take account of the diameter of fissile particles and their distributions in the fuel pellet in the design?

A. With the sintering process we use for the fabrication of fissile pellets, we have in principle no risk of advant of a phase different from the UO_2 - PuO_2 solid solution. Nevertheless, in fact, we have already had advent of UO_2 as a second phase in some of our fabrications. We don't think that this phenomenon is very important for RAPSODIE, which is a small reactor with a ever-negative sodium coefficient. The same conclusion can be applied to the JEFR, in our opinion.

(ii) Q. Cracking phenomena of fuel pellet is not under consideration on the JEFR design. How can we take account of this phenomena in the fuel design if necessary?

A. About cracking phenomena, we suppose that the gap which primitively exist between fuel and cladding is uniformely distributed across the pellet, as porosities the total volume of which is equal to the volume of the primitive gap. We take this into account in our thermal calculations and in our estimations of swelling.

(iii) Q. Please show us the outlines of experiments and results of the irradiation studies that you had carried out on the fuel pellets and elements of Rapsodie.

A. Joint to this paper is a digest of results which have been obtained on RAPSODIE fuel elements. This digest has been presented as a paper at the recent conference of WASHINGTON by M. ANSELIN (p.J. N 1)

4.2.2 Gas release ratio

1) Comment

No comment

2) Question and answer

(i) Q. Please show us the gas release rate from the oxide fuel as the design criteria which you adopt in France.

A. We take the curve from report number 4B/presented in UKAEA London Conference. These figures are in good agreement with our preliminary results.

4.2.3 Gap distance

1) Comment

(i) Going further with the fuel, it appears that the design of the pins are perhaps not the best: the gap between fuel and cladding seems too small and the criteria for clad probably not very adequate. Gap conductance seems too optimistic.

(ii) The pad theorical gap (0.1 mm) is the same as in Rapsodie.

But this gap was not found between a large part of subassemblies in Rapsodie mock-up and reactor. This is due to several small mal fabrication within normal tolerances. However, we have been able to load and unload Rapsodie subassemblies without any difficulty.

2) Questions and answers

(i) Q. In the JEFR design, the pellet diameter is 5.5 ± 0.05 mm and the inner diameter of the clad is 5.6 ± 0.03 mm. Please give us your comments for these values.

A. We think that a so calculated value of gap must be really considered as a very minimum value.

(ii) Q. The gap distance of the JEFR is decided such that it must not change from zero to negative by thermal expansion under nominal operational conditions. Please give us your comments for this gap.

A. We cumulate the void provided inside the pellets (difference between theoritical density and real density of the sintered pellets) and the volume of the hot gap (hot meansing gap calculated after thermal dilatations of the pellet and the clad) to calculate the total volume which is available for the accommodation of the swelling effect.

(iii) Q. Please show us the criteria on the gap distances between spacer pads.

A. On Rapsodie, there is a theoretical clearance of 0,1 mm between two adjacent pads. Actually, owing to the defects in subassembly manufacturing, a large number of the pads are in contact each other after core loading at cold temperature, but we got no difficulties for removing or inserting subassemblies into the core, owing to their large flexibility.

A number of dummy assemblies must be manufactured, then controlled and placed on a dummy core support plate grid. Then, it must be measured what are their real positions in the subassemblies bundle. Some must be introduced or removed from this element of core support plate to check the loading and unloading conditions. Results closely depend from the manufacturing tolerances on the subassemblies.

(iv) Q. How do you take account of the swelling effects for deciding gap distance in the design?

A. We think that Japan workers will have some difficulties during the cladding operation (putting pellets inside the clad) with such values. The minimum gap (which is in JEFR 0.02 mm) must be at least 0.05 mm. Reasonables values can be 5.5 + 0.05 mm for the pellet diameter and 5.63 ± 0.03 mm for the inner diameter of the clad. If the values established now in the JEFR are adopted many fuel pins will be rejected because impossibility of normal cladding.

(v) Q. What kind of factor do you consider for the decision of the gap distance? Please give us these original data?

A. Generally speaking we consider that swelling effect is the major factor to determine the gap distance.

4.2.4 Behavior of fission products

1) Comment

No comment

2) Questions and answers

(i) Q. The solidable fission product gases such as C_s, etc. accumulate in fuel pellets. How should we consider to these in the design? Are there any experimental results for these behaviors?

A. We dont consider the problem of solidable fission product gases in our design. We never had trouble with them till now; it is possible that for higher burn-up (50,000 MWt/Tonne) solidable fission products can bring some trouble.

4.2.5 Design criteria of the clad strength

1) Comment

No comment.

2) Questions and answers

(i) Q. About the design criteria for the clad strength, the method of the combination for the following items.

1. the stress by fission product gases,
2. the thermal stress,
3. the stress by swelling,
4. the creep,
5. the thermal shock on the scram,

⑥ the thermal stress caused by the axial temperature distribution,

⑦ the low cycle fatigue.

A. We take into consideration only the stress produced by fission gases and the thermal stress. It would be very interesting to consider the stress induced by swelling but till that day, we never found a calculation method available for these stress. Different trials have been made in foreign countries but in our opinion none can give interesting results.

About the creep, we suppose that it is only induced by stress produced by fission gases and that the other stresses which act on the cladding have no influence on the creep rate.

It would be interesting to take into consideration the thermal shock on the scram, but we think that it is not possible to apply the conclusion of Section III of ASME Code, the cladding material having properties completely changed by neutron irradiation.

We don't take into account thermal stress caused by axial temperature distribution and low cycle fatigue.

(ii) Q. ① Please give us your comments for the limitations in the JEFR that the thermal ratchet should not yield by the combinations of the inner stress and the radial thermal stress.

② If we design the fuel pin by following criteria which you have recommended.

$$\sigma_t + \sigma_p < \sigma_y$$

σ_t : thermal stress

σ_p : pressure stress

σ_y : yield stress of the caldding material

it is necessary to lower the linear heat rate of the fuel pin by large extent from the current design of JEFR fuel pin.

Have you designedthe fuel pins of the Rapsodie Reactor with this criteria?

Please explain the reasoning of recommending this criteria.

A. You apply these equations

$$\sigma_t < 2\sigma_t$$

$$\frac{1}{4}\sigma_t + \sigma_p < \sigma_y$$

When applying these equations, you authorize plastic deformation during the first cycle and no further plastic deformation. This is quite reasonable for out of pile utilizations. But for a fast reactor, you must consider that plastic deformation will not be produced during the first days inside the reactor, when the cladding material is still ductile, but when σ_p is sufficiently high that is to say, cladding is "old" and, consequently brittle. Then, if you impose a plastic deformation, it could lead to a cladding rupture. So, as for Rapsodie we apply the equation $\sigma_t + \sigma_p < \sigma_y$ which give us the guarantee of never imposing a plastic deformation.

被覆管のラチェットについて： Ratier JFER について $\sigma_t + \sigma_p < \sigma_y$ の式を使用し試算を下記の通り行なつた。

$$\sigma = 70\ \mathrm{Kg/cm^2} = 0.7\ \mathrm{Kg/mm^2}$$

$$\sigma_p = \frac{P \times D}{2 \times e} = \frac{0.7 \times 6}{2 \times 0.35} = 6\ \mathrm{Kg/mm^2}$$

$$\sigma_t = \frac{E\,\alpha\,\Delta\theta}{2(1-\gamma)} = \frac{16000\ h \times 20 \times 10^{-6} \times T}{2 \times 0.77} = 8.2\ \mathrm{Kg/mm^2}$$

$$T = \frac{430 \times 0.35}{\pi \times 6 \times 0.22} = 36^\circ\mathrm{C}$$

故に $\sigma_t + \sigma_p = 14.2\ \mathrm{Kg/mm^2}$ とする。

Cadarache にあるＳＳ材についての σ_y の表によると 1 4.2 kg/mm² は elastic を remain するぎりぎりの値である。$\sigma_t + \sigma_p < \sigma_y$ を採用するのならば，多少の余裕をみて被覆管厚をもう少しうすく，又，reservoir volume をもう少し大きくすればよい。

この式を recommend する理由は

OMWD/T においては $\begin{pmatrix} \sigma_t \doteq 8 \text{ Kg/mm}^2 \\ \sigma_p = 0 \end{pmatrix}$ で no elastic deformation with cycling であるが,

5% Burn up においては $\begin{pmatrix} \sigma_t = 8 \text{ Kg/mm}^2 \\ \sigma_p \sim 3 \sim 4 \text{ Kg/mm}^2 \end{pmatrix}$ でcycling のもとで plastic deformation

を起こす。 neutron dose による material embrittlement creep deformation swelling deformation が起きる。このことは London conf. paper (1966) 4B/2 P.4 に記述されてある。

(iii) Q. ① The limitation for the creep is 1 % for the radial direction in the JEFR. We adopt the assumption that the stress which yield the creep depends only on the inner stress by fission products, and that the inner stress rises in proportion to burn up (operation time). Please give us your comments to these assumptions and show us the examples of analysis.

② You mentioned that it was a little bit too optimistic to allow fuel claddings 1 % creep strain.

Would you please explain this in details?

A. We think that 1 % calculated as the creep deformation caused by fission gas stresses is optimistic, because we have made rupture tests by internal gas pressures on irradiated claddings (claddings of Rapsodie fuel pins).

〔注 第2次概設では最大(公差の最も悪いもので316溶能化処理材として) 0.12%である。このときの $\sigma_p = 7 \text{ kg/mm}^2$〕

(iv) Q. ① Please show us the following data a) ~ d) for deciding the design criteria of the clad strength.

a) relation between yield point or tensile strength and allowable stress,

b) actual values and original data for yield point and tensile strength,

c) creep data and these experimental data,

d) creep rupture data and these original data.

② How do you consider the irradiation effects on the design for the items above-mentioned?

A. Mechanical characteristics of the clad.

For applying the equation $\sigma_t + \sigma_p < \sigma_y$ we take for σ_y the yield point value as measured by out of pile tests. For creep data, we apply an equation of the type

$$V = A \sigma^m$$

σ being the stress applied by fission gas

A, n constants

A, n are determined experimentally, before any calculations on specimens of claddings, taken from the batch which will be used for the fuel pins themselves. For safety purposes, we don't use the true experimental value of A, but three times this value 3 A. We think that such a method will cover different uncertainties such as neutron irradiation effect on creep laws.

(v) Q. We would like to know your experiences of the quality control for fuel clad material.

A. Joint to this paper is a note presented at the ASNT National Conference International Session (Detroit October 14 - 17th 1968), and related to Eddy current application to non destructive testing of fast reactor fuel - subassembly clads (P.J. n° 2).

4.2.6 Structure of the fuel elements

1) Comment

No comment.

2) Questions and answers

(i) Q. ① Please give us your comments for the integral fuel element which contained upper and lower axial blanket elements and the core elements in one canning tube.

Why do you adopt the separated type? In Phenix also?

② The fuel element structure is adopted so as to release the axial stress of the clad and of the fuel by forming the spaces between the axial blanket and the core fuel in the JEFR. Though such considerations are not paid on the few other design examples (G.E., etc.), how do you think these problems?

③ About the draw-out processing at the edges, the conclusion had not been now obtained for the best processing in the JEFR. Please show us its processing methods and the inspection methods if you have any experience.

④ Please show us your experiences of the welding and the inspection methods.

A. The two devices integrated and separated types, are quite possible.

The separated type minimizes the coolant pressure losses and simplifies the fabrication problems of the fissile fuel pin.

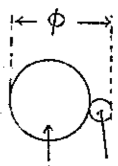
The integrated type minimises the distance between fertile and fissile fuel and increases the breeding ratio.

For Rapsodie, we adopted the entirely separated type, but in Phenix, the lower axial blanket is integrated, upper axial blanket beeing separated.

For future reactors, the definitive choice will be made on economical considerations. We have not very well understood the meaning of questions 2

(ii) Q. Please give us your comment on the Accuracy of fuel pin dimensions and wrapper tube inside dimension.

A. Rapsodie では pin の max. dimension で, wire の min. dimension を採用し $20^\circ C$ で no gap である。 40,000 MWD/T 迄の経験では問題を生じていない。



pin, wire

〔fuel pin + wire〕の直径を ϕ とし，その製作誤差を $\phi^{+\varepsilon_1}_{-\varepsilon_2}$ とする。

Wrapper tube 内面距離を $H^{+\varepsilon_3}_{-\varepsilon_4}$ とすると

$\sum_i [\phi_i + \varepsilon^{1}_{i}] = H - \varepsilon_4$ でよい。 (at cold condition $20^\circ C$)

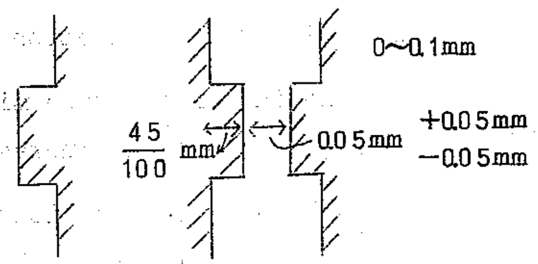
4.2.7 Fuel assemblies (Wrapper tube)

1) Comments

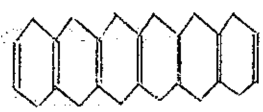
(i) You propose stellite for pads spacing the subassemblies. This is probably a correct design. We use pads stamped by hexagonal wrapper strain, with chromium deposit on the pads.

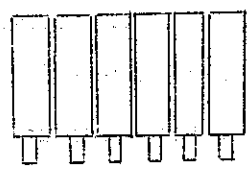
Our design allows more strain in reactor and we have seen some change of the distance between two opposite pads in spent fuel subassemblies. But we are not yet able to tell whether it is better or worse to permit or not this type of strain. We are not sure chromium deposit which we use is necessary. We did it as more conservative for Rapsodie, but it is essential that deposit quality is good.

パット部はstick 防止のためには chromium の電気メッキをやつた方がよい(約5cm位)，しかしspecification およびquality control はむつかしい。フランスでは2社にやらせたところ一方は駄目であつた。

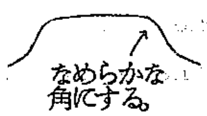


cleaningの問題も考慮しておく必要がある。Rapsodieパット間のgapは右図のとおりでgapを設けたのはbowing のためのみである。bowing については右図のように6〜7本アセンブリを並べてテストをやつてみた。





パット部は右図のようになめらかな角にしておくこと。



(ii) We recommend you to make a prefabrication of a sufficient number of dummy subassemblies and to check their dimensions in the purpose of knowing real tolerances. You can also load these subassemblies in reduced support plates with a small number of subassembly location, in the purpose of checking their behavior and introducing and removing subassemblies in real conditions. This test may be done in air. It will permit you to be sure of subassembly fabrication tolerances.

This test is particularly useful if you adopt design differences as pad materials or pad location in comparison with Rapsodie.

(iii) You propose to locate pads on the angles of the subassembly hexagonal wrappers. We have always set the pads at the middle of the six walls of the hexagonal wrappers. We did that at the first time because it was easier for fabrication and we did not study another solution because the first one was well working.

It is probably possible to locate pads on the angles, but it seems to us that this solution shows a little bit less flexibility in subassembly rotation during introduction into the subassembly bundle.

2) Questions and answers

(i) Q. The grid type and the spiral wire type are discussed for the spacer in the JEFR. Please give us your comments for these spacers based on your experiences.

A. We don't have any experience on the grid type for the spacer. We have only experience on the spiral wire type and spiral fin (incorporated in the cladding). These two devices are quite correct. For the spiral wire device, the wire is only tied on the end plugs of the pin, with no intermediate welded spots on the cladding. These device has been tried on Rapsodie type fuel pins in the Rapsodie Reactor. It has been tried also on Phenix type fuel pins in sodium loops. No remarks.

The spiral fin type has been tried on standard Rapsodie fuel pins. It gave satisfaction. We don't use it for Phenix because difficulties in obtaining claddings with spiral fins in such lengths with good tolerances.

(ii) Q. Please give us your comments on the following items about the spiral fins and the spiral wire;

① Considerations for the thermal expansion and swelling. How do you take the clearance between the wrapper tube and the bundle of fuel elements.

② Are not there the problems of the damages for the clad by the galling in the case of the spiral wires and fins? How do you consider the counterplan for these problems if there are ? any?

③ Please show us the criteria for deciding the spiral pitch and lead.

A. ① For Rapsodie, we consider the maximum allowable dimensions for diameter of the clad + height of the fin. We suppose that all pins have this dimension and adjust clearance to make possible the entry of 37 such pins in a hexagonal array in the minimum diameter hexagonal tube.

② in our knowledge, there is no problems of galling produced by fins or wires. In the two cases experimented fuel pins in Rapsodie never reveal such damage.

③ the diameter of the wire (or distance between two adjacents fuel pins) has been adjusted in Phenix to maintain a definite percentage of circulating sodium inside the core (35 %).

The pitch of the spiral wire has been established after a compromise between pressure loss and mixing of coolant between the different cells.

(iii) Q. Are there the data on the fretting corrosion between the grids and the clads in the medium of the sodium?

A. In our opinion, there is no risk of fretting corrosion between clads and grids. We have never seen any evidence of such a corrosion, on spiral fins, on the interior wall of the external hexagonal tube, for instance.

(iv) Q. Flow mixing factor

① Please show us your experiments, measurements and analyses.

② What value do you estimate for the mixing factor on Rapsodie. (We've assumed zero on the JEFR)? Please show us its base.

A. Calculations have been made and have shown a great deal of mixing between the different unit cells. Some experiments have been done, injecting hot water in a mock up subassembly crossed by a flow of cold water. These experiments have shown a certain mixing. After examinations of irradiated Rapsodie fuel elements, one can say that peripheral fuel pins located just against the hexagonal tube are significantly colder than central pins, the difference of temperature beeing not clearly precised at this day.

For Rapsodie, we suppose that each unit cell of coolant acts independently of the others. So we suppose practically no mixing.

(v) Q. Please give us the calculation methods and the experiments for the vibration of the fuel pins and fuel assemblies.

A. Vibration

i) We did not calculate vibration of the pins or total subassembly of Rapsodie. We just checked there was no vibration during hydraulic test, by very simple methods as looking at pins or listening to noise.

We are presently developing vibration measurement methods for the future, because sodium velocity is larger in large reactors.

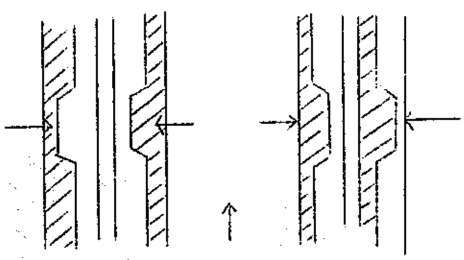
(vi) Q. How did you put the spacer pad on the wrapper tube by welding, spot welding, or some other method?

Please explain the reasoning of placing the spacer pad on the flat surface of the wrapper tube rather than placing on the corner of hexagonal tube.

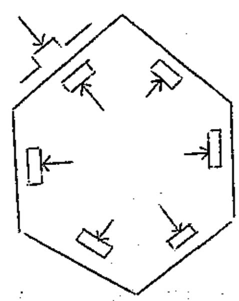
A. See the comments

(vii) Q. Thickness, method of attaching of spacer pad?

A. embout 法 (swanged) による。押し出すだけである。(下図参照)



wrapper tube
の内側(たて断面)

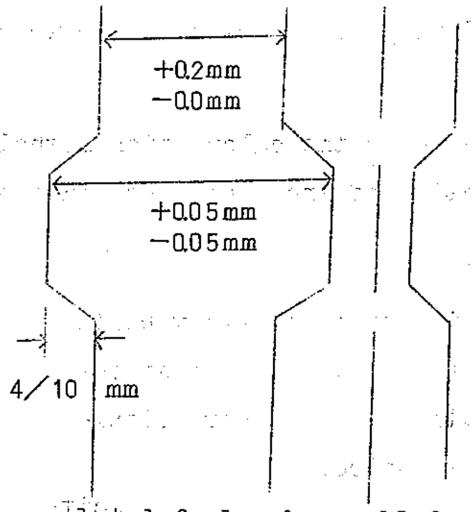


(平断面)

(viii) Q. Manufacturing accuracy of wrapper tube?

A. flat 部に対し $49.8^{+0.02}_{-0}$ mm を考えた。

しかしパット部に対しては ± 0.05 mm を考えた。



(ix) Q. Accuracy of completed fuel subassembly?

A. Rapsodie では各 subassembly の top における中心のずれを下記のように押えた。

2.5 mm for fuel
3.0 mm for blanket
3.0 mm for steel

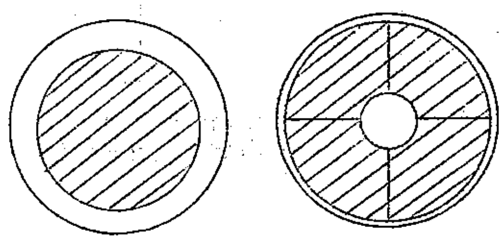
この値の制限のみでよい。

Rapsodie ではこの部分の welding に問題があつたが，製作後の検査としては上記のしんの“ずれ”のみでおさえた。

3) Information

(i) burn up による熱伝達の変化

fuel subassembly の burn up が進むにしたがい中央に穴ができ，clading と fuel pellet の clearance が小さくなり熱伝達率は良くなる（pellet の再結晶現象）



4.2.8 Handling head

1) Comment

No comment.

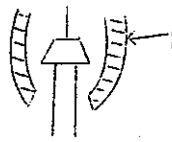
2) Questions and answers

(i) Q. Is any treatment such as hard facing necessary for the gripper of fuel handling machine?

Please show us the actual examples on this problem.

A. We use stellite (or similar material) for the gripper fingers.

We do not use stellite for subassembly heads because they are used three times only in reactor for a fuel subassembly and twice for another subassembly.



Stellite を使えば hard facing の必要はない。

(ii) Q. Please show us the criteria how do you take the flow rate through the handling head for the purpose of measuring the outlet temperatures on the tops of subassembly.

A. We designed the Rapsodie subassembly head so that its mechanical resistance was good. Afterwards, we determined flow rate through handling head by water test measurement. It is narrowly dependent on subassembly flow rate. It is about 4% of the flow rate of a subassembly located at the core center and less for other locations.

特に criteria はない。

(iii) Q. Please show us the accuracy of the outlet temperature of the subassembly affected by the surrounding sodium temperature.

A.

4.2.9 Spacer

1) Comment

(i) You discuss possibility to use either grids or spiral wires as spacers for fuel pins.

We have always used spiral pins or spiral wires and we have no experience of grids. We got no trouble with spiral pins or spiral wires, but we have a tendency for using wires because the clad quality is better.

Sodium mixing in the subassembly is favoured by spiral wires and this is another argument for this solution.

2) Questions and answers

(i) Q. The spacer pads in the JEFR are put on the same height of the core assemblies.

Please give us your comments for the spacer pad positions.

A. The position of the pads should be determined in order to obtain a zero or negative reactivity coefficient. This position does not matter for subassembly handling.

(ii) Q. Please give us your comments for the material of the spacer pads.

A. In Rapsodie, subassemblies are provided with chromium plated pads. It is perhaps needless, but we have no experience without chroming the pads.

Note: The quality of the chromium plating must be excellent and it is necessary to control that cleaning is good after chroming.

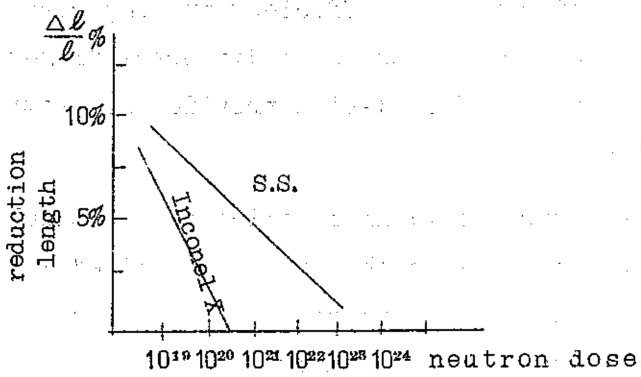
(iii) Q. You mentioned that it was undesirable to use Inconel X as a grid material because of its heavy irradiation embrittlement.

Would you kindly explain the actual data, if available?

A. We mentioned it is undesirable to use Inconel X as a grid material.

In our opinion, Inconel X knows great deterioration of mechanical properties even after a short irradiation by thermal or fast neutrons. Ductility, as measured by a mechanical rupture test at temperatures in the range 550 - 750°C becomes zero. If your grid has to endure some deformation it is not possible to use such a material. Why not use 316 stainless steel?

Inconel X の embrittlement について, Inconel X による embrittlement は比較的低 flux の条件で起き, grain boundary に crack が生じる。



4.2.10 Orientation and supporting method of periphery on the core elements

1) Comments

(i) In Rapsodie, the subassemblies are positioned in a good angular direction by the loading machine. We dropped this principle in the Phenix design to get a simpler loading machine. The subassemblies are loaded in any direction and rotate under the couple due to special cams located at their top and bottom. This procedure was successfully tested and leads to a simplification of the subassembly transfer mechanisms. But it must be carefully designed and tested.

Many various designs of this device can be found without difficulty, but mechanical tests are necessary first in air, then in sodium in connection with fuel loading machine test.

Rapsodie assemblies の orientation は3方向の任意性があり,一方向にのみ決めた guideは持つていないが, 現在迄非常にうまく操作出来, 1回の trouble を除いて困難なことは経験しなかつた。assemblyの固定のために guide をつけると loading machine がもつと複雑になつて長時間の Test が必要となろう。

(ii) In Rapsodie, we designed "interchangeability keys" at the subassembly bottom just to prevent loading of any subassembly in a location where cooling is not sufficient in comparison with heat production. We use for that tubes of different diameters in the subassembly lower part. This design provides no restriction to subassembly rotation.

You designed keys providing restriction subassembly rotation. It is not indispensable since this is not realized in Rapsodie without trouble. But, we think it is possible if you allow some rotation flexibility when subassembly is introduced into the subassembly bundle.

(iii) Special cams

Many designs can be made for these cams. Our experience is bound to larger subassemblies for Phenix and other power reactors and cannot be directly used to a smaller reactor.

2) Questions and answers

(i) Q. Please explain us the counterplans for the orientation of the core elements in relation to the insertion method of the core elements.

A. We are not sure to have well understood these questions.

In our reactors, the subassemblies are either oriented with help of the handling chain or oriented by themselves by the self guiding system discussed in the "comments".

(ii) Q. Please give us your comments for the key method in the JFER.

A. a) to avoid subassembly loading at position where it would be overheated when reactor is in operation. For that reason, we have also keys on Rapsodie. It is not proved that they are absolutely necessary, and administrative instructions may be sufficient. But it is more conservative.

b) to avoid subassembly rotation. The Rapsodie nozzle keys allow a free rotation and no inconvenient resulted. However, the keys you designed seem very acceptable.

(iii) Q. The core and the blanket parts are surrounded by the reflectors being fixed, on the base of which the removable reflectors and the blankets are orientated in the JEFR. Please give us your comments for it.

A. This system is not absolutely necessary. If it exists, it must be carefully checked to know if it does not lead to a positive reactivity coefficient.

3) Information

(i) Phenix 用燃料装入キー

Phenix も燃料装入用のキーはあるがJEFR のようなキー溝を切つての方向指定方式はやつてこない。

Cadarache で subassemblyを見たところ，6角の各角に長さの異なるキーが6ケ付いている。これは燃料装入時，周囲の6本の上端にひつかゝらないようにするためである。subassemblyの方向は6方向いずれも入り方向性はない。キーの高さはかなり高く，キー溝を切る必要のない高さである。このキーはまた燃料装入された状態でのラツパーチューブの間隔を保つ役目もしている。

4.3 Reactor components

4.3.1 Design of the reactor components

1) Comment

No comment.

2) Questions and answers

(i) Q. About the reactor components on your experience: the core support plate, the support structures, the core upper structures, the other core inner structures, the reactor vessel, the upper shield plug;

Please show us on the following items.

① the design conditions,

② the basic philosophies on the design conditions and on the reasons by which you have decided structures,

③ the test items and these procedures on the tests,

④ the presumable problems which have been yielded on the each way to the manufactures, the constructions and the operations, and the means for solving these problems.

A. ① ② A study of the operation of each reactor component is done by using dynamic calculations. The hardest conditions at nominal operation are investigated and the transients are determined. By this way, conditions are assigned to the components.

a) normal / transient / abnormal } の3つの条件について operation analysis を行なう必要がある。temperature を求め、さらに temperature change の量を求める。

b) nominal / transient } の両方について stress の計算をする。

c) ASME section III case 1331

基本的な考え方は以上 a), b), c), の通りに進めればよい。

③ See the Development tests where a testing program is discussed.

a) geometrical test

b) hydraulic

c) high temp. Na 中での test

d) mock up test ... flow change condition での response time などをやる必要がある。

④ We consider that the main problems, which could arise, are as follows:

a) welding specification and control

welding の control が問題, 特に specification と material について注意が必要。

b) surface treatment such as stellite deposit for the subassembly supports on the core support plate, chroming of mechanisms, etc...

必ず使用する材料は makerから試験片をとりよせて Na 中で test をする必要がある Rapsodie 建設段階では Cr deposit, stelite deposit など経験した。

c) adjustment of the position of the rotating plug. The positionning errors should be acceptable for the extreme positions. ... subassembly中心軸の tolerance はいろいろなところからくる。Rapsodie の control rod のずれは最大 6 mm 程度でこの量で充分 scram 出来る。

d) adjustment of the verticality of the control rods mechanisms in the guide thimbles of the plug.

e) nuclear cleanliness during reactor construction.

(ii) Q. Please give us your comments for the reactor components in the JEFR based on your experiences.

A. Our two main remarks are given in the "general comments"

① it is better to have no displacement on the core cover plate,

② the sodium supply to the core grid plate is too complicated.

4.3.2 Vessel

1) Comments

(i) The vessel diameter can be decreased if rotating fuel storage is dropped, as we propose, and if core cover plate is less broad by cancelling the spider arms, for example.

(ii) Water tests of the upper part of reactor vessel

These tests are necessary to determine:

① Level of sodium outlet nozzles.

Sodium outler nozzles must be at a level as high as possible to improve sodium mixing, especially during scram. But argon entrainment at the nozzles must be avoided. That occurs if outler nozzles are too close to sodium free surface.

ノズルの位置はガスの方から考えた場合は低い方が良いが thermal shock mixing 効果を考えた場合は高い方が良いという矛盾がある。
Rapsodieの Na レベルとノズル中心の高さの差は約1 mである。

② Sodium velocity distribution

It is expected that sodium horizontal velocity is large from subassembly outlets to vessel wall under core cover plate. That may result into a large temperature gradient along the vessel at the boundary between cold and hot sodium.

If estimated gradient is large, the vessel has to be shielded by baffles.

③ Temperature distribution during scram

By using tracers in water tests, delays of sodium flow from subassembly outlets to different parts of vessel wall can be determined. Temperature distribution during scram can be calculated from this test and proper thermal shocks shielding baffles determined.

④ Vibration of components located in the upper part of the vessel

In France, all these tests are made by a specialized hydraulic laboratory "SOGREAH" (in Grenoble).

(iii) The lower part of the JEFR vessel is very complicated. We are pretty sure that a simpler design is possible and would be better.

A simpler design must particularly be sought for any part of the structure connected with the reactor vessel since the integrity of this vessel is essential during the reactor life. This implicates that you cancel the adjustment of the blanket flow rate.

If you follow our opinion, we think you have two main ways to develop a new design.

In the first way, high pressure sodium is fed directly into the two plates supporting the subassemblies. The vessel bottom is at low pressure and a leakage around the lower plate of the subassembly diagrid feeds the low pressure plenum.

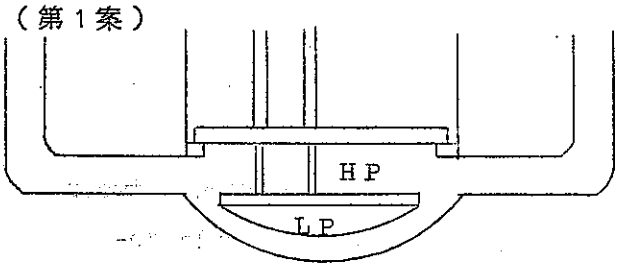
The working condition of the vessel is good, because low pressure is established in the vessel but technological difficulties rise to get a low rate for the low pressure flow.

In the second way, the vessel bottom constitutes a high pressure plenum. A low pressure plenum is obtained by using a third plate and it is fed by the subassembly feet leakages. In this design, cool sodium pipes may be connected to the bottom of the vessel as it is foreseen presently in the JEFR project. But vessel must support high pressure which is certainly admissible.

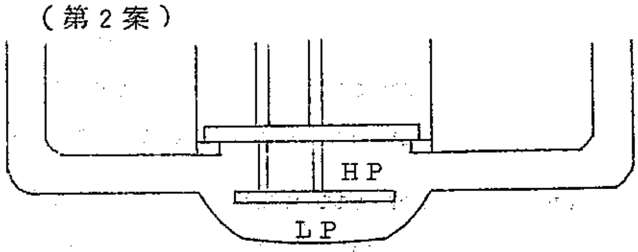
Many variations can be found for the two general designs we have described above and which are only ideas to begin with.

If an emergency sodium circuit is wanted, it can pass through the vessel at a level higher than main sodium outlets and flow down to the high pressure plenum through a pipe located in the reactor primary vessel.

(第１案)



(第２案)



① thermal shield や reflector の冷却は Naturel convection によれば十分で，forced convection は必要ない。

② melt down panel は必要ない。酸化物が melt したときには温度が非常に高いので，どんな材料でも保持できないであろう。

③ inlet nozzle での thermal shock は大きくない。

④ H.P. region は熱に対して追従が早いが，L.P. region に対しては遅い。しかし L.P. region を小さくすることによつて避けられる。

⑤ Rapsodie の Mock up test では $100^{\circ}C/1.2$ sec の実験を行なつたが何も問題はなかつた。しかし，この test は vessel の上部について行なわれたものである。

⑥ 動特性関係の側から thermal shock の数値を出したときには必ずこの値を check するように云うこと。

上の2つの案は基本的な考え方であつて，これを発展させることにより，いろいろな可能性がでてくるであろうが，要するに Vessel 下部を simplify させることは大切である。

(iv) The number of pipe nozzles is important since there are (from pp. 280 and 281):

4 nozzles for main cooling
2 nozzles for blanket cooling
1 nozzle for over-flow pipe
2 nozzles for emergency cooling
1 nozzle for drain system.

The most part is connected to the same half shell of the vessel.

Every small pipe connection has a pretty large importance, because it has to be enclosed in a double tube. As room must be left in the double tube for allowing differential expansion, the external diameter of the double tube is large enough.

Moreover, if there are fewer nozzles, reliability is increased.

For these reasons, we see another advantage to drop special sodium supply to blanket. It would also be more conservative to connect the two emergency cooling nozzles at levels higher than main cooling outlet nozzles.

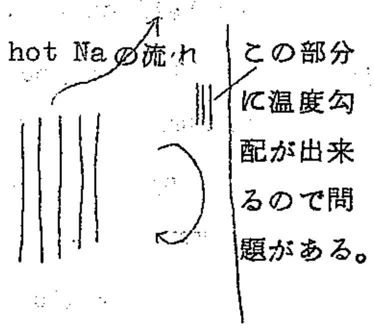
(v) Much attention must be paid to avoid parasitic sodium flow between thermal shields. In the Rapsodie reactor mock-up, we could determine that half of the thermal shields were, not only useless, but yet prejudicial because there was parasitic sodium flowing between shield plates due to holes necessary for shield plates supports. This sodium flow produces non symetrical temperature distribution and new cause of stresses. The problem was not completely solved for Rapsodie and an excess of thermal shielding was left in this reactor.

We presume that thermal shielding is useless in the lower part of the primary vessel, because temperature transients are probably very soft at sodium inlet, but this must be checked.

thermal shock の transient は Rapsodie の場合，実験の結果，-1 ~ -2°C/sec が max. で，thermal shock absorber は必要ない。たゞ Inlet nozzle から入る冷却材の上部および下部の温度差を避けるための nozzle 部分への薄い thermal shield は必要だろう。

入口ノズルに buffle を設けておくことは良いが数は少なくした方がよい。Rapsodie は少し多過ぎた。入口での thermal shock についてはこれよりも thermal stress (thermal buffle にあけた穴にかゝる) の方が大きいので問題ではない。

炉心の周囲でも右図のような局部的自然対流が生ずるので炉心上端の高さに相当する Vessel 壁には特に温度勾配ができる恐れがあるので注意する必要がある。



(vi) support plate の温度変化に対する追従性

遅れは JEFR の方が Rapsodie より大きいだろう。Rapsodie は目下計算中である。

2) Questions and answers

(i) Q. If you have any critical limit value on the radiation dose for the vessel materials, how value do you adopt for it?

A. Austenitic steel used for Rapsodie (Virgo 14 SB, batch n° 1) has been irradiated with a fluence of 5×10^{20} n/cm^2 by neutrons with an energy higher than 1 MeV at 550° C. Its ductility was reduced by 40-50% which is still very acceptable.

(ii) Q. How do you consider the effects of the sodium on the material strength?

A. In normal conditions, we do not consider the sodium effects on steel. We recently tried in our calculations of large plastic deformation to take into account ductility reduction caused by action of the carbon contained in the sodium. (these calculations were conducted for the case of very important explosive accident).

(iii) Q. Please show us the calculation method of the effectiveness on the shield plates for the thermal shock, and the comparisons with the experimental values.

A. If there is no convection between baffles, the conductibility calculations are accurate and sufficient. But the efficiency of the baffles depends very much from the design details. The forced convection between the baffles must be reduced as much as possible. This has not been sufficiently studied for Rapsodie where we merely installed more baffles than necessary. In the future, we intend to put obstacles realizing an hydraulic resistance between the baffles.

(iv) Q. How do you treat the effects of the sodium in the upper plenum area for calculating the thermal shock?

Please show us also the comparison with the experimental value if you have.

A. The sodium which is above the sodium outlet nozzles is not interested by thermal shocks, especially by cold thermal shocks during scrams. That was checked by tests in the Rapsodie vessel mock-up.

(v) Q. We think that many experiments and analysis on the transient thermal stress, the low cycle fatigue, and the creep of nozzles of Rapsodie were conducted.

Please show us the numbers and titles of the reports and magazines on these items.

A. We don't know fundamental tests in France or elsewhere concerning the thermal fatigue with long cycles during which creep could occur. We worried very much thermal shocks in Rapsodie and we installed more baffles than necessary. Experience of the vessel mock-up showed that a part of these baffles is of no use because there are pretty important flows between the baffles.

(vi) Q. Please show us the pass of the preheating gas flow in the jacket.

A. Nitrogen enters by the two sodium outlet pipes and flows out by the one sodium inlet pipe.

(vii) Q. JEFR reactor vessel will be preheated by nitrogen gas. May we have your comments on working sequences or time schedules of the sodium charging and the cold start-up (usual start-up) after the sodium was charged?

A. The Rapsodie vessels were preheated during about three days prior sodium filling. This interval of time doesn't matter very much and may be longer.

Preheating was done by nitrogen at 250° C. Some thermocouples were located in the steel main bulk (like for instance the baffles) and close to the subassemblies. The vessel was filled when the measured temperatures reached 150° C.

The reheating of the vessel with frozen fodium is not foreseen. Experience shows that it would be very dangerous owing to the sodium expansion during melting.

(viii) Q. The reactor vessel of JEFR will be enclosed by the jacket.

May we have your comments on the items of inspections and the methods of inspections after the vessel and the jacket was completed?

A. The tightness of Rapsodie primary vessel was vacuum checked by helium test prior the jacket assembling. In order to control double wall tightness space between the two vessels was over-pressured under some hundred grams and the outside leaks were detected because the double wall doesn't resist to vacuum.

(ix) Q. Please list up the technical problems on the reactor vessel at the each stage of the construction, the installation and the operation and show us the processes to deal with these problems.

A. No special difficulty was met during vessel fabrication. It can even be noted that quality of the weldings was excellent. However, we must add that the small diameter tubes connected to the primary vessel gave more difficulties than it seemed initially.

As a matter of fact, they are located inside a double wall, the diameter of which should be rather large in order to allow easy assembling and ulterior differential expansions. This leads to rather important difficulties during detailed design and construction.

(x) Q. Please give us your comments for the structure, the design criteria, etc. of the vessel in the JEFR.

A. See 4.2.5 questions and answers (2)

(xi) Q. We would like to know your experiences on the following items, and please give us your recommendations.

① support structure of reactor vessel

② deformation of reactor vessel and measuring device

③ welding, heat-treatment and inspection

A. ① The support structure of the vessel has to be cold in order to minimize any change of its own geometrical shape. Consequently this structure must be mechanically and thermally independent from the vessels.

② Thermal deformation of the reactor vessel can be due to two causes:

a) thermal assymetry in the upper part of the vessel coming from an unbalanced cooling inside the rotating plugs (this is the case in Rapsodie)

b) thermal assymetry in the vessel part under the upper shielding.

This effect was important on the Rapsodie vessel probably because there is only one inlet pipe resulting into assymetrical thermal conditions. Some additional thermal insulation was then installed outside the Rapsodie vessels, to protect them from heating by inlet sodium pipe. By that way, the horizontal displacement of the bottom of the Rapsodie vessel was reduced from 13 mm to 6 mm. This final value seems quite convenient for reactor operation. These remaining 6 mm displacement are probably due to the assymetrical cooling of the two rotating plugs. Measurements were made by optical device for horizontal displacement and by quartz rods for vertical ones.

(3) Welding procedure has to be carefully tested in the same conditions than the actual welding conditions. The Rapsodie vessel was not thermally treated after welding. Welds were 100 % controlled by radiography.

(xii) Q. May we have your comments on the sodium over flow design.

A. We found no effect of fas entrainment by overflow system. If there would be a gas entrainment in the overflow, probably gas could easily escaped in the overflow tank.

3) Information

(i) サポートプレート'

Rapsodie のサポートプレートは LE GUELLE で製作した。
オランダで作らせたのは失敗した。

4.3.3 Core upper structure

1) Comments

(i) A great attention has to be paid to the subassemblies outlet thermocouples which can give an actually good feeling of the behaviour of the fuel elements.

(ii) Core upper mechanism and core cover plate

We were told that core cover plate could not be fixed because:

a) the external guide tube of the control rod mechanisms has to be moved up and down before and after refueling time.

b) For that, control rod mechanisms are supported by core upper mechanism and follow its vertical displacements.

c) Control rod mechanisms cannot be drastically modified because a prototype was already constructed and water tested.

For Rapsodie, we designed and tested a core cover plate with specifications very similar to that of JEFR. After many troubles, we had to simplify the operation. Now the Rapsodie core coverplate is fixed and, however, the subassemblies temperature measurements are very satisfactory.

The supplementary work due to the proposed modification must be compared to eventual difficulties of reactor operation.

Rapsodie では cover plate 下端と assembly head 上端との間の距離は 8 ~ 10 cm 位離れている。もし cover plate を fix すると，control rod mechanism も fix すべきである。

(iii) It is necessary to plane tests for checking introduction of thermocouples into guide tubes supported by core cover plate.

(iv) In Rapsodie we have no finger to maintain alignment of the core cover plate which supports thermocouples. A device was only provided to adjust alignment of the core cover plate during reactor construction. This was very useful.

We think you would simplificate design by dropping these alignment fingers and their supports. But, do not forget to check vibration eventuality of free core cover plate.

If you do that, it is possible to set the core upper mechanism in a large orifice of the small rotating plug so that it can be removed without removing the small rotating plug.

Moreover, as it is noticed above, diameter of the primary vessel may be decreased if the outside dimensions of the core cover plate is decreased and the fuel rotating storage is dropped.

(v) Dip plates

We do not see why this dip plate is necessary. In Rapsodie, we could have a quiet sodium surface just with core cover plate.

In every case, you have to test the dip plate effect in a water model.

Generally speaking, we do not like to put in the reactor something which is not strictly useful.

Check also eventual vibration of the dip plate.

Hold down の下部機構が dip plate の役目をするので, dip plate は特に必要ないのではなかろうか。どうしてもつける必要があれば，上部機構の一部を Na の中に入れるようにしてはどうか。

dip plate については計算は困難であるのでその効果は mock up test によつて確かめねばならない。

2) Questions and answers

(i) Q. Please show us the structure to measure the channel output temperature, and explain us how do you decide the sodium flow for the detection areas.

A. See the questions and answers 4.3.2 - (2)

(ii) Q. By how much accuracy are these temperatures to be measured?

A. In Rapsodie, the required accuracy of measurement is of $\pm 1^{o}$ C.

(iii) Q. Please point out the problems for the structure of the temperature measuring regions in the JEFR.

A. We already said that we think that the core cover plate may occur. However, we made tests on a vessel mock-up where a stress gauge had been placed in order to make measurements if necessary. The gauge was in the argon, inside the support column bearing the core cover plate. If no vessel mock-up is foreseen, the proper frequency of the core cover plate must be determined as well as the flow excitation frequency is tested by help of a hydraulic test.

(iv) Q. Please explain us the counterplans for the vibration caused by the coolant flow on your experience.

A.

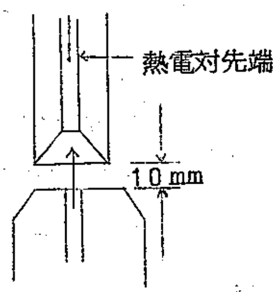
(v) Q. How should we make the design standard of thermal shock resisting plate, setting in the spider assembly of hold down mechanisms?

A. We didn't have this problem and we don't well understand this question.

3) Information

(i) In Rapsodie, thermocouple pocket thimbles are filled with inert atmosphere in the purpose of avoiding sodium oxydation in case of tube failure.

(ii) 右図のように Rapsodie の場合，集合体上面と上部機構が 10 mm 離れていれば nominal flow の条件で subassembly の actual temp の影響の 70 % 位である。（但しこれは最も悪い条件においてである）
nominal flow でないときには恐らく 50 %以下であろう。



4.3.4 Core support plate

1) Comments

(i) You have to be aware that core support plate fabrication is difficult. The specified tolerances are small and welding produces strains. Moreover, the tubes which connects the horizontal plates have to be surface treated where they are in contact with sodium.

But you have specially to pay attention to the two meters radius layer of stellite which you want to be deposited.

For that problems, choose attentively your fabrication and ask for preliminary tests.

(ii) In case of temperature transient of inlet sodium, delays are different for main core sodium flow and blanket flow. This may produce temperature difference between parts of core support plates.

We are not sure that is a problem because time constant of support plate parts may be much more larger than delay difference. But you have to check it.

(iii) In your report, you write the core support plate is designed to be removable. If you want to be really sure to be able to remove it, you have to design carefully to details of clamping. You also must make test of bolt removing in sodium.

The principle of a removable support plate has been very much debated in France. It is more conservative but it complicates design. Only the central part of the Rapsodie support is removable.

(iv) Support plate support structure

The curved form of this structure is probably to divide melted fuel in case of accident. This type of device was designed for Rapsodie as long as metallic fuel was envisaged. But if oxyde is used, fuel melting temperature is such as the curved plate would be melted too.

It is more probable that oxyde would be frozen in lower parts of subassemblies before going down to the curved plate.

2) Questions and answers

(i) Q. Please show us the criteria for the deformation of the core support plate.

A. The limiting deflection of the core support plate corresponds to the maximum permissible error of the control rod verticality.

(ii) Q. Please explain us why do you take the limitation value of 1/10,000 on the inclination at the position of control rods on Rapsodie.

A. This 1/10,000 slope value has been early imposed. It was the smallest slope that the maker could realize. Tests showed that the mechanisms of the control rod can operate in much worse conditions. On the other hand, the slope certainly has varied in Rapsodie when the vessel was distorted owing to the thermal gradients.

(iii) Q. Please show us the criteria of the fitting tolerance for the core elements.

A. As a rule, it must be the best possible to get a good positionning of the subassemblies. A lower limitation is due to a possible fretting of the subassembly nozzle by differential expansion. If radius clearance is below 3/10 mm, surface treatments are necessary in order to avoid any jamming (for example stellite deposit).

(iv) Q. Please show us how to treat the effects of corrosion, the erosion and the swelling by neutron irradiation on the fitting parts of the core elements.

A. The neutron flux is small on the bottom. It is lower than $1,2.10^{19}$ n/cm^2 for neutrons with energy higher than 1 MeV. No swelling is predicted. The erosion should not occur when the subassembly is well positionned, as there is not leak. This has been checked with water. Furthermore, the subassembly supports of the core support plate are treated with stellite, in order to obtain a harder material in the reactor. There is no reason for particular corrosion on this part.

(v) Q. How do you measure the sodium leak flow rate to the outside of the core elements (outside the wrapper tube) through the fitting parts? If you carried out some model tests for it, please show us your method to measure it?

A. Leak is measured by water tests with Reynolds analogy.

(vi) Q. How do you consider the effects of the irradiation (deterioration) on the core support plate design?

A. The fluence is small lower than 10^{20} n/cm^2 for neutrons of an energy over 1 MeV.

(vii) Q. As we have adopted the constant-inlet-temperature control system, the fluctuation of inlet temperature will be small. Therefore the effect of temperature of bottom support plate is little worth consideration. We have doubt about the calculation of temperature coefficient using perturbation method in spite of the fact that Rapsodie's temperature coefficient is large. Would you explain the calculation method in detail?

A.

(viii) Q. Complete structure of the core support grid plate is required. Drawings are preferable.

A. Design and fabrication of core support grid is dependent on manufacturer, know-how and equipment. We discussed tolerances specifications with him by taking into account the tool machines he planned to use. Detailed design is in the responsibility of manufacturer.

4.3.5 Inlet nozzle

1) Comment

(i) The fabrication of the spherical plenum at the bottom of the primary vessel is certainly difficult.

2) Questions and answers

(i) Q. How many difference are there between the calculated flow rate and control flow rate caused by the manufacturing error? Please give us your opinions based on your experiences how to estimate the error on the leak flow rate from the narrow gap between these nozzles and the support plate and how to adjust the flow rate.

A. We estimate the real flow rate through a subassembly is within 3% of the wanted flow rate which was calculated.

During tests, we measured no flow leakage at the location where subassembly is supported by the horizontal plate, but subassembly was correctly positioned on its support.

In Rapsodie, we do not have a leakage between nozzle and support plate at the nozzle bottom, because subassemblies are axially fed. This kind of leakage can be easily measured by water test.

(ii) Q. What methods do you use to prevent the erosion of the nozzle and the support plate?

A. As we wrote above, if the subassembly is in a good position, there is no leakage at the location where it is supported. From our experience, a subassembly comes in good position by itw own weight. However, it can be always imagined that some small foreign particle is located between a subassembly and its support. Also, we use stellite deposit on the supports. We do not use stellite for nozzles because there are frequently changed.

(iii) Q. Please give us your comments for the inlet nozzle and core cooling method in the JEFR.

A. We agree with the general design of sodium distribution to subassemblies.

Some work must be devoted to be sure that there is no cavitation due to restriction holes in the nozzles.

4.3.6 Cooling of the blanket

1) Comments

(i) Separate cooling pipes for blanket could be dropped

PNC representatives pointed out 5 major reasons to keep on blanket flow rate adjustment, resulting into a blanket independent sodium inlet and a more complicated design of the reactor vessel lower part.

① Water tests of vessel lower part can be dropped since it is not necessary to predetermine flow distribution between core and blanket.

② Present unknowledge of subassembly pressure drop.

③ Unknowledge of cold sodium leakages.

④ Ability to adjusting core and blanket flow rates to follow core and blanket evolutions or modifications.

⑤ Large design change non compatible with present reactor schedule. The last point has not to be discussed here. For the 4 previous points it can be answered:

(a) Water tests are not very expensive or very long. Moreover a rough calculation of flow distribution can be made at the beginning and confirmation tests can be ran during the first stages of the vessel fabrication. The orifices dimensions can be known later.

(b) Subassembly pressure drop can be fastly determined since hydraulic tests were already made on bundles.

(c) Leakages can be reduced by a simpler design and their accurate knowledge becomes less important.

(d) Core and blanket evolutions can be taken into account in hot channel factors.

Core and blanket flows can be adjusted by modifications of subassembly nozzles in case of core or blanket modifications.

We maintain our recommendations that the lower part of the primary vessel has to be redesigned.

A modification of the free internal piece of the vessel lower part was communicated to us. We do not think that a guarantee can be given for using non removable segments in sodium during reactor life. Therefore, the new design may be better than the previous one, but does not seem very satisfactory.

(ii) In the same memo DRP.S 68/7101, we expressed also some doubt about usefulness of using a special cooling of the blanket which must be regulated. This design affords possibility to continuously adjust flow rate to power generated in blanket. This procedure is not indispensable since we have always possibility of changing sodium flow distribution in Rapsodie, by changing orifices located in the lower part of the fuel and blanket subassembly.

However, we are conscious that procedure proposed for JEFR is more flexible. But this advantage is paid by two disadvantages.

① Three more pipe nozzles are necessary in the reactor primary vessel, with double tube, preheating sodium, leak detection, etc...

② If you have a special sodium supply to blanket and if you feed core subassemblies by lateral holes in their lower parts, between the two support plates, you cannot get a simple design for the lower part of the reactor primary vessel. And you can see that subassemblies are axially fed in the Fermi Reactor where blanket flow is regulated. But subassembly hold down system is a problem in this case and gave trouble to Fermi people.

On the other hand, one of your arguments in favour of flow regulation in blanket, is the possibility to compensate sodium leakage flow inside primary vessel. We are going to propose you, here after, several vessel designs limiting sodium leakage.

It seems to us that, if sodium leakage is small enough, it is better to take into account the maximum possible flow leakage in the calculation of temperature distribution and hot spot factors. If error is found to be surprisingly large after reactor start-up, it is always possible to correct it where subassemblies are changed.

In Rapsodie (and Phenix), we have a special cooling of the blanket but we use a by-pass of the core cooling, avoiding a supplementary pipe crossing the reactor vessel. We are conscious that the JEFR device permit a continuous adjustment of the core and blanket flow ratio following the change of the power distribution due to irradiation.

口頭で聞いた blanket 流量調整の欠点は下記のとおりである。

(1) 構造が複雑になりモックアップが大変である。

(2) fuel assembly の圧力損失は大して大きくない（流量調整が困難）

(3) 炉心下部の blanket 領域への leakage flow は非常に少ないと思うし正確な値をつかみ難い（流量調整が困難）

(4) 流量計の誤差が入る。

(iii) ① long irradiation 後の flow を adjust するための情報はどのようにして得ることができるか。（ブランケット領域）

② Valve で control することに，どのような長所があるか。

以上2つの問題をM Leduc より出されたが，彼の意見では blanket 領域の flow を Valve で control することには反対で，もしどうしてもつける必要があるのならばvalve の error，flow estimation の error，power estimation の error を絶対に忘れるな………とのことです。現実問題としては，これらの errorsを正しく評価することは難しい。 commercial reactor に対してはこの Type の control は場合によればよいかも知れないが， experimental reactor にはよくない。

Temp.

max. fuel temp.

max. clad temp.

max. Na temp. without blanket flow control

max. Na temp. with blanket flow control

$500^{\circ}C$ outlet

$370^{\circ}C$ inlet

もし blanket flow control をしていれば左図のように control なしに比較して max. Na temp. は多少下るかも知れないが，その差は valve の error，flow meter の error，power estimation の error などと比較すると意味のあるものが得られるか，どうか疑問に思う。

Rapsodie および phenix では blanket 領域の control はせず， blanket の highest burn up 時の power に合わせた flow を最初から流すことにする。

2) Questions and answers

(i) Q. The orifices are used in the Rapsodie, on the other hand regulating valve is used on the JFER. Please give us your comments for these control methods.

A. See "General comments"

Flow adjustment by valve is certainly more flexible than with orifices as in Rapsodie. But this system is more complicated and consequently seems less reliable.

(ii) Q. How do you estimate the flow change by the change of the fluid resistance during the fuel life? And show us how to consider the thermal balance on that time.

A. No detailed calculations were made to know the flow variations in blanket subassemblies, due to swelling of the pins. As for core subassemblies, we only expect to get more experience by actual operation of the reactor. Measurement of pins diameters are made using irradiated blanket subassemblies. Up to now, not any noticable swelling was found. If we had to take some swelling into account, the sodium flow rate would have to be some bit increased at the end of the life of subassembly. If this flow rate used at the beginning of the life of the subassembly, there would be some overcooling, and the hot point factor would be some bit increased. Of course, a flow rate adjustment by a valve could take that factor into account by proper adjustment in case of necessity, but this does not seem very necessary.

(iii) Q. Please show us the method of flow measurement for the core and the blanket on your experience.

A. For core and blanket subassemblies, pressure drops versus flow rate are measured with water models by Reynolds analogy. Temperature measurement for the sodium temperature at the outlet of each subassembly gives a broad correlation with this preliminary pressure drop measurement during reactor operation. For Rapsodie, a special subassembly with flowmeter was built and was built and was intended to be put in the central position of the core. It has never been used because we had a great confidence with the previous hydraulic results, and it was not necessary to lose time for such a measurement on the reactor.

(iv) Q. Please give us your opinions if you have any comments for selecting the materials on the JEFR.

A. See "Comments"

4.3.7 Upper plug

1) Comments

(i) Structure of rotating shield plugs

The design concerning rotating plugs shielding is yet in discussion in France.

In the Rapsodie case, it consists of a closed box constituted by the shell and two horizontal plates which are welded together. The different channels for loading machine, core upper structure, etc... are made of tubes welded on the two horizontal plates. This design is simple, but thermal expansion between top and bottom of rotating plugs.

Up to now, no trouble rose from that, but it is sure that stresses are very high at the connection between tubes and plates.

(ii) Gaps between rotating plugs and vessel shell

The rotating plugs of the Rapsodie mock-up and reactor itself, worked up to now without important trouble. Once in the reactor mock-up and once in the reactor itself, rotating plugs were jammed when temperature was lower than sodium freezing temperature. Rotating plugs operated again after heating. Therefore, we think some sodium was deposited between the two rotating plugs and the large rotating plug and the vessel shell. But this deposit does not seem to be a trouble source as long as temperature is high enough.

We are not able to generalize this result to another clearance form or dimensions, but it may be thought this result is reproductive for a design not too far from Rapsodie design. The nominal clearances of Rapsodie rotating plugs are, from 21 to 48 mm.

Evidently, the real gap dimension is varying very much around the nominal specification. Basic tests are now being started in France for a better knowledge of this problem.

(iii) Adjustment of rotating plugs position during construction

Rotating plugs must be provided with a device allowing to center them and to adjust verticality of the control rod supports.

(iv) Rotating plugs drivers

① A large conservative factor must be used in designing torque of the motor which will drive rotating plugs. Multiply at least by 2, the calculated torque.

② We do not use brakes for stopping the two rotating plugs of Rapsodie. We make only a manual adjustment of the rotating plug positions but that can be get by a low motor velocity.

(v) Liquid metal seals

The JEFR liquid metal seals present two main differences with Rapsodie ones:

① The JEFR rotating plugs are lifted up before rotation. Therefore the central blades of the liquid metal seals are vertically translated in the liquid metal. This is not done in Rapsodie.

② A cooling of liquid seal is provided in the JEFR design. There is not cooling in Rapsodie.

The good operation of the Rapsodie liquid metal seals was get after several years of test run by GAAA and CEA laboratories. It is possible that several changes in the design of procedure of these seals may result into drastic operation differences. For example, we guess that the monitoring conditions between the liquid alloy and the wall are important.

Because of this remark and the two main differences between JEFR and Rapsodie liquid seals noted above, we think it is necessary to plane a large development and test program of the JEFR liquid seal. If the Rapsodie experience was intended to be used for JEFR, design and operation procedure might be modified as little as possible.

(vi) Rotating plug cooling

The simplest cooling design must be searched. The sketches showed on pages 330 and 331 do not provide sufficient information to determine whether the construction will be easy.

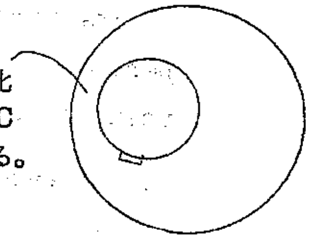
We made just a remakr: in every plug, you supply cooling gas at one inlet and total gas flows through the same circuit.

As gas is heated during its trip, a side of the plug will be at higher temperature than other side, unless gas flow rate is very large.

In Rapsodie, we use vertical coils with parallel gas flows.

Rotating plug の温度差による変形について運転中は温度差が生ずる（約５０℃）が Rapsodie では今までに変形を起すような trouble はなかつた。

反対側に比較し５０℃位高くなる。



2) Questions and answers

(i) Q. Please show us how to decide the gaps between large and small rotating plugs, between the vessel and the plug, and between the holes in the plugs and the insertion materials, etc. and show us the examples on your experiences.

A. Basic rules for sodium vapors or aerosols migration and deposits in the gaps are not fully understood. Gaps dimension and thermal gradients are surely of importance.

In Rapsodie, there is surely some sodium in the gap around the rotating plugs, as these plugs cannot rotate if they are too cold. But with some heating of the plugs, there is no problem for moving them. The gaps around the rotating plugs are from 21 to 48 mm.

It is quite obvious that these values are only means values. Surely, due to improper construction or to thermal expansions there are large deviations from these nominal values. It is necessary to provide some mean of heating for the rotating plugs. Electrical heating seems the most simple procedure. For mechanisms located inside the plugs the bottom part shall be free.

(ii) Q. How should we estimate the driving torque in rotating plugs?

How is the presumption of the rotating resistance, in case that sodium vapor condensed between plugs and the vessel wall?

A. This is largely dependent of the design of the seal of the rotating plug. Concerning the drawing torque, it is very advisable after calculation to keep a large safety margin and the make experiments to check the calculated value.

(iii) Q. Please give us the items and these procedures on the method of the inspection on the base of your experiences.

A. For inspection of the frozen seal, around the rotating plugs, we use mirror, periscopes, and fibre glass sticks (putting these sticks through some holes). The best results were obtained by periscopes putting lighting device by another hole than the one used for the periscope.

(iv) Q. Please show us the examples of the shield composition and these assembling method, and give us your comments for the JEFR design.

A. Concerning the shield composition, the JEFR documents are not very clear. It seems that there is on JEFR steel baffles at the bottom, then graphite layers in closed boxes, then borated graphite, cooling nitrogen, and finally steel. But we do not see exactly how these different parts are mounted together. In Rapsodie, all these shieldings elements are located in a stainless steel shell entirely closed on the reactor vessel side. So, there is no problem with sodium vapors inside the rotating plugs. But, thermal stresses are important, especially in the guide tubes of the mechanisms (these tubes are welded at the top and at the bottom to the upper plate and the bottom plate of the plug). Better solutions are envisaged and could probably be found.

(v) Q. Though we have used the method that all the graphite in rotating plugs are canned by using metal, is it necessary for us to do so?

A. We have no experience with no-canned graphite under sodium vapors and under radiations.

(vi) Q. Please show us the examples of the material composition of the liquid metal seal. Why do you select this composition?

A. Concerning the frozen metal seals, following criteria were taken into account:

- low melting point
- good wetting for stainless steel
- no shrinkage hole at freezing

- low expansion at melting
- low oxydation
- low cost

Pb-Bi eutectic was chosen after tests.

(vii) Q. Please show us the conditions of the following application on the liquid metal seal:

① the seal depth,

② the temperature conditions for each parts,

③ the protecting conditions (to protect the oxidization etc.,),

④ the leak value

A. The frozen seal of Rapsodie was designed, developed, and tested by GAAA. Up to now, it works with very good results, but the development test were difficult and it is possible that minor modifications could change the results. Our advice is as follows:

Either you could adopt the design and operating conditions developed by GAAA for Rapsodie, or you can make your own tests for your system taking into account that these tests could be time consuming. Comparing the Rapsodie at the JEFR system, we have to point out two differences that could be of importance:

① in JEFR, some forced gas cooling is provided. We have no such cooling.

② in JEFR, the vertical blade can be moved upwards and downwards in the liquid metal seal. It is not the case in Rapsodie system. Protection of the liquid metal seals by oil seems very suitable.

(viii) Q. Please show us the purification and the insertion methods of the liquid metal for the seal, and give us your comments to the JEFR.

A. Wetting conditions of the walls of the seal can be important. But this is not absolutely evident. The most important is to make proper and careful tests to get a good practice of the filling procedure.

(ix) Q. How do you think of the reliability about freezed seal and welded bellows?

A. The question does not seem very clear. For rotating plugs, frozen seals is very reliable.

(x) Q. Please show us the heating and the cooling methods of the liquid metal for the seal and give us your comments to the JEFR.

A. At Rapsodie, the seals are electrically heated, as it is proposed for the JEFR. For cooling, see "Comment" (v)

(xi) Q. Please show us the problems of the seal on your experiences during the operation.

A. After development and test of the seal, the main problem was the possibility of oxide formation at the surface and proper protection against oxide.

(xii) Q. Please show us which is the best method for the seal.

A. We have only to recall that in Rapsodie, the frozen metal seal is working very well up to now. Other kinds of seal are under development, but we have not yet sufficient results to propose any other system at the place of the frozen seal system.

(xiii) Q. Please show us the cooling system of the upper plug.

A. Cooling of the rotating plugs is made by nitrogen coils located in the upper part of the plug. The shape and the position of the coil are designed to give as homogeneous cooling as possible. The documents on JEFR gives only limited details on the cooling system of the rotating plugs. It seems that this problem is still under discussion and under design. We can only recommend to use a solution as simple and as flexible as possible.

(xiv) Q. Please show us the counterplans when there are some trouble with the cooling system.

A. We have no counterplans for any trouble with the cooling system of the rotating plugs.

(xv) Q. We intend to use hard facing cromium plating in the part of that it is presumed that much relative sliding happens on metalic contact faces in sodium by thermal expansion.

How do you think of it?

A. The question is not very clear and the answer could depend of the precise use of this chromium hard facing. We do not see exactly where you intend to use this chromium plating. Generally, the quality of chromium plating is largely dependent of the shape of the piece and of the quality of the execution. It is very difficult to control this quality. The measurement of the plating thickness is in itself a difficult job. We can only recommend to be very careful with chromium plating utilization. Up to now, we use this technique only for removable components.

(xvi) Q. Is it necessary to use metalic o-ring in sealing gasket used on the top face of the plug?

A. We never used metallic O'rings as gaskets on the top face of the rotating plugs. Some experiments would be needed if you plan to use them. We use only rubber O'rings on the top of the rotating plugs, as these gaskets are on cold parts of the system.

(xvii) Q. How do you think of that paint used on the top face of the plug should we have the specific conditions?

A. We use no painting of the rotating plugs.

(xviii) Q. How do you think of that oil used in bearings of plugs and oil jack should have specific conditions?

A. This problem has to be solved with makers, taking into account precise operating conditions. Generally, the only important conditions is the operating temperature.

(xix) Q. May we have your comments on the following items.

1. dip-plate
2. design of jacking up and rotating mechanism.

A. We do not use a dip plate and this system does not seem necessary. If you use it, it would be very suitable to make tests on hydraulic model to study sodium flow and vibrations of the mechanical devices. We have no specific remakrs, but the drawings are not very detailed.

(xx) Q. Shield Material

① Canned graphite and canned boron-graphite are to be used as the shield in a rotating plug of JEFR.

We understand that some gases are formed by heat and nuclear radiation in the above materials.

We would like to know whether the volume of the formed gases is considerably large and special caution is paid to the can structure for releasing gases or not.

② We would like to know your opinion regarding formed gas volume by neutron radiation at rather low temperature such as a range between 150°C and 50°C.

A.

(xxi) Q. Packing

What kind of material is used as packing material in the rotating plug for Rapsodie?

A.

(xxii) Q. Freezed Seal

We would like to know the following matters for Rapsodie plant and your opinions based on the experience of Rapsodie.

① Freezed metal (We hear Bi-Sn eutectic alloy is used in Rapsodie)

② Leakage of gas through "Freezed Seal" when melted.

③ Temperature range of melted freezed seal under the operation of the rotating plug and the max. allowable temperature of melted freezed seal.

④ Use and thickness of silcon oil on the surface of freezed metal to prevent oxidation.

⑤ Necessity of the exchange of freezed metal and method of the exchange if that is the case.

A.

(xxiii) Q. Type of the connector for thermocouple.
(We have a plan to use Canon Plug for C.A thermocouple)

A.

(xxiv) Q. System of power supply and signal take-out to and from double rotating shield plugs.

We are designing to hang down the cables from cable towers for power supply to the equipments on the rotating plugs and for the taking-out of control and measurement cables.

But it needs to use high cable towers for getting full flexibility of large bundles of cables.

And the Cable towers hang just over the rotating plugs. It makes very difficult the access and maintenance of fuel handling equipments.

Please give us some coments on these subject.

A.　We have no experience of cable towers.

We use support wheels and got no trouble.

(xxv)　Q.　① Please show us the leakage flow rate at seal of main pumps in Rapsodie.

② Please show us the design values and actual values at normal operations of leakage flows of argon gas at rotating plug, at freeze-seal and at door valve of fuel exchanger, etc... and the method for preventing these leakage in Rapsodie.

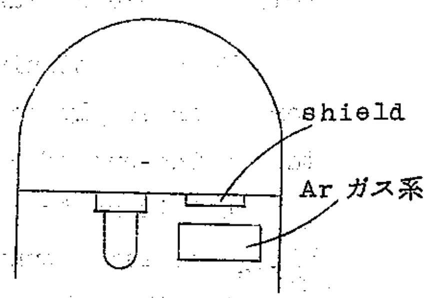
N.B.: On EBR-II the leakage value is 580 cc/day and will adopt that it will be 500 cc/day on JFER. But we have no reason for this value.

A.　Rapsodie では0である。
actual value も0である…………これは over pressure test で確かめた。

There is no measurable leakage of argon gas, when the reactor is in operation, at rotating plugs, freeze seal, pump seal.

Ar のリークの大きいのは書かれている部分よりも Ar 系の pipes や valves あるいは FFD 関係のラインからであろう。

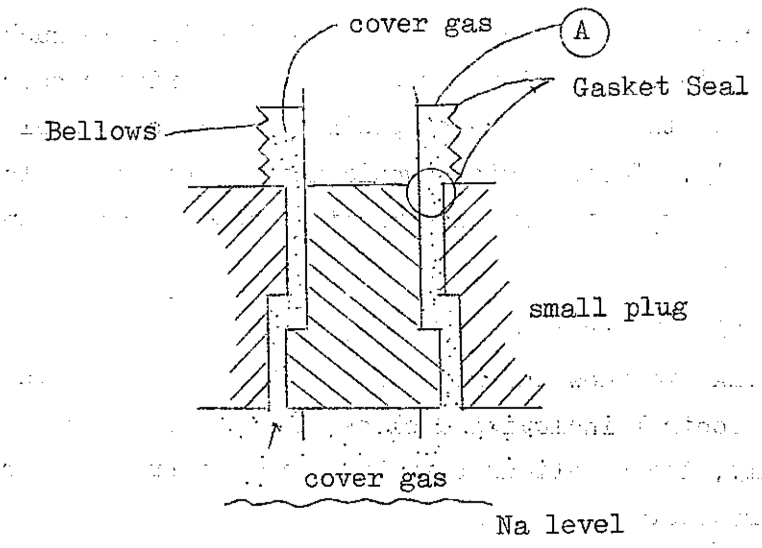
Rapsodie では Hazard report については pin rupture の仮定はとり入れていないが実際には現在 clad が破れリークが起つているが Ar ガス系の上に shield を置いて放射能を防止している。こうすれば activity は運転中上昇するが運転停止後数秒間で急下降する。



燃料棒破損はどの位起つてもよいと考えるかの質問に対しては誰も推定できないとのことであつた。

Rapsodie では active な Ar ガスは全部1次系に戻し close された circuit 内で循環させている。

(xxvi) Q. Present design is shown schematicaly in the following drawing.



The gas in the bellows is the same gas which covers sodium level of the reactor.

The bellows are fixed upper and lower end (flange) to the flange of the core upper mechanism and the upper shield plate of the small plug respectively.

The gas is tightend by the gasket seals which are inserted both upper and lower flange connection. (see above figure)

Please point out if there are some problems.

We consider if necessary, some seal elements should be attached at point A in the above figure, and fresh argon should be filled in that space.

Please give us your comment.

A. We have no definite answer for this question which is yet studied in France. Two phenomena are considered:

- The first one is a possible sodium deposit into the upper part of the gap. Natural convection would transfer Na aerosols up to the colder part of the gap, and, there, sodium would deposit. Sodium transfer is certainly function of gap thickness and temperature gradient. We intend to develop a program to solve this problem.

- The second phenomenon is an eventual fission gas transportation, accelerated by natural convection and resulting into a gamma activity above the operation level of the reactor. It can be difficulty solved except on reactor itself. In such a position, the best solution consists probably on designing conservative procedure as those you are proposing: additional seal and fresh argon supply. These devices would not be put on reactor as soon as criticality, but provision might be made to add them if they become necessary.

(xxvii) Q. Rotation Control of Plug

We like to know the rotating angle detection method and the rotation control including a block diagram of control circuit. Especially, the relations between control room and operation site is requested.

A. We designed and put an automatic angle detection of rotating plus, using a magnetic device. This system did not work pretty well and now we use only a direct reading of plugs positions by an operator remaining close to rotating plugs during refueling.

(xxviii) Q. Deflection of Plug

We like to know the calculation results on deflection of plugs and the experimental results. Have you ever done the measurement of thermal expansion?

A. We guess you mean thermal deflection of rotating plugs.

All the design of rotating plug was conducted so that supports of control rod mechanism are in a cold part where thermal deflection is small.

Thermal deflections of the other parts of rotating plugs were estimated from calculated temperature distributions. There are some thermocouples in the rotating plugs and, as we told you, we saw that temperature distribution is not symetrical in a horizontal cross section. Temperature distribution estimations must be causiously used because geometrical forms are not simple and heat transport phenomena in gaps are not basically known up to now.

We have not solved the problem and we are worrying that only rough pessimistic approximations can be made.

3) Informations

(i) Liquid metal seal

① Bi-Sn eutectic alloy is used in Rapsodie.

② Leakage of gas is zero when seal alloy is melted.

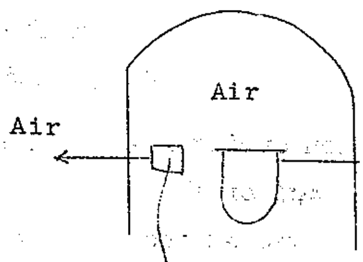
③ The temperature range of melted alloy is about 170-180°C during refueling time
No maximum allowable temperature is specified.

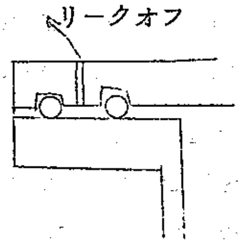
④ Silicon oils are good protection against oxydation but must be carefully tested.

⑤ It is very useful to design possible exchange of Bi-Sn alloy.

(ii) プラグ上面のガス漏洩率

Rapsodie では漏れは0である。上面の activity は小さく，ガス漏洩との関連はないとしている。コンテナのベンチレーションはガス漏洩を考えていない。アルゴン系の雰囲気は空気で減圧している。（下図）
制御棒その他の固定フランジのシールはOリングを2重にして中間よりリークオフをとつている。（F図）





4.3.8 Core barrel structure

1) Comments

(i) Space between thermal baffles of the reactor primary vessel

In the case where thermal baffles are used to protect vessel against thermal chocks, if distance between plates is large, natural convection by small loops is possible and conservatively, only steel thickness must be contemplated in calculations.
Layer gaps result into an easier heat transfer by natural convection than smaller, but are easier for construction.

As we are not able to calculate accurately the play of natural convection, we recommend to make this conservative hypothesis.

All this reasoning is true only if parasitic forced convection was carried out.

We do not see why you have divided layer of shield in three pieces.

(ii) Core barrel structure の酸素および酸化物の除去について

Na を充填し1度ドレンすれば十分除去できるが実際は自然循環で循環しているので必ずしもドレンする必要はない。よどんだ部分は hole を設けておいた方がよい。

4.3.9 Labyrinth structure

1) Comments

(i) Laybrinth structure has stellite deposits on lateral and lower faces. The fabrication will be certainly difficult.

(ii) The internal piece of the labyrinth structure seems to be free. The setting up will be difficult and has to be carefully studied.

(iii) If the outside piece gets some deformation during reactor operation, is it possible to assure that the lower part of the internal piece will always be in contact with sphere located at the primary vessel bottom?

In every case, contact will be loose and cavitation and erosion must be feared.

(iv) You have to check eventually of vibration of internal free piece.

(v) ラビリンス

原案，東芝案ともフランス側は反対，製作は可能であろうが，問題はどうして20年間をgarantee するかである。また修理も困難であろう。

4.3.10 Control and safety rod drive mechanism and dash pot

1) Comments

(i) We have never tested a control rod and its driver mechanism, where rod is released during scram. But we made a conceptual design of this type of rod and mechanism, and pointed out four main difficulties:

① The reliability of the rod scram must be assured. Therefore, the problem of correct guiding of the rod in its hexagonal wrapper is important. For that, a pretty large clearances must be provided between rod and wrapper.

② The spring located between hexagonal wrapper can and rod, must have constant characteristics. We have no irradiation experience of behaviour of nickel alloys under neutron flux in France, but, from, foreign information, we know that they become brittle.

③ The rod must be reliably released by the gripper in case of scram. The gripper design must be tested and improved as soon as possible. The JEFR conceptual design seems good, but only test may confirm this guess.

④ Control rod must be provided of a sodium dash pot. Calculation of a dash pot is always approximative. Water tests must be run.

(ii) Hexagonal wrapper guide must be clamped in the subassembly support plate, but it must be removable. We did not understand how that is made in JEFR.

In Rapsodie, a "baionnette" system is used, but the six surrounding subassemblies must be removed before the hexagonal guide.

(iii) Control rod and mechanism must not be rigid when they are connected so that some relative displacement of control rod guide and mechanism support is possible.

(iv) We did not see the detailed drawings of control rod mechanisms. But it appears to us that their design is probably difficult because of room necessary for irradiation loop and because of the complex design of the core upper mechanism.

From our experience, very much attention must be paid to conditions of removing and dismantling of control rod mechanisms.

2) Questions and answers

(i) Q. Please show us the problems which had been taken place on the way to the development.

A. Our main recommendations are the following:

① as far as possible, keep the mechanisms in the cold region

② as far as possible, use lubricated pieces in the argon region.

That is made on Rapsodie, but our mechanisms are working in fairly cold regions.

③ use clearance as large as possible

④ If some bellows are used, they must work in compression (no extension) during fast drop (scram)

(ii) Q. How much degree of the maximum deviation do you tolerate between the core center and the upper plug center for installing the drive mechanisms in the Rapsodie design?

A. In Rapsodie, the misalignment between the axis of the channel of the control rod mechanism and the axis of the top of the guide tube of the rod is $\leqslant 3$ mm at room temperature.

(iii) Q. Please show us the counterplans necessary on the structure design for the deviation of the position above-mentioned, and show us the relation between that deviation and the inclination of the core support plate of 1/10,000.

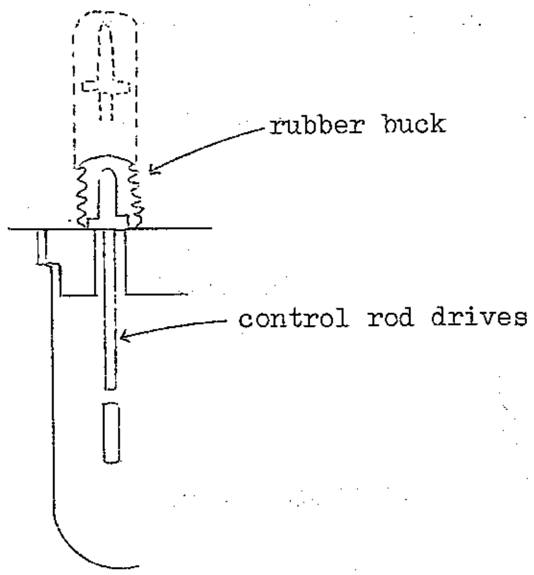
A. See 4.3.6 questions and answers (2)

(iv) Q. Please give us your comments for the JEFR design.

A. See "Comments"

(v) Q. The removal machine of control rod drive mechanism is not considered yet. Please give us information about it.

But we can consider some rubber buck which envelopes whole of drive mechanism.



A. We have no experience of this type of rubber buck, but we see no reason not to use it.

Do not forget, however, that the lower part of control rod mechanism is active, especially if stellite, or another cobalt containing alloy, is used to harden the gripper fingers. A gamma shield must be put between gripper and operation or operation must be remote.

Up to now, we use closed and shielded coffin for removing anything from reacror.

(vi) Q. How much value is the acceleration of spring for Safety rod insertion at scram, except gravity?

A. Spring is decompressed within 66 mm.
Max. acceleration は 4 g で 300 m sec で scramする。400m secなら3gでよい。

Rapsodieでは springは mechanism 上部の Arガス雰囲気中 (20°C) にあるが, JFER design では Na雰囲気中にあるので多少問題があるかも知れない。Enrico Fermi 炉の場合も Na中にあるが, 比較的うまく動いている。しかし, 実際に使用する迄に多数回の Test が必要であろう。

(vii) Q. Shock absorber for dropping of safety rod is dash pot only, isn't it?

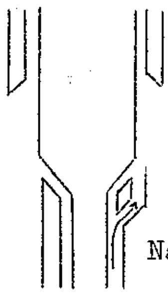
Is Spring used for it:

A. dash pot は mechanism 上部の Gas 雰囲気中に 1 つあるだけで，下部にはない。一般には 2 つの方法がある。 i) drop rod mechanism をもうける場合は mechanism 中に dash pot があり， ii) drop rod のみの場合は rod 中に dash pot がある。 Spring は使つていない。

(viii) Q. How does Na discharge from safety rod channel when safety rod drops:

Does it have special mechanism for Na outflow from the channel?

A.



Rapsodie の dash pot は左図のとおり
The safety rod is designed so that dash effect is small

Na の沈げを設けている。

(ix) Q. Relative motion between control rod and vessel

As you have pointed out, this will probably occur during reactor power up. But, for instance, if power was increased with a constant period, it would be automatically compensated by control rods.

But the relative motion would be zero during rated full power operation and it would take pretty long time for vessel to expand and contract in the case of accident. Therefore, there is no necessity, we think, to consider the relative motion to transient analysis during rated power operation.

What do you think of it?

A.

(x) Q. The upper guide tube is sealed by a metal "O" ring (type 316 stainless steel or Inconel X) or alminium sheet packing at the lower surface of the bolted flange of the control rod drive mechanism housing (fig. 1).

Please give us your comments.

A. We use only rubber O'ring because temperature is low in such a case.

Metallic O'ring can be used, but it is possible that surface quality must be very good.

(xi) Q. It is conceivable that sodium vapor deposits in the annular cavity between upper guide tube and inside surface of the hole of hold down plate. We are afraid that the guide tube might be stucked and beoome difficult to withdraw for maintenance because of sodium deposition.

Does any consideration have to be taken to prevent the guide tube from stucking, for example, heating or argon gas blowing? (fig. 2)

A. There is sodium deposit between control rod drive mechanisms and their guide tubes in the small rotating plug of Rapsodie. Some supplémentary force was used to remove control rod mechanisms after a special preheating. Clearance was about the same as that you intend to have in JEFR between mechanisms and guide tubes in the small rotating.

We have now a tendency to increase this clearance, so as that located between the gripper command tube and the gripper support tube.

(xii) Q. The outer extension shaft is sealed by graphite or alminium-bronze grand packing and its stroke is 900 mm.

The flow rate of argon sealant gas is 160 cc/min.

Is there any problem for such design? (fig. 2).

A. Why don't you use bellows since your mechanisms are not working during scram?

4.3.11 Hold down device

1) Comment

(i) hold down 機構を fix するようにすゝめる。

(ii) hold down 機構の押し拡げ方式はやめた方がよい。

hold down を押し拡げ式をやめ上部押え式にした場合右図のように周囲 6 assemblies のうちの 2 本が内側に傾く可能性があるが周囲の assemblies は flexibility をもつているので問題はない。

(iii) hold down を炉内に入れる前に下記の状態でテストをやる必要がある。

① cold
② room temperature
③ hot nitrogen
④ hot sodium

2) Questions and answers

(i) Q. Please give us your comments for the hold down system in the JEFR.

A. The hydraulic locking system adopted for the JEFR is good because simple. Furthermore, it leads to a subassembly sodium supply by lateral holes in the bottom of the subassembly, which reduces the pluggings hazard of the core subassemblies.

(ii) Q. Please give us your comments for the hold down systems on your experience.

A. In Rapsodie, we use clamped subassembly feet which operate satisfactorily up to now, but its development was difficult: some spring blades are provided to get a supplementary force against lifting.

5. 燃料交換系統

5.1 General comments

(i) Transfer machine (from core to internal storage) and loading-unloading machine could be made more rustic

We understand the advantages you see to remove the spent subassemblies from the reactor as soon as possible after irradiation. But we think that this principle leads to many important difficulties:

- you have to unload the subassemblies from reactor to storage during reactor operation. Consequently, sodium is hot and very active. We have no experience of subassembly loading in such conditions. A very extensive experimental program has to be developed.

From our tests and Rapsodie experience we guess that sodium deposit will be a hard problem at high temperature, and we disagree to get this problem for a first reactor.

- The subassembly will be hot and its cooling will be more difficult. You have to cool the subassembly pot during its transportation. The flow diagram of the argon cooling system is very complex.

Such a cooling system was designed and constructed in the Rapsodie loading machine. It could never be used.

In conclusion, we think that it is better to wait more to remove the spent subassemblies and to avoid any cooling system.

この種の design は世界中でどこも経験したことはなく，最初の高速炉の経験にもちこむことは数多くの trouble をもたらすであろう。

Rapsodie では燃料輸送中 cooling はしない。 storage rack で冷やした後に行なう。

(ii) We do not think that the internal storage is useful. We propose simplifications for the fuel exchanger. We suggest a modification of the cooling system of the fuel transfer machine.

But note that the drawings which were joined to the consultation documents do not permit us to make remarks on details of the machines, such as guidings, bearings, shaft supports, gaps and tolerances and so on.

(iii) The fuel loading system uses tubes diving in the reactor. If pressure is accidently increased in the reactor and sealing at the tube top has a failure, primary sodium raises through the upper shielding and flows on the operation levels.

In the purpose of avoiding such an accident, holes have to be bored in the tubes above the maximum sodium level, in the argon cover gas part. Thus, there is always pressure equilibrium inside and outside the tubes.

2) Question and answer

(i) Q. You mentioned that the breakage of the fuel assemblies would be more likely during transportation than in reactor.

Would you please tell us the causes of the damages and in what part those damages would likely to occur?

A. Breakage eventuality of fuel subassemblies

When fuel subassemblies are located on the subassembly support plate, no breakage cause can be found.

During a subassembly transportation, it is always possible to envisage that a gripper may accidentally open and release the subassembly. This accident occurred several times during transportation of new assemblies in the fuel loading building of Rapsodie.

Moreover, once a dummy subassembly was released from the fuel loading machine located above reactor mock-up and fell down to subassembly support plate. Hexagonal can was strained but subassembly could be removed by removing first the neighbouring subassemblies. The dummy fuel and blanket bundles were without damage.

From our experience, we think there is no hazard in case of a subassembly fall during transportation because the hexagonal wrapper is a good protection for the bundles. But the subassembly may get some deformation and it is more prudent not to use it in the reactor.

3) Informations

(i) 燃料交換系

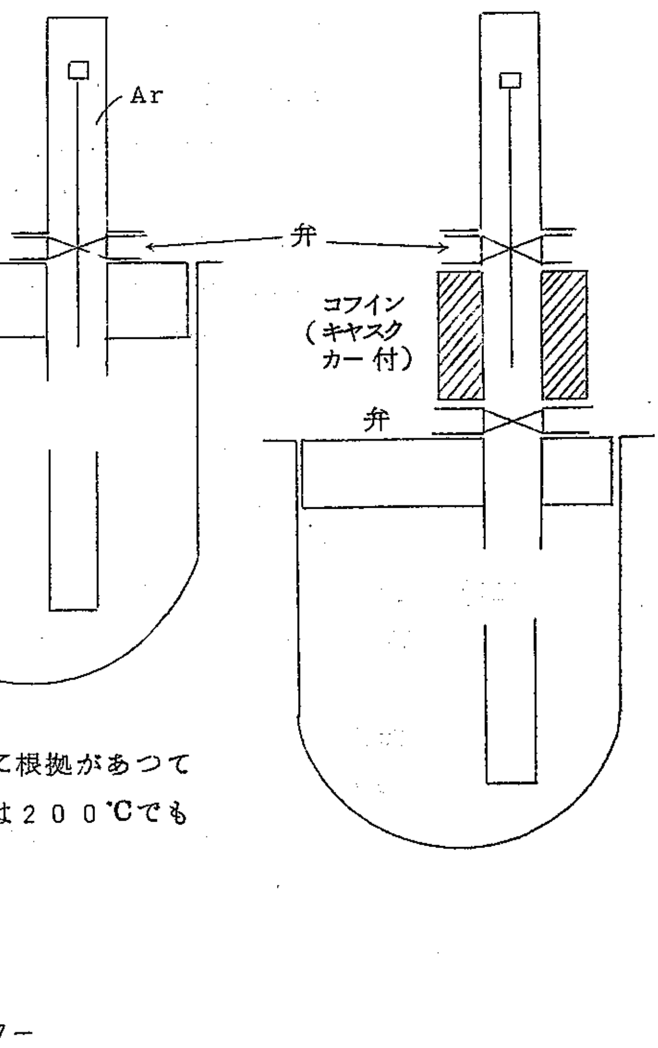
① Rapsodie の燃料取替について

a) Vessel 内の炉心の周囲に12 subassemblies まで貯蔵できる炉内貯蔵施設がある。

b) これを中継とし炉心と炉内貯蔵施設の間，炉内貯蔵施設と炉外の間を別々に交換する。

c) 炉心と炉内貯蔵施設の間の交換は，以前はシール，冷却，運転室，自動操作機能の付いた交換機を使用していたが，故障率が高く運転費も高いので簡素な別の交換機を製作し下図のように小回転プラグ上に置き燃料交換を行なっている。Na 中なので冷却の必要はない。
seal には metal ring （材質，cast iron）を使用。
新交換機は Ateliers et Chantiers de Bretogne で製作。

d) 炉内からの使用済燃料の取出しおよび新燃料の装入は古交換機を使用することもあるが右図のように輸送用コフィンを用いその上に新交換機を置いてコフィンと炉内貯蔵施設の間の交換を行う。
３５日冷却後の１ subassembly 当りの熱出力は約４００w なので特に冷却は必要ない（ＪＥＦＲの見積は大き過ぎる）



e) 燃料取替時のＮａ温度は１５０℃である。これは特に根拠があつて決めた温度なのでＪＥＦＲでは２００℃でもよいだろう。

② 運転中の燃料交換機の取扱いについて

Rapsodie では取外しておく。またつかみ部をアルコール洗滌するようになつている。

(ii) Rapsodie ではステライトの bearingに対しても test を行なつた。

5.2 Fuel exchange machine

1) Comments

(i) It seems that PNC has not determinated whether the fuel exchanger must remain in reactor vessel during operation. In such a case, the lower part of the fuel exchanger will be actived, specially if stellite is used, and special shielding has to be provided during fuel exchanger removing for maintenance.

In every case, the fuel exchanger removing system has to be designed in the same time as the machine itself. By this way, some changes in the fuel exchange may be convenient to reduce the dimensions of the part which has to be transported in special conditions.

(ii) The actions (from the actuators to grippers, hold down device, sensing device, etc...) are transmitted by rigid tubes and bars. We agree with this general concept.

Numerous coaxial tubes are used to transmit the different movements. We are not able to make valuable remarks on details because the drawings are too general, but this may be a source of difficulties.

駆動が直線なのはよい。

フランスではギャップのとり方ガイドなどでかなり trouble を経験している。

(iii) Cooling and heating system is provided for the fuel exchanger. We do not understand the usefulness of the cooling.

Heating must be provided in case of sodium deposite between fuel exchanger and its channel, but an electrical heating is more convenient.

Rapsodie では直径約３００mmのプラグでも冷却していない。

(iv) When the subassembly foot is inserted into its hole in the support plate, its velocity is 10 mm/sec in JEFR. We use 1 mm/sec. in Rapsodie. This last velocity is too low, but leaves the contact pieces in very good condition. You must make a test to see if it is the same for 10 mm/sec. velocity.

Rapsodie では 5 mm/sec 位に変える考えである。この速度は安全性とも関係があるので機械屋だけでは決められない。

(v) The fuel exchanger has a hold down system to keep in good location the 6 subassemblies surrounding the extracted subassembly. There is such a system in Rapsodie and it gave many troubles. From our experience, we can make the following recommandation to you.:

- It does not seem useful to push away the head of the six subassemblies when the hold down system is acting. The force transmitted by the pads located on the subassembly hexagonal can are very low.

  Moreover, if it is pulsed on the heads by inclined surface, there is some hazard to hasp a head by an angular part of the hold down system.
- It is absolutely necessary to be sure to get a good force limitation when the hold down system is pushing on the heads. If it is not, the subassembly heads can bend.
- It may be more convenient, not to push on the head, but to have just the lower face of the hold down system at some very small distance of the subassembly head. Then, if a subassembly is lifted in the same time as the extracted subassembly, it is stopped and it falls again after extraction of the central subassembly.

  By this way, any damage possibility of the subassembly heads is avoided.

(vi) Core element orientation may be dropped, if special cams are provided on the upper and lower parts of the subassemblies.

But, if you want to maintain it, it is relatively easy with your design, since you use two rotating plugs and a vertical fuel exchanger.

(vii) The advantage of using a sensing device does not seem clear to us.

In the JEFR design, the sensing device has two plays:

- It must reknow whether the subassembly head is in good position before gripping it,
- It must indicate that the head is in good position when the gripper released it.

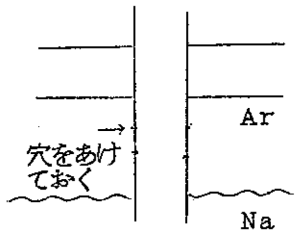
For the first play, the gripper can be used. It must have two "fastened positions", one with a subassembly head in the gripper, another without a subassembly head. This last position gives a different indication by a special switch.

For the second play, the gripper can be raised a little after having released the subassembly head (by 10 cm for example). Then the gripper is fastened and if it is in position "fastened without subassembly head", it is sure that the subassembly was left in right position in the reactor.

(viii) fuel assembly をつかむ機構が非常に複雑なように思う。

(ix) 取替機の制限重量を２００kg にしたのは悪くないだろう。
Rapsodie では 1 assembly の２倍で設計した。

(x) machine の答を通つて Naが上つてくる恐れがあるので途中に穴をあけておく必要がある。
Rapsodie の mock up test では２回このような事故を経験した。



2) Questions and answers

(i) Q. Please give us your comments on the following items.

① Due to the difference of the thermal expansion, some deviation will be arisen between the fuel exchanger axis and the center line of fuel channel. In this design, this deviation is adjusted by using the location control system which contains a programms of the thermal expansion compensation.

② We consider it is necessary to provide a door valve between a rotating plug and a fuel exchanger so that it should seal the argon gas and the radition from the core during removing and fitting operation for maintenance.

③ As the maintenance and inspection of the activated fuel exchanger is extremely difficult, we consider it is better to remove the machine from the reactor when the reactor is under operation as that of the PFR in the U.K.

A. Effectively in many cases some deviations would exist between the axis of the transfer machine and the axis of the fuel subassembly to be discharged. These deviations could come from different causes as explained in the following lines. Some of the deviations can be known by measurement or calculation, and the deviations due to thermal expansions, etc... This kind of deviations can be taken into account for proper calculation of the rotation of the shielding plugs corresponding to every subassembly location. Another kinds of deviation are of the random type. They can be due to some inaccuracy in the position of the rotating plugs. They can also be due to some bowing in the fuel subassemblies, or to some asymetrical thermal gradients in the transfer machine, etc...

In any case, this problem is an important point for proper design of the fuel transfer machine. That means that the gripper system should be able to work with some angular deviation and some misalignment of the axis of the fuel subassembly compared to the axis of the gripper.

(ii) Q. On the determination of the position of the fuel exchanger by plug.

① How much is the accuracy of the positioning in your experiences?

Please show us the design value and its actual value, and kindly show us your design philosophy.

② How about the confirmation of the fuel insertion?

A. A door valve is necessary for removing the transfer machine. In Rapsodie, the transfer machine is always removed from the reactor after loading-unloading operations. In Phenix, the transfer arm is designed to be kept inside the reactor vessel during reactor operation. But, it is yet in discussion although

Phenix is provided with an internal neutron shielding. The maintenance of an irradiated machine would be a difficult job. So, our advice is that it would be better to remove the transfer machine of the JEFR during reactor operation, at least for the first years of running.

Provided careful measurements and proper corrections are made after construction, the positionning of the axis of the transfer machine at the level of the rotating plug could be obtained with an accuracy of the order of 1 mm. For any loading-unloading operation, the unknown deviations between the gripper and the subassembly can be essentially due to:

- the uncertainty of the actual position of the top of the subassembly (gaps, bowing, etc...)
- the thermal deformations on the transfer machine.

Finally, concerning the transfer machine and especially the gripper of this machine, it would be quite unsuitable to look for a better positionning than could be the positionning of the top of the subassemblies. The only solution is to design the machine taking this fact into account.

Concerning the level of the top of a loaded subassembly, this level can be measured within a margin of a few millimeters by putting the gripper on the top of the subassembly and comparing this position to any reference.

(iii) Q. On the moving components in sodium,

(1) Please show us the data which were obtained by testing the operation of gears, cams, bearings, etc., in sodium, if any.

(2) Please show us the data which can be used in the design of the moving components of core fuel storage rack in JEFR, if any.

A. From our experience with mechanisms working in sodium, we can express some important general laws, as following:

- the design of the mechanisms has to be as simple as possible.
- large gaps are necessary for moving pieces, working in sodium.
- the results are generally better with mechanisms working in sodium than with mechanisms working in argon with sodium vapors.

- for rotating or sliding pieces with small gaps (bearing for example) some proper plating is necessary. In that case, materials like stellites give very good results.

(iv) Q. ① What is your design criteria?

② What is your fundamental phylosophy for determination of the structure and for design criteria?

And, how about your procedure of a fuel exchange and the time used in the fuel exchange?

③ What is your items and methods for development tests of the mechanisms?

④ Please show us your problems which appeared during design, construction and operation, and these resolution methods.

A. The main criteria for a design are as following:

a) simplicity is necessary for the mechanisms in sodium. This simplicity can be obtained on the one hand by avoiding any functions which are not absolutely necessary, and in other hand by using separate machines for the functions which are not to be made together.

b) for any machine, the working parts and the control system have to be designed as far as possible to work outside of the sodium and outside of the argon. Large clearances are very important.

c) all the mechanisms are to be removable for maintenance.

After construction long testing is necessary, first in air, then in hot argon, and finally in sodium in as actual conditions as possible.

For Rapsodie, the preliminary tests showed clearly that the machine was very complicated. The main simplifications consisted to not make use of the forced argon cooling and to change the automatic control into a manual control. Moreover, a new machine as simple as possible was built, firstly as a spare machine. But this machine will be used in a next future as the normal machine.

(v) Q. Please point out the problems about the mechanisms of the JEFR fuel exchanger from your experiences.

A. It is difficult to give a precise advice due to some lack of detailed drawings. It seems however that this machine could probably be made simpler. For example, the finger indicating that a top of a subassembly was inserted inside the gripper could probably be dropped. We can recall that the detailed design of our fuel handling machines was made by our makers. These makers would be in better position for a complete examination of detailed drawings.

(vi) Q. ① Please point out the problems about the following items on JEFR fuel exchanger,

a) axial seal
b) bellows
c) behaviour of sodium in narrow distance
d) pre-heating

② On the shaft seal

a) Please show us its backdata, if any.
b) How about your countermeasure for adhesion of sodium vapour?

A. Sealing problems can be solved by two rubber seals with argon injection between the two seals, provided that this sealing system is in a cold part of the mechanisms. Bellows are better located in sodium than in argon. If any bellow is located in the argon part of the reactor vessel, sodium deposits would occur, turning into some sodium oxide which could enhance bellows operation. So, it is generally important that a reheating would be possible in the colder parts, where some sodium vapors deposits can occur.

(vii) Q. Seal on Hold Down Shaft

In our design, the sealing mechanism on the hold down shaft of a refueling machine in composed of a O-ring, two mould packings and a retainer as shown in attached sheet (Ref. JEFR FBDO-10084).

The material to be used for O-ring and mould packing are as follows.

O-ring	:	Teflon (or silicon rubber)
Mould packing	:	Asbestos (or Teflon)

During refueling operation, the hold down shaft immersed in liquid sodium is in up and down motion and comesin contact with air after passing through the seal mechanism.

We are afraid whether there is a possibility of sticking at the seal by deposit sodium on the surface of the shaft and damage to the sealing material as a result.

We would like to know your view on this matter and to have your recommendation on other suitable sealing method, if any.

A. We have a seal of this kind used in same conditions on the new simple refueling machine used for Rapsodie. We got no trouble with sodium transportation by the shaft.

However, Rapsodie sodium is not yet contaminated by solid fission products and adsorption of these fission products on shaft surface can be imagined.

A seal of this type is also used on the Rapsodie first loading machine.

Detail design of such seals were made by private companies (Hispano-Suiza and A.C.B.)

(viii) Q. Seal Gas Flow between Hold Down Shaft and Pre-heating Sleeve

We think, it is not necessary to make the flow of seal gas in the gap between the hold down shaft and the pre-heating sleeve during reactor operation, if there is enough capacity of electric heater in the preheating sleeve to melt deposit sodium on the surface of the shaft.

Please let us know your view on this matter.

A. Seal gas flow between hold down shaft and preheating sleeve.

We agree with you. But gas supply at the top of pre-heating is necessary in case of accidental tightness failure of the seal.

On other hand, it is useful to carefully discuss advantages and disadvantages bound to removing of refueling machine after every refueling time.

3) Information

(i) Rapsodieの simple な fuel handling machine について

① 従来のmachineは非常に複雑であり，従来のものがtrouble を起したときのback up 系として製作した。

② simple で冷却もないが shield に問題がある。

③ しかし徐々に考え方が変つてきて fuel exchange にこの machine を使用し loading-unloading に従来のを使用する考えである。 将来は loading unloading にも新しい simple なものを開発し，2つの simple な machine を使用することになろう。

④ 従来の machine の運転費1年分でこの machine 1機製作可能。

5.3 Fuel loading-unloading machine

1) Comments

(i) The most important remark concern the cooling system of the fuel transfer machine. You have designed a direct cooling of the sodium pot by argon either in the coffin or in the reactor loading channel. This design leads you to have a complicated argon gas system with the following disadvantages:

- argon is active and the argon loop has to be tight
- sodium moisture separator must be used to eliminate sodium vapors and aerosols before the blowers.
- argon cooler and heater are necessary.

We propose you a simpler concept, as you see on joined figure.

The fuel loading channel is in active argon since it is in connection with reactor. But there is no forced convection of argon gas. The sodium pot, carrying the fuel subassembly is cooled by radiation and some argon natural convection. It exchanges its heat with an air cooling loop, surrounding the tube which closes the argon atmosphere.

Another air cooling may be provided around the reactor loading channel in the rotating plug. It is probably not useful during normal operation and has to be used only in case of failure of the driving system, when the sodium pot is stopped in the channel.

The air loop may be opened, but air must be heated at about 150°C before entering the coffin, in the purpose of avoiding sodium freezing inside coffin.

Our proposition is based on our experience of sodium pots (see 5.2 Fuel exchange machine)

(ii) Your design of two rotating coffins is not bad. However, it seems to us that it is possible to have only one coffin, transporting either the channel plugs or the subassemblies. By this way, you can make an economy.

(iii) In every case, you can transport the coffins by using the reactor building crane and you increase the flexibility of your system. You avoid to take permanently room for the rotating system.

In your design, you have to take into account, the maintenance of the fuel transfer machine and, mainly, decontamination in case of reparation. It may be easier to decontaminate a coffin transported by crane in every part of the building.

(iv) The figures which we have, do not permit us to make valuable detailed remarks. However:

a) it seems that the grippers are heavy and important especially the plug gripper;

b) the shielding of the upper part of the reactor loading channel may be a problem when the plugs were taken out and the full transfer machine is not located above the channel.

(v) If you use such a machine to extract fuel subassemblies at high temperature during reactor operation,

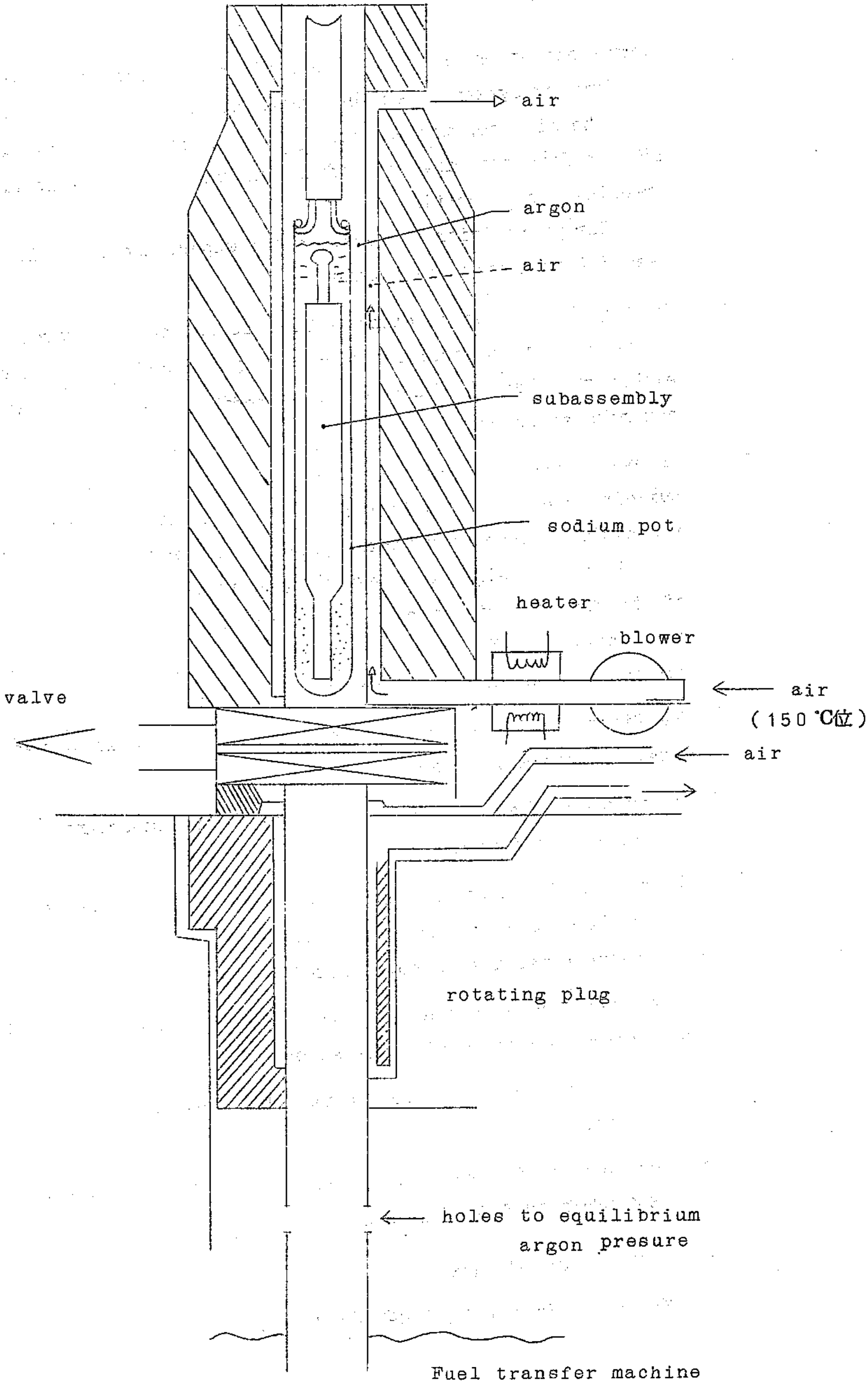
a) we are not able to give you valuable information in such conditions,

b) you have to foresee a very important testing program.

多くの問題が生ずる恐れがある。

(vi) ポット内の Na の activity

停止後は急速に低下するので問題ではない。



Fuel transfer machine

cooling scheme

2) Questions and answers

(i) Q. We are now in mind of adopting a fuel transfer machine of the bridge type which can freely access the top of the reactor. In this design, we consider the charge face level of the top of the reactor is to be lifted up to the level of the top of the control rod drive mechanism.

A. No comments. But we have to recall that a fuel subassembly loading-unloading during reactor operation does not seem very convenient and we think that this procedure should be dropped.

(ii) Q. On cooling the fuel in the fuel trensfer mechanism,

① If you have experience that you had removed sodium vapour contained in gas in the case of gas cooling, please show us the method of its removal.

② Please point out the problems in the case of JEFR.

A. Forced gas cooling of the subassembly during its stay inside the machine is a very difficult task. Many troubles can occur (sodium entrainment plugging of argon pipes). It would be better to have cooling only by natural means (radiation, conduction, natural convection). This would be made easier of unloading is made during reactor shutdown and if the subassembly is discharged in sodium pots. An easy cooling (or heating) of the machine could probably be obtained by an air loop (air flowing between the outer wall of the internal part of the machine and the biological shielding around it).

Some experimental tests of this cooling system (sodium pots with electrical heating in a shell at the same dimensions as the machine) is necessary and easy to make.

See "General comments"

(iii) Q. What kind of counter-measures do you have for the sodium adhering to the fuel to drop into the fuel transfer mechanism in your experiences?

A.

(iv) Q. On cooling a fuel during its transfer

Please give us your comments for the JEFR design.

A.

(v) Q. Please show us your safety criteria used in the fuel transfer system design in your experiences.

A. The safety of the fuel handling is essentially based on administrative control (cards, tables, etc...). However, concerning the number of fuel subassemblies to be loaded, a neutron counting is made during loading operation. This operation is automatically stopped if some definite value of the counting rate is reached.

5.4 Internal storage

1) Comment

(i) The reactor internal storage is rotating. The driver shaft is crossing the reactor vessel at the worst location, just above the sodium level. and its reparation or change are very difficult.

But, moreover, we think it is bad to put in the reactor vessel mechanisms which are not absolutely necessary and we propose to use normal subassembly supports for the internal storage.

(ii) This storage is probably not useful. The subassembly decay heat is about 10 kW after 12 hours since reactor was shut down, A sodium pot is used for subassembly transportation. From our experience a sodium pot can transport more than 10 kW if the pot wall is at 600°C, The fuel clad temperature is then about 630-650°C.

Ar ガスの自然対流による冷却でも fuel clad の温度は６３０℃以下に押えられるだろう。

(iii) If there is an internal storage, it has to be not rotating
Rotating rack は Cadarach でも Na 中で数回の bearing test を行なつているが原因のよくわからない trouble をよく経験している。したがつてフランスではこの種の design はとり入れないことにしている。

(iv) 貯蔵ラック固定については賛成である。 But, if it is rotating, every turning and hiding part must be removable. This is particularly true for balls and rolls acting in the bottom of the vessel. We do not know any material whose the operation can be guaranteed during a long time in sodium.

On the other hand, the command shaft and gears are positionned at the worst location above the sodium level. Sodium deposit may lead to a jamming of the mechanism by oxide formation. This has to be changed in every case and it does not seem useful to to discuss the mechanism details.

JEFR の rotating rack は機構が複雑で故障した場合容易に取出せない。特に下部は unremovable であり問題がある。フランスではこれに関する実験はやつたことがない。guiding ball の取替えは難しいだろう。

2) Question and answer

(i) Q. We would like to know your experiences on the design and handling of the rack.

A. Rack for structural material irradiation test pieces.

This problem is dependent of the position where irradiations are to be made. In Rapsodie, we designed a lot of irradiations subassemblies to be placed at the position of core subassemblies or blanket subassemblies or steel reflector subassemblies. It is, of course, also possible to provide special positions for material irradiation as you plan to make it in JEFR. But an important problem is to be careful of a sufficient value of the neutron flux and to look for proper access by handling machines.

Irradiation elements must have a shape sufficiently close to the thermal shape of a subassembly to be easily handled by normal handling machine. If these irradiation elements must have a special handling machine, this machine has to be developed and checked.

5.5 Transfer rotor

1) Comment

(i) The transfer rotor vessel is annular and has be designed for having a minimum sodium capacity. This is an advantage from the point of view of sodium supply. But this advantage has to be discussed in comparison with heat capacity. In case of failure of transfer rotor cooling, when the rotor is filled by spent subassembly a larger sodium mass may be useful to allow a longer time for reparation.

(ii) At the bottom of the internal shell, there are guide rollers, which are not removable. As we mentioned above, we do not know presently surface treatments or material deposits allowing a long time operation in sodium.

Therefore, this guide rollers have to be removable. But we think it must be possible to have no guiding inside the transfer rotor vessel and to locate every mechanism above the seal.

(iii) Argon pressure and supply are not mentioned in the fuel loading system volume, but do not remember:

1. argon may be at pressure slightly higher than building pressure

2. argon may be active in case of clad failures of spent subassemblies and argon purge must be foreseen,

3. argon over pressure may be supplied between the first seal (O ring) and the second seal (gland packing) to avoid any leakage of active gas to the building.

2) Questions and answers

(i) Q. We consider now the driving mechanism of the transfer rotor should be located at the outside of the transfer rotor room for easy maintenance.

A. It would be very suitable that most of the mechanisms would be in air and would be accessible for maintenance.

See "comments".

(ii) Q. We consider the procedure for the access to the transfer rotor as follows.

① Replace the active sodium to the fresh one in the rotor vessel.

② Circulate the fresh sodium in the rotor vessel.

③ We would like to know your views on the reliability of the guide roller operating in sodium which is attached at the underside of the rotor.

A. Circulation of fresh sodium would be less necessary if all mechanisms can be easily removed.

(iii) Q. It would be better to avoid the use of guide roller at the bottom of the rotor, as they cannot be removed.

A. See "Comments".

5.6 Sweeper

1) Comment

(i) We designed and fabricated a sweeper for Rapsodie and, after long discussion, we do not use it because we consider it more dangerous than useful.

If the core cover plate cannot touch any subassembly head, the vertical position of a subassembly can be changed only by the fuel exchanger. But we saw above one can control easily that the gripper do not faulty carried away the subassembly head. However, once, in Rapsodie a subassembly head was gripped by the hold down mechanism of the fuel exchanger.

On the other hand, sweeper is including bearings and drivings in sodium and argon. It is very sensitive to jammings and if it jams in horizontal position, it is very difficult to remove it.

Up to now we preferred some risk on the subassembly location to a larger risk on sweeper operation.

2) Question and answer

(i) Q. We would like to know your opinion from the viewpoint of maintenance for the design of the sweeper which is not the removable structure in the vessel.

A. A sweep arm has been built for Rapsodie, but it has never been used on the reactor. In fact, the troubles due to the sweep arm itself can be more important than the troubles it can eliminate. If there is a sweep arm, it is necessary that it would be removable.

5.7 Door valve

1) Comment

(i) We have no special remark on door valve design.

At CEA, we have the habit to order the fuel loading valves to a specialized valve manufacturer and use standard valves as often as possible. By such a way, we do not design valves by ourselves.

2) Questions and answers

(i) Q. We consider the radioactive sodium adhered to the door valve must be taken off and we are intending to adopt a method to blow off the hot argon gas on the contaminated door valve for removing the adhered sodium.

A. On EBR-II and on Rapsodie, some sodium falls on to the door of the valve, during each unloading. Decontamination by hand is made after each operation.

(ii) Q. As the air in the connection void space between the charge discharge machine and the door valve is a little quantity, we consider this air is allowed to enter into the reactor vessel.

A. Some argon purging into the intermediate part, between the door valve and the bottom of the machine, is convenient before opening and closing the valves.

5.8 Gripper

1) Comment

No comment.

2) Questions and answers

(i) Q. We are taking account of the following operation signals at least for the design of fuel exchanger and the charge discharge machine.

① gripper arrived

② jaws engaged

③ gripper loaded

④ gripper oriented (only for fuel exchanger gripper)

Please give us your comments

A. No comments.

(ii) Q. On the hinge of the gripper of the fuel exchanger

① Please show us the backdata for selection and determination of the structure and the materials, if any.

② How about your countermeasure for the sodium adhesion to stop its operation?

A. From our experience with mechanisms working in sodium, we can express some important general laws, as following:

a) the design of the mechanisms has to be as simple as possible.

b) large gaps are necessary for moving pieces, working in sodium.

c) the results are generally better with mechanisms working in sodium than with mechanisms working in argon with sodium vapors.

d) for rotating or sliding pieces with small gaps (bearings for example) some proper plating is necessary. In that case, materials like stellites give very good results.

On Rapsodie, no counter-measures are provided against sodium adhesion. If the gripper is removed outside of the reactor and becomes cold it has to be reheated before resuming to operate and as far as possible it has to resume its work in sodium. If it has to work in cold atmosphere, it is necessary to clean it, what is only possible after dismantling.

3) Information

使用中の gripper は Rapsodie ではアルコールで洗浄した。場所は container にグローブボックスを置き Ar を入れ，その中で行なう。運転中は取外した方がよい。

5.9 Periscope

1) Comment

No comment.

2) Questions and answers

(i) Q. ① Which method is better from the view-point of reliable operation for the inspection of the core inside?

a) Periscope method

b) Inpile TV method

② We would like to know your view whether the clear sight is obtainable through periscope or ITV which is cooled by blowing the cold argon gas of $30^{\circ}C$ into the reactor.

A. The use of periscope gives very good pictures from the inside on the reactor vessel above the sodium vessel, if sodium temperature is not more than about $200^{\circ}C$. Of course, temperature conditions are to keep in mind for the choice of the periscope, and it is very convenient to use a system without any sticky paste on the glass pieces. (French maker, Clavé, builds such periscopes). We have also a good experience with TV cameras. But some cooling is necessary. Furthermore the use of TV is not so easy and the cost is higher than for periscope.

For periscope and for TV, the quality of the picture is largely dependent of a proper lighting. Some holes have to be provided in the shielding plugs to insert the lighting devices.

5.10 Installation for new fuels

1) Comment

No comment.

2) Questions and answers

(i) Q. On the installation for test:

Kindly show us the items and the methods by which test a new fuel in Rapsodie.

A. On Rapsodie, the new subassemblies are controlled by comparison with their respective fabrication cards. The following operations are made:

a) measurements of main dimensions (height, bottom part, etc...)

b) weighting

c) control of the pressure drop by gas blowing, to detect any foreign pieces inside the subassembly

(ii) Q. On the storage of new fuels

Kindly show us its method in the case of Rapsodie.

A. At Rapsodie, new subassemblies are stored inside the transport containers (aluminium).

(iii) Q. On washing and drying

Kindly show us your methods of washing and drying new fuels in the case of Rapsodie and Phenix.

A. There is no cleaning of the subassemblies on the site, the only cleaning is made after fabrication.

(iv) Q. On the fuel management

Kindly show us your method of the fuel management in the case of Rapsodie.

A. At the construction, a special card is filled for each new subassembly giving all its characteristics. On the site, another card is prepared giving all operations concerning this subassembly (control operations, positions in the core or in the storage, etc...)

Before each loading-unloading operation, a detailed program is written. For each operation, one people is conductiing the operation, and another is controlling it (for example, the positionning of the rotating shield plugs is adjusted by one operator and controlled by another operator).

6. 冷却系統設計

6.1 General comment

1) Comments

(i) All the circuits seems to be too complicated. They include too many components such as E.M. pumps, cold traps, plugging indicators, tanks, auxiliary coolings, etc...

We think that these service loops have to be re-designed with the purpose of using the minimum number of pumps, plugging-indicators and storage tanks.

例えば

① E.M. Pumps

Purification system などまだ詳細にはみていないが，electro-magnetic pump が多過ぎる。 Rapsodie および Phenix では，それぞれ合計3ケの E.M.P. しか持つていない。 Phenix の場合，現在検討中であるが，2ケの E.M.P. になるかも知れない。 JEFR の design のように機器が多いと maintenance，修理などが大変である。 loop service 系統は出来る限り単純化して．最小必要限度のものでよいと思う。

② Na 受入れ line のタンクの数

JEFR では3個もつているが Rapsodie は貯蔵タンク1つである。

③ 機器へのArガス供給管

JEFR では1次系ポンプ，熱交換器など各機器へのArガスの供給管はいずれもバッフアタンクから取出しているが， Rapsodie ではバッフアタンクからの取出しはArガスの母管で機器の近くでこの母管より各機器へのArガス管を取している。

(ii) Use of NaK has some interest only in so far as an alloy being liquid at ambient temperature is chosen. But the use of NaK is never to be recommended because NaK is expensive and always more dangerous than sodium.

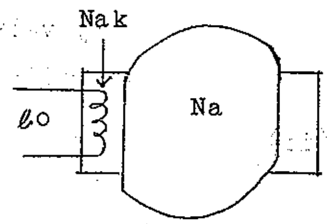
In the secondary loop of the emergency cooling system NaK allows to decrease the terminal heat exchange temperature without freezing. But NaK nearer eutectic alloy would assure a lower freezing temperature and avoid pipe preheating. On other hand, if NaK is used, primary sodium may eventually freeze in IHX, which is not possible when secondary coolant is sodium.

Although NaK has these advantages, for an emergency cooling system designed as yours, we would prefer use Na for the secondary loop.

なお JEFR の設計では NaK を使用しても 70°C 以上に保つための予熱装置を設けておかなければならないので設備費は Na に比較して安くはならないだろう。

Phenix では purification system に liquid organic を使用する予定である。

EBR-II では右図のような cold trap を使用している。 NaK を使用しているのは配管破損により直接 Na がよごれないためである。



(iii) A special attention may be paid to the tanks:

Tanks: Have a special look when hot sodium must be dropped into a cold tank (or cold sodium into a hot tank). Special thermal protection must be provided on the nozzles when there is a large temperature difference between tank and sodium.

2) Question and Answer

(i) Q. Operation of sodium circuits

The JEFR will have two main primary cooling loops. It is planned that only one loop will be operated during reactor shutdown to remove decay heat.

Is it all right to operate only one loop from the view point of sodium characteristics, or does it cause trouble?

A. We do so in Rapsodie.

6.2 Primary main system

1) Comments

(i) On the primary loops of the JEFR, there are sodium valves either on the main flow or on the blanket flow.

We did not used valves on the primary loops of Rapsodie because we have thought that failure hazard is larger for valves and we have feared maintenance difficulties. Phenix being a pool reactor, we do not need valves on the primary cooling. Therefore, we have no experience of the use of valves in a large active sodium circuit.

o if any valve could be avoided on main primary loops, it would be more reliable

(ii) The primary sodium loops have an important length: 34 m from reactor to IHX; 17 m from IHX to pump, 50 m from pump to reactor.

In Rapsodie, the equivalent length are for one loop about 16 m from reactor to IHX: 9 m from IHX to pump, 30 m from pump to reactor.

These lengths differences may be partly due to diameter difference (0.484 ; 0.391 ; 0.340 m in JEFR and 0.308 ; 0.208 ; and 0.314 ø out in Rapsodie). We could not check whether there is some difference in calculation methods.

2) Questions and Answers

(i) Q. Would you tell us some factors limiting the increase of the coolant temperature difference between the inlet and the outlet of the core? For example, the thermal stress, the thermal shock, the effect for the nozzle, etc. and please give us the reason.

A. The inlet-outlet Δt of the reactor is limited by the hot point of the clad. The loop and structures of the reactor can certainly support higher Δt because they can be protected from thermal shocks for instance by baffles. The allowable limiting ΔT_s of the circuits and structures have not been evaluated since knowledge of these values is useles.

(ii) Q. Do you consider anything caused by the corrosion or the erosion of the materials for deciding the sodium velocity in the core?

A. There is no corrosion or erosion in the loops where the sodium velocities are rather low. This problem only exists for the core especially for the clad.

(iii) Q. Would you show us the design criteria for the outside pipings of main cooling lines?

A. The criteria for the design of the external piping of the main circuits are as follows:

a) the total double piping is divided into sufficiently small parts so that the volume of two consecutive parts is smaller than the volume of sodium surmounting the outlet pipes of the reactor. The core cooling is consequently realized when a leak occurs at the common point of two parts.

b) The maximum temperature difference between piping and double wall is 200°C in case of reheating of a loop where sodium was frozen.

c) The double wall tightness is checked by soap-bubbles and NH_3 testing.

The double wall is not calculated for resisting to vacuum.

(iv) Q. Would you give us your recommendation on the cavitation in sodium flow based on your experience?

A. For cavitation, as more accurate informations are missing the same questions as for water are presently used.

Na は水より pesimistic になるだろうがほとんど水相当と考えてよい。
キャビテーションについては Phenix の Na の流速が Rapsodie より大きくとつてあり少し問題になるかもしれないので検討中である。criteria にはキャビテーションを考慮しておく必要がある。
この問題の基礎テストについてはミシガン大学の Hammit が研究を行なつている。

(v) Q. Would you give us your comments on the thickness of main pipes that is 10 mm?

A. Piping thickness is 4 mm for the main piping except 7 mm in the loop part which is inside the safety vessel where a common pipe is used for the two circuits. Let us recall that the piping diameters are of 200 to 300 mm. The elbows are reinforced for 500 mm ∅ pipings an about 7 mm thickness is foreseen.

Consequently, we have the feeling that the JEFR pipings thickness should be between 4-7 mm, of course in accordance with the mechanical stresses.

薄い程 thermal shock に対して強い。

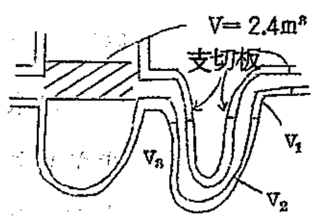
(vi) Q. Cleaning at site after welding

Was the primary sodium system cleaned by acid or other cleaning agent? I am concerned of the residual chemicals or remaining water in the crevice or cavities in the system, what is the general practice of cleaning after field welding of the sodium loops in France?

A.

3) Informations

(i) Rapsodie では double containment 系の配管をいくつかの領域に分けそのうちの最大の 2 つの領域において inner 管より outer 管へリークがあつても (支切板でのリークを考慮) 炉心の Na レベルが出口管より下らないように設計した。即ち ~< V/2 , したがつて領域の最は 1.2 m^3 である。



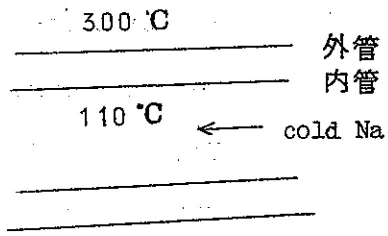
leak test は NH_3 ガスを使用して行なつた。H_2 は leak が多く不可能である。

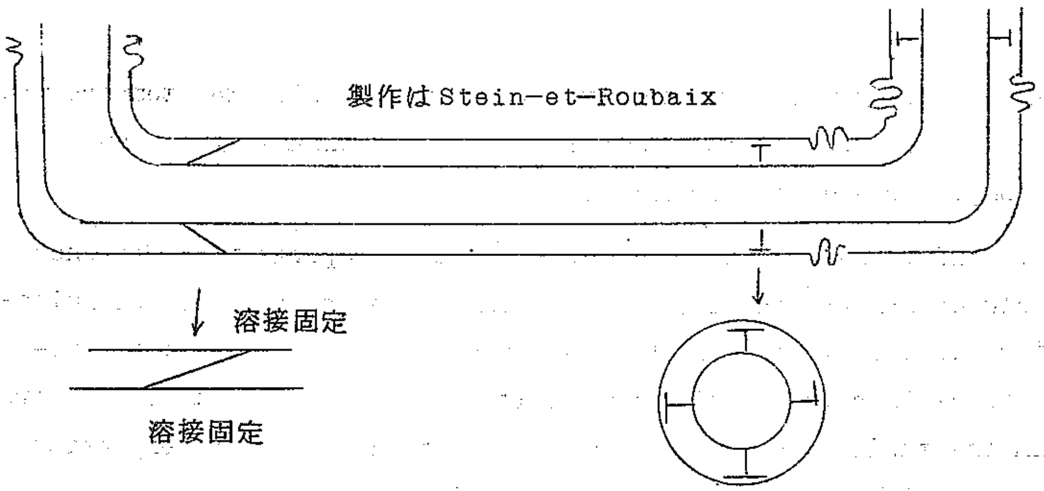
double contaiment の外管の肉厚は 2 mm。

1次系内管設計圧力は 125 m Na ≃ 10 kg／cm²

圧力は 2 kg/cm²

preheating のときの内管と外管の温度差は約200℃位生ずる。loop の配管は下記のとうりで外管はベロー式になつている。





E.Fermi はもつと簡単な方式で double contaiment の外管を電気的に加熱している。

(iii) Rapsodie 主配管（内管）

　　内径　　300mm

　　外径　　308mm

たゞし安全容器内の炉心下部入口管に限り

　　内径　　300mm

　　外径　　314mm

にした。これは安全容器内は修理不可能なためである。

6.3 Secondary main system

1) Comments

(i) The pipe lengths seem to be important.

See the comments of primary main system

(ii) The use of jackets for the main secondary sodium does not seem necessary, since sodium is not active.

2) Questions and Answers

(i) Q. Would you show us the criteria how to decide the temperature differences between the primary and the secondary system and the example, based on your experience?

A. If there is a steam production, the temperature difference between primary and secondary sodium is set by a global optimisation of the plant. In the case of Rapsodie, or JEFR, this difference is rather arbitrary but it has an upper limit due to stresses in the tubular sheets or stresses due to differential expansions between tubes, if the pipes are straight without expansion-bends.

Rapsodieは 30°C 対向流方式の制限温度差は最大５０℃位である。管厚を薄くし形状を複雑にすればもう少し大きくできるが長さには関係しない。

(ii) Q. Would you show us the criteria of the decision on the flow of the secondary system and the example based on your experience?

A. If there is a steam production, the secondary flow rate results from an optimisation. When there is no steam production, it is also rather arbitrary. For Rapsodie, the secondary flow has been decided to be the same as the primary one, without precise criterion.

General remark: The air cooling admits a relative great allowance in the choice of the operating temperatures, since the mean temperature of the cold source may be changed, which is not possible if steam is produced. For example, Rapsodie operates at temperatures which are different from the nominal temperatures of the design.

実験炉特有の criteria は持っていない。水冷却，動力炉と同じと考えている。

なお Rapsodie では2次系について Na fire による消火は不可能なため，containment の中だけ2重管にした。

Phenix は2重管は全然使用していない。

6.4 Emergency system

1) Comments

(i) In the present design, the emergency cooling system is very complicated and its reliability would be poor (it includes 4 pumps and 2 air blowers).

The leakage hazard of this cooling system is probably more important than that of the main sodium loops, because of the use of electromagnetic pumps, for example.

Moreover, the penetration of the emergency primary loop through the reactor vessel are located at a level lower than that of the main sodium loops penetrations, which is not conservative.

小さいノズルは大きいノズルより破損の危険性が高いことを忘れてはならない。

(ii) These loops are operating exceptionally and must be designed to avoid sodium boiling in case of accident. They can be calculated for minor characteristics by allowing more important temperature gradients, but they must be very reliable and always ready to operate.

(iii) The decay heat cooling system is more dangerous for the reactor than the normal cooling system, because it is welded on the vessel at lower level. On the other hand, it is not working by natural convection and so it is not a true ultimate cooling system.

We would propose to use a entirely different principle and to study:

- either a cooling external to the reactor vessel,
- or a small heat exchanger immersed in the vessel and connected with a NaK loop operating by natural convection.

Moreover, we think it is possible to permit exceptional high temperatures during very improbable and scarce accidents and, therefore, to use a smaller ultimate cooling loop.

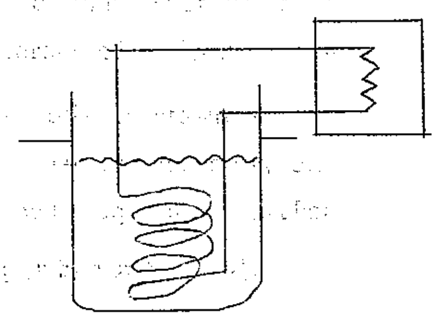
(iv) It is extremely desirable to use natural convection instead of forced convection if it is possible. By this mean, the emergency cooling system can operate without any power supply and is more reliable.

(v) We think it is better to use NaK coils immersed in the peripherical part of the reactor vessel. The NaK flow must be due to natural convection if it is possible.

In the case where there are difficulties due to the NaK activity, a secondary loop may be added, using also natural convection.

This solution was envisaged for Rapsodie, but too late and was dropped only because of the constructions schedule.

即ち，右図のようなＥＢＲ－Ⅱ型にしてはどうかと思う。熱除去は自然循環。なおこの方式では緊急時だけでなく常時循環しておく方が好ましい。

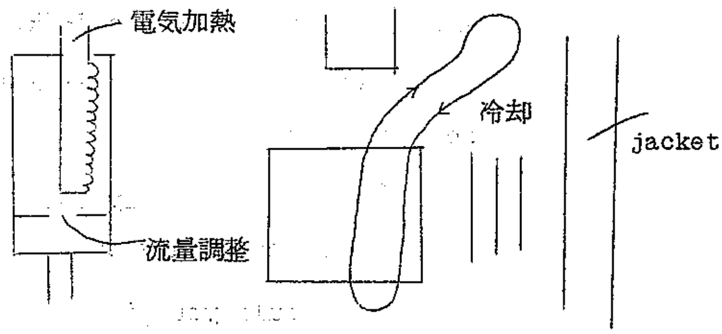


(vi) We allow a sodium temperature above maximum nominal temperature for the vessel (about 700°C), but this figure has to be discussed. We use thermal inertia of reactor and, as decay heat is decreasing with time, maximum accidental temperature is got after several hours, with a safety cooling system of relatively small capacity.

(vii) JEFR への close loop の提案は緊急系としての提案で停止時使用は考えたものではない。緊急系はその目的のみに使用すべきである。Fermi 炉は容量的に natural だけでは十分でないので組み合わせてやつている。Rapsodie は natural convection で十分除去可能な設計にしてある。停止時には main pump を低速で運転する（Fuel handling 中も）

Rapsodieで事故時達するであろうと予想している最高温度は７００℃である。Plenixも７００℃である。

事故時vessel内で起る
自然対流は右図のように
なろう。左図のような実
験装置を作つて実験を行
なつた。



(viii) 緊急時の容量は1%位で十分であると思う。温度は少々上つてもよいと思う。クリープを考えれば700℃位までよいと考えている。
JEFRの2ループにする変更はよくない。trouble が増加されるだけである。

2) Questions and Answers

(i) Q. Would you show us the scheme and the design criteria on the emergency system based on your experience?

A. See "comments".

(ii) Q. Would you point out if there are any problems on the scheme and the criteria of the JEFR?

A. See "comments".

(iii) Q. The NaK is used in the secondary system of the JEFR. It is the reason to prevent the solidification of the fluid caused by the miss operation of the blower. Would you give us your opinion if there are any comments?

A. If NaK is used, there is a possibility of freezing sodium inside the primary circuit instead of freezing sodium in the secondary circuit if it is not used. That would not be better. It is not justified to worry that freezing might be dangerous in the terminal heat exchanger during normal operation because of the important thermal inertia of the Na circuit. In Rapsodie, the fast return to zero point of the fans blades position which had initially been provided for this purpose has been dropped.

(iv) Q. (1) There are one intermediate heat exchanger, one air cooler, two pumps for driving, two valves and two blowers on the emergency system on the JEFR.

Would you show us the criteria on the emergency system based on your experience?

(2) There is only one emergency system on the JEFR. Do you consider it necessary to install two systems or one sufficient?

(3) Would you give us your comments on the 7 m.m. pipe thickness of the emergency cooling system?

A. See "comments".

As we do not agree with the design of the circuit and propose its modification it does not seem useful to discuss these questions.

6.5 Purification system

1) Comments

(i) Only one plugging indicator is necessary for every auxiliary circuit. Valve system may allow to use it for measuring plugging temperature either at inlet or outlet of the cold trap. A spare plugging indicator may be added, but is not indispensable if the first one is easily reparable.

(ii) We agree with setting two cold trap on the primary auxiliary loop, one of which only is operating. It is prudent to design removable cold traps and to foresee a third spare cold trap in case of plugging of the operating one.

(iii) We have not a real formulation of the minimum sodium flow rate of the purification loop. But the Rapsodie purification flow rate seems to be convenient and we intend to extrapolate it for larger circuits by considering wall surfaces of the loops.

By using this rule, the flow rate chosen for JEFR seems to be rather large. But this is more conservative and we do not recommend to change it.

JEFR の cold trap の容量２０m³／h は大き過ぎる。１５m³／h で十分だと思う。high flow の場合，冷却用のブロウの速度または容量が問題となる。フランスは容量についての式および criteria は持つていないが

Rapsodie	5 m^3/h	(total quality sodium 35 m^3)
Phenix	30 m^3/h	(total quality sodium 1,000 m^3)

(iv) It is not necessary to have an argon connection at the top of the cold traps. It is more convenient to design just a trapped argon volume above the sodium level for allowing a free expansion when sodium is melted after freezing.

(v) In Rapsodie, cold trap are cooled by air because the cooling power is not very important. It is the simplest way.

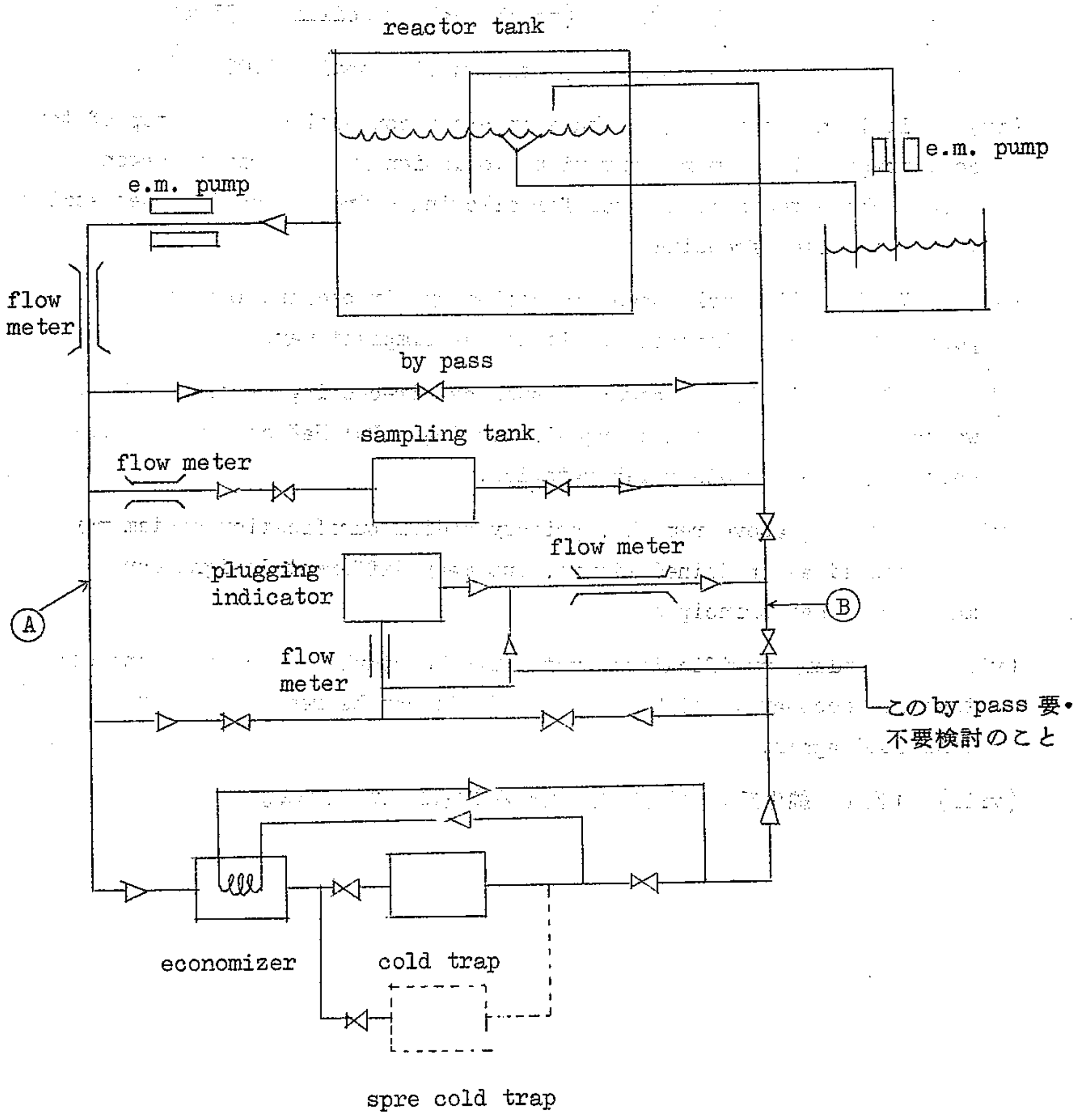
If the dissipated power is such as air-cooling is not sufficient, we have a tendency to use organic cooling. But NaK cooling is successfully used by other laboratories.

(vi) With the above remarks, primary sodium purification system may be sketched as on joined figure, but many different designs can be made from that principles.

(vii) Secondary purification system can be simpler since only one cold trap is necessary. If it is plugged, it can be removed, cleaned and connected again.

(viii) １次Ｎａ純化系を格納容器内に入れる変更には賛成である。

Primary sodium purification system
example of flow diagram



Notes: 1. A spare plugging indicator may be added to the circuit

2. Hot trap system or rhometer or any other device may be added in parallel between line Ⓐ and line Ⓑ if it is useful

2) Questions and Answers

(i) Q. Would you explain to us your methods, basing on your experience, of the purification system especially on the inlet and outlet positions?

A. The Rapsodie cleaning system is similar to the Fermi one. The sodium is picked from the overflow tank and flows successively through the e.m.p., the cold trap and returns to the reactor primary vessel. This system works pretty well but other designs are certainly possible if they are not more complicated.

(ii) Q. Would you give us your comments on the JEFR design of this system?

A. Modifications of the JEFR design are proposed in the "General comments": it is possible to use only one pump, one plugging indicator and two cold traps.

In your design, attention should be paid to the cold sodium return into the hot exchanger and especially to mixing, thermal stresses at the piping connection on the exchanger.

(iii) Q. The design criteria of the JEFR is as follows:

- o Purification system flow = system inventory x 0.002 (l/min.).
- o The sodium passing time of the cold trap is approx. 5 - 6 mins.
- o Would you show us your criteria of this system, basing on your experience.
- o Would you give us your comments on the criteria of the JEFR.

A. The Rapsodie cleaning flow is about 5m3/h for 30 t. of sodium in the primary loops. But when two sodium circuits are compared for cleaning flow, stainless steel surfaces and Na-air contact surfaces have also importance.

There is no precise criterion, furthermore the greater is the purification flow, the faster is the initial cleaning. For Rapsodie about 15 days are needed. The JEFR flow is certainly very sufficient and may be, a little bit large when it is compared to the Rapsodie one. But this is conservative.

(iv) Q. Would you tell us the methods of the purification, (or the disposal) of the sodium being contaminated by the fuel rupture, based on your experience?

A. For the small sodium quantities removed during components dismantling, i.e. some kg, washing water carries away fission products to the effluents circuits.

In Rapsodie, there is no continuous sodium purification from fission products except probably a certain trapping with help of the cold trap. Let us remember that no real cladding failure occurred and that, up to now, there are not yet fission products in the sodium.

(v) Q. Would you give us your comments on the construction of the cold trap (Drg. FBDO - 20013) with regards to the performance?

A. Up to now, we always made a bad use of cold trap volume aimed to oxyde trapping. The sodium oxyde collects as a thin crust blocking prematurely the flow instead of filling the volume intended to trapping.

Similar observations have been made on EFFBR whose cold traps are different from the Rapsodie ones. We cannot say if the system designed for JEFR is better. Testings are necessary.

(vi) Q. We are using the type that has a free surface on the cold trap, do you think it is necessary to use the free surface?

A. The argon pipings connected at the top of the cold traps must be dropped. Only a closed argon volume should be put of the cold trap, as a safety system against sodium successive freezing and melting.

Provide a small by-pass hole in the top of the cold trap for continuous argon draining from the inlet to the outlet.

(vii) Q. Would you show us the following items on the NaK purification?

a.) the saturated concentration curve.

b.) the design criteria of the purification components.

A. We don't have any saturation curve for NaK. We consider the problem exactly as for sodium.

使用しているのは Liquid Metal Handbook (old one) にある diffusion cold trap についての記述のみである。
saturation curve は特に必要ないのじやないかと思う。plugging temperature だけでよいだろう。

(viii) Q. We have settled the minimum temperature of Na-K cold trap, 70°C. How do you think of that?

A. In the experimental 1 MW and 10 MW loops, the NaK cold trap could operate with a temperature from 50 to 100°C. It does not seem useful to lower temperature under 70 to 80°C. At 100°C, an appreciable quantity of oxyde is probably still solved in NaK, since an accidentally fast plugging occurs when a part of the NaK piping becomes colder than 100°C.

(ix) Q. Do you have any data of impurity solubility for NaK?

A. See the questions and answers (vii)

(x) Q. On primary sodium purification system in your consultation with PNC representatives on December 12th at Cadarache, you suggested the purified sodium should not be returned to the over flow tank. Please let us know the reason why it should not be returned to the over-flow tank?

A. During discussion it was proposed to you to return sodium directly to reactor and to avoid a second electromagnetic for circulating sodium from overflow tank to reactor. But this modification must be considered in the general modification of the auxiliary sodium circuits as we have proposed to you.

3) Informations

Rapsodie のプラギングメーターの数は１次系２次系とも各１個である。プラギングメーターの故障はほとんどないと考えている。

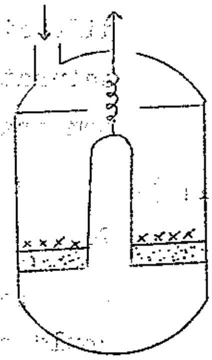
(i) Cold trap について

Cold trap の数は Rapsodie と同じである。 Rapsodie では１次系は系統に２個設置し１個使用１個予備，他に１個を常時取替えられるようにもつている。２次系は各ループに１個ずつである。

Rapsodie では active になる前に２回取替えたが，それ以後運転に入つてからは取替えた経験はない。

JEFR 別案………運転以前は小さい cold trap を多く持つか，大きいのを設置し運転に入る直前に小さいものを2個（各ループ）づゝにする………は非常によいと思う。

(ii) Rapsodie の cold trap はよくないしよい設計方法も知らない Rapsodie では下部に oxide がたまり易く詰りやすい。



(iii) Contaminated Na の purification は考えていない。

6.6 Charge and drain system

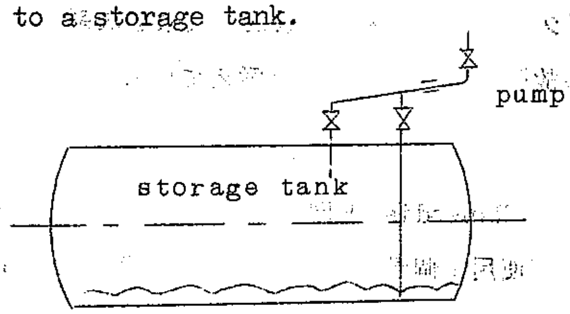
1) Comments

(i) Your sodium purification and filling and drainage are very complex. You use many more components than we do for Rapsodie and Phenix.

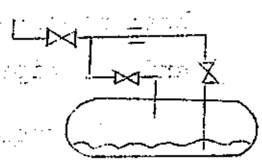
(ii) The intermediary tanks between transportation containers and primary sodium storage are not useful because it is not necessary to know exactly the sodium contenance of primary sodium storage on primary main loops.

(iii) The pipe length including bellow valves or e.m. pump must be drainable, to avoid bellow or pump conduct failure in case of sodium freezing and melting.

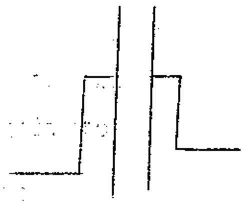
In this purpose, we generally separate filling pipe and drainage pipe connected to a storage tank.



貯蔵タンクからポンプまでの配管の変更について



ＪＥＦＲ方式ではドレンしたとき完全にドレンできたかどうか確認する方法がないので上記のような型式にしてもよいし，図のような系統にしてもよい。



配管とタンクの取付部はRapsodieでは右図のような方式は熱膨脹をさけるため右図のような型式で取付けている。

(iv) The loops are more advantageously filled at 110°C instead of 200°C. By this way, less sodium oxyde is transferred to the loops.

(v) It is dangerous to use argon over pressure to transfer sodium from storage to primary system or from primary system to storage, because there is a hazard to get some sodium flowing up through a vertical pipe immersed in sodium and spreading on the reactor operation level, if the closure is good at the top of the vertical pipe. We got this accident once before Rapsodie criticality.

We recommend you to use a pump to fill or drain sodium primary system.

2) Questions and Answers

(i) Q. What centimeter is proper to the diameter of the pipes for the drain system.

A. For Rapsodie, we use a pipe with an inside diameter of 36 mm. しかしこの値は特に根拠のあるものではない。

(ii) Q. Would you show us the data if you have any data on the adherence of the sodium on the surface of the components and the pipes after drain?

A. We have no precise data concerning the adherence of sodium. We specify only a 3% slope for the piping.

(iii) Q. Would you tell us the results if there any datum which have been measured gamma-ray dose on the circumference of the components and the pipes after drain?

Do those values consist with the estimation values of the sodium remaining?

A. During sodium loop reparation, it was showed that activity due to corrosion products is much more important than sodium activity consequently nothing can be deducted for sodium.

Rapsodie ではこの放射能を考慮するようなことはしなかつたが, Phenix でやつたところ遮蔽が非常に厚くなつた。大きさの criteria はないがクレーンで持ち上げる程度にする予定である。計算の必要がある。フランスでは他国のデータにもとづいて計算しておりRapsodieのデータ待ちなので数値は出せない。

(iv) Q. If there is no data of (ii) and (iii) would you tell us how to estimate the degree of the adherence of sodium?

A. See the questions and answers (iii)

(v) Q. The raw sodium from the chemical production company includes potassium and oxygen. Would you tell us the method of the purification control of them when they are loaded initially?

A. Rapsodie has been filled with sodium purified from its calcium (about 10 ppm instead of 300 to 400 for raw sodium). No special purification is done, excepted a filtration at filling.

Kは問題とならない。むしろCaの方が酸化物はNaに溶解しないので問題である。Rapsodieでは酸化物の除去にはSUSのフィルターを使用している。

(vi) Q. Would you tell us the degree of the oxygen remaining on the metal surface of the components and the pipes?

A. No test has been made. The following result has been obtained from UKAEA 1 gr O_2 by square meter of stainless steel surface (London Conference on FBR, 17-19 May 1966)

(vii) Q. We use the electromagnetic pump on the drain in the JEFR. Do you think of any possibility that we could not use the pump by the gas which would enter into the pump?

A. The piping of the e.m. pump must be inclined so that argon may not be accumulated inside the pump. It is also necessary to check that the NPSH at the pump suction is correct; from this view point, the induction pumps are generally better than the conduction pump.

Rapsodie では問題はなかつた。EMPの pipe には slope を持たせておくことを忘れないように。

6.7 Level control and syphon breaker systems

1) Comment

No comment

2) Questions and Answers

(i) Q. Would you tell us the system of the level control in the vessel and of the syphon breaker based on your experience?

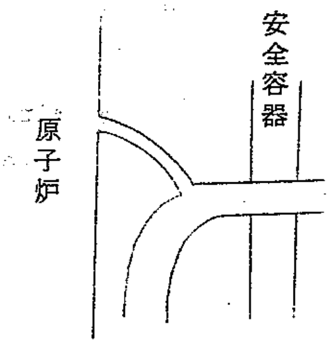
A. The Rapsodie level is approximately controlled with help of the overflow system. The pipe of the siphon-breaker connects the inlet siphon top to the argon volume which is inside the reactor vessel; the disadvantage of this system is that a small pipe is located inside the safety vessel.

It becomes non removable after the first reactor power start up. That is a weak point. It would certainly be better to locate the siphon-breaker pipes outside the safety vessel in an accessible zone for repairing.

JEFR の over flow 制御方式は非常に良い。Rapsodie の syphon breaker は一次系の Na の clump に対しては有効であるが小 leakage（フランスでは大破断は考えていない）に対しては役に立たない恐れがある。

バルブを置く考え方には賛成できない。原則的にループにはバルブを置くべきではない。

Rapsodieではsyphon管の破損は考えられない。



(ii) Q. Would you give us your comments if there are any comment on the over-flow system to control the level in the vessel of the JEFR?

A. Some disadvantage of the JEFR overflow system is that is requires additional pump added to the purification pump.

(iii) Q. Would you give us your comments if there are any comment on the syphon-breaker system of the JEFR?

A. A more theoretical than real disadvantage of the siphon-breaker designed for the JEFR is that it may be unefficient if the argon piping of the primary pump is plugged as a matter of fact, in this case, argon cannot enter into the siphon since the pump argon volume becomes under pressured.

(iv) Q. Would you tell us the structure of the syphon breaker and the detection method of the failure based on your experience?

A. In Rapsodie, there is a permanent control to insure that the siphon-breaker is not plugged, by using a sodium flowmeter as we wrote down above.

This flowmeter is provided with an alarm for minimal flow.

syphon breaker の管は30 mm φ の細管(標準品)を使用している。

(v) Q. Would you tell us your experience if you have had some troubles during the operation?

A. There was no trouble up to now.

6.8 I.H.X.

1) Comments

(i) You have designed an intermediate heat exchanger with cross-flow by using baffles.

This leads to a bigger IHX without any real advantage because the heat exchange is very good outside the tubes in parallel flow. Moreover, we have no experience of this type of heat exchanger.

We propose you to re-design your heat exchanger. You will get a component smaller, cheaper, and easier to build.

(ii) For sodium-sodium heat exchangers, parallel flows inside and outside bundle, without baffles, are recommended because the sodium-wall heat exchange is very good.

The dimensions and shell pressure drop are lower. The fabrication is easier and the IHX is less expensive.

(iii) For parallel flow heat exchangers, Lubasrky' formula is used inside and outside tubes. It includes a convenient safety margin of about 15% on temperature drop.

(iv) Temperature expansion problems are difficult. In Rapsodie heat exchangers, bent tubes were used. But now, it is estimated stresses are high in bent part of tubes and straight tubes are being designed for new IHX.

The company "Stein et Roubaix" is specially competent in such a design and has the necessary computation codes.

The IHX of Rapsodie was designed by Stein et Roubaix.

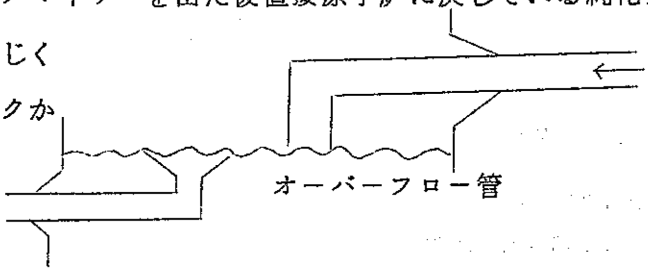
(v) Special attention must be paid to the IHX cleaning after removal. We are not sure that insulation is well set at the top of IHX when sodium cleaning is envisaged.

(vi) Sodium test of a prototype IHX is not necessary. In opposite, water test is absolutely necessary to check coolant distribution between tubes and vibration absence.

(vii) 純化系の戻りラインがＩＨＸに入つているのは問題である。しかしオーバーフロータンクへ戻すという別案はさらによくない。

cold Na とhot Naの mixing の問題, thermal fatigue の問題やその他の問題をよく検討する必要がある。

Rapsodie ではエコノマイザーを出た後直接原子炉に戻している純化系への取出しはＪＥＦＲと同じくオーバーフロータンクからである。



2) Questions and Answers

(i) Q. Would you tell us the formulas and the method for the calculation to the overall heat transfer coefficient being used?

A. The Lubarsky's equation ($Nu = 0{,}625\ Pe^{0{,}4}$) is used inside and outside tubes. It gives an about 15% safety margin for a heat exchanger without baffles, with flow parallel to tubes and a good flow distribution between tubes.

(ii) Q. Would you give us your comments if there are any problems on the structure of the JEFR? (thermal stress, thermal shock, drain of sodium, maintenance, etc.).

A. See "comments"

(iii) Q. Would you tell us the formula which you are using for the non-stationary heat transfer to evaluate the thermal shock?

A. These calculations are performed by the manufacturers "Stein et Roubaix" in the case of Rapsodie. The calculation codes belong to them.

(iv) Q. The shell side pressure drop of the heat exchangers are predicted by the same formula for the heat exchangers with water or oil in the shell side with the same construction.

Would you give us your comments on the validity of the use of the formula for usual fluids to sodium?

A. The same equations as for water are used. Uncertainty remains mainly for the particular pressure drops such as inlet and outlet, antivibratory belts, etc...

(v) Q. ① What do you think of the idea that make the temperature of secondary cooling system fall more than 20°C and decrease the number of Intermediate heat exchanger tubes?

② Would you give us your recommendation on the quality control for Intermediate heat exchanger tubes and for piping materials based on your experience?

A. See 6.3 - 2) questions and answers (i) & (ii).

(vi) Q. Intermediate heat exchangers for EBR-II, Enrico Fermi and Rapsodie have no baffles or tube support plates. It is against the TEMA standard. I think this is to minimize the shell side pressure drop. Are heat exchangers without tube supports safe from the view point of tube vibration?

A. The Rapsodie intermediate heat exchangers are provided with tube spacers against vibrations.

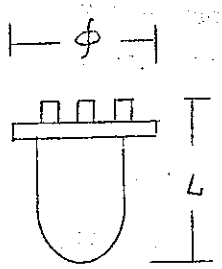
As we told you in our comments concerning JEFR cooling system, it is necessary to run a hydraulic test to check tube vibrations and to determine vertical distance between tube spacers.

3) Information

(i) Rapsodie Phenix の removable 部の概略大きさ

Rapsodie ϕ = 1,305 mm, L = 6,300 mm, P = 10 MWt

Phenix ϕ = 1,550 mm, L =11,500 mm, P = 90 MWt



なお製作に先立つてRapsodie Phenix ではhydraulic test をやつているチューブの取付部はかい管方式である。

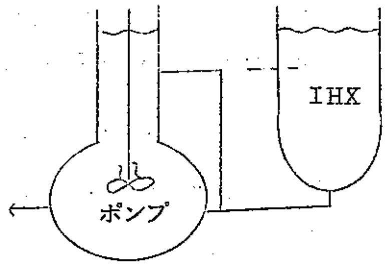
t = 16°C
(現在の使用温度差)
t = 28°C

6.9 Pump

1) Comments

(i) Cold sodium coming from hydrostatic bearing leakage must be returned to pump inlet. In every case, it must not be mixed to sodium at different temperatures.

Rapsodie には現在はこのようなバイパスを持つていないがForticimo 計画ではつける予定である。



(ii) A sodium over flow is provided to drain sodium coming from leakage of the hydrostatic bearing. This design may produce argon entrainment into sodium loop.

It is better to drain sodium by a line connected to pump shell at a level lower than the free sodium level. This last level will vary with operation conditions, but the variations can be limited by introducing a convenient pressure drop in this line. The pressure drop is difficulty computable and tests are necessary.

(iii) Sodium hydrostatic bearings were lengthly tested by CEA for different prototypes constructed by several companies and especially by Hispano-Suiza. At first glance, the JEFR hydrostatic bearings seem pretty long.

The items (ii) and (iii) are very closely connected to the fabrication know-how.

(iv) The pump removal must be very much studied. The part which is disassembled with sodium contaminated pieces must be as small as possible. The driving part must be removed easily and rapidly. We could not understand how it is foreseen in your design.

2) Questions and answers

(i) Q. Please show us the period of maintenance for mechanical pumps of Rapsodie.

N.B.: We uses that the maintenance will be performed to exchange the mechanical seal after the draining of the sodium and the pulling out of the pump before or after 10,000 hours operation.

A. The pumps of Rapsodie are in operation for 12,000 to 16,000 hours. In October 1968, one of the primary pump has been examined and did not show any trouble.

The mechanical seal on the top of the pump can be changed without removing the pump.

Rapsodie pump の maintenanceは mechanical seal の下部に shield plug があり direct な放射能は防げる。mechanical seal を取り出した後には back up sebl をはめ込み Ar ガスのリークを防いでいる。

取替時 Ar ガスのリークは少しはあるが新しい Ar に取替えてからやれば少々もれてもたいした問題ではない。

maintenance の period は定期的には行なつていない。今までの経験は下記のとおりである。

① 1967年 power up 前に1度 check して改造した。

② 1968年10月6日 primary pump の1つを check のため取出してみたが何ら悪いところはなかつた。(11,000時間運転後)

他のポンプはやらなかつた。

1967年の改造では hydraulic bearing を少し修正し gap を大きくした。(400μ)

使用温度は通常運転400℃(原子炉入口温度に等しい)

テスト 520℃

でやつた経験がある。

CEA-R3406 P.65参照 この文献ではポンプの使用温度は450℃に限定しているが，現在では最高520℃までの試験を実施している。

(ii) Q. Please show us the R.P.M. of pony motor and main motor of mechanical pump of Rapsodie, and the power supply for driving (kW).

N.B. On Fermi D.C. battery is used for one loop and A.C. batteries are used for the other two loops. On JEFR pony motor will be installed.

A. In case of electrical power failures, the pumps are supplied by batteries which allow a speed of around 1/10 of the nominal speed. Details are given in the CEA report n° R 3406.

Rapsodie には pony motor をもつていない。

(iii) Q. Bearing of Main Pump

a) Concerning materials for a bearing shaft and a bearing liner, Mr. Inoue's #2 report said "colmonoy is used for a primary pump and sterite for a secondary pump". We consider these are surface materials of the bearing shaft. Then, what material is used for the bearing liner?

b) We like to know a diameter and an axial length of the bearing.

c) Are there any grooving on the liner surface? If so, please show us their shapes.

d) "Gap 400" is a radius or a diameter? And this value is same both in the primary and the secondary pumps?

e) Please tell us the detail compositions or the material standard Nos. of colmonoy and sterite.

A. Detailed design of the bearings is in manufacturer field. We are just discussing materials for which we have testing apparatus in sodium in Cadarache.

For Rapsodie, we use stellite for secondary pumps and colmonoy for primary pumps. Stellite is giving better result, but it is more activable under residual neutron flux,

3) Information

(i) Rapsodie ポンプのハイドロスタテイック・ベアリング

1次系はコルモノイ，2次系はステライトを使用した。

1次系をコルモノイにした理由はCoが出て炉心をまわるのをさけるためである。

ライナーは，径200mmのとき，高さ200mm程度である。

ライナーにはグループが切つてある。

6.10 Main cooler

1) Comment

(i) The air cooler design with parallel units connected in series, implicates a complex sodium circuit.

2) Questions and Answers

(i) Q. We use the gas to pre-heat the main cooler of the JEFR. Would you give us your comments about that?

A. For Rapsodie, it has been concluded by studies that the safest and the most flexible preheating might be realized by electrical resistances located under the finned tube bundle, but it is necessary that the air duct may be closed at its two ends in the heat exchanger because that reduces considerably thermal losses during preheating.

(ii) Q. Would you give us your recommendation on the design of the structure of the main air cooler?

A. It is prudent to divide each heat exchanger into module easy to remove in case of sodium leak. Attention must be paid to tube vibrations and to possible corrosions by heat insulation wetted by rain. The Rapsodie's heat exchangers have been sheltered because important corrosion occurred in the experimental 10 MW exchanger which was outdoor.

(iii) Q. Would you give us your recommendation on the control meahanism of the air cooler based on your experience?

A. The air flow of the exchanger is adjusted by motor speed variations in the experimental 1 MW circuit and by orientation of the screw vanes in the Rapsodie ones. The regulation in the case of Rapsodie is manual. A self regulation was installed but is not used because it is not sufficiently accurate. Its accuracy is only about 1°C. The operator must manually operate with a variation of the temperature in the range of some tenth of degree because of the reactor temperature coefficient.

6.11 Valve

1) Comment

(i) Bellow valves: The sodium which is inside the pipes supporting bellow valves must be removable instead of avoiding any sodium freezing in the bellows.

2) Questions and Answers

(i) Q. Would you show us the structure of the 6" ϕ valve being used (flow control valve, grobe valve, check valve)?

A. Until now the following valves had been successfully used:

a) Bellow valves with Y form up to 200 mm diameter.

b) Some isolating gate valves with direct flow up to 100 mm diameter with frozen seal are used only in inactive sodium.

c) Non return valves of 200 mm diameter are used with the pumps. For future reactors isolating gate valves of 500 mm diameter and regulating butterfly valves of 400 mm diameter are constructed and tested.

The large pumps will be provided with non return valve of 500 mm diameter, similar to the Rapsodie ones "Globe valves" have never been used in France.

(ii) Q. Would you explain to us the reliability of the valve being used in France and the frequency of the maintenance?

A. Reliability

- Valves type a): reliability is approximatively equal to 100% except when used in bad conditions such as working with frozen sodium or when sodium is frozen or melted with insufficient care.
- Valves b): reliability is equal to about 95% but with an experience of only two years. Difficulties are possible following a long time operation especially for the sliding shaft in the frozen seals.
- Valves c): reliability is equal to 100%.
- Large valves are presently developed. Difficulties were met in Rapsodie with the drive parts of the valves.:

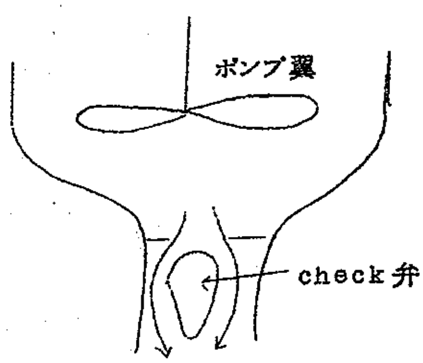
Cardan system, cable, etc..

These parts need a maintenance and especially lubrificationù during reactor shut-down.

(iii) Q. Please show us the structure and reliability of the check valves being installed at the outlet of the primary pump in Rapsodie.

N.B.: Are not there any troubles that the check valve would not open smoothly in the case of the natural convection or of the low speed flow by the pony motor if the check valve is installed to shut down perfectly in the case of the inverse flow? If the check valve would open in this case, would the hydraulic resistance be very large?

A. Rapsodie の check valve は float type である。
Check valve については trouble は経験していない。
Check valve の作動については mock up を作り十分確かめて使用した。



製作に際し water hammer を avoid するように設計した check valve が (Na) 作動したとき少しはリークはある。

フランスでは高速炉に関しては他の type の check valve を使つた経験はない。

maintenance に関しては check valve は pump と一体となつておるので一緒に引き出せる。

check valve の抵抗は他の部分に比較すれば非常に小さい。

(iv) Q. Please show us the speed, the angle and the power of rotation for the rotating plug of the upper part of the core on Rapsodie. Show us the method for connection of the cables in the case of rotation.

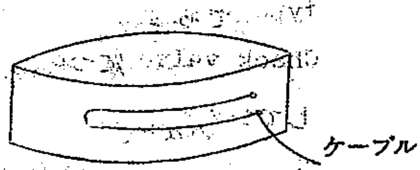
N.B.: We have adopted 1/20 r.p.m., 360° for the outer and inner rotating plugs, and 3.75 kW for the outer plug and 1.5 kW for the inner plug on JFER. We are now discussing that the rotating angle of the inner plug will be limited at 180° (± 90°) to make the treatment be possible for connections of many cables, and that the rotating speed will be increased at 1/10 or 1/5 r.p.m. to make the time of fuel exchange shorten.

A. The angles of rotation for the Rapsodie plugs are:

large plug: 360° with overlapping
small plug: 180° with overlapping

The speed adopted for JFER (1/20 r.p.m.) corresponds to the low speed of the Rapsodie plugs. We have also a high speed which is about ten times higher. The power for rotation seems to be adapted, but depends on various factors, such as friction coefficients etc...

wire はかなりの長さをためておく方式をとつている。



Rotating plugの回転角を合わせるのはマイクロメーターまたはノギスのようなもので合わせている(前はautmaticであつたがうまくいかなかつた)。

Rotating plug の control は control room から行なつている。

6.12 Argon gas system

1) Comments

(i) Ar ガス系は非常に複雑すぎる。もつと単純化すべきである。

Argon circuit can be simplificated by joining components by argon equilibrium pipes instead of primary every component to the pressure regulating tank.

(ii) Rapsodie のガス圧力調整は手動でやつている。自動制御装置ももつているが非常に安定であるので使用していない。hot layer があるので手動で十分間に合つている。Rapsodie は1日2～3回チエツクし手動調整するだけである。

However, we cannot recommend to you to drop definitely argon pressure automatic control. Argon flow rate from and to the reactor must be considered and we do not have a definitive opinion for the future.

(iii) Figures of argon supply to pumps and control rod mechanisms which were considered to JEFR are very large. Argon flow is a source of troubles and must be reduced as low as possible, because, it is very difficult to separate sodium from argon when argon is extracted from reactor and is becoming cold. This results into argon circuit pluggings.

Therefore, we recommend to you:

a) to modify control rod mechanisms, for example by using bellows (not any argon supply). We do not see why bellows are not used, since they do not operate during scram, when bellows were already used during scram in other reactors.

b) to modify pumps and to use pumps with an oil lubrificated mechanical seal. Hispano-Suiza has fabricated for us pumps, the argon leakage of which is zero.

(iv) Argon supply to reactor and argon extraction from reactor must be provided by two separated pipes because:

a) A reactor purge must be possible in case of large fission gas release:

b) If one of the two pipes might be plugged, the other could be used as a spare.

(v) It is more convenient to avoid argon underpressure at the inlet of argon compressor in the purpose of eliminating possibilities of air introduction into argon circuit.

(vi) Plugging of hot argon pipes is one of Rapsodie problems. This was not noticed in other sodium cooled reactors because sodium outlet temperatures were lower or argon was not circulating. This problem is narrowly bounded to the argon clad rupture detection, for that it is necessary to have argon circulating.

It is absolutely necessary to be able to clean argon pipes by sodium in the part of loop going from the reactor to the location where argon is cold. This can be by different ways, for example by raising sodium level.

Rapsodie ではＡｒ系はＮａ管とし設計した。温度は２５０℃まで上昇可能洗浄は１年に１回やつている。配管は傾斜をもたせた。

(vii) Sodium vapor trap is cooled by organic liquid. We do not think it is useful. For Rapsodie, only air natural convection is used. If it is not sufficient, an air blower can be added.

Pay very much attention to the possibility of introducing organic liquids into sodium. This can lead to carbon transportation by sodium and initiate plugging in loops or reactor.

(viii) 回転プラグからのＡｒリークの測定方法としてはactivity の monitor 法が最も良い。

(ix) Ar ガスの消費量について

最大でも ３～５ l/min であろう。 これは container の床上にも漏れているが換気があれば問題ではない。

2) Questions and Answers

(i) Q. 1 Would you show us the criteria of the cover-gas pressure? Would you give us your comments if there are any problems about the criteria of the JEFR?

(2) Would you tell us the regulating method and the design criteria of the cover-gas pressure based on your experience?

A. Criteria adopted for Rapsodie during construction

a) helium tightness

b) pressure slightly higher than pressure in the reactor building in order to avoid the air entrance following small accidental leaks. If such a leak occurs, gas activity allows its detection.

c) Pressure range sufficiently wide in order to allow a simple regulation (+ 30 ± 20 mbars)

d) pressure regulation with recycled argon to reduce argon consumption

e) possibility of washing the pipes with hot sodium where a deposit of sodium vapor is possible.

f) Safety valves calibrated at 1 bar in order to protect the components.

After operation the following modifications have been adopted:

- manual pressure adjustment twice or three times a day instead of self-regulation. The operation is safer and there is no under pressurized device
- improvement of the pipe washing system to make it more efficient and easier to use
- installation of mercury hydraulic valves to avoid that the argon pressure may accidentally raise the sodium level above the shielding plugs in any component. The maximum pressure becomes $\simeq$ 160 mbar instead of 1 bar.

It has been established that it is very difficult to avoid any leak. Consequently it is better to suppress every device which is under pressurized relative to air. On the other hand an argon leak to air atmosphere is easy to detect owing to gas activity. Because it was wanted to avoid any under pressurized device, pressure self regulation was dropped. Manual pressure regulation is done by injecting periodically fresh argon into reactor by actuating the compressor. Spent argon coming from clad failure detection is now rejected to the stack, because the clad failure detection circuit is partially under pressurized compared to air and may be contamined by it. Previously, this argon was recycled, resulting into oxygen introductions.

(ii) Q. ① Would you tell us your opinions if there are any comments on the consumption quantity of the gas of the JEFR?

② Would you tell us the regulating method of the gas consuption quantity, based on your experience, as follows?

(example)

service	radiation	consumption quantity
ⓐ for purging during the fuel handling		
ⓑ for the seal of the control rod axes		
ⓒ for the seal of the pump axes		
⋮		

A. The most part of the consumed argon corresponds to the sampling of CFD ($\simeq$ 1 m^3/h) and to fuel and components handling operation especially during argon purge.

There is no gas supply flow neither for the pumps nor for the control rod mechanisms.

Ar ガスの使用量はRapsodie では1 m^3/h (JEFR 70m^3/day)

Ar ガスを流すところはtrouble のもとゝなるし経費も高くつくのでできるだけ少なくすべきである。

Rapsodie の pump にはoil mechanical seal を使つておりleak 量は0である。oil mechanical seal がよい。

制御棒にはベローを使いleak 量は0である。

(iii) Q. Would you give us your comments on the number and the volume of the Argon gas tank of the JEFR?

A. Tanks number and volume may be reduced compared to the design adopted for Rapsodie. The following items may be proposed:

- suppression of tank T8
- only one tank instead of tanks T7 and T13. (The corresponding volume is about 30 m^3 in Rapsodie)
- one other tank instead of tanks T25, T26, T27 (The corresponding volume is about 4 m^3 in Rapsodie)

(iv) Q. Would you give us your comments on the gas refining system? Would you tell us the criteria of the system?

A. The argon purification system of Rapsodie consists of a simple NaK bubbling-device including a NaK aerosol trap.

(v) Q. ① Would you tell us your method, based on your experience, of the gas disposal in the case of fuel rupture?

② Would you give us your opinions about that the cover-gas being contaminated by the fission products are transfered to the decay tank?

A. In Rapsodie, there are two desactivation tanks. The volume of each is about 10 m^3. They have never been used up to now. The desactivation in the buffer-tanks, corresponding to T7 and T13, being yet sufficient for rejection to the stack.

It must not be forgotten that fission gas leaks from fuel have not been important until now.

Rapsodie ではフイルターにチヤコールフイルターを使用している。

Rapsodie の原子炉のＡｒ系配管は２本あり１本は通常用（１m^3／h）他の１本は緊急用でパージ用として使用している。パージ用を設けておく必要があるだろう。

(vi) Q. Would you tell us your opinions on the efficiency of the cleaning of the vapor trap and the gas piping by sodium.

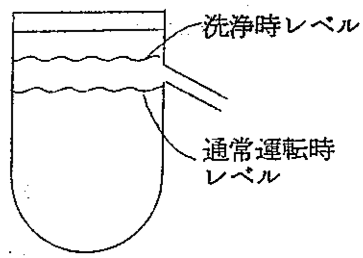
A. The efficiency of cleaning by sodium is very good and the operation is absolutely necessary if the plugging of the argon piping appears. The period between two cleanings depends very much on argon purity, i.e., on oxygen content and it is now impossible to give any rule.

(vii) Q. It is supposed that the cleaning will be required once a year. Would you give us your comments on the time interval between cleanings?

A. The efficiency of cleaning by sodium is very good and the operation is absolutely necessary if the plugging of the argon piping appears. The period between two cleanings depends very much on argon purity, i.e., on oxygen content and it is now impossible to give any rule.

Rapsodie は運転に入つて18ケ月間の間に3～4回やつた。しかし臨界になる前はやつていない。

洗浄方法はＡｒ系の配管が右図のように通常Ｎａレベルよりやゝ高い位置にありかつ傾斜をもたせているので洗浄時Ｎａレベルを上げるだけで洗浄可能である。



(viii) Q. Although we have no experience on the cleaning by hot sodium, we also have another idea that it will not be necessary to have the vapor trap adjacent to the equipment when the gas lines are cleaned by sodium. Would you give us your comments on the idea?

A. This question has not been well understood. What "equipment" do you mean?

(ix) Q. How do you think of that sodium vapour trap is needed in the gas outlet of sodium and Na-K system tanks?

How should we settle a design standard?

A. A sodium vapor trap is required every time argon must be removed from a hot sodium circuit, let us say more than 350°C. The trap connects the part of the circuit included in the sodium circuit and the part of the argon circuit which is cold and clean.

(x) Q. ① Would you tell us the information on the linear velocity of Argon gas in the Rashing Ring Packed tower?

② Would you tell us the information on the checking method of the end point of the tower?

③ Do you think it is necessary to take off sodium oxide (Na_2O) as product continuously?

④ Would you tell us the information on the minimum size of the Rashing Ring? (for avoiding the clogging the tower)

⑤ Would you tell us the information if there are some materials of the eliminater of the liquid sodium as entrainment?

⑥ Would you give us your comments on the equilibrium relation between sodium oxide and oxygen in Argon cover gas.

A. On Rapsodie, we use a NaK bubbler filled with KNITT mesh, followed by a sintered stainless steel filter. Flow is 6 m^3/h. Different devices are now in development for larger flows, but we still have no experience of them.

(xi) Q. Equipment in argon gas with sodium vapor.

Please let us know cares to be taken or special provisions to be made for the equipment in argon gas with sodium vapor.

A.

6.13 Pre-heating

1) Comment

(i) Loops preheating is designed for an equilibrium temperature (250 to 300°C), but the loops are filled with sodium before equilibrium is realied.

時間の節約になるし温度の均一化にも役立つ。

2) Questions and Answers

(i) Q. ① What degrees do you recommend the pre-heating temperature of the components in sodium, based on your experience?

② Would you tell us your opinions if you have any comments on the pre-heating temperature of 250°C of the JEFR.

A. Reheating temperature must be from 120 to 300°C relative to the components:

a) 120°C for the sodium storage tank and the cold traps

b) 150°C for the large components

c) 150-250°C for the pipings. Large temperature range may simplify preheating design

d) 200-300°C for the small pipings of the sodium purification circuits

After experience, the electrical preheating system appears very reliable if the electrical resistances are of good quality and not too much loaded; 1/4 of the limit power when resistances are used in air with natural convection, is only authorized for piping preheating. The installation of a gas preheating system is expensive because it requires complicated double-piping where free flow must be insured. The double wall has important diameter and need room in the cells where active sodium circuits are located.

The gas preheating cannot be used in case of sodium leak from the internal pipe.

(ii) Q. The reactor vessel and the main primary sodium piping of the JEFR are designed to be pre-heated by hot nitrogen gas. While other equipment and piping containing sodium will be pre-heated by electrical resistance heaters.

Would you give us your comments on the reliability and the economic comparison of the two preheating methods?

A.

(iii) Q. Would you give us your opinion if there are any problems on the pre-heating system when it is used as the emergency system?

A. The disadvantages of this solution are:

- It cannot be used in case of sodium leak.
- It needs an important power for the gas flow.
- It needs installation of large gas valves at high temperature.

6.14 Double tube

1) Comment

No comment

2) Questions and Answers

(i) Q. Would you show us the supporting of the double tubes.

A. Principle:

The sodium pipe is centered in the double pipe either by welded fixed anchorings when no relative displacement is admitted, either by centering rings sliding inside the double wall. Bellows on the double wall absorb the differential longitudinal expansions.

The pipe supports, springs or counterweight system is realized at anchoring or guiding points. The anchoring is done by conical sleeves for better distribution of the stresses due to radial differential expansions between piping and double wall which may be at different temperature.

see 6.2 the questions and answers (iii).

(ii) Q. Would you give us your recommendation on the construction of the double tubes?

A. The prefabrication and the preliminary tightness tests of the sections must be done so that the number of in-pile tests should be reduced as much as possible for the following reason:

- at each section of the piping which must be tested in the reactor building, a supplementary portion of the double wall must be provided to receive two end closures necessary to close the internal volume during tests.

(iii) Q. The primary sodium cooling system is clasified into the following categories.

1. Main circulating system.
2. Emergency cooling system.
3. Auxiliary piping systems, for instance, sodium purifying system, sodium charging system, plugging meters etc.

Among them, what is the system that requires the double wall piping from the safety consideration?

A. a) The actual objective of the double wall is to limit the quantity of sodium which may leak in order to insure the cooling of the reactor, so, all the auxiliary circuits can be realized without double wall provided that a leak cannot lead to draining of the primary circuit.

Therefore, for Rapsodie, a double wall has been placed around the small draining pipes until the section which is frozen during normal operation.

b) The fire protection realized by double wall can also be done by the nitrogen filling of the cells where active sodium circuits are located, or at least the cells where sodium normally flows such as purification and overflow circuit cells because there are the most dangerous locations.

(iv) Q. How high must the design temperature be divided for outer wall lead wire penetration parts, in the case when sodium design temperature is 550°C? Is it 550°C or lower than that?

A. It has been established that the leak detection plugs located on the double walls of Rapsodie are not sufficiently reliable because their contact with high temperature sodium following a leak may cause their failure. Consequently, it is necessary to design all the materials of the double wall for the sodium maximum temperature.

(v) Q. Please show us where you use the double tube for the coolant system, and show us the design criteria in Rapsodie.

N.B.: We are now in mind of adopting double tube for the primary systems and for the emergency system, and adopting single tube for the secondary system. We had considered on the early design that the double wall bubes would be offered for the leak detection of the sodium and for the passage-way of heated nitrogen. But we are now discussing on the assumptions that the outer tube would not break even if the inner one would break, and we consider the maximum credible accident for this trouble. We consider the hypothetical accident in the case that the both of the inner and the outer tubes would happen the guillotine breakage.

A. The double wall pipes are used only for the primary cooling system, and for the part of secondary cooling system located inside of the reactor building. The double tube has been designed essentially in order to detect the sodium leakage. The external tube is divided into a series of separate compartments in order to limit the leakage. In case of rupture of the primary system of Rapsodie inside the safety vessel, additional sodium will be injected in the reactor vessel to maintain the sodium level over the core and allow circulation.

理由は sodium fire が起つた場合 reactor building の外では消火作業もできるが中ではできないからである。1次系でも2重管にしているのは主配管のみで purification system などの小さいのは2重ではない。

purification system は over flow tank から取出し reactor に再注入している。

Phenix でも2次系は一重の予定である。

Outer pipe の design criteria は inner pipe からのリークがあつても十分こ
これを保有できる設計になつている。

(vi) Q. Thermocouple wells for double walled equipment.

Did you use double walled wells for thermocouples in double walled equipment in Rapsodie plant?

A. No, we don't

(vii) Q. The double tube structures are adopted and the leak detectors are installed on the primary and the secondary pipes as the counterplans for the leak on the JEFR.

Would you give us your opinions if there are any problems on the JEFR?

A. This question is not well understood.

For Rapsodie, we have the following system on the primary circuits:

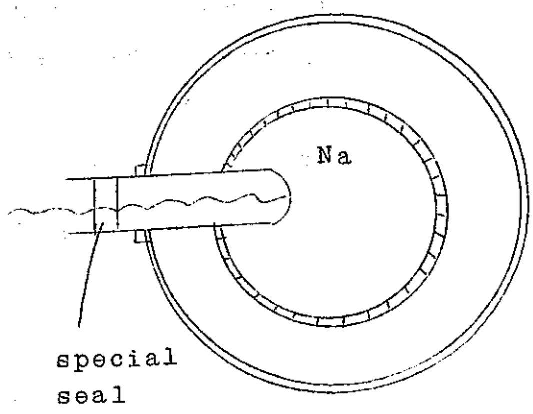
- plugs on the double walls
- insulated wire detection system on the auxiliary pipings

On the secondary circuits, there are insulated wire detectors.

These insulated wire detectors are presently tested in order to improve their reliability and to reduce their cost.

3) Information

Rapsodie で用いている2重管へ取付けは右図のとおりである。



6.15 EMP & EMF

1) Comment

(i) It does not seem very useful to double the e.m. pumps. It is better to use a more reliable pump. There, several valves and pipe lengths are dropped.

2つ以上の場合，他の停止中の保温の問題が生じてくる。

Rapsodie では primary に1つ secondary に1つである。事故時の Na レベルの保持は reservoir tank に圧力（ N_2 左）をかけて炉心へ流れ込む方式をとつている（この方式では reservoir tank を container の外に置けるし炉心より高い位置に置く必要もない）。reservoir tank は常時加熱 Na レベルはタンク－4993 mm 炉－4460 mm である。

(ii) EMF を tank の上に置くのはよくない。

electric current が切れた場合，中が空になつて使えなくなる。また EMF をタンクレベルまで下げるのはポンプの近くでのリークを考慮し，さらにポンプ用タンクおよびバルブを設ける必要があるのでよくない。

Rapsodie ではオーバーフロータンクの中に設置してある。かなり高価ではあつた。

(iii) モーターは conduction type はよくない。

induction type では low pressure で作動できる。

Rapsodie では直流の induction type を使用している。

2) Questions and Answers

(i) Q. The double sealing wall piping is often used in sodium loop.

In the case, what structures and materials are used for EMP and EMF, especially for coil cooling method, electrical load wire penetration parts in Europe?

A. a) GAAA company manufactures double walled flowmeters.

b) On Rapsodie only one e.m. pump is used, immersed in the overflow tank. This pump is double walled to separate the electrical part from sodium, but there is no cooling; the electrical part is designed for withstanding a 550°C maximum temperature. This double walled d.c. pump could be used without being immersed. It is also manufactured by GAAA Company.

(ii) Q. Installation or layout of EMP and EMF in the sodium piping.

High operating temperatures of the sodium system cause large thermal expansions and stresses in the piping. EMP and EMF are the weakest components in the system, but they are heavy because of their electrical structure such as coils.

Therefore how to install and support them are big problems in the fast reactor plant.

What is your practice in setting and supporting EMP and EMF to avoid large thermal stresses on them?

A. Heavy electromagnetic pumps are fixed. Small electromagnetic pumps have a sliding support allowing displacement only along one axis.

Electromagnetic flow meters are supported by the pipes. If the pipes are in horizontal position, elastic supports may be used.

(iii) Q. Sealing method and materials for electrical leads of double walled EMP and EMF.

Safety features require double walled EMP's and EMF's for reactor services. If the inner wall fails, hot sodium, 500°C for example, will be in contact with the conduits or electrical leads.

Let us know the construction and the sealings materials for the penetrations of electrical leads through the double walls.

A. We never used an electromagnetic pump with a double wall.
For the electromagnetic flowmeters, the coils are outside the double wall.

(iv) Q. Saddle type EMF for high temperature services.

The coils of saddle type EMF's are exposed to high temperature atmosphere.

① Do you cool the coils by forced circulation of gas, or,

② Do you keep the coils cold by setting heat insulation between the pipes and coils, or

③ Do you have any special high temperature resistant coils?

④ How do you insulate the heat and what is the insulation materials when ② is applicable?

⑤ What type of electrical insulating materials do you use when ③ is applicable?

A. From figures, we saw during argonne meeting in November 1968, we think this type of electromagnetic flow meter is different from those we use. The coils of our electromagnetic flowmeters are not cooled by forced convection. Special insulation is used by the manufacturer "GAAA", when temperature is high.

(v) Q. Multiple EMF's for reactor protection system.

Spaces are limited in the containment vessel. Therefore, it is very difficult to install multiple electro-magnetic flow meter heads in a pipe line within the containment vessel. Did you install multiple EMF heads in a pipe line for redundancy in reactor protection system? How did you handle this if you did not use multiple heads.

A. In Rapsodie, we have just one electromagnetic flowmeter which is not removable and it is not working now because of an electric circuit failure.

In such a case, we recommend to use only one permanent magnet and 3 couples of electrodes.

But it is better by far to use only removable electromagnetic flowmeters.

3) Information

(i) Rapsodie は EMF と permanent magnetic flow meter の2種類を使用している。EMFの電力は1KVA(100mmφ管)である。

7. 計測制御設計

7.1 General comments

1) General organization and philosophy of instrumentation and control

It appears that your philosophy is quite different regarding instrumentation and control:

- for instrumentation you intend to use well established techniques, and on this we agree,
- for control, a wide use of computer technique is envisaged, so that JEFR should be one of the most automatized reactor in the world, and we suggest some more manual and simple way of operation.

If JEFR were, say, the third fast reactor built in Japan, we should agree with the proposed organization, the design of control you have made beeing good except for little details. The misfit, from our point of view, lies rather on a question of philosophy.

We shall see now in more details the different parts of the system. We sometimes refer in what follows to two papers given to your representatives:

(ref. 1) JEFR - réponse au questionnaire, Instrumentation et contrôle

(ref. 2) JEFR - Remarques sur l'instrumentation et le contrôle.

2) Neutron detection instrumentation

(i) General organization

For both control and safety, you foresee several instrumentataion channels:

1. 2 channels for start up with fission chambers using time scalers, logarithmic counters and period meters on a range of about 6 decades
2. 3 channels for intermediate power with fission chambers using logarithmic amplifier, period meter and Campbell method for linear measurement on a range of about 5 decades.

(3) 3 channels for linear power with compensated ionisation chambers using linear amplifier on a range of about 3 decades.

Each detector can be moved vertically with an accuray of 1 mm thus allowing possible variation of the flux received on 8 decades.

There are probably as many different organisations for neutron instrumentation as reactors built. The system chosen is quite classical for start up and for linear power.

For intermediate power channels the use of Campbell mechod may be a good solution if experimental before during sufficient time on a reactor. The delay time of such a system (we found during recent experiment 100 msec for level and 4 seconds for period) is large alternative solution may consist to enlarge the range of linear channels up to five decades with efficient gamma shielding. Another important remak we can do on your system is the following:

the flux must be measured on nearly 12 decades (one decade below the source to half a decade over the nominal). It is hard to cover such a range with only two different kinds of channels, for logarithmic and period measures. Derived from your organization, two alternative solution may fulfilled sufficient overlapping:

1st - Placing behind each compensated ionisation chamber a log amplifier and its associated periodmeter (the use in parallel of log and linear amplifier is possible with some precautions).

So the range of measure of the so called intermediate power channels may be extended toward the lower valves.

2nd - Using the intermediate power channels for the whole range of 12 decades by vertical displacement of the detectors from the medium plane of the core to a level where the flux is 8 decades below. In this case, 3 fixed positions may for instance be used and the counting method employed. The so called start up system may be suppressed.

We prefer the 1st solution, for safety reasons.

(ii) Source

We agree with the use of an external source placed in the radial blanket as we guessed.

It is of great interest to examine if what we call the "inherent source" chiefly due to 240 Pu with a new core and later increased by generation of 242 Cm, has a sufficient strength so that it is possible to avoid any external source. From another point of view much more sensitive detectors than fission chambers can be used for start up (for instance BF-3 counter: gain 100, but with more severe problems of temperature and gamma flux so that the "inherent" source may be used. With or without external source, an important problem is the possible error on the estimated neutron flux, reactor with all control rods inserted (= stopped) before the first criticality.

(iii) Detectors

We agree with your position concerning detectors cooling: it is not necessary for fission chambers (we use in France fission chambers specially built so operate up to 400°C and in a near future up to 600°C) but it is wise to cool ionisation chambers, even designed for hot temperature. We suggest the use of nitrogen or air rather than helium as coolant, chiefly for leakage problems.

Gamma shielding around the detectors is necessary or at least helpful for two reasons: precision of the neutron measurement and damage (life shortened) for the detector itself. Lead or other gamma absorbing materials (such as tungsten alloys) may be used.

(iv) Deviation detecting average circuit

This system seems rather complicated due to the fact that the mean value of the 3 channels is used for automatic control of neutron flux.

If no such automatic control is done, as we suggest later, the D.D.A.C. can be simplified and used only so indicate the failure of a channel (in case of two failures however the reactor must be automatically stopped).

(v) Neutron safety circuits

Safety and control information are issued from the same channels in your design. For start up and intermediate power, we do so, but not for the linear power range. Our philosophy is that safety and control must not interfere during long time and we prefer adding two more channels to control linear power. Two more remarks concern the interest of:

- an ionisation chamber, far from the core, for measures in case of accidental power excursion (with the appropriate channel)
- the measure of reactivity (a rapide step in reactivity may indicate the beginning of an accident such as fusion of a sub-assembly, etc...)

3) Control and safety rods

We discuss here only the general organization for the use of control rods, calculation and mechanical structure being seen else where.

The design of JEFR envisage twelve control rods:

- 2, for fine control (0,5 $/rod)
- 6, for coarse control (0.67 $/rod)
- 4, for safety (2 $/rod)

Both safety and coarse control rods may be scrammed.

It appears clearly that this organization derives directly from EFFBR principle following which any rod used for control (fine or coarse) must have a value less than 1 $ and even as near as possible of 0,5 $ so that never prompt critical situation may occur due to accidental fast removal of one and perhaps two control rods.

This was the major safety preoccupation in earlier times when the behavior of fast reactors were greatly suspected near prompt criticality.

Now, a best knowledge of such situations has been gathered from various theoretical studies and from experience so that which a good mechanical design for rod mechanism and some safety precautions it is possible to consider that the use of values higher than 1 $ for control rods is possible. So, the total number or rods may be reduced (almost divided by 2) by using multifunction rods (safety and control) having the same characteristics (value, speed, etc...). The advantages of such a conception are briefly listed here after:

- its less expensive
- a better package of reactivity for burn up (factor 2) is obtained
- a better package of reactivity for safety (factor 2) is gathered
- there is faster insertion of rods already partially inserted in core in case of scram
- an easier detection of stuckings on jammings is possible

We suggest too that the use of special rods for fine control is perhaps not necessary, even if there is a regulation on temperature or flux.

4) Process Instrumentation

(i) Sodium instrumentation:

The sodium instrumentation which is proposed in the JEFR conceptual design is in general agreement with the one we use in Rapsodie and we have just to make four remarks:

a) We agree with using chromel-alumel thermocouples and platinum thermoresistances: thermocouples measure core and sodium loop temperature, and thermoresistances are used for thermal balances.

b) In Rapsodie, we have no sodium pressure measurements and we do not need them for normal operation as long as the pump characteristics are known. We are not in favour of such measurements in a reactor.

c) Strain gages detectors can be located on external side of the reactor vessel. In this case, they may be not unremovable.

In the case where you envisage to put strain gages inside the reactor vessel, they have to be removable we do not know any good solution for such a use that we do not recommend to you.

d) In Rapsodie, there are just temperature measurements above the fuel subassemblies. Some blanket subassemblies are instrumented because they are under the core cover plate. This procedure is not very logical since a plugging may cause a sodium accidental heating even in the blanket zone. But the cost of a large cover plate is too high when the vessel diameter must be increased to accommodate it, as in Rapsodie.

e) Detection by delayed neutrons:

- Moderator: graphite is not the best moderator and it could be suitable to use another material, if there are no special difficulties with temperature, or if the gamma level cannot give a problem with gamma-n reaction

- Outside radiations shielding (gamma and neutrons):

This point has to be kept in mind in order to get better signal/noise a ratio.

It could be better to have more than one BF-3 counter. Under certain conditions, sensitivity could be largely increased by using more than one counter.

Important parameters not given in the project, are:

- sodium volume inside the detector system
- sensitivity of the VF-3 counter
- transit time of the sodium
- dimensions of the moderator-block
- flow rate of the sodium.

f) Detection by gas fission products

It does not seem appropriate to make a gamma measurement on solid fission products by means of a NaI scintillator. Beta measurement by proper scintillator with a simple electronic system seems more useful. By means of a detector with rotating electrode differential measurement is possible.

Important parameters not given in the project are:

- sodium trapping (vapors and aerosols) on the argon, before the gas reaches the detector system

- volume of the chamber for ions collection
- clean gas purging under the scintillator
- flow rate and temperature of the argon gas coming from the reactor

- transit time of the argon gas
- collection time or measurement procedures
- detection efficiency or sensitivity to Kr_{88} for example
- signal analysis method: ictometer or spectrometer.

Your design is more conservative, but probably more expensive.

g) The distribution of temperature measurements in the reactor and sodium loop seems good. But the thermocouple number is large. We guess a large part of these thermocouples are located on the vessel.

It would be worth while trying to discuss the play of every thermocouple and to reduce their number and, by this way, their price.

h) For the instrumentation of the primary sodium purification, see remarks on cooling system.

i) The argon gas pressure regulation seems too complex.

(ii) Failed Fuel detection system:

We agree with the general organisation for JFER failed fuel detecting system where both delayed neutron detecting method and cover gas method are used independently. Some details are discussed p. 4 and 5 (réf. 2).

We have two general remarks to add:

- Important studies and experiments are to be done for failed fuel detecting system, for both methods
- The problem of failed fuel identification and localisation system (inside or outside the reactor) is not yet resolved, and an important effort is here also to be done.

It would be important, at last, to pay attention to different ways of surveying the contamination of argon and sodium circuits due to defective fuel elements.

5) Control system

(i) General:

We discuss here after points at of stated in chaper 4 paragraph 1.

a) we agree with the use of control rods for changing the power level of the reactor.

b) the reactor sodium coolant inlet temperature should be manually, and not automatically, hold constant.

c) the maximum time rate of change of reactor sodium coolant outlet temperature of 50°C/h is very low for operation (150°C/h is possible without problem).

d) the use of a constant sodium coolant flow rate (equal to nominal value) is good during start up and power level rise or change. The use of variation of sodium flow rate is not necessary, so that constant speed motors can be used for sodium pumps.

e) OK (primary and secondary Na flows equals)

f) OK (use of a heater for reaching the isothermal temperature of 350°C).

(ii) Automatic versus manual operation:

The use of on line computer system is discussed later on (see paragraph 7). We do not think that automatic reactor operation is necessary for reactors like JFER, Rapsodie which are not connected with electrical plant. More, we do not plan such a feature for our Prototype Phenix.

(electrical power 250 MW).

In fact our thought is that a best knowledge may be gathered on reactors of new type if manual operation is done rather than automatic one.

There is no real difficulty for manual operation with a fast sodium cooled reactor. Stability is high, changes in temperature are slow and there is no need for frequent changes in power level for fuel irradiation.

(iii) Regulating devices:

The same philosophy may be applied for regulation. There is no real need for power (neutron flux), inlet or outlet core temperature regulation.

In Rapsodie, we built a neutron flux regulating device and a temperature (outlet sodium temperature of air exchanger) regulating system, we finally do not use them because variations of power and sodium temperatures are week.

Furthermore it is rather hard to get sodium temperature regulation performing well and some unexpected effects may arise in particular situations.

6) Safety system:

For the safety system of JFER, six different automatic actions are envisaged: isolation, fast scram, slow scram, set back, alarm level 1, alarm level 2; the first two beeing out of the computer functions while the others beeing actuated by the computer according to our philosophy, and what we do, we can raise the following remarks on this system:

a) the number of different automatic actions is much too high, some of them may be manual (isolation, set back) in every cases and other in some cases (slow scram).

b) as far as possible; slow scram must be actuated by separate analog system and not by the computer.

c) the number of abnormalities actuating fast scram must be minimum. In several cases slow scram may be used instead of fast scram.

d) fuel assembly outlet coolant temperature which is actuating both slow and fast scram, must be treated by the computer system because it's not in practically possible to do it by analog way. So, great care, must be taken (periodic check non interference with other programms and so on) for this treatment.

Others functions of the computer in the field of safety are discussed in the next paragraph.

e) as a general rule the use of 2/3 is good for scram but not always recessary for slow scram.

f) trip level must not be too close to each other (i.e. 105% and 108% for neutron flux too high).

7) Computer system:

(i) Computer functions:

We guess that main computer functions for JFER are the following:

a) Date logging (normal and emergency records)
b) On line performance calculations
c) Automatic start up and power level change
d) Sequence interlock (for start up and power)
e) System analysis in emergency cases
f) Experimental data analysis

We think that this is a very ambitious task and that probably no such system works at present time, though no doubt it will be the case in a near future. It is just the step before a fully automatized plant and there is a great interest in building it.

The question his in the opportunity of developping it for a reactor of a new type. Whose operating procedures and accidental evolutions are unknown (or just predicted by theoretical studies), for the first start up of which an important effort is to be expected if classical control methods are used so that extensive use of on line computers may considerably enhance the difficulties by adding the proper problems of a new control method to those lying in the plant itself.

Considering the heavy load assigned to the computers, considerable problems of software are to be expected (use of machine language, connection between computers, number of reactor situations, number of functions, ...)

We suggest:

a) A large use of computer system for data logging, for normal and emergency records, and specially for subassemblies temperature outlet.

b) A very limited on line performance calculations (i.e. the integrated power and no more), normal performance calculations being made on external computers (the same used for other theoretical studies not necessarily located on the site).

c) Neither automatic start up, nor automatic power level change, because normal manual operation is simple and because normal frequency ofsuch operations is low.

d) A very limited sequence interlock for start up and power, with great confidence for the use of operation check up lists, the computer performing a very general supervisory control function with very limited action (i.e. on the control rods).

e) No automatic system analysis in emergency, the operator having the whole responsibility of doing it.

f) The use of a separate small computer for experimental data analysis if possible located on the site.

(ii) Organisation:

Use of two computers, sharing the work, is good. Connection between them must be limited for sake of simplicity. It is perhaps more reliable to treat in parallel in both computer the most important date (i.e. core temperatures and 10% of monitoring data).

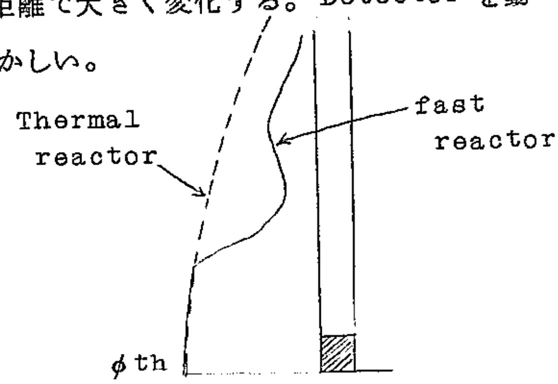
8) Questions and answers

(i) Q. 2)の(I) quite classical という意味について説明されたし。

A. Intermediate region に Campbell method を使うことは充分気をつける必要がある。Phenix には使うつもりはない。

(ii) Q. (2) の(I)の終り) decade に関する Safety reasons とは何か。

A. Thermal reactor 中における flux は Vertical Axis に対して linear に変るが、fast reactor では小さい距離で大きく変化する。Detector を動かす場合に正しい測定をうることがむずかしい。



(iii) Q. 3)の部分でRapsodieの現状について話してもらいたい。

A. Rapsodieのcontrol rodは現在5本で、全部Safety Shim rodであるが、将来は、6本共同でするであろう。

引き抜き速度　：　0.3 ㎜／sec.

挿入速度　：　2　㎜／sec.

1本づつ可動であるが、挿入方向では全部同時に出来る。引き抜き方向では同時にはできない。

one rodのmax. insertion ratioは1.5 pcm/secである。

Phenixは6 rodsが同じ機能を持ち、operation, safety, scram and slow scramの役目をもつ。

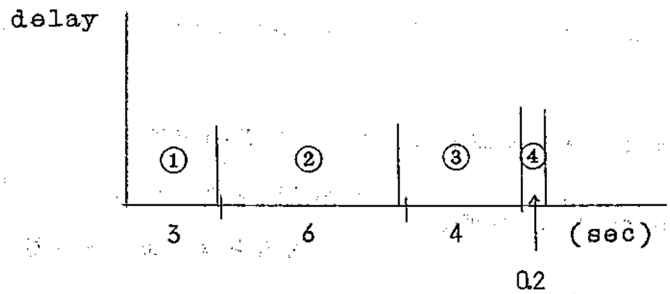
(iv) Q. (4)、(i)、g) thermocouple number in largeとあるが、具体的にはどの部分のことか。

A。 Rapsodieではvesselのまわりに16ケ、Shield plug内に100ケ設置してある。

炉心上部に設置されているthermo－coupleは、1燃料集合体あたり、Rapsodieの場合は1本、Phenixの場合は2本である。Rapsodieには合計85本設置されている。1 cycle/secで測定し、calculatorには2 secおきにsignalを送る。

Phenixの場合は2つのsmall computorによりand回路でscramするが、Rapsodieではthermo－coupleが集合体当り1つしかないので、2回目のsignalが規定値をこえたときにscram signalとなる。

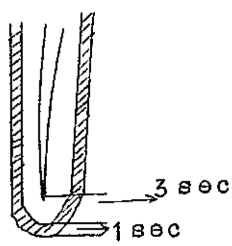
Thermo－coupleによるdelay timeは下図の如くなる。



① subassembly内のthermal capacityによるもの。この値はblanket領域では10 secである。

② thermo－coupleによるもの。

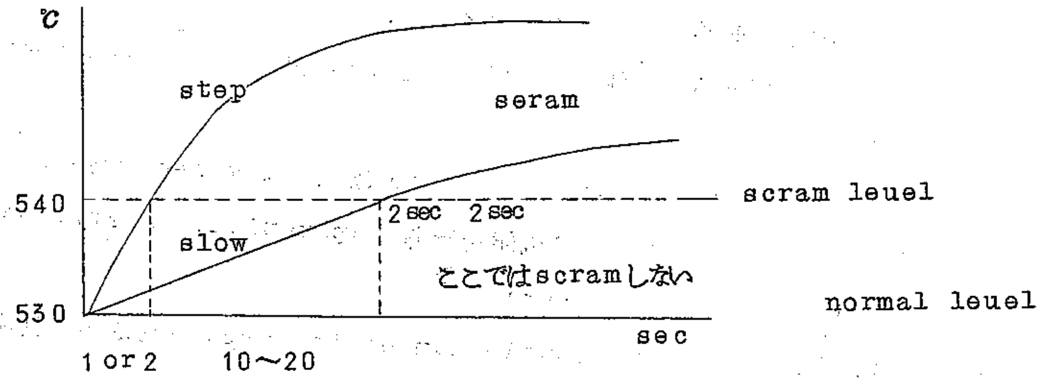
熱電対挿入管によるものは右図の如く計4 secとなる。



③ 3 sec ＜ signal ＜ 4 secの範囲でcomputorに伝わるもの。これはreconformationも含む。

④ unrached によるもの

温度対時間の関係を示すと下図の如くなる。



step 状の場合、1 ＋ 4 ＝ 5 sec で scram

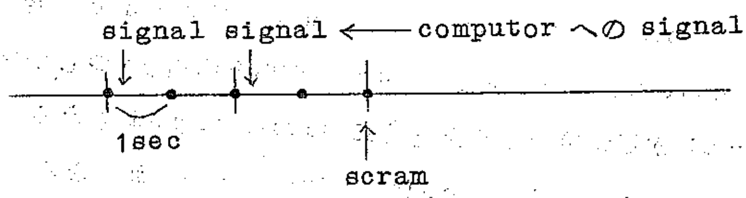
4 sec は signal が computor のところに来てから。

scram action が起こる迄の時間

slow step の場合、1 0 ＋ 4 ＝ 1 4 sec で scram

Computor に signal が送られてから scram に対する迄の時間は $3 < t < 4$ (sec)

となる。下図を参照せよ。

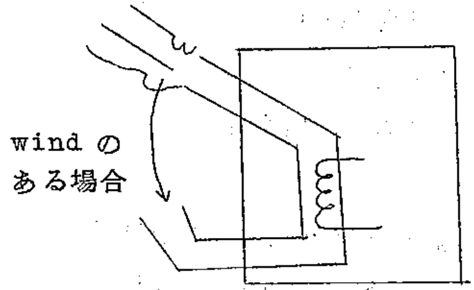


(V) Q. (5)、(iii)) power and sodium temperature の variation について実際例を示めせよ。

A. power の変動：2 4 M W運転時に± 0. 1 M W の変動

inlet temperature の変動：{ 風がないとき ± 0. 2 ℃以内 / 風があるとき ± 4 ～ 5 ℃ }

no wind の場合



7.2 Neutron counters

1) Comment

No comment

2) Questions and answers

(i) Q. In our design, the fission counters for high temperature are used without cooling as the detectors for startup and intermediate power range. Do you think that the endurance for high temperature is sufficient?

What kind of causes are there in the counting error of high temperature? If it is necessary to cool the detectors, how much is the most probable temperature for cooling?

A. For a maximum temperature of $300^{\circ}C$ no cooling is necessary for fission chambers like british chambers mentioned in JEFR report or CFU-6 chambers made in France. In such temperature conditions good operation of these chambers is not a problem and it would be no errors for the counting.

However, for Rapsodie, we were very happy to have standard channels provided with cooling for all detectors (fission chambers and ionisation chambers). So we could use for the reactor start up, CFUA2 fission chambers which are not designed to operate at high temperature, but which have a better sensitivity than CFU3 chambers, which were initially intended.

(ii) Q. In our design, the detection of neutron in the intermediate power range is performed by the vibrational method. Is it considered to be suitable for reactor control? In addition to this method, is it necessary to use the counting detection formula? What is your countermeasure for the life of the detectors (F.C.) which are used in the vibrational method?

A. We do not know the vibrational method, but we suppose that it is closed to what we call the fluctuations method (also called Campbell method). We think that this measurement technique is quite suitable in the range of intermediate fluxes. However, we have to say that up to now we have only limited experience of operation of such instrumentation especially with a reactor. If such an instrumentation is used, it would be sufficient by itself and does not need to be implemented by impulsion chambers. But it would be convenient to provide some sufficient overlapping with low level and high level channels, which does not seem to be a problem. To keep at a sufficiently low level, the noise due to fission products activity in the detectors, it would be suitable to put the detectors outside the high fluxes (by a factor of 10^3) by some appropriate system during nominal power operation of the reactor.

(iii) Q. In our design, the γ-ray compensated ion chamber used for the linear power instrumentation is cooled by helium gas. At the atmosphere of the high γ-ray levels in the fast reactor, is it possible to measure the linear power by this method? Kindly show us your comments about the problems in this case, especially the helium leak, and about the other suitable formula for cooling from the comparison with the case of Rapsodie.

A. It is quite possible to use compensated ionisation chambers for power linear channels.

Helium cooling seems an expensive solution and helium leakage could be a problem. At Rapsodie, the neutron chambers are not cooled by themselves. Only the thimbles in which they are inserted are cooled by nitrogen blowing (standard nitrogen used for cooling the shielding of the reactor). Temperature is under 50°C.

No corrosion was observed on the detectors after two years of operation.

(iv) Q. We would like to use a CIC up to five decades from the full power range from the point of view of control design (CIC has better response and less vibration than the pulse counting method or the Campbell method). Is it considered to be possible by the γ-ray shielding?

A. It is possible to use a compensated ionisation chamber on a range of 5 decades, but due to the high gamma fluxes, an appropriate gamma shielding is to be provided to protect the chambers.

If there is some cooling of the chamber, a lead shell around the chamber is the best solution (a lead thickness of the order of 5 cm gives an attenuation of the gamma fluxes by factor of 10). For temperatures above 1001C, the piece of lead is to be put in sealed container, or another material with higher melting point is to be used (for example, denal which is a tungsten alloy).

(v) Q. Characteristics of fission counters and γ-ray compensated ion chambers are shown in Volume 7. Which is the best (not only the tabulated ones) for neutron instrumentation? Kindly show us this reasons and the situation in which these counters are used in France? And is it considered to be possible to use the γ-ray compensated ion chamber R-12 without cooling? Kindly show us what is the reason if it is so and what is the problem in this case?

A. Considering the characteristics of the fission chambers and of the compensated ionisation chambers given in the volume 7, these chambers seem very suitable. We have no large experience with foreign material and use essentially detectors made in France. Especially we have no experience with the RC 12 chambers from Plessey UK, but the technology of these chambers seems suitable for an operation at 300°C.

A catalogue of neutron detectors made in France has be given to Mr. INOUE.

(vi) Q. Kindly show us the proper methods for γ-ray shield to detectors. And show us your opinions about the junction of a cable with a nuclear counting detector, and about preventing to weaken the joining position, in comparison with the case of Rapsodie.

A. At Rapsodie, the detectors are located in special thimbles inside the concrete shielding. Nitrogen cooling is provided and a lead shielding is provided around the thimble, at the level of the detectors. Electrical connections with the detector are an important point, and the operating conditions of this connection have to be considered (temperature, radiation level, gas atmosphere). These parameters have a great importance for problems such as sealing and oxydation resistance of the connections. Finally, the electrical characteristics of the system could be largely dependent of the good solution of these problems. At Rapsodie with temperatures of 50°C and with nitrogen cooling, the connections seem not to be a problem. Only some difficulties was encountered due to a poor access to the cables inside the shielding plugs of the channels.

Handling of the neutron chambers are made by the rotating crane of the reactor building. Two chambers are handled together. The system which is removed consists of: the chambers, the shielding plug, and about 5 meters of special cable (radiation resisting cable). After removal of the chambers, the shielding plug is put in a decay pit and another shielding plug is placed on the thimble.

The activity level of the chambers is sufficiently low to permit manual handling for maintenance and replacement.

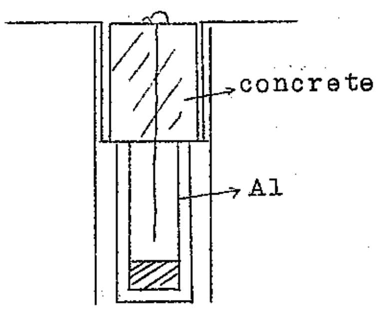
(vii) Q. On the countermeasures to cable against high temperature its contamination by the irradiation in the reactor,

What is the problems?
What kind of cables should be used?
How often is the cable exchanged?
How about the cable exchange method?
And how about the equipment for exchange? Show us your present situation about the problems mentioned above.

A. Long exposure to high temperature and radiations can turn out to some evolution of electrical and mechanical characteristics. As a result, insulation and electrical dielectric properties can be lowered. Fairly good results have been obtained with fibre glass insulation. The life time of the insulation and consequently of the cable is of the order of 2 years.

At Rapsodie, the only part of the cable which is exposed to radiations is inside the shielding plug. Removal and handling on of this shielding plug is some bit difficult, due to its weight and its length.

Cable change method; shut down 後2日も待つと detector のところで 10 r/h 位に下るので実際の操作は no shield できる。特殊な machine は必要ない。



detector をつけたプラグをそのままとり出し、ささえ容器に入れる。

(viii) Q. Show us your present situation about the following items.

1) correlation of the detector life time with the neutron and γ-ray level at the position of the detector in full power

2) frequency of detector exchange

3) method of detector exchange

A. The frequency of replacement of the detectors can be very different from one reactor to another one, in connection with specific working conditions. Some orders of magnitude can be given: for fission chambers which are disconnected during power operation of the reactor, the replacement frequency seems higher than two years. For compensated chambers working under neutron flux of 10^9 n/cm^2.s and gamma flux of 10^5 Rh^{-1}, this frequency is of the order two years.

(ix) Q. Self-powered detector, fission counter and boron coating incore monitor are considered as the incore detectors.

Which do you think is best?

Is there any suitable methods for the fast neutron measurement in the core?

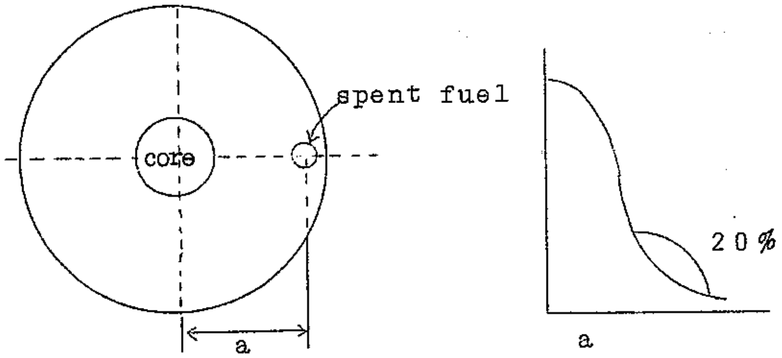
What degrees of informations are necessary from the core?

A. Concerning in core detectors, the conventional method is to us fission chambers and boron chambers. Using these detectors, the advantage is a low response time, but the disadvantage is a fairly high loss of sensitivity due to the burn-up inside the detectors. No large experience has up to now been obtained with self-power detectors. With these detectors, response time is longer, but larger life time could be expected. At Rapsodie:

- we used fission chambers with several fissible elements. We used also activation detectors.
- we measured radial and axial distribution. Of course, others useful measurement could have been envisaged.
- fission chambers need connecting cables and in Rapsodie, this is easy only using the central position in the core and some positions in the blanket. In these positions, detailed measurement can be made by continuous displacements of the detectors from the bottom to the top of the channel. For that purpose, special guide tubes with shielding plugs are provided inside the rotating plugs of Rapsodie. So, there is no problem with sodium, argon or shielding. It should however be noted that these measurements were made at low sodium temperature and low power level.
- other measurements were made inside the core of Rapsodie with instrumented subassemblies, in which activation detectors were positioned. After irradiation, these subassemblies were unloaded and the detectors were measured. This technique needs long preparation for the special devices which are need, but gives a good relative accuracy of the measurements.

fission counter が best である。 Self-powered detector はまだ response がよく知られていないので高速炉では余り使つていない。 boron coating はドーンレイ炉が使つていると思う。最も大事なことは、もし可能ならば power density を測ることだと思う。 炉の出力は outlet temp. および flow で測ればよい。 Control rod cal. については余りひんぱんには行なつていない。Rapsodie では '68年 1月2日に calibration したのが最後である。

Rapsodie では thermal power 20時間ごとに check している。



Rapsodie で spent fuel を blanket of 外側領域に入れた場合，その位置での total flux は20%増加する。しかし原子炉全体への影響は neglisible である。

原子炉出力の信頼性については flow meter および temperature measurement からくる error を合計して5%位と考えている。flow meter の error が2%であることは test で明記した。

(x) Q. We should like to know the following information about neutron detectors, such as CFU-6, which is available in JEFR.

a) bidder's name and address
b) cost and delivery schedule
c) specification, characteristics and technical data
d) detailed design requirements and arrangement drawings (location) of neutron detectors in Rapsodie and Phenix plants

(xi) Q. Neutron detectors at their as installed location.

1) receiving neutron flux level and gamma flux level during rated power operation and their ambient temperature.
2) receiving gamma flux level during shut down and ambient temperature.

A. à pleine puissance sur les chambres on a:

flux neutron : $5. 10^9$ N/cm^2 sec.

flux gamma : 10^4 R/h.

7.3 Process instrumentation

1) Comment

No comment

2) Questions and answers

(i) Q. In our design, the electromagnetic flow meter is used to measure the sodium flow. Show us the suitable method for its regulation. And also show us the measuring method of sodium flow, the variation of this flow meter with years and its confirmation method in France.

A. For electromagnetic flow meter using electromagnets a good stability of the power supply is necessary. It is appropriate to use a stabilized alimentation with constant intensity. In the case of flowmeters with permanent magnets it is important to avoid any shock and to keep away any metallic structure which could have demagnetising effect. Following these rules, we observed no meaningfull losses of the magnetic induction. Another type of flowmeters such as orifice type can be used especially for calibration of electromagnetic flowmeters. But we did not use them on Rapsodie.

(ii) Q. In our design, the following three kinds of detectors are used as the liquid surface detectors.

1) electrical resistance on-off type liquid surface detector, one point indication

2) electrical inducement fixed coil type liquid surface detector, continuous indication

3) NaK enclosed diaphram pressure difference liquid surface detector, continuous indication

Kindly show as your comments whether these detectors are proper to the JEFR from your experience of liquid surface detectors in the plant of Rapsodie.

And concerning the liquid surface detectors of continuous indication in Rapsodie, kindly show us your information about the experience of use, the accuracy, the confidency and the response speed for temperature variations.

A. Discontinuous level measurement by electrical probes: these detectors are working satisfactorily on Rapsodie. There are no sensitive to temperature variations and are very suitable. Continuous level measurement by mutual inductance detectors: only one detector of that type was installed on Rapsodie. But it was removed after some mechanical defect. These types of detectors are now under development but are not in operation in Rapsodie.

NaK enclosed diaphragm pressure difference liquid surface detector: we have no experience on this type detector.

Continuous level measurement with resistance type detectors: detectors of this type are installed on Rapsodie but are not working very well, due to their sensitivity to vapor deposits. It should be noted that the effective operation of the Rapsodie is very convenient, using only discontinuous level measurement by electrical probes (see above).

1) がよい。 Rapsodie の場合は liquid surface detector は直接 scram 系には接続していない。 operator に alarm を与え, operator が scram をする。

discontinuous type を上下2つと continuous type を1つもっている。

(iii) Q. In our design, the leak detectors are the electrical transmission type. How much is the reliability to the measurement by this type detector? And kindly show us your present situation of the use of leak detectors in France.

A. Leak detection by electrical plugs is very suitable for double jacketed wall pipes. Electrical detection by two parallel electrical wires is more suitable for pipes without double wall (secondary loop of Rapsodie). These two types of electrical detectors are very reliable, but many electrical defects can occur at the beginning of the operation.

Rapsodie では primary system にプラグ型（自動車のプラグと同じもの）を100ケ位設置している。同じ近い位置に2つの plugs を置いて miss detection をしないようにしている。plug の位置は配管の傾斜の低い方にある。secondary system には全 line にわたつて並列型のものを置いている。重要なことは，これらの電線がpipe などに contact しないように注意することだ。detector にかける電圧は約 24 volt 位で，battery を使う。

(iv) Q. Kindly show us the attention and the reliability for the setting position of the pressure gauge, in addition to the present situation of the use in France (examples of troubles and these resolutional methods and the best type your recommend).

A. On our experimental circuits, we use pressure detectors with bellows. This system was developped by the CEA, and is now built under licence by GAAA.

In Rapsodie there is absolutely no sodium pressure measurement on the primary and secondary loops.

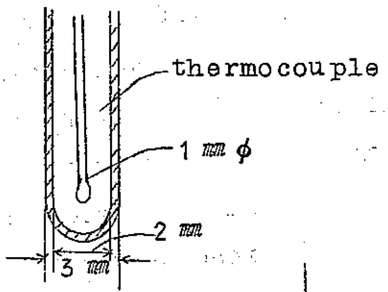
Pump の electrical loss などの検出は loop の flow rate, loss of electrical power などからとればよい。pressure measurement は trouble のもとになりやすい。

(v) Q. In our design, the setting position of the thermo-couple neighbours upon the most upper part of a sub-assembly. When the core temperature is measured by this thermo-couple, kindly show us its reliability in addition to some examples of troubles.

A. The thermocouples located at the sodium outlet of each sub-assembly at Rapsodie are chromel-alumel thermocouples with stainless steel sheath and magnesia insulation (made by SODERN and SACM). The outside diameter of the stainless steel sheath is 1 mm. The thermocouples are installed inside narrow thimbles from which they can be removed. The reliability of these thermocouples is very good: 84 thermocouples are in operation, and during 14 months only 4 of them failed.

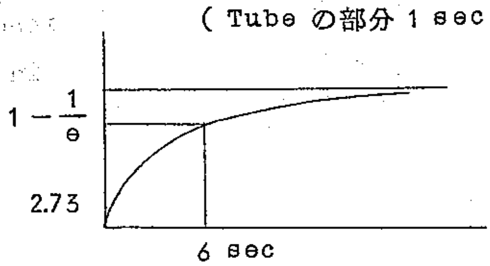
The most usual defects were due to the electrical connection or to mechanical shocks on this connections. It is to be noted that the results of the measurement with these thermocouples are very good without any change of the position of the core cover plate. But this is largely dependent of the detailed design of the sodium outlet of the subassembly and of the design of the bottom of the core cover plate. So, a precise water checking of the sodium flow at the outside of the subassemblies is necessary, to be sure that each thermocouple gives the actual value of the sodium outlet temperature of the corresponding subassembly.

14ケ月の運転期間で、84本の thermocouples のうち、4本が failure を起したのみである。Rapsodie ではとりかえが可能なようになつている（左図）。先端を溶接していない。



この thermocouple はGAAAが製作した。
Time response は全体で6 sec である。
Thermocouple の先端を fix して response time を早くすることにより重要なことは取換え可能なようにすることだ。

（Tube の部分 1 sec）



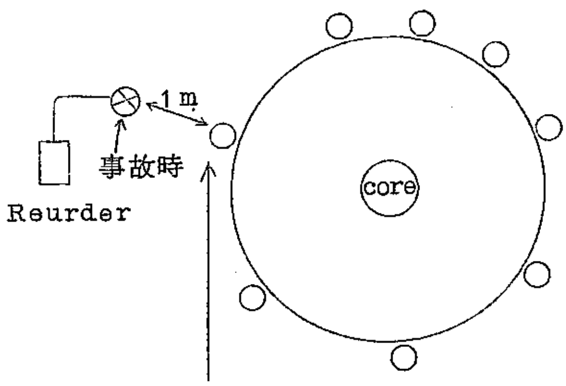
(vi) Q. Kindly show us what should be considered about the radiation damage of the detectors which concern to sodium and these limitation for use, in addition to the present situation for use in France.

A. At Rapsodie, not any kind of detector are in contact with sodium. Especially, the thermocouples at the outlet of each subassembly are located inside stainless steel thimbles. And so there is absolutely no problem of sodium contamination for these detectors. Up to now, it does not appear any defect due to radiation. Others detectors are not exposed to high radiation level and so there is also absolutely no problem of radiation damage.

(vii) Q. Do you admit the necessity to use process instrumentation detector, cable and so on as the explosion proof?

A. Generally we dont use explosion proven material in Rapsodie. Only for some thermofluids (organic fluid) circuits we use some explosion proven materials. In that case we follow the normal rules for standard industrial realizations. Considering the case of large nuclear accident, some neutron chambers with special electronic equipments and some thermocouples are located inside the concrete shielding far from the reactor vessel. These detectors could give some indication in the case where after important accident all the normal detectors would be out of work.

explosior の damage に対して特殊な specification は持つていない。しかし実際は flux と2つの thermo-couple により検出できる。右図のように事故時に全部の detector が破損したときには、⊗印の位置（1m位外側）にある detector（CCS）により測定可能なようにしてある。



（通常の detetor 位置）

(viii) Q. What is your opinion about using many MI cables?

A. We do not know the MI cables. For all detectors we use individually protected cables (metallic sheath to avoid any outside perturbation of the signal). This type of cable is necessary for all low level measurement which are inserted into the computers. As these low level measurements concern about 90% of the total of the measurements, this type of cable is used all over the reactor (at Rapsodie).

(ix) Q. Do you admit the necessity to set a failmonitor on each instrumentation channel?

In case of this, is it all right to use only alarm as failmonitor of out put signal?

A. We do not think that it is necessary to install a failmonitor system on each instrumentation channel. We use such a failmonitor system in any case on all safety measurement (neutron measurements, temperature measurements at the outlet of the subassemblies, sodium temperature measurements on the main cooling circuits). In case of a failure on some instrumentation channel an alarm signal is given to the operators. Some others instrumentation channels are also provided with a failmonitor system. But in that case, if some failure occurs, only a small red lighting signal appears on the face of the corresponding rack.

(x) Q. We consider it's impossible to change the wirings in the safety vessel after several operations.

May we have your comments on the material for the wiring.

A. We do not understand exactly what are these wirings. We suppose that this question concerns cables of the neutron chambers. For these cables, temperature is more a concern than the radiation level. The cables used seem to have a good resistance under irradiation. These cables are made from the following materials (from inside to outside):

- electrical wires, PVC ribbon (polyvinyle chloride) PVC sheath, metallic flexible sheath, and external polythene sheath. This type of cable can withstand an integrated flux of thermal neutrons of about 10^{15} n/cm^2.

At Rapsodie, we used another type of cable for high temperatures. This cable can withstand an integrated flux of thermal neutron of n/cm^2 and an integrated flux of fast neutrons of 1,6.10^{16} n/cm^2. The operating temperature for this cable can be permanently about 200°C. In these cables, the electrical wires are insulated with fibre glass with silicon, then by a stainless steel sheath, and then by an external insulating sheath from fibre glass.

(xi) Q. Accuracy and measuring range of electromagnetic flowmeter. Flowmeters may be of permanent magnet type and of saddle (coil-energized) type.

A. The accuracy of the permanent magnet flowmeters is about 2% to 3,5% (when they are not under the influence of iron masses being close of them). These flowmeters are principally used for safety reasons. Induction flowmeters are used in case of difficult approaches or when there are no safety reasons.

(xii) Q. Number and location of Na leak detectors, particularly around reactor vessel and primary coolant system. Their reliability; did you ever have their maloperation?

A. Leak detectors on primary loop: $\simeq 50$

There are 2 leak detectors for the reactor vessel. Some trouble occurred at the beginning, but at this time, there are less than one failure per month.

(xiii) Q. Type of Na level meters.

What type of Na level meter is used to be connected to either low or high Na level scram circuit?

A. The discontinuous level detectors are used in sodium vessels only for alarm. There is no safety action related to level measurements.

(xiv) Q. What type of pressure guages are used for the primary circuit and also for reactor vessel, if any?

A. No pressure gages.

(xv) Q. experience for plugging indicator

A. The plugging indicators work very satisfactorily (see Report CEA R-2522)

7.4 Fuel failure detector

1) Comment

No comment

2) Questions and answers

(i) Q. In our design, both the delayed neutron detection method and the cover gas method are used for the broken fuel instrumentation. What kind of methods are there on the determination of the position of the broken fuel channels? And kindly show us the response time of these detectors and the background data.

A. a) To our knowledge, in any existing fast reactor a method to locate broken fuel element has never been experienced. At Rapsodie, a special device is provided for this localization, but only after unloading of the subassemblies from the reactor. Up to now this device has never been used. It consists of a pit in which the subassembly would be heated to detect fission products coming from any broken fuel pins.

b) The response time of the delayed neutrons detectors is approximately the transit time of the sodium from broken subassembly up to the detector. The response time of the electrical collection detector (on the argon gas) is of several minutes.

With Rapsodie, our experience for broken fuel detection is very limited, as up to now we got no failures of any pin clad. For example, the delayed neutron detectors was previously tested and calibrated, but it has never got occasion to really work. We find that the detector system on the argon gas follows pretty well the activity of the argon with reasonable delay (of the order of 5 minutes), but other analysis techniques using longer period fission gas products could be more sensitive. Generally speaking it has to be recalled that up to now at Rapsodie we got some fission products probably due to some previous defects in some clads, but we got absolutely no mechanical failure of any clad.

Rapsodie ではどの燃料が broken しているかを検出する方法はない。out of pile の方法による。Ar gas detector で検出している。現在燃料破損はあるがたいしたことはない。検出された Ar gas の max. activity は2～4 c/m³であつた。normal operation での Ar gas の flow は1 m³/hr である。stuck 中の detector は reactor operation とは関係がなく、building と関係している。

(ii) Q. The sensitivity for the broken fuel detection is affected by various condition or by various atmospheres. Can we expect the sensitivity to detect at least one broken fuel pin with the delayed neutron detection method under the conditions that the fissionable materials from a few ppm to a few tens of ppm are mixed into the sodium, and that the γ-ray level is very high?

A. Clad failure detection by delayed neutrons using short life elements (longest periode 55 sec.) will be largely dependent to the transit time, that means to the time necessary for the sodium to go from the fuel up to sensing device.

At Rapsodie, the sensitivity of installation is about 100 pulses per second for 1 cm^2 of bare fuel, with a transit time of 12 or 15 seconds.

For a given sensitivity, the minimum detection level is largely dependent of sodium and clads contamination. It seems difficult that this level could be less than one bare pin, when the fuel contamination in the sodium is above one or a few p.p.m.

7.5 Control

1) Comment

(i) It is probably ambitious and not suited to such a reactor to envisage any automatic operation. Manual control is very easy and safer.

2) Questions and answers

(i) Q. Concerning the sodium flow control, the constant flow which corresponds to the setting power is taken in our design, but the flow control characteristics are only analysed until 50 % power and 50 % flow because of the capacity of our simulator.

In addition, it is considered to be necessary to decrease the flow until about 10 % of it in a low power operation.

Then, are there any problems which can not be considered in a large flow?

We are planning to control the cooling system by the variable flow control system in future. Then, is it sufficient in order to deal with the change of the flow over the large range if only the changes of the heat transmission coefficient and of the time constant of the flow are measured?

A. Dynamic studies with variable sodium flow is very useful to fix the operation and control rules and to get a good knowledge of the stability of the reactor. Sodium flow variation has an influence of the sodium levels, heat transfer coefficients, delays, mixing, thermal counteractions, pressure drops, etc....

So it is very useful to make such dynamic studies especially for low flow rates, taking into account the variations of the above mentioned parameters.

(ii) Q. In the control system of the JEFR which is shown in Vol. VII, we are planning to control the reactor core and the reactor cooling system independently. Is it considered that the better control characteristics are obtained if the both control systems are united?

A. In Rapsodie, there is no automatic control neither for the power, nor for sodium temperature, and this automatic control seems absolutely no necessary. So we can say that these two parameters (neutron control and temperature control) are separate, but they are only manually control. If we had to use any automatic control, it would be only to compensate low effect of burn-up, outside temperature variations, wind variations, etc... Generally speaking we have to recall that by our experience, we think that manual control is definitely better than automatic control.

(iii) Q. In our design, the reactor outlet temperature changing speed is less than 50°C/hr. What degrees of the temperature changing speed are suitable for the actual reactor operation with considering the thermal shock?

A. A temperature changing speed of 50°C/h seems pretty low (for Rapsodie 150°C per hour).

At Rapsodie, several limitations exist concerning the sodium temperature:

(1) Sodium heating without nuclear power. This heating is made before start up of the reactor, by means of electrical heaters (1 MW) installed on the secondary sodium loops.

Due to the power of these heaters, the temperature raising speed is about 30°C/h.

This sodium heating is made up to reaching the nominal value of the inlet temperature of the reactor (about 400°C).

(2) Power raising: only for an easy operation of the reactor, it seems convenient to keep the inlet temperature of the sodium at its nominal value (400°C at Rapsodie).

For that purpose, the temperature raising speed we use at Rapsodie is about 150°C/h.

(3) Allowable thermal shocks on mechanical structures are not known with a good accuracy, but are surely largely higher than 150°C/h. Finally the temperature changing speed during power raising is essentially a matter of an easy operation for people in the control room.

アナコンの解析で75，150，300℃/h の3つについて行なつた結果、300では control が困難であつたので、その次に早い150℃/h に設定した。150℃/hの基準だと原理的には40分で power up が可能であるが、いろんな step で温度を saturation させる必要があるので、実際には平均3時間位で power up している。1時間ででも可能である。

(iv) Q. In our design, the reactor is started up after the sodium loops are heated using the pre-heating system (1 MW). What kind of formulas of reactor operation are proper to the JEFR? And, is it possible that the reactor operation is performed with heating the sodium loops by the nuclear heating step by step after the reactor has started up only using the keeping warmth system? In this case, what kind of problems are there on the control system?

A. We think that a reactor start up with sodium heating by nuclear power is possible. But at Rapsodie, the normal rule is that start up and power raising of the reactor are made with inlet sodium temperature at 400°C. There are two reasons for that:

① Sodium temperature is lowered only for loading-unloading operation. Before a new start up sodium cleaning by cold traps is undertaken. This cleaning is as better as sodium temperature is higher (at higher temperature we get a better solubility and a faster dissolution of sodium oxide and of other possible impurities). So a temperature of 400°C seems very convenient for that cleaning operation. And after this cleaning is completed, there is, of course, no reason to decrease sodium temperature before start up.

② For operating people, it is surely easier to make the power raising with fixed inlet sodium temperature than with variable temperature.

(v) Q. In the analysis of the reactor kinetics, the nuclear and the thermal characteristics are analysed with a few approximation models. How much was the error of the calculations in comparison with the experimental results? What kind of modifications were necessary to the models? Kindly show us the informations about the problems mentioned above, concerning the reactor core, the coolant loop, the intermediate heat exchanger, and the air cooler, respectively.

A. Regarding the reactor kinetics, we found at Rapsodie a good agreement between calculation and experimental results, as far as neutron characteristics are concerned. Concerning the thermal and mechanical effects, the calculations were sometimes adjusted taking into account the experimental results. Some effects are not completely understood, for example, the power coefficient evolution. We can say that mixing factors and heat transfer coefficients are better than expected, as well as time constants due to sodium volumes. Temperature difference between primary and secondary sodium in the intermediate heat exchanger is less than expected. But the thermal capacity of the sodium-air heat exchanger is greater than expected.

(vi) Q. There are two loops of the cooling system in the JEFR, but only one loop is considered in the design of the control system. Is it necessary to consider the interference between two loops?

A. This problem is largely dependent of the control system (for pumps and core) and of the power supply system to the pumps motors. In the case of any failure concerning one pump, the situation is very different from a power operation with half nominal flow rate. The behaviour of the installation is very different if there is check valves on the primary sodium loops or if not.

(vii) Q. What meters are suitable to set on console?

A. On the control panel of Rapsodie, indicators are provided for the following measurements:

- control rod position
- power
- reactivity
- period

- dP/dt
- pumps flow rate - pumps motor amperage
- flow rate and position of the air vanes on the sodium-air heat exchangers.
- reactor inlet and outlet temperatures
- secondary sodium temperature at the inlet and outlet of the terminal air heat exchanger.

(viii) Q. How should we interpret about the basic concept of panel & rack consistence in main control room?

A. Panel and rack consistence in the main control room is not a problem. For large nuclear power plant there are always a main control room and auxiliary room. In the auxiliary room are located the computers, the auxiliary computers, and all the racks containing amplificators electronics and measurement dispatching systems. In the main control room are located only measurement panels synoptics and control systems. For smaller installations, all the racks are generally located in the main control room. Generally they are put all together and are not necessarily to be seen from the control panel. The indicators on the front face of the racks are not used to operate the reactor.

(ix) Q. Is it right to record sodium flow of main cooling system, coolant temperature at outlet & inlet of reactor and coolant temperature at main cooler outlet in console?

A. It is very convenient to provide recorders for the main parameters which are of importance for reactor operation. So it is easier to follow the evolution of these parameters. It is useful to record inlet and outlet temperatures at the reactor and at the heat exchangers, as well as the flow rate of the cooling medium. Furthermore, these parameters being logged by the computers, it is always possible to operate the reactor only with indicators. The evolution of any parameter can always be followed using the computers.

(x) Q. We would like to get the following detailed information on Rapsodie

a) operating procedure
b) reactor control system and interlocks
c) detailed drawings of control room panels
d) control characteristics of cooling blowers
e) plant control system, specially, of coolant temperature and flow, including automatic control system
f) computor system and its function in plant control (specially safety function of computor)
g) coolant flow after reactor scram
h) set back actions, if any.

(xi) Q. one control rod worth and withdrawal speed.

A. The speeds of control rods are:

0,33 mn/sec. for going up
3 mn/sec. for going down

For fine control, the speed is 0,3 mn/sec. for up and down.

(xii) Q. Control of reactor and plant

1. Method of load control (At JEFR, the air blower system is controlled to keep the reactor inlet Na temperature constant. What is controlled at Rapsodie, by receiving what signal?)

2. Method of reactor power control (Are movement of control rods automatically controlled? If so, by what signal are they controlled?)

A. For Rapsodie, the power is raised with a constant sodium temperature at inlet.

The operators watches at:

- the sodium inlet temperature,
- the sodium temperature at the outlet of the final heat-exchanger.

For raise to full power in one hour, this temperature must be lowered by 20°C in one hour. The operator can check the measurements of $\frac{d\ \text{Power}}{dt}$ and $\frac{d\ \text{Temperature}}{dt}$

In Rapsodie, the control rods are moved only by the operator or the safety system.

(xiii) Q. Show us following data if they are available:

① Neutron fluctuation (noise) level (peak to peak) and its frequency spectrum.

② Reactor-inlet-Na-temperature fluctuation (noise) level and its frequency spectrum.

③ Reactor-outlet-Na-temperature fluctuation (noise) level and its frequency spectrum.

④ Primary Na flow fluctuation (noise) level and its frequency spectrum.

A. Le niveau de bruit est très faible $< 0{,}5\%$ pour les mesures neutroniques, de là 2% pour les mesures de température et débit. Les expériences faites n'avaient pas assez de précision pour donner un résultat autre que la réponse du capteur. Mais d'autres expériences doivent être faites incessamment.

(xiv) Q. Operation method of the coolant system

Do you operate the primary and secondary coolant at full flow when you approach to criticality?

Would you explain the control method of air coolant flow relating to the operation sequence of cooling system during reactor power up?

A.

7.6 Safety system

1) Comment

No comment

2) Questions and answers

(i) Q. At the time of the scram by the safety operation, whether you cut off the plant control circuit or operate it as usual? Which is better? Why is it so?

A. At Rapsodie, there is no automatic control. In case of scram, the blowers of the sodium-air heat exchangers are automatically cut off. But, the vanes of these air heat exchangers are left open, so that 4 MW can be removed by air. These vanes on the casing of the air heat exchangers are closed about 15 minutes after the scram.

(ii) Q. The items of the reactor protection system are shown in Sep. Vol. XIII. What kinds of safety systems were operated by the abnormal situations in Rapsodie? And kindly show us your opinions about the safety systems of the JEFR.

A. The safety system of Rapsodie is made by two steps:

1 In the case of scram signal, the control rods are dropped into the reactor.

2 In the case of stop signal (in less than one minute), the control rods are inserted at normal speed into the reactor.

Up to now, we got no trouble with safety system of Rapsodie. Some minor modifications were made to get a greater simplicity. Comments about the safety system of the JEFR will be given in the final report.

(iii) Q. Capacity and number of unit of emergency-used battery. List of name of each load and its electric burden connect to the battery.

A. I. - For emergency batteries, the autonomy is about half an hour. In fact, at the beginning of operation, the autonomy was 45 minutes.

	Nombre	Capacité	Chargeur	Spécification
batteries: 24 volt	4	342 A/H (12 el)	160 A	83 A/30 mn
: 48 "	2	450 A/H (23 el)	250 A	200 A/30 mn
: 127 "	2	266 A/H (62 el)	80 A	(50 A/30 mn 175 A/1 sec.
: 260 "	2	600 A/H (120 el)	350 A	
poney motor: 12 volt	2	80 A/H		

(iv) Q. List of names of scram and slow scram.

A. 1 - Criteria for alarm, set-back or scram (additional list) -

A - Sodium outlet temperature from subassembly 2/3

10°C above threshold : alarm

20°C above threshold : set-back

30°C above threshold : scram

B - Safety control system malfunctioning:

scram in case of electricity supply failure

C - In case of electricity supply failure on the control rod actuating motors: scram

2 - Acoustic and visual signalisation -

Corresponding to each set-back or scram, an alarm is given simultaneously, both acoustic and visual.

For some analogical measurements which raise safety actions, there is always a first threshold giving only alarm before set-back or scram.

Several (1000) other alarm instruments are inserted in the control system.

7.7 The others

1) Comment

No comment.

2) Questions and answers

(i) Q. In France, how much do you estimate the MTBF (mean time between failures) of the detector, the control system, and the computer in use? Kindly show us its information in addition to the situation of use up to now.

A. For our instrumentation (all silicium transistors) the MTBF (mean time between failures) was estimated at about 8,000 hours. But larger life times have been obtained, and the MTBF is probably larger than 8,000 hours. More precise values could be only given after longer experience.

The MTBF of the systems is depending of the working conditions. Practically, the MTBF could be a few years.

For the computers, the MTBF is essentially a matter of the type of the computers.

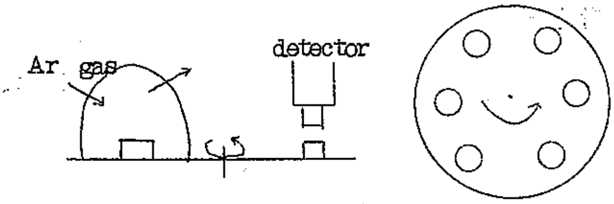
(ii) Q. To measure the fuel burn up, we apply the destructive method and the non-destructive method in which the γ-ray from the fission products is detected and measured. What degrees of accuracy are expected by the non-destructive method?

A. Concerning the fuel burn-up, the degree of accuracy by gamma-rays measurement is, by our experience, of the order of 10 %. However, we think that this technique is essentially useful to defect any defect on the fuel pins, more than to make any burn-up measurement.

(iii) Q. In the cover gas method, the wire of the precipitate detector either moves or doesn't move. Which is the better? In addition, is there a suitable method for the cover gas detection?

A. We think that the best solution is to use an instantaneous collection detector with a rotating electrode, so that no make a differential measurement of the gas fission products concentration.

Wireはない。他の Type のものを使つている。

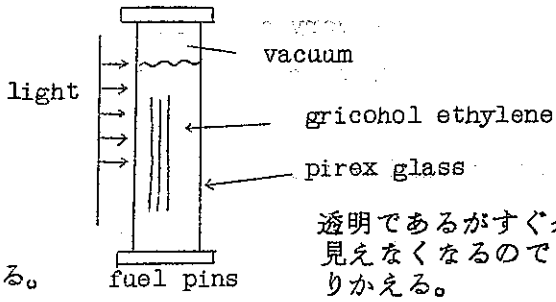


electro-static collection
により highpoleutial なドータ
― nuclide を吸着させて、右転し
左図の右側の γ detector （シン
チレーター）で測定する。

3) Information

(i) Leak detection

真空に引くとF.P.が出る。
Decay heat により沸騰している
気泡とは容易に識別できる。クラッ
クの大きさは取出してから別に調べる。



透明であるがすぐ茶色になり
見えなくなるのでときどきと
りかえる。

8. 遮 蔽 設 計

8.1 General comment

1) Comments

(i) too good な design と考える。

(ii) vessel のまわりの borated graphite は graphite でよいのではないか。内圧の問題がある。

(iii) secondary loop の Na の activation に注意するように design 基準をとること。

(iv) Fission products contamination must be considered when auxiliary circuit shielding is designed. We do not have a large experience of that in Rapsodie since no opened clad failure occured up to now. We use poor informations from litterature and we do not believe very much into our calculations. We will know actual data only from reactor operation.

2) Questions and answers

(i) Q. Kindly show us your comments on the phylosophy for the shield criteria of JEFR, especially on the following items, in addition, what is your phylosophy in France?

① temperature condition for the concrete
② optimization for the general shield design
③ radiation damage

A. ① Temperature conditions on concrete

The temperature allowable for ordinary concrete is about 80°C. The principal reasons of this limitation are the weakening of mechanical properties and the loss of free water. In order to restrain the stresses, it is advisable to reduce the thermal gradient. For some special concretes (like the rare earth concrete of Rapsodie) the temperature can be higher in case of a special treatment before operating the reactor, such as preliminary drying. In the case of Rapsodie's special concrete, the admissible temperature is 200°C.

(2) Optimisation for the general shield design

The shielding materials must be able to protect against radiations during the whole life of the reactor, so as to insure the access for maintenance in all foreseen places. Especially, access must be allowed to the secondary circuit, where the radiation dose must not exceed 2,5 mrem/h. For that purpose, all shielding properties of the materials must be conserved during reactor's life.

The main criteria of choice for materials are:

- good and well known holding under irradiation (swelling, thermal conductivity, etc...)
- in case of internal pressure (borated materials), this pressure must be compatible with cladding resistance. Consequences of a cladding failure must be carefully checked (sodium in contact with materials).

All mechanical components (support plate, vessel, fuel cladding) must support radiation effects during their whole lifte.

(3) Radiation damage

Radiation damage on cladding, support grid, vessel, shielding elements, must be estimated with testing of samples of the corresponding materials in irradiation facilities. Results of testing must be extrapolated to the real conditions existing in the reactor (temperature, neutron spectrum, environment).

(ii) Q. What was the problem in the shield ability test which was performed in Rapsodie after the completion of the reactor? Kindly show us the problem concretely.

A. This matter is discussed in the special report already transmitted (report CEA n° 3626).

(iii) Q. In the shield design, is it necessary to consider the case in which there are the leaks of the primary sodium to the N_2 gas system?

A. Leaks of sodium in the N_2 system

This case must be considered as an incidental failure and consequences must be examined.

(iv) Q. Shielding design criteria, minimum permissible temperature of graphite and maximum temperature of concretes in Rapsodie.

8.2 Shield materials

1) Comment

No comment.

2) Questions and answers

(i) Q. On the special concretes (serpentine concrete and rare earth concrete), kindly show us your opinions from your experience in France:

1 mixture of concrete, aggregate, water, etc. and physics of the aggregates,

2 On the reactor operation, it is considered that the concrete temperature may be pretty high.

Then, how about the loss of the water conserved in concrete and how about the change of the strength of the concrete?

3 physical constants of the concrete and its shield constants,

4 how to determine the shield constants (experimentally or theoretically?).

A. 1 As an example, we give hereafter the composition per cubic meter of the special rare-earth-concrete used in Rapsodie.:

Cement Lafarge (fused cement)	460 kg
Rare earth (Yttric)	150 kg
Corindum (0,2 -2 mm)	600 kg
Corindum (2,5 mm)	400 kg
Serpentine 5/15	1,000 kg
Water for mixing	170 kg

More details are given in the joined report (BIST 110), in the first article.

2 The concrete of Rapsodie can support a maximum temperature of 200°C. At this temperature, it retains the water pertaining to the serpentine (100 l/m^3) and a part of the water corresponding to the hydratation of the cement (environ 100 l/m^3 also). At this time, 120 l/m^3 of the cement remain in the concrete.

The variation of mechanical characteristics versus the temperature are the following:

Geometrical contraction (shrinkage)	(60°C)
After 28 days	87 μ/m
After 90 days	140 μ/m
After 180 days	295 μ/m

Young module of elasticity

after 28 days	450,000 bar
after 90 days	475,000 bar

Mechanical strength:

after 28 days	700 bar (crushing)
after 90 days	700 bar (crushing)
after 28 days	66 bar (tensile flexion)
after 90 days	68 bar (tensile flexion)
after 5 months at 200°C	

Contraction	837 µ/m
Young module	86,000 bar
Crushing strength	655 bar
Tensile strength	19 bar

(3) Physical data

Mean thermal linear dilatation coefficient ($T \geqslant 75^{\circ}C$)

alpha = 12.10^{-6} /°C

Thermal conductivity (between 40°C and 150°C)

1,2 to 1,5 Cal/m/h/°C

(4) Shielding data

The calculation method must be tested in neutron spectra which are the same as in the reactor for the considered materials. Propagation experiments have been made with activation detectors for the special concrete of Rapsodie in spectra harder than the Rapsodie spectrum.

(ii) Q. Shielding materials

Please let us know special requirements on specifications of the following shielding materials if any:

a) Borated graphite,
b) Graphite,
c) Serpentine concrete,
d) Calcium borate.

A. a) Borated graphite: the boron rate must be specified (f: 5% ± 1%)

The density is related to the boron rate.

The specification depend on the chemical form of the boron introduced (for instance B_4C).

b) Nuclear graphite:

Following characteristics must be specified:

for instances

density	(f.i. $d \geqslant 1{,}68$)
total void rate	(26% f.i.)

thermal expansion

ashes rate (or sum of impurities transformed into oxide) f.i. 98 ppm

boron rate (0,12 ppm f.i.)

capture cross section for thermal neutron (3,76 mbarn)

c) Serpentine concrete:

You must specify:

- density
- water content
- water remaining versus temperature
- maximum operating temperature (it depends on the special treatment performed before operating the reactor, f.i. heating to 250°C)
- crushing strength, tensile strength and bending strength, shrinkage

d) Calcium borate:

The mineralogical and chemical nature of the borate must be precised. Generally, we speak about Colemanite ($3\ B_2O_3$, $2\ CaO$, $5\ H_2O$)

The mean content of B_2O_3 is fixed: 43 - 44%, as well as water content (18%) and size of grains.

(iii) Q. Though we intend to settle impurity of graphite in rotating plugs as follows, is there any trouble about it?

Fe	24 ppm	Si	220 ppm
Na	12	Ca	36
K	12	Ti	24
Cr	2	Al	60
Mg	24		

A. Impurities

Impurities mentioned in the questionnaire seem not to raise special difficulties.

(iv) Q. Kindly show us the reasons why the corundum was used as the aggregates in Raprosie, and the volume ratio of the corrundum in the concrete mixture.

A. Corundum has been used in order to increase the concrete density and to have a small thermal expansion coefficient.

(v) Q. You have chosen rare-earth graphite as the shielding material instead of borated graphite.

Would you please tell us available reports in which we would know material descritpion and fabrication method?

A.

(vi) Q. Please let us know special cares to be taken in the installation of graphite blocks if any.

A.

(vii) Q. Please let us know the relationships between boron content and shielding effect of borated graphite.

A.

(viii) Q. Did you experience any trouble with graphite blocks surrounding the safety vessel of Rapsodie?

What were the reasons and how did you solve it if you had trouble?

A.

(ix) Q. The dimensions and arrangement of graphite used in Rapsodie.

A.

3) Informations

(i) ボロン入黒鉛のキャンニングの必要性、使用温度等

Rapsodieの場合、ボロン入黒鉛は、ブロック毎のキャンニングは行つていない。プラグ容器内に裸のブロックを積上げ、フタ板により全体を密閉したのみである。内の雰囲気は組入時の空気であり、ヘリウム or 窒素ガス置換とか真空にするとかの操作はしていない。何ら問題はない。

使用温度はウイグ＋エネルギーの蓄積がこの程度なら小さいので、２００℃以上に保つ必要はないと考える。

（Rapsodie は５％Ｂグラファイトを使用したが、２％Ｂでよいと思う）

8.3 Compatibility

1) Comment

No comment

2) Question and Answer

(i) Q. Kindly show us your information about the compatibility between sodium and graphite, or borated graphite.

A. Experiments are under way at Cadarache on this subject.

We consider two very different cases:

- irradiated graphite (before coming in contact with Na
- non irradiated graphite

8.4 Codes and cross sections

1) Comment

No comment

2) Questions and answers

(i) Q. Kindly show us the present situation of the adjustments of the shield calculation codes and the shield constants in France. (Particularly for ones which have been developed and adjusted for fast reactors)

A. The calculation programmes are compared to the experimental results obtained by mock-ups of the shieldings which are placed on the Harmonie source reactor.

The programme used up to now is the NIOBE transport programme (spherical, polycinetic). A Monte-Carlo code normed TRIPOLI is under completion.

(ii) Q. Please let us know the informations on sodium-graphite reaction investigated in Rapsodie project.

A.

(iii) Q. Please let us know the composition and characteristics of special shielding concretes used in Rapsodie.

A.

9. 材　料

1) Comment

(i) You propose to use inconel X in subassemblies for springs or grids. We made no irradiation of inconel X and we think you have to find information on the behavior of this material under neutron flux in literature. It is possible that it becames brittle. An irradiation test is probably to be made.

インコネルＸはフランスのサーマル中性子での照射実験で3.8×１０^{22}の照射で伸びが１〜２％位になつた。

フランスとしてはステンレスまたはハステロイの方がよいと考えている。

(ii) 炉　容　器

a) 熱処理温度８５０℃はクロムカーバイドの析出に対しては十分高い温度であるがσが急速に減る恐れがある。

最高熱処理条件はよくわからないが８５０℃で行うならσについてのテストを行う必要がある。

b) フランスでは中性子照射に対する制限は特に定めていない。

c) ステンレス３１６と３０４の混用はＣの含有量がほゞ同じなのでマストランスファー上からは問題になるようなことはない。

(iii) 被　覆　管

クリープレートは照射材と非照射材との間に大差があるとは考えていない。

Rapsodie ではＡＳＴＭ４２０−３１６ステンレス鋼を使用

10. 燃料使用中検査施設

1) Comments

(i) We are in complete disagreement with the general characteristics of your fuel facility:

- we do not think it is prudent to re-use spent pins after having dismantled subassemblies and we are pretty sure it is impossible after having washed the pins. Our experience showed us that normal sodium cleaning is preduciable to thin stainless steel pieces such as bellows and clads.

the philosophy about periodic watching of the fuel element is very different from ours. We think that scheduled destructive inspection of one or two subassemblies after say each 10 000 MWd/t gives a good understanding of the general behaviour of the fuel. It would be of course very interesting to examine one subassembly from time to time and put it again into the reactor, but we think that the foreseen procedure is too ambitious and anyway not necessary. Furthermore, that would mean only non destructive examination, and consequently only limited results.

Moreover, we never put fuel pins into the core after dismantling, cleaning, and washing. Such a operation could be possible, but would need special care and attention.

(ii) Fuel reloading program

In the JEFR conceptual design, it is proposed to examine pins after a determined burn up and to re-load them into the reactor. We were in disagreement with this program because:

(a) We do not know up to now a method to predetermine by pin examination when a clad rupture will happen. Thus, risk is not reduced.

(b) Reactor must be designed to allow some clad failures, which must occur without trouble even for normal operation or maintenance.

(c) We do not permit to reloadany pin extracted from reactor because some sodium exidation and clad corrosion may occur during unloading, cleaning and reloading operations.

Now, it is proposed by JEFR representatives to examine only some pins from every subassembly, and then to reload into the reactor a subassembly with both old and new pins.

We do not agree with this solution because:

(d) Reloaded pins can suffer some oxidation and corrosion as previously noted.

(e) In a same subassembly, there will be pins of different burn-ups. Swelling of the more irradiated pins may occur in a different way. Temperature measured at a subassembly outlet is not representative of all pins. And there is an heterogeneous power distribution in such a subassembly.

(f) It is more convenient to consider groups of subassemblies, such as these subassemblies have location not very diffe-rent from the burn-up point of view. A subassembly of every group is unloaded after a said burn-up and its examination gives valuable data for every subassembly of the group remaining in the reactor.

In every case, fabrication of new subassemblies in-cluding spent pins must be made in a hot cell and affords difficulties larger than subassembly dismantling.

- - - - -

(iii) You have to separate your facility into two smaller facilities: one for the subassembly dismantled and the other for the clad and fuel examination. The first one is connected with the reactor and has to follow the reactor operation and be able to receive the spent subassemblies when it is necessary. The second one is a more basic facility where the work rythm must not be in a narrow connections with that of the reactor.

(iv) You could largely simplify your fuel facility by using an irradiation politic similar to that of Rapsodie. You load a relatively important number of subassemblies containing a same type of fuel pin and you cut and examine subassemblies from time to time, for example after every 10 000 or 15 000 MWd/t.

In conclusion, we propose you to have only a cell bounded to the reactor where you can cut the subassemblies and make an external examination of the pins and to cut the pins and examine the fuel in a second hot cell separated from the normal operation of the reactor.

FMF はこのまゝ製作すれば非常に大きく複雑な施設になるだろう。１９６８年秋のＡＮＬ meeting ＵＳＡ，France，ＵＫＡＥＡの engineer の間で一致した意見は「照射燃料を検査後原子炉へ再挿入することは不可能である」ということであつた。

2) Questions and answers

(i) Q. We are planning to test all assemblies in this installation for monitoring the core fuel assemblies under irradiation. Kindly show us your opinions about the propriety of this plan in correlation with your plan of fuel monitoring in Rapsodie.

A. First of all I think it is not necessary to check or to test all assemblies. What we do for RAPSODIE seems to be a natural approach of the problem of behaviour of irradiated fuel assemblies we check one or two assemblies at different steps of burn-up. It is quite likely indeed that:

a) at the beginning of their irradiation, if one assembly is checked to be safe, all the other assemblies having reached the same burn up have a good chance to be safe also.

b) at the end of the expected life of an assembly, or a little before, if a majority of fuel pins appear to be dangerous for the safety of the reactor and must be changed, it is likely that the surrounding pins will soon reach the same dangerous state, as well as the other fuel assemblies at the same step of burn up.

In these conditions, it does not seem to be a good solution to change the failed pins and to put new ones instead, because it is quite likely that the pins not yet failed will probably dail soon also and oblige the dismantling of fuel assembly after only a short burn-up of new pins.

(ii) Q. In our design, we are planning to resolve, to test, and to recompose 10 core fuel assemblies per week. From your experiences of hot cell operation in France, kindly show us your opinions about the probability for this work to be performed.

A. We have not here in France a hot cell facility to compare to what F.M.F. will be. I think that the only valuable comparison should be made with EBR II Fuel Reprocessing Facility. I don't know exactly the number of fuel elements they can dismantle, chech and refabricate per day.

Our actual experience, with the rather small hot cells of A.D.A.C. concerns the examination of 2 fuel assemblies per week and we have not yet any experience of remote fabrication.

What is expected for Phenix reactor where there won't be either remote fabrication, is one fuel assembly per day.

In any case, 10 assemblies per day is a rather large number. It will take much room and many equipments. Since, of course, the total operation on one assembly will last much more than one tenth of a working day, it means that many fuel assemblies will be present in the facility at a time (at least 10, may be more.)

In this case, it is expected that criticality limitations will be a problem. They are already a problem in A.D.A.C. hot cells and in E.B.R.II. facility, where, to my knowledge, the total amount of fissible products is much smaller.

(iii) Q. In our design, if any pins are normal when an assembly is resolved and tested, these ones are recomposed and used once again. Kindly show us your opinions whether this phylosophy is proper or not from the point of view of fuel monitoring plan and of fuel safety.

A. See above our opinion about your principles of testing the testing the fuel pins. What we may add here, is that, in our opinion, it is very difficult to see immediately which pin of an assembly is faileed without dismantling the whole assembly. For instance it is certainly very unlikely to be able to see any details on a X ray radiography of the whole assembly.

Remember that two of the main points to check are the swellin of fuel pellets, and the gazeous fission products pressure, which both can lead to a cladding failure.

Also a good idea of thermal behaviour of fuel itself is given by X scanning.

The result of these tests is not immediate nor obvious. So all the fuel pins should be tested. And, again, if some of those pins appeared to be at the end of their life, there is no reason why the next pins would be far from the end of their life.

Thus, these other pins should not be recomposed and used again in the core.

(iv) Q. In our design, core fuel assemblies can be resolved and be tested to use these ones in core once again. Principal reason why this phylosophy is taken is to test integrity of fuel assemblies under irradiation as well as fuel pins. In Rapsodie, what kind of tests were performed for checking integrity of your assembly design? From the results of your tests, how much is the minimum number of the fuel pins which have to represent a fuel assembly (consisting of about 90 pins/(SIA))? What kind of irradiation tests and of post-irradiation tests can represent integrity of an assembly with the minimum number of fuel pins?

Kindly show us your opinions about our questions above mentioned.

A.

(v) Q. On the problems of the storage of post-irradiated fuels, kindly show us your comments from your experience in Rapsodie,

a) method of the storage of post-irradiated fuels

b) critical problems in the room for storage

c) methods of the storage of fuel assemblies which have the leakage of fission products

d) cooling method in storage and its problems

A. Storage of post-irradiated fuel subassemblies in Rapsodie

a) Method of storage of post irradiated fuel
Subassemblies areput in pots containing inert gas. Pots are tighly closed by a cover and are immersed in a water pool.

b) Critical problems in the storage

A minimum distance must be provided between pots.
Pitch of support structure is about 25 cm.

c) Storage of fuel subassemblies which have fission product leakages

As they are stored in closed pots, spent fuel subassemblies which have fission product leakages must follow the normal procedure.

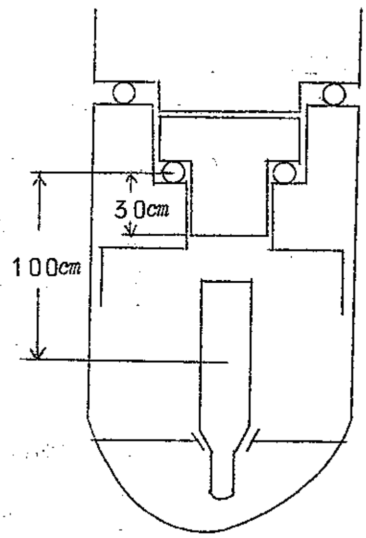
d) Cooling method in storage

Pool water is cooled, without any problem.

Phenix の使用済燃料貯蔵ポットは右図にに示す。

1 pot には 1 subassembly を入れシールには O リングを使用する。

Rapsodie は pot はもつていない。



(vi) Q. On the problems of the mixed oxide fuel irradiation in a fast reactor, kindly show us your opinions,

① experiments, calculations and calculation formula of decay heat

② cooling method in the fuel handling mechanism, the fuel transfer mechanism and the fuel transfer capsule.

A. Transportation of mixed oxyde fuel irradiated in a fast reactor

① Decay heat

See 3 - 5 Decay heat carve

② Cooling method during unloading

a) fuel transfer machine

Subassembly remains in sodium and it is cooled by sodium.

b) fuel handling machine

5. 燃料交換系参照

(vii) Q. On the problems of the F.P. leakage from an irradiated fuel assembly, kindly show us your opinions from your experience in France,

① detection method

② selection method of particular fuel pins having F.P. leaks from many pins which compose a fuel assembly.

A. F.P. leakage from an irradiated fuel

① Detection methods

Two global detection methods are used in Rapsodie:

- Delayed neutron detection in sodium.
- Fission product detection in argon by electrostatic collection of solid fission products transported by argon.

The sensitivity of delayed neutron detection was checked by introducing a bare uranium pin in sodium.

When fission gas was released from some pins, only argon detection gave a signal.

There is no mean to select the failed subassemblies in the reactor. They are identified by analysis of water which is used to clean sodium before subassembly dismantling. We got good correlation between these analysis and fission gas releases

② Selection of failed pins

Pins are punched after subassembly dismantling and fission gas volume is measured. By this way, empty pins have been identified.

Moreover, every pin is visually examined.

We recall you that we have not get an open pin failure up to now.

(viii) Q. On washing the sodium which adhere on fuel assemblies and fuel pins, kindly show us your opinions from your experience in France.

1. quantity of sodium which adhere on the fuel assembly
2. situation of the adherence of sodium
3. method for sodium removal and its problems
4. degree of sodium-vapour reaction in the case of the sodium removal by dry vapour
5. method for removing sodium oxide which is remained and adheres

A. 1 The quantity of sodium depends very much on the geometry of assembly. It depends also on the cooling time, because the hotter are the fuel pins, the smaller is the quantity of sodium.

In Rapsodie, after 40 days cooling on a 40 000 MWd/T assembly the quantity of sodium is less than 50 grams.

Maximum quantity was observed for cold assemblies, having a low burn up. It was then about 100 g.

2 Sodium adheres of course mostly on cold parts of assembly (foot, head, grids, fertile pins, hexagonal cladding) but also on fuel pins where it seems to be liquid but present as a thin film.

3 We use wet nitrogen, then rincing by water of the sodium hydroxyde formed. No problem about washing.

Severe corrosion problems of fuel pins would occur if washing were not well performed.

4 We use no vapour (or steam). It is rather a cold mist of water in nitrogen.

5 It is some times necessary to brush the fuel pins when they are dry.

(ix) Q. Kindly show us your opinions whether it's possible to use the X ray radiography in a hot cell to the fuel assemblies and to the fuel pins which have been irradiated for a long time, and kindly show us your opinions about the neutron radiography and its problems.

A. It is certainly possible to use X ray radiography in a hot cell, even on high burn up assemblies.

The point is that you cannot see through a whole bundle of pins, except the fact that one pin would have moved from its grid.

However one can see very easily what is inside of fuel pins, if there is only one layer of pins that is, only the thickness of one pin to check.

About neutron radiography, we have not yet a good in-cell experience of such an examination. But we have very good pictures of neutron radiographies of pins obtained in a reactor. It is certain that one can see much more by this means than by X-ray radiography. And once more, neutron radiography should be used only on pin layers and not on a whole assembly.

(x) Q. In the case of a series of post-irradiation tests of many samples in a short time in a hot cell, what is the neck from the point of view of equipments and these operations? If there are any necks, which equipments should have any abilities to resolve these necks? Kindly show us your opinions from your experience of the hot cell operation in France. If it may be permitted, we hope to know the results of the working time study in the hot cell operation in France.

A. Rather difficult to answer this question. There are necessarily necks, because some operations are slower than others. For instance γ scanning and dimensional examination of fuel pins are slower than bubble leak detector. X-ray radiography is also rather slow. But if there is room enough and if sufficient equipment is available, there might be no problem.

A further discussion about the planned equipment of F.M.F. would be necessary.

(xi) Q. In the case of the remote reconstruction in a hot cell, kindly show us your opinions about the following problems,

a) method of the confirmation of the situation of a reconstructed fuel assembly

b) reliability of flow test and its problems

c) reliability of vibration test and its problems

d) method of the confirmation of the reliability of the welding works which have been performed in a hot cell.

A. We have a certain experience of reassembling of some separate components of prototype irradiation assemblies, but not of roal reconstruction.

a) We intend to reconstruct some prototypes assemblies some day. Which means that we think it is possible.

b) No experience in cell.

c) idem

d) No experience yet of welding works in hot cell. We intend to do it some day. We think it is possible. E.B.R.II is a good example.

3) Information

(i) ADAC の fuel pin のキズの検査方法（fiscion gas の検査方法）

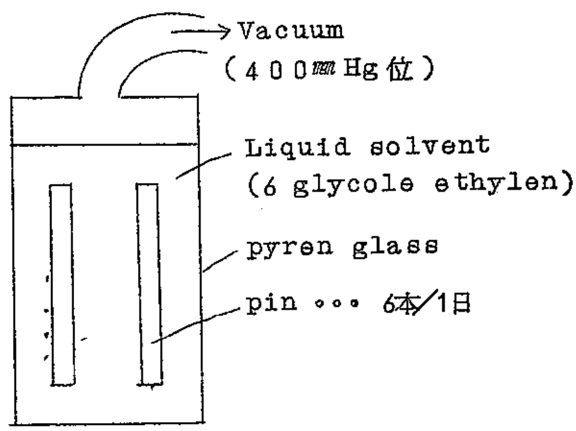
右図のように fuel pin を liquid solvent の中に入れ容器を真空に引きそのときの泡の状態を観察することにより failure の有無を検出する。

failure のあるときは

1点から気泡が出る

failure のないときは

全体から気泡が出る



検出前の fuel pin の洗浄は

0℃位の wet nitrogen（水と N_2 の混合物）で洗浄する。

検査時間は現在

failure のあるとき　　約30分／6本

〃　　ない〃　　約　5分／6本

1 cell の critical 量は6 fissile kg（約3 subassemblies に相当）である。

(ii)　Burn up

Rapsodie は設計では30,000 MWD/T での取替えであつたが、現在50,000 MWD/T で12 subassembly ずつ取替えることを計画中である。現在まで ADAC で取扱つた subassembly の最高 burn up は約40,000 MWD/T である。

(iii)　Cut の前に行う一般的な Check

a) Geometrical check …… bowing, twisting, diameter など

b) Visual inspection

c) Metallugical check

d) Weight measurment

e) Cladding failure detection

f) Neutron diffraction

g) Micrography

11. 放射性廃棄物処理系および炉サービス系

Radioactive Waste Disposal Systems and Reactor Service Systems

1) Comment

no comment

2) Question and answer

(i) Q. Would you please explain the following item.

a) How do you estimate the radioactive nucleide and their quantities (volume & Curies or weight) on gases-, liquid- & solid radioactive waste.

b) How do you treate (waste) radioactive contaminated sodium, specially sodium in Cold trap?

c) How do you estimate the radioactivity of primary sodium and cover gas under the normal operation condition and also under the operation condition with some number of fuel pin failure?

A.

(ii) Q. We would like to know also with regard to radio active wastes as following:-

1) What kind of waste and its activity,

2) Quantity,

3) Treatment way.

A. Concerning radioactive waste, see the reports which were given at Cadarache to PNC experts.

3) Information

P 382 3) Waste disposal 参照

12. 格納容器および建屋施設

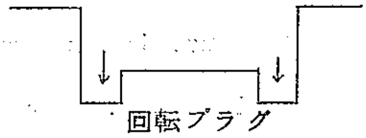
1) Comments

(i) 500 cc/day は非常にオーバーだ。Rapsodie では0、もし少しでもあれば炉を停止して修理する。

(ii) 一次Na純化系を格納容器に入れることについては agree する。

(iii) Operating floor の変更について

Operating floor を持ち上げたのは賛成できないが、回転プラグまわりの床は作業に支障のないように十分広くとる必要がある。



2) Questions and answers

(i) Q. We would like to know the structure and the atmosphere under the operating floor around the rotating plug on Rapsodie.

N.B.: When the hypothetical accident would happen, do you consider that the sodium would leak to these parts?

On the JFER, we are considering that we will have to prepare something for these parts.

A. Operating flow 下の屋根裏部分の構造と雰囲気：

Vessel 囲りについてはCEA−R3406 P.18 を参照されたし、この図で緑色がN_2系、黄色はAr系を示します。Operating flow 下部については非常に複雑だが、Rapsodie の資料参照のこと。Vessel まわりの雰囲気はN_2（記号①で示す）であり、ナトリウム配管系の部分はdepleted air（O_2 5%）（記号②で示す）であり、その外側は炉室の空気（記号③で示す）と同じである。原則として炉室の空気圧は（③）大気圧よりも高く、Ar圧は炉室圧よりも高い。又窒素圧①はdepleted air ②圧よりも高い。圧力の順序はAr系＞N_2系＞depleted air系＞炉室空気である。Rapsodie では maintenance を行なつた場合、通常運転開始前にdepleted air 領域の tightness がほどこされる。

仮想事故時にはVessel が破壊して、N_2 の雰囲気となつている安全容器内にNa が流出するとしている。

(ii) Q. Please show us the size of equipnent hatch and personal air lock on Rapsodie.

What kind of equipment do you transfer through this hatch or this air lock?

N.B. : JFER: equipment hatch : 4 x 5 m

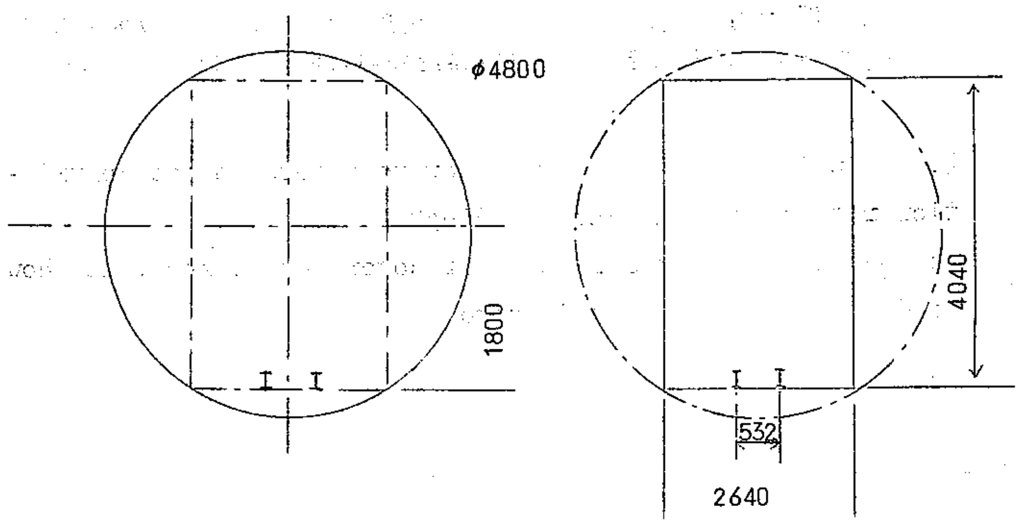
we can transfer the casing of outer rotating plug as the largest one through this hatch

personal air lock : 1.85 m x 0.75 m x 3.82 m (hight) (wide) (length)

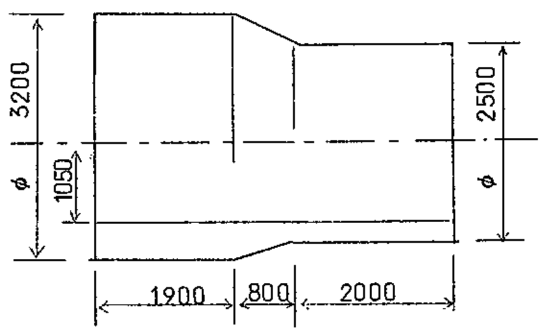
we wonder if it is too small

A. Rapsodie のエクイップメントハッチ、パーソネルエアロックの寸法は第12−1図に示す。

SAS A MATERIEL



SAS A PERSONNEL



(iii) Q. Please show us the condition of the ventilation (cycle of ventilation, temperature, humidity) and the purification method of injection air of the container in Rapsodie.
Please show us the recirculation system of the ventilation in container if you have in Rapsodie
N.B. : JFER : 2 to 3 times/hr - 38° C in summer

A. Normal conditions of temperature and humidity are insured by the ventilation system of Rapsodie. (air flow between 1500 m3/h and 4500 m3/h).

ＪＥＦＲの２～３回／ｈｒは大き過ぎる。
Rapsodie の格納容器内容積　　　２５,０００ m³
上記の空気流量のときの温度は２０℃、湿度は３０％
Rapsodie では放射能棟と原子炉容器との間の door が開かれるときは、air の flow を上記の値より大きくして原子炉容器側から放射能棟の方へ空気が流れるようにしてある。格納容器内の換気再循環は行なつていない。

(iv) Q. Would you please show us how to determine the Ventilation rate of the reactor building?
We wonder if the leak rate of Ar cover gas is zero, and how to detect the leak rate is zero?

A.

(v) Q. Is thermal insulation attached around the seal part of the equipment hatch door?

A.

(vi) Q. Design condition of the reactor containment is required. Applied code or regulations, if any.

A.

1. Comparison on the analytical methods between in Japan and in France

1) Question

To show the methods of analysis, the assumptions being set up, the constants for calculation and the results which are analysed in France, on the same items as all ones which have been carried out in Japan.
To show also us these reasons in the cases that there are any remarkable differences on the methods of analysis, the constants, the results, etc., between in France and in Japan.

		In Japan	In France			Reasons in the case that there are remarkable differences
		Methods of analysis. Constants set up Results	Methods of analysis (Comparised the assumption)	Constants being set up	Results	
Stabilities		see Sep. Vol. III				
Bowing effect of fuel subassemblies		see Sep. Vol. III				
analysis of accidents	Reactivity insertion accidents ① Start-up accident,	see Sep. Vol. III				
	② Control rod drawal accident on power,	"				
	③ Fuel loading accident,	"				
	④ Cold sodium accident,	"				
	⑤ Sudden increase of the primary coolant flow,	"				
	⑥ Bowing effect of fuel subassemblies	see Sep. Vol. III				
	⑦ Fuel slumping accident,	"				
	Loss of coolant flow. ① External power loss,	"				

	② Loss of pumping power in the main primary system,	"					
	③ Loss of pumping power in the main secondary system,	"					
	④ Loss of power of blower on the main air cooler,	"					
	⑤ Local blockage of coolant flow,	"					
analyses of ac-cidents	⑥ Leakage in the primary system,	see Sep. Vol. III					
	⑦ Leakage in the secondary system,	"					
	⑧ Rupture of the pipe in the main primary system,	"					
	⑨ Rupture of the pipe in the main second-ary system,	"					
	⑩ Rupture of the pipe in the emer-gency primary system,	"					
	⑪ Rupture in reactor vessel,	"					
	⑫ Rupture between high and low pressure plenum in the core inlet,	"					
	⑬ Failure in the control valve of blanket coolant flow,	"					
	⑭ Failure of the waste gas disposal facilities	"					
Meltddown accident		"					
Sodium-air reaction		"					
Hazard e-valuation		"					
Hazard e-valuation from plu-tonium							

(2) Answer - Comparison of the analytical methods -

Table (1) shows the comparison of the main features of the two reactors.

The dynamic code used to study the behavior of the reactor is of digital type (see our memo Dec. 4, 68) and describes the reactor core and the two independant cooling loops (Rapsodie).

In the following, we will give the incident studied for Rapsodie, and some remarks on the incident studied for JEFR. (See also paper by J. FOURNIER at Aix Conference)

a) - Bowing effect of fuel subassemblies -

Assumptions made for the calculation of the various reactivity effects due to the successive subassembly deformations, are given thereafter (3)

The differences with JEFR calculations are not originated in the theory (almost the same) but in the choice of the spacer pad positions and the initial clearance between two adjacent spacer pads. These two values were optimized in Rapsodie, in order to fulfill the following theoretical criteria:

- compact core at spacer pad level for nominal operation
- no contact at the hexagonal can heads, before compaction of the core at the spacer pad levels (in fact there is no contact up to nominal power)
- minimization of the total effect due to a hypothetical buttressing of the subassemblies.

TABLE 1 - Reactor data

		JFER	RAPSODIE
Delayed neutron effective fraction	$10^{-5}\,\Delta K/K$	455	532
Delayed neutron group ratio		0.0289	0.034
		0.2125	0.210
		0.1850	0.189
		0.1481 ?	0.391
		0.3829	0.140
		0.0426	0.036
Decay constant	s^{-1}	0.0132	0.0128
		0.0323	0.0316
		0.129	0.122
		0.334	0.322
		1.38	1.39
		3.73	3.83
Prompt neutron life time	s	$1.88\text{x}10^{-7}$	$1.11\text{x}10^{-7}$
DOppler coefficient $T\frac{dk}{dT}$	$10^{-5}\,\Delta K/K$	-.93	0
Fuel axial expansion coefficient	$10^{-5}\,\Delta K/K\ {}^{o}C$	-0.26	- 0.20 (*)
Sodium expansion coefficient	"	-0.76	- 1.44
Structure expansion coefficient	"	-0.90	- 2.21 (**)
Fuel heat conductivity	$W/{}^{o}C\text{xcm}$	0.029	0.027
Clad and structure heat conductivity	"	0.195	0.208
Coolant heat conductivity	"	0.72	0.690
Fuel density	g/cm^3	9.3	10.19
Clad and structure density	"	7.82	7.73
Coolant density	"	0.864	0.847

(*) Fuel free to expand inside the cladding.

(**) This coefficient takes into account the effects due to steel and support plate expansion, and the spacer pad effect.

TABLE 1 - Reactor data (continued)

		JFER	RAPSODIE
Specific heat of fuel	j/°C xg	0.32	0.327
" " of clad and structure	"	0.577	0.590
" " of coolant	"	1.283	1.267
Heat transfer coefficient of He gap	w/cm2x°C	0.848	0.667
Maximum power density of average channel	w/cm3 of fuel	1260	1020
Maximum power density of maximum channel	w/cm3 of fuel	1820	1255
Over-all peaking factor		1.70	1.45
Flow rate of coolant	m/s	4.1	3.6
Coolant inlet temperature	°C	370	410
Safety rod reactivity	10^{-5} $\Delta K/K$	3642	~ 7000
Scram delay period	ms	200	200

These criteria allowed us to place the spacer pads 7 cm above the core midplane and to design the subassemblies with a 50 μ clearance between two spacer clads close one to the other.

b) - Start-up accident -

Rapsodie -

The accident, calculated for Rapsodie, is the withdrawal of a safety rod (0.37 ¢/s) or the withdrawal of a control rod (41 ¢/45 s) from a given power of 2 W, primary and secondary flow set at nominal flow.

JEFR -

The reactivity insertion accident chosen seems to us pessimistic. You suppose, as a matter of fact, three successive malfunctions:

(a) - unwanted withdrawal of one safety rod

(b) - this safety rod is operated by a control rod mechanism (and so the reactivity insertion rate is multiplied by 3)

(c) - Usually, one must begin the reactor start-up by progressive safety rod withdrawal, when control functions are effectively separated. But then, the reactor cannot be made critical, only if all the safety rods are withdrawn !

Moreover, one does not understand why the reactivity insertion rate you obtained is increased by 40 %.

In the second case where the threshold scram is set at 108 MW, for a given flow of 1 % of nominal flow, it is abnormal not to consider a threshold on the outlet Na temperature of the subassemblies, or on the comparison between flow and power. A threshold of this last type could allow to stop the reactor in this case.

c) - Control with rod/drawal accident at rated power -

Rapsodie -

The accident considered for Rapsodie is the safety with rod/drawal (0.37 ¢/s) or the control with rod/drawal (41 ¢ in 45 s) from the nominal power, primary and secondary flow set at nominal values. Thus, the power increases up to 2.4 % /s in the case of the control rod.

JFER -

One can make the same remark here, as for the start-up accident, as the calculations of the reactivity insertion rate are concerned. The time constants of the cover-plate thermocouples in Rapsodie are larger (about 3 s). A threshold scram on the power-flow ratio would be faster than on the thermocouple values.

d) - Cold sodium accident -

Rapsodie -

We took into account the increase of the air flow from 11 to 320 % for an initial power of 3 MW, primary and secondary flow being set at nominal values. This accident could not happen in Rapsodie, we put mechanical stops in order to limit the amount by which the blower blades can be opened.

The inlet temperature then decreases by 140 °C at 2.7 °C/s.

JEFR -

A larger variation of the inlet temperature than those considered herein could be got by an increase of the air flow, for instance, from 10 to 120% for an initial power of 10 %, primary and secondary flow being set at nominal values. But, this accident could not seriously affect the reactor, we think.

e) - Sudden increase in the primary coolant flow -

Rapsodie -

We consider the following accident : Initial power is 2 W, Na inlet temperature in loop n° 1 is 450 °C and the Na flow is minimum; in the loop n° 2, the Na temperature is 150 °C and the flow is raised linearly from 0 to 140 % of the nominal flow within 2 minutes.

In fact, this accident is impossible because operators are not allowed to operate the reactor with only one loop. Moreover, valves of the primary pumps are machined with a hole, so that the Na temperature is homogeneous inside the unoperated loop.

JEFR -

We think, that one has to take measures in order to avoid this type of accident.

f) - Loss of coolant flow - External power loss -

Rapsodie -

In case of loss of electrical power, nothing happens during 2 to 3 seconds, the pumps speed regulation keeps the flow constant and the rods do not move. Then, the rod drops are initiated by the pumps' cut-outs and the primary and secondary flow decrease with the group inertias and the air flow with the blower-motor inertia. Primary and secondary decreasing flow versus time are given in第安一1図 One can see that approximately the primary flow decreases by a factor of 4 at the end of 2 minutes and the secondary flow decreases by a factor of 4 at the end of 2,5 minutes.

The diesel starts to take up the emergency electric load; a primary Ward-Leonard group is automatically switched after a maximum of 30 s for a pump speed of 300 t/m (the nominal speed being 1000 t/m). If none of the diesels starts, batteries, possessing an autonomy of 1/2 hour, supply the two primary pumps at 100 rpm.

The experimental results obtained when none of the two diesels starts, show that:

- the power reaches 10 % of its initial value 15 s after the beginning of the rod set back;

- the reactor inlet and outlet sodium temperature and the subassemblies sodium outlet temperatures are given on 第・安－2図.

JEFR -

The decay time constants of secondary flow and air flow seem to be very small. It seems that there is a mistake on the scale of the power curves, so that the temperature so-called core outlet temperature be the maximum coolant temperature.

g) - Loss of pumping power in the main primary cooling system -

Rapsodie -

The experience made is the primary pump cut-out which is the most probable accident. A rod set back is initiated; the global reactor flow remains pratically constant during 15 s. Na temperature at the subassemblies output remains always below the nominal values.

We have studied by calculations the case of a primary pump jamming. This jamming has been represented by a pump stop in one second. It has been supposed that shut-down is provoked at the initial instant. It is thus found for the maximum sodium temperature and the cladding an increase respectively of 45 °C and 40 °C after 2 seconds.

JEFR -

The 50 % flow step is not realistic. The time at which appear the maximum temperatures seems to be short and the maximum temperature obtained seems to be high.

h) - Loss of pumping power in the main secondary cooling system -

Rapsodie -

The experience done is the cut-off of a secondary pump which is the most probable accident. The case of a secondary pump jamming has not been studied by calculations. The rods set back is initiated and the air blower is cut off. If it was not cut off, the sodium air exchanger outlet temperature would decrease at a rate higher than 50 $^{o}C/min$.

JEFR -

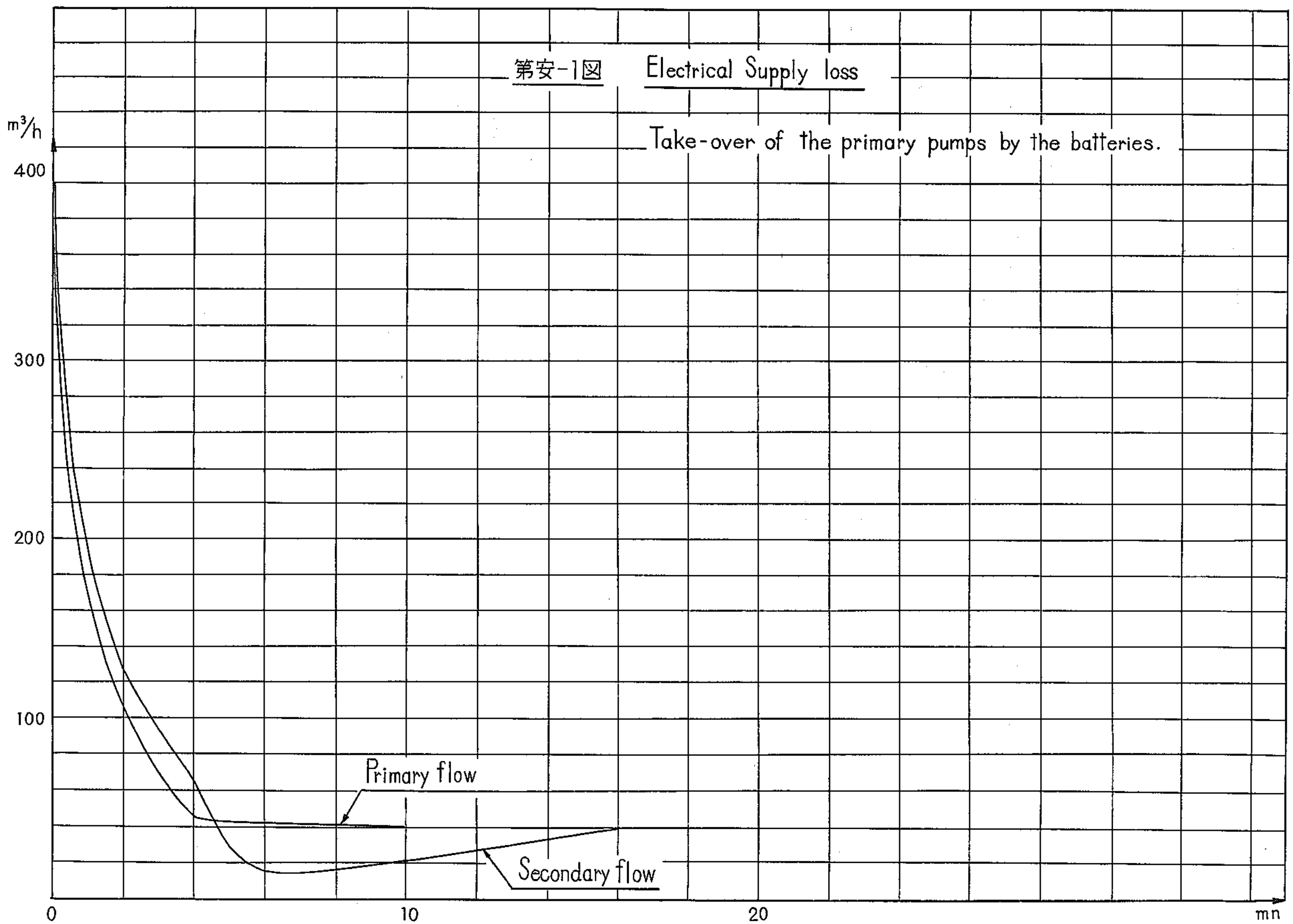
The calculation assumptions made do not define the secondary flow law and the action on the air flow. It is not described if the incident appears on a single or on the two loops simultaneously.

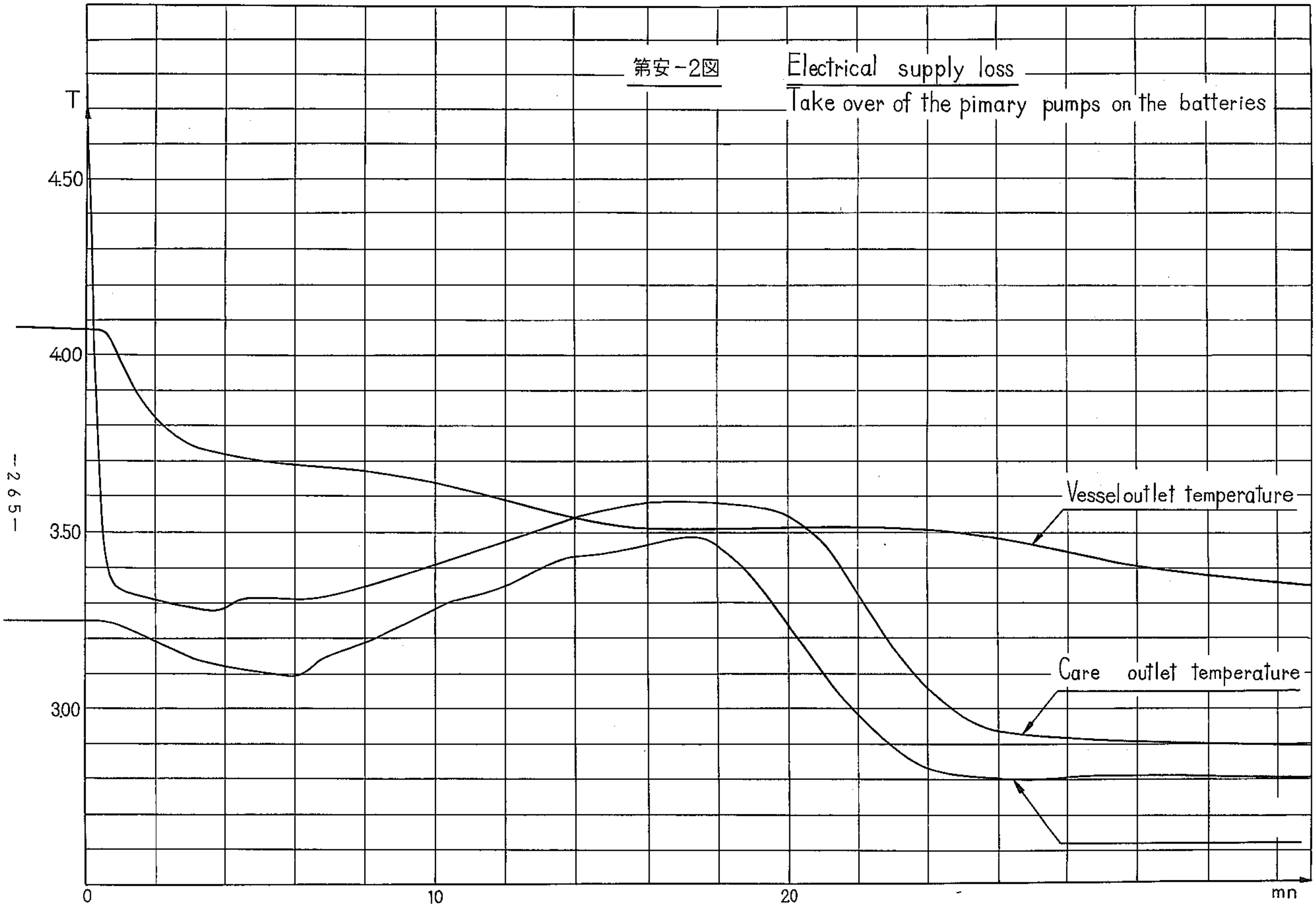
For this type of accident, the zones of the reactor to be protected are not the core but the structures and the exchangers.

i) - Loss of power in the main air blower -

Rapsodie -

The experience done is the cut off of the blower motor. The blower blades are reset to zero in approximately 15 seconds. The set back is initiated by the cut off signal. This incident is of no consequences on the plant and the possibility is foreseen to shunt all safety actions on such incident.


第安-1図 Electrical Supply loss
Take-over of the primary pumps by the batteries.
m³/h
400
300
200
100
0
10
20
mn
Primary flow
Secondary flow


第安−2図
Electrical supply loss
Take over of the pimary pumps on the batteries
T
4.50
4.00
3.50
3.00
0
10
20
mn
Vessel outlet temperature
Care outlet temperature

2. Stabilities

1) Comments

Rapsodie は基本的には JEFR と同じ解析手法を使つている。但し次の3つ comment の与えたい。

(i) bottom support plate による効果・・・・重要なのは一次系を通しての feed back による温度の影響が support plate にも影響を与えるためである。

Rapsodie では support plate の isothermal coefficient は通常運転時で

compact の場合 全 isothermal coefficient のうち10%以上

compact しない状態では 〃 30%位

である。danger 係数から purterbation により計算から求めた温度係数は

compact の場合 4×10^{-6}

compact しない状態 14×10^{-6}

位である。

(ii) 制御棒の relative motion による効果・・・・vessel の expansion について JEFR では無視しているが relative motion の効果を考慮に入れる必要がある。何故ならば rotating plug は vessel に対し相対的な動きをするからで

制御棒が入つているときは vessel は(−)の影響を与え

〃 〃 いないときは 〃 (+) 〃 る

また feed back は internal には1秒 outernal には分のオーダーなので計算が非常に困難である。

(iii) stabilities について・・・・EBRの事故以来および今後共問題を生じないであろう Bowing による(+)の効果は pad の位置の設計により避けうるし、(−)の効果をもたせるように十分設計できる。

2) Quations and answers

(i) Q. How to introduce the bowing effect of the fuel subassemblies in stability analysis in France?

A. - Coefficients for local gradients pcm/$^{\circ}$C/cm ($1\ pcm = 10^{-5}\ \Delta k/k$) are calculated in order to take into account:

- the "fanning out" (before core compaction and without subassemblies heads in contact)

- the bowing (effect of radial gradients on a compact core and without subassemblies heads in contact)

- the "buttressing" (effect of radial gradients on a compact core with subassemblies heads in contact)

If it is supposed that the inlet sodium temperature in the core is constant, these different coefficients for local gradients may be converted to temperature coefficients. They are taken into account by the stability code calculation in the same way as the other temperature coefficients.

Anyway, we remind you that bowing coefficients and buttressing coefficient are made negligible, after correct optimization of the spacer-clads positioning.

(ii) Q. Does not the instability happen in the whole of the system including the control system in Rapsodie? Or, from these reasons did it happen that you couldn't adopt some control scheme?

A. - In Rapsodie, we do not have, for the time-being, any automatic control (all automatic controls were actually suppressed, as being not necessary, or even harmful).

(iii) Q. Are there consistencies between the results of the calculation and those obtained by the experiments in Rapsodie?

What are the reasons if they are not consistent?

A. - The table, hereafter, compares calculated and experimental results:

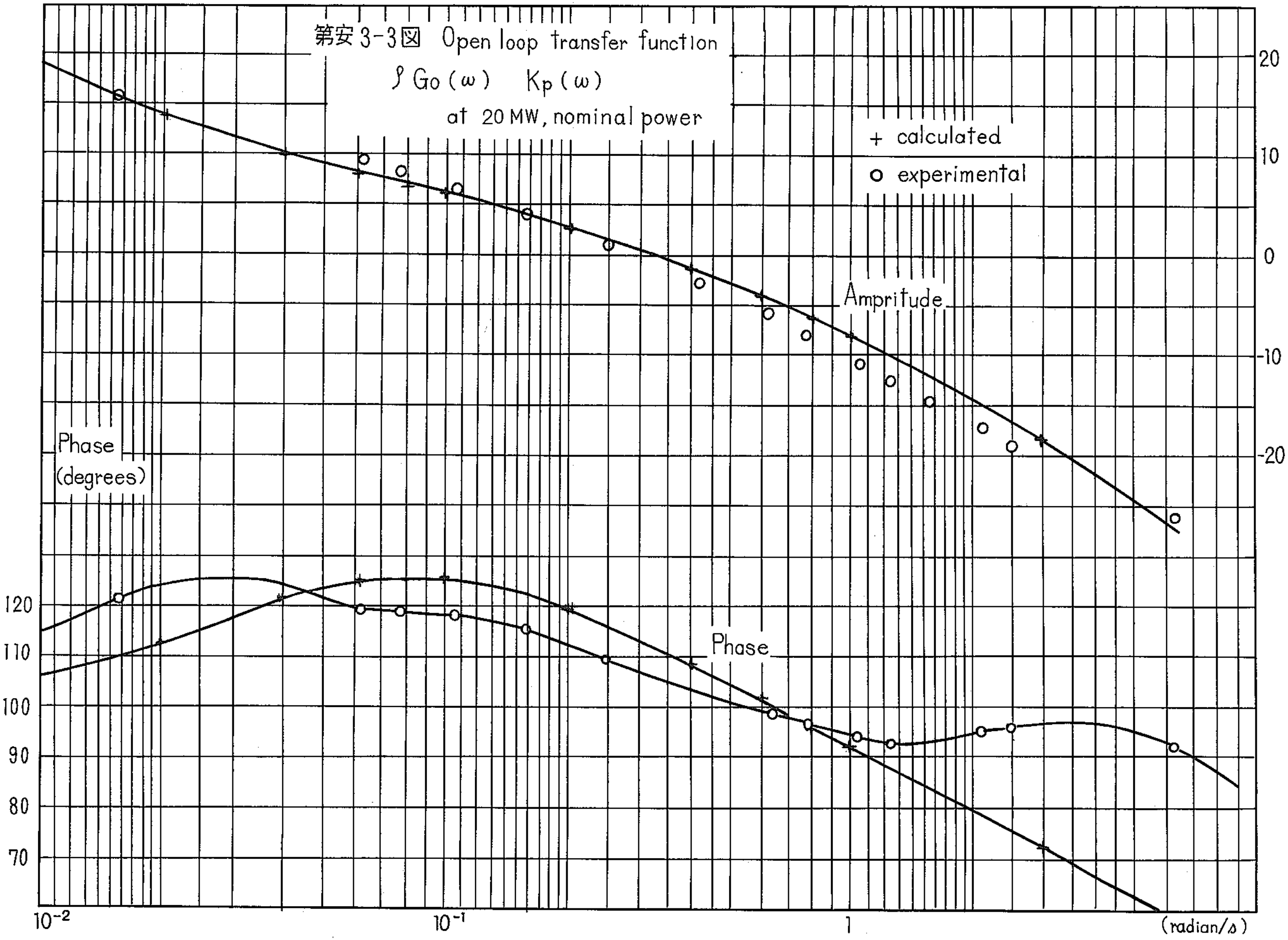
	calculated	experimental
Isothermal pcm/^{o}C	-4 to -3.6	- 3.7
Integrated power coefficient		
(pcm) - new fuel (*)	- 510	~ - 650
- used fuel (**)	- 210	~ - 180

(*) calculated by assuming the fuel free to expand in its cladding and that this expansion is due to the central temperature of the fuel oxide pellets.

(**) calculated in assuming that there is no more fuel expansion.

It seems that the reactivity effect due to the core height increase was underestimated.

Fig. (3) shows the open loop transfer function obtained experimentally and calculated for nominal conditions. The phase margin is calculated to be 112 degrees and the experimental phase margin was measured to be 105 degrees.


第安3-3図 Open loop transfer function
ρ Go (ω) Kp (ω)
at 20 MW, nominal power
+ calculated
o experimental
Amprifude
Phase
Phase (degrees)
20
10
0
-10
-20
120
110
100
90
80
70
10^{-2}
10^{-1}
1
(radian/s)

3. Bowing effects of the fuel subassemblies

1) Comments

(i) bowing analysis で(+)の効果をもたないように、pads の間隔、Assembly 間の間隔を調節出来ないか。pad がくつつく前に Assembly 先端がくつつくのがよくない。Power Coeff は Burn up とともに減少するので出来るだけ設計をうまくして反応度の(+)の効果はもたさないようにしておくこと。

(ii) 図面によると JEFR の場合 assembly の upper part の方が先にくつつくので(+)の影響があるようだがもし可能なら設計変更した方がよいと思う。

Rapsodie においては Clearance はパット部 0.05 mm、upper part 部 1 mm である。多少の positive でも避けた方がよい。運転と共に出力係数が減少していく傾向がある。

2) Questions and answers

(i) Q. What assumptions do you adopt on the calculational model to obtain the bowing configuration in Rapsodie?

A. - The reactivity effects are calculated from the danger coefficients table and from the local changes of material composition per cm^3.

The changes of local material composition are calculated from mechanical and thermal distorsions of hexagonal cans.

When rising power, it is assumed that the three following states appear successively:

- the subassemblies are set in the support plate (without clearance) and are free to expand without any contact between them. This is what we call the "fanning out" coefficient, which is calculated from the free distorsion curves due to radial thermal gradients;

- the core is compact at the level of the spacer pads and there is no contact at the hexagonal cans heads. Then we define the following coefficients:

- the temperature coefficient of the spacer pads, which is due to the expansions of the hexagonal sections of the cans in the spacer pads plane;
- the bowing gradient coefficients which are due to the effects of radial gradients in the hexagonal cans which are

supposed to be fixed in the support plate and to have a fixed axis at the spacer pad level. This total bowing effect may be equal to zero, if the spacer clads are correctly located.

- the core is compact at the level of all the spacer clads and the pessimistic assumption is made that all the can heads are in contact. We then define the hypothetical gradient coefficients due to the "buttressing" of the hexagonal cans. This effect is quite hypothetical and may be also equal globally to zero, if the spacer clads are correctly located.

On the other hand, we have verified by calculation, that no contact at the hexagonal can head can occur before the whole core is compact at the spacer clad level. The calculated results are given hereafter:

fanning out coefficient	- 0.44 pcm/MW
bowing	+ 0.06 pcm/MW
spacer pads (-1.11 pcsm/°C)	- 4.74 pcm/MW
buttressing	- 0.25 pcm/MW

(1 pcm = 10^{-5} $\Delta k/k$)

(ii) Q. Did you carry out the mock-up tests on the bowing configurations and their nuclear effects?

To explain us those outlines if you did.

A. - measurements of the thermal and mechanical distorsion curves have been made on real hexagonal cans. These measurements were in good agreement with the calculations made. No measurements of reactivity effects, as related to these distorison curves, we made.

(iii) Q. You have measured the thermal bowing of the fuel assembly. Would you please tell us any reports which cover experimental apparatus and procedure?

A. Measurement of subassembly thermal bowing

For Rapsodie we had measurements of subassembly thermal bowing made in "Laboratoire National d'Essais" (LNE). (See the internal report: Note Technique L.N.E. Th. 101-41 Rapport Partiel N°2 du 23 mai 1961 "ETUDE EXPERIMENTALE de la FLEXION THERMIQUE d'un TUBE a SECTION HEXAGONALE") This test was run by heating and cooling two opposite faces of hexagonal cans with rubber flot capacities where water was flowing. By this way, temperature differences between two opposite faces. Different outside conditions were imposed to the dummy subassemblies.

The measured bowing was in very good accordance with the computed one. For that reason, we do not consider it is useful to run this test again.

It must be noticed that the main uncertainties in thermal bowing estimation are due to temperature distribution in the hexagonal can, especially if helical wires are used and mixing test as well for computation.

(iv) Q. How to take accounts of these results? Are there any modifications being improved on the design?

A. - this question is of no interest to us.

(v) Q. Bowing effect

Would you explain the reasons why bowing cause decrease of power coefficient with burn up?

A.

4. Accidents analysis

1) Comments

(i) descriptionのp55 にある 850°C は高過ぎるように思う。Rapsodieでは 800°C を採用した。

simulation calculationの方法は基本的には日本と同じだと思う。time constant, delay constantも同じだと思う。違うのはアナログとディジタルの解析手法の違いだと思う。Rapsodieについての経験より下記の事項に注意する必要があると思う。

① ポンプの熱容量をneglectして計算したのはよくなかつた。

② upper plenumに対してはmiscing time constantを採用する必要がある。

③ 熱交換器のplenumの熱容量を無視して計算したのはよくなかつた。

(ii) reactivity insertion accident

① start up accident

Rapsodieでstart up accidentの検出のために使つている計器はPeriod meterだけでＪＦＥＲのようにrangeを切替えてscram point を順次上げていくような方式はとつていない。同種の計器の scram pointは２０ＭＷの１０％増に置いたまま起動する。

ＪＦＥＲ方式では切り替え忘れや切替時間など別の問題が起きてくるのではないかと思う。切替中のtimerが必要とならないか

② Cold sodium accident

フランスでは１ループ運転中他の停止中のループを急に起動することによつて起る事故のことをいつている。ＪＦＥＲの１ループ模擬ではこのような現象は出てこないので２ループで計算しておく必要がある。

(iii) Loss of coolant flow

Descrption Volume Ⅲ Fig５７のLoss of external accident時のoutlet temperature of coreはもつと高くなるのではないかと思う。

coreのupper partにある熱電対はlong timeの変化に対してよいが、１本の燃料がmeltするような急激な場合は熱電対では追従できない。この場合にはsodiumがboilingし、sodiumが移動することによつて負の反応度係数を与えscramできる。この現象のpropagationは周囲の６本に影響を与えるだけでstopするだろう。

2) Questions and Answers

(i) Q. How do you judge the philosophy which we adopt to the fuel rupture in the case of the hazard evaluation?

To show us the criteria which you adopt in France on this matter.

A. - In case of Rapsodie, for the safety point of view, we do not have definite criteria for fuel rupture in accidental circumstances (see paper on Rapsodie safety at the Aix Conference)

(ii) Q. Do you take in the view that the center melting of fuel pin may be admitted? What is the ground if it may be admitted?

A. - Fuel melting at the pin center is not allowed on normal operation. But, if under accidental circumstances the fuel melts at the center of only a few pins, this situation probably should not be dramatic, provided that irradiation of these pins does not continue and that the damaged subassemblies are unloaded.

(iii) Q. How do you consider the burn out phenomenon on the design of a fast reactor?

Is it necessary to set up the same criteria as in BWR?

A. - We do not consider the burn-out phenomenon. But, if one has to consider this phenomenon in the future (this is not obvious, since the phenomenon which occur with boiling are not the same, as compared to all the water reactors), one will do that only when conclusions on boiling studies will be available.

(iv) Q. What value do you adopt to the helium gap conductance in France?

What is the base of that value?

How do you treat this value on the transient case though we consider it becomes smaller in case of the startup accident especially?

A. - 0,667 $w/cm^2/^oC$ for all our studies on fast reactors - this value, known with a bad accuracy, comes from experiments. We do not change this value in the transient case studies.

(v) Q. To show us the scram trip and its delay time being estimated for each accident, and these grounds in France.

A. The time constants of the detection chains are given in table 2. The time between the trip and the beginning of the rod motion is 0.2 s. We did not make a systematic study of this delay for each accident.

TABLE 2 - Safety thresholds

	logic	time constant	threshold	action	
linear power	2/3	<10 ms	25 MW 26 MW	alarm scram	
logarithmic power	2/3	<10 ms	30 MW 35 MW	alarm scram	
Period log meter	2/3	variable	30 sec 10 sec	alarm scram	
Seism	2/4			scram	
Flow power ratio	2/3	<1 s	1.265 1.33 1.66	alarm set back scram	
Subassemblies outlet temperature (computer)(*)	1/84	<4 s	+ 5°C +10°C +15°C	alarm set back scram	
γ Radio protection	2/3	<10 ms	25 mrh 125 mrh 250 mrh	alarm set back scram	
linear minimum counting	2/3		2 c/s	scram	inhibited during reactor in power
linear maximum counting	2/3	~10 s	50 - 500c/s		
linear maximum counting	2/3	<1 s	8000-80000c/s		
logarithmic maximum counting	2/3	<1 s	10^5 c/s 1,2 10^5 c/s	alarm scram	
period meter associated with the impulsion chambers	2/3	variable	30 sec 10 sec	alarm scram	

T A B L E 2 (continued)

	Temporisation	action
cut-off primary pump	2 s	set back(**)
" secondary pump	2 s	"
" air blower	1 s	"

(*) This safety threshold is given by a digital computer and permits to detect an abnormal subassembly. Measured value is compared to the computer calculated one.

(**) This set back is stopped when the flow to power ratio becomes lower than 0.1.

Addition:

① Actions de securite a ajouter a la liste donnee par M. J. LADET -

A - temperature sortie assemblage (traitement analogique) logique 2/3 (2 fois)

seuil : Alarme $10^{o}C$ au-dessus du normal
DB $20^{o}C$ " " "
AU $30^{o}C$ " " "

B - bon fonctionnement des chaines neutroniques lineaires de securite
logique : 2/3
action : "Scram" si il manque l'alimentation electrique 24 V
ou la THT
ou une liaison electrique mauvaise

C - manque d'alimentation electrique sur les moteurs de barre lorsqu'il y a une commande de D.B. (set back):
action : scram

② Signalisation sonore et lumineuse -

Pour toute action de securite (scram ou set back) il y a en meme temps declenchement d'une alarme sonore d'un voyant general et d'un voyant particulier sur un synoptique.

Pour les mesures analogiques declenchant une action de securite automatique, il y a toujours un seuil declenchant uniquement une alarme sonore et lumineuse avant le declenchement de l'action de securite elle-meme.

Il y a aussi de nombreuses autres alarmes sonores et lumineuses sur toute l'installation (de l'ordre de 1000).

(vi) Q. Give us comments about the items and these values on the scram and the alarm systems on the JFER? To show us also the table of these on Rapsodie.

A. It seems that for Rapsodie the number of safety interlocks is adequate. In particular, the comparison of power and flow which does not exist at JFER seems to us very interesting. Compared to Rapsodie, we notice no seisms safety device. Besides that, many incidents, considered as minor for Rapsodie, give rise to a scram in JFER. Moreover, any incident on secondary or auxiliary loops must not initiate a scram, but at most, a set back.

(vii) Q. What do you use as the back-up system of the electric power in Rapsodie?

A. All the electrical supply for the control is furnished by floating batteries (1/2 h autonomy). Primary pumps are backed up by the batteries at a speed of 100 t/min. (nominal speed 1000 t/min). Two diesel enginesusually not in operation, but with fast start up, can back up the electrical supply, within a 12 s delay, which is necessary for the plant in a non-operating state. Usually, one primary pump can be supplied up to 30 % of its nominal speed. If none of the diesels starts up, decisions have to be made within one hour.

Rapsodie では下記の原則で考えている。

① loss of current

② scram

③ 数分間は forced convection power circuit により循環する (by big wheel)

④ バッテリ on 1 時間はもつが下記のジーゼルが働くので事実上は必要がなくバックアップと考えている。

⑤ ディーゼルによる電源供給１０

秒位で供給可能

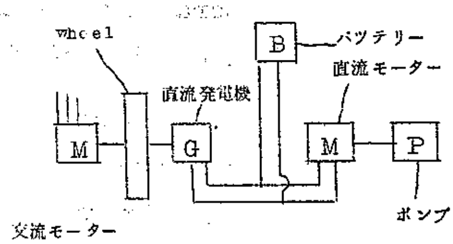
なおこの間３０分後にNa を

heat する必要がある。

電源系統は右図のような

WARD−LEONADD System

を採用している。



(viii) Q. By what criteria do you decide the reactivity worth of safety rods in France?

A. For Phenix, we decided to have a negative reactivity safety worth of about 10 s. This worth allows to quickly decrease the fission power down to 10 % of its initial value. This seems to us quite sufficient for all the incidents which are currently foreseen. This reactivity worth must be higher than any unwanted one, which could be inserted.

(ix) Q. Is it all right to have nothing to consider the back up system of the safety protection system (safety rods)?

What is that reason?

A. - We are not sure we understand this question. No back-up safety system is deemed necessary in France.

(x) Q. Is it unnecessary to install the emergency cooling system?

Are the two loops necessary if installed?

- For Rapsodie, a nitrogen loop, normally used to preheat the vessel, allows to evacuate a part of the power, when the primary loops are out of service. We did not double this loop, but there are two nitrogen blowers. The two primary loops are necessary, even with an emergency cooling, in order to prevent a very fast stop such as a primary pump jamming, for example; as a matter of fact, in the case of only one primary loop, the jamming of a primary pump could give a very fast stop of nearly all the primary flow.

(xi) Q. How do you consider the calculational errors of the reactivity coefficients from the view point of the accident analysis? What percentages of the calculational valves of reactivity coefficients do you use on the analyses? Show us each reactivity coefficient for each accident analysis?

A. - We did not consider the inaccuracies of the calculated danger coefficients, because they are difficult to determine. In general we try to take for the fuel axial expansion the pessimistic assumption, that is to say, the fuel bound to the cladding. In view of the experimental power coefficient of Rapsodie, it seems normal not to take into account the fuel axial expansion since this effect disappears after a certain time of fuel burn-up.

(xii) Q. Is the axial expansion coefficient of the fuel able to be expected if the cracks happen in the fuel pellets at the high burn up condition?

A. - We did not consider the inaccuracies of the calculated danger coefficients, because they are difficult to determine. In general we try to take for the fuel axial expansion the pessimistic assumption, that is to say, the fuel bound to the cladding. In view of the experimetal power coefficient of Rapsodie, it seems normal not to take into account the fuel axial expansion since this effect disappears after a certain time of fuel burn-up.

(xiii) Q. How do you consider, the changes of the peaking factor by the arrangement of the control rods on the accident analysis in France?

A. - We did not take into account the influence of the control rods position on the power distribution inside the core. In fact, this correction has to be taken into account in the hot spot factor calculations.

(xiv) Q. How do you introduce the hot channel factor in the analysis of the transient behaviour?

A. - The calculation code used allows us to study the transient behavior of a channel called "hot channel". This channel does not influence the behavior of the reactor; this means, that it is not taken into account for the reactivity, flow and sodium outlet temperature calculation. Hot spot factors used are the ones corresponding to nominal operation.

(xv) Q. Is the rupture of pipe in the main primary system supposed on the accidents analyses in France? Are not there any troubles in this case that the core would be exposed of the coolant or the coolant would flow to the reverse direction? Or especially are any countermeasures taken for these troubles?

A. Despite of the precautions taken, the rupture of the main primary system is not excluded. We have to avoid absolutely:

- the core draining
- the sodium fire

Sodium pipes get into the vessel at a higher level than the core position. They are equipped with a siphon-breaker within a nitrogen atmosphere.

All emergency cooling loops with nitrogen are placed around the vessel and must allow the extraction of decay heat in case of a total flow loss.

In the same way, the vessel rupture is considered. One must maintain the subassemblies always inside the sodium. A special tank (50 m^3) allows us to inject sodium inside the vessel. By doing so, the level of the sodium, inside the safety tank, will always be higher than the level of the subassemblies' heads.

(xvi) Q. Is the rupture of the vessel supposed in France? What is the reason if it is not supposed?

A. Despite of the precautions taken, the rupture of the main primary system is not excluded. We have to avoid absolutely:

- the core draining
- the sodium fire

Sodium pipes get into the vessel at a higher level than the core position. They are equipped with a siphon-breaker within a nitrogen atmosphere.

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In the same way, the vessel rupture is considered. One must maintain the subassemblies always inside the sodium. A special tank allows us to inject sodium inside the vessel. By doing so, the level of the sodium, inside the safety tank, will always be higher than the level of the subassemblies' heads.

We do not have a reliable model, allowing us to describe an eventual propagation of the sodium boiling.

(xvii) Q. Is it unnecessary to suppose the accident of the channel blockage? What is the reason if it is unnecessary?

A. - The channel blockage is a quite credible accident.

Thermocouples located at the Na-outlets of the fuel subassemblies allow a continuous temperature checking and allow to localize an undercooled subassembly. In a more advanced stage, the blockage, which would give rise to sodium boiling and fuel melting, will give reactivity changes. We do not have in Rapsodie an alarm signal for a given the shold, but the reactor must be shut down if any unexplained variation of 20.10^{-5} k/k is observed.

Detection is very difficult when we have only an clementary channel which is locally plugged. The fluctuations which may be observed at the outlet of the subassemblies, might give indications.

We do not have a reliable model, allowing us to describe an eventual propagation of the sodium boiling.

Q. $20 \times 10^{-5} \triangle k/k$ の根拠は何か

A. これは 2 - 3 centsに相当し、administrative Valueである。実際には one subassemblyの boiling 時の反応度変化分と等価である。sodium boiling, fuel meltなどについて正確なことはわからない。fast blockage は reaceivity meterにより、slow blockage は thermocoupleにより検出される。

この値は2〜3年前には$10 \times 10^{-5} \triangle K/K$と決められていた。しかし、inlet tempがupしたり、blower, vesselなどの条件変化により、scramをする経験があつて2倍の値とした。

(xviii) Q. How do you design on the detection if the accident of the channel blockage happens?

A. - The channel blockage is a quite credible accident.

Thermocouples located at the Na-outlets of the fuel subassemblies allow a continuous temperature checking and allow to localize an undercooled subassembly. In a more advanced stage, the blockage, which would give rise to sodium boiling and fuel melting, will give reactivity changes. We do not have in Rapsodie an alarm signal for a given there should, but the reactor must be shut down if any unexplained variation of $20 \cdot 10^{-5}$ k/k is observed.

Detection is very difficult when we have only an elementary channel which is locally plugged. The fluctuations which may be observed at the outlet of the subassemblies, might give indications.

We do not have a reliable model, allowing us to describe an eventual propagation of the sodium boiling.

(xix) Q. How do you take account of the propagation of the boiling to the channels of the radial direction when the boiling happens in one channel?

To show us the reliable methods of the analysis.
How do you check up experimentally?

A. - The channel blockage is a quite credible accident.

Thermocouples located at the Na-outlets of the fuel subassemblies allow a continuous temperature checking and allow to localize an undercooled subassembly. In a more advanced stage, the blockage, which would give rise to sodium boiling and fuel melting, will give reactivity changes. We do not have in Rapsodie an alarm signal for a given theshold, but the reactor must be shut down if any unexplained variation of 20.10^{-5} $\Delta k/k$ is observed.

Detection is very difficult when we have only an elementary channel which is locally plugged. The fluctuations which may be observed at the outlet of the subassemblies, might give indications.

We do not have a reliable model, allowing us to describe an eventual propagation of the sodium boiling.

(xx) Q. Is it necessary to be supposed the fuel slumping accident? How many pins would happen the slumping at one time if it is necessary?

A. We do not believe in the fuel slumping accident. One cannot visualize what type of accident could cause a simultaneous compaction of many columns.

(xxi) Q. The positive reactivity is inserted caused by the increase of the primary coolant flow or the cold sodium accident when the revolutions of the primary or the secondary pump increase suddenly. What countermeasures do you adopt to prevent the flow rising suddenly?

A. We must avoid this type of accident. More precisely, in Rapsodie, two cases can be considered:

1° - Unwanted start up of the pump of a primary loop not in operation:

a) - it is impossible due to administrative procedures, to make the reactor critical with only one loop in operation

b) - valves of the primary pumps are machined with holes, in order to have an inverse flow in the loop, the pump of which is shut down. By doing so, the temperature differences between the two loops are never large.

2° - Unwanted increase of the flow rate of one pump: in normal operation, the two primary pumps' flow is set to nominal flow, whatever the power level is. The pump controls are equipped with stopscrews but, in any case, the flow could be increased by only a very small amount above the nominal flow.

(xxii) Q. By what means do you detect the fire caused by the sodium leakage in the air cooler?

To show us the principles of the aerosol detector if this detector is used.

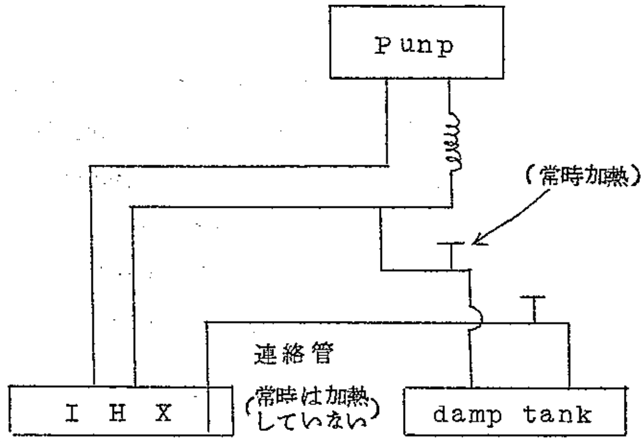
A. In the case of Rapsodie, we have detection plugs for sodium leakage in each Air-Na exchanger. Moreover, projectors are pointed on the air outlet of these exchangers during the night, in order to be able to see, even by night, if there is a sodium fire. For the time-being, we do not have developed aerosol detectors.

(xxiii) Q. The sodium fire would happen when the leakage or the rupture would occur in the secondary system. To explain us the methods of fire fighting or the counterplans to prevent the accident to the minimum.

A. A pipe for emergency drainage is provided in each secondary loop. After reactor shut-down, that pipe allows us to drain within a short delay, the major part of the damaged secondary circuit.

少量のときはまずreactor scram(manual)し，次に連絡管を加熱し（約2時間）落とす。

しかし大量のときは加熱の時間的余裕がないので，scram (manual)後damp用配管の弁を開けdamp tankに落とす。しかしこのときは全部damp することはできない。



(xxiv) Q. Would the sodium solidify in the air cooler if the secondary flow goes down suddenly?

To show us if the special counterplans adopt for it.

A. More precisely, the air blower is shut down automatically in case of secondary pump stop. Moreover, one avoids the natural convection of the air by closing the blower slides in a manualcontrol way.

(xxv) Q. The primary or the secondary sodium temperature must not go down abnormally in the case of the scram operation. Therefore the secondary sodium flow and the air flow in the air cooler must go down suitably. How do you carry out these controls?

A. When the reactor power is down, the only imperative action to do is to shut down the air blowers. It is not absolutely necessary to stop the secondary flow.

Scram後も1次系主ポンプはそのまゝ運転を継続する。2次系は air flow を停止し15分後に予熱し始める。

(xxvi) Q. We are now dicussing on the small design change (+) that the decay heat of the core would be able to be removed by natural convection when all power supplies would cut off. We are now in mind preferentially of adopting the method by the natural convection though it make accompany the influences for the other parts.

How degrees of importance do you pay for the heat removal by natural convection in France?

(+) The difference of the heights at the positions of centers between of the core and of the IHX will be result of going up of the sodium level in the reactor vessel and of shortening the height of IHX.

Q. A great degree of importance has been paid to the heat removal by natural convection in the Rapsodie design as well as in the Phenix design.

自然循環による除熱に対しては非常に重要視している(high level importance) Rapsodieでは完全に除熱できるように設計してあるRapsodieにはpony motorはなく異常時の冷却はバッテリーによりmain pumpを運転する。バッテリにより運転するポンプの速度は非常に遅い。自然循環による冷却だけで充分と考えているのでバッテリによる方はadditionalと考えている。

main pumpが両方同時に故障することは考えていない。

バッテリによるpumpの容量は約1/10位だと思うが、CEA-R3406報告書のP·115，FigV-13を御参照下さい。

(xxvii) Q. Is the heat removal possible by the natural convection in Rapsodie? If it is possible, please show us the difference (++) of the heights between the center of coolant gravity in the core and that in the IHX, the heat removal values by natural convection, flow, the temperature conditions, the pressure drop (+++) of coolant, etc ..

(++) Is it 0.65 m in the case of Rapsodie?

(+++) It is about 5 kg/cm2 for the core part and about 6 kg/cm2 at nominal flow condition if the IHX is included on JFER.

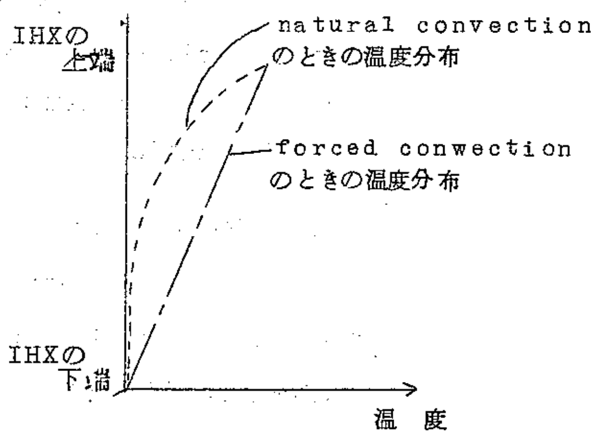
A. The removal of the core decay heat is possible by natural convection in Rapsodie. The difference of height between the core center and the IHX center is 1.175 m. Other informations are given in the CEA report n° 3406, p. 112-113-114.

Pontierによる"LE DE MARRADE DE RAPSODIE"を参照せよ。

①炉心とIHXの幾何学的中心高さの差(Rapsodie)は0.65mではなく、1,175 mm である。自然循環が行なわれているときの熱的な coolant gravity の炉心およびIHXの高さの差については一口には云えない。

②自然循環によつて除去される熱量については(温度条件についても同様)CEA−R3406、P112のFigV−11およびP113のFigV−12を参照されたし。

③圧力損失はnominal operationでは1.2kg/cm² であるが、natural convectionのときにはflowにdependするので同様に上記の文献を参照されたし。



④natural convectionのときのIHXの温度分布は右図のように強制循環時とは異なる分布になるのでもう少しhead 差が小さくともnatural convectionによる熱除去は可能である。

(xxviii)Q. How do your consider the counterplan in the case of the rupture of tube in Rapsodie? Especially would not you have the troubles that the cooling through the core would not be impossible when the tube between the outlet of the primary pump and the inlet of the vessel would rupture? N.B.: We have check valve just near the safety vessel in JEFR. The core flow would be assured by the other one loop in the case of the rupture between the pump and the check valve.

A. The double wall pipes are used only for the primary cooling system, and for the part of secondary cooling system located inside of the reactor building. The double tube has been designed essentially in order to detect the sodium leakage. The external tube is divided into a series of separate compartments in order to limit the leakage. In case of rupture of the primary system of Rapsodie inside the safety vessel, additional sodium will be injected in the reactor vessel to maintain the sodium level over the core and allow circulation.

2つの可能性が考えられる。

1つはouter pipe が破断しない場合……N_2 による冷却が続けられる。

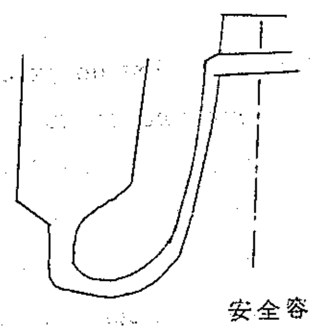
他の1つはouter pipe 破断した場合……reservoir tank からナトリウムが流れ込み炉心上部より上のNaレベルは保てる。

(注 空間部はJEFRではGraphiteがつまつているがRapsodieではN_2が入つているだけで何もつめていない)

Check valveの位置はRapsodieではpumpの直ぐ出たところというよりポンプの1部と考えている。(integral type)

配管破断についてはフランスではリークは考えているが破断は考えていない。

Rapsodieではたとえ安全容器の外側で破断があつてもサイフオンブレーカーが働きNaの流出は防げる。



(アメリカの安全基準を知つているかとの質問に対し)知らない。

light water reactor の場合とは管内の圧力が大いに異なる。

(フランスでは最大事故として何を考えているかとの質問に対し)

配管破断事故：

①事故が起つたとき、Phenixの場合。Primary tankは3kg／cm²、Primary Containerは6kg／cm²と仮想しているが、これらの値はまだfixしていない。

②Phenix designの場合、最大事故について、中央の1集合体のみがmeltすると仮定するのか、7本がmeltするとするのか、或いは全炉心がmeltするとするのか、現在議論している最中である。

(xxix) Q. Fuel rupture temperature

Would you explain the reasons how you have estimated 800°C for the clad rupture point?

You have not considered, we think, the hot channel factor in the transient calculation. Do you mean that the "800°C" is applied to norminal clad temperature of the hottest channel?

A.

(xxx) Q. Electromagnetic Flowmeters of the primary loop keep their function when the external power is lost?

A.

(xxxi) Q. Would you tell us what kind of accidents on control rod failures are analysed?

And also we would like to know your plan of development on the back-up system for B_4C rod control system.

A.

(xxxii) Q. In your consultations with PNC Representatives on the reactor safety on December 2nd at Cadarache, you stated that the ejected sodium the ruptured reactor vessel could be contained within the safety vessel should the core melt-down accident occurr in Rapsodie.

May we have your answers on the following questions? ;

① By what method was the energy absorption mechanism calculated for the explosion energy of the Rapsodie core especially the mechanism which absorbed the energy towards the bottom of the reactor vessel?

② By what means has the integrity of the safety vessel been proven for the explosion energy of the Rapsodie core? Has it been done by an analytical method or an experimental method?

Please show us the method, especially the method which was used to prove the integrity of the bottom of the safety vessel and the concrete structure.

③ We suppose that the reactor vessel is supported by the radial keys or similar structure to the horizontal motion caused by the seismic force. If such a mechanism is provided will not the safety vessel be impaired by such a rigid structure when the core melt-down accident occurs?

④ Did your design and install decay heat removal system to be used after the accident?

By what means will the heat be removed? Please let us know the capacity, coolant and the location where the system is installed in the block-pile.

(xxxiii)Q. Would you please explain how to estimate the effect of pin rupture relating to design of fuel?

(xxxiv) Q. How do you estimate the disturbance on Cooling System by the introduction of large gas bubbles into core?

5. Core meltdown and containment

1) Comments

Descrintion に記載されている「AX－1」の code は10年前の code で非常に複雑な code である。Cadarache では R－Z 2次元 code を開発し非常に簡素化して取扱つている。この code によると 2 Zones の解析が可能である。

Decription の Fig. 82 の 2,000 joul／g は Cadarache で使用している値の約2倍で少し大き過ぎないか。

Doppler 効果は少し小さ過ぎるように思う。解析手法は基本的には良いと思うが effective energy は大き過ぎるように思う。

Storrer の melt down accident に対する個人的 general comment ………完全な core melt down というものは起り得ないであろう。何故なら負のドプラー係数や負の反応度係数をもつているしまた loss of coolant でも scrum できるし独立の scrum 系統を別々にもつているのでまず scrum する。

この melt down accident については実験をやつてみる必要があるが金がかかり過ぎる。また計算による方法は誰も知らない。TNT explosion による方法は vapor の escpansion ではなく hot gas の expansion であるので実験をやつてみる必要がある。燃料の模擬についてはフランスでは oxide ではなく金属で模擬している。

(i) Maximum accident and containment philosophy maximum accident accident and containment philosophy

① - Maximum hypothetical accident philosophy

We do not apply in France the terms "maximum credible accident" and "maximum hypothetic accident" because we cannot say what is credible or not. We think that the aim in assessing safety is ultimately to apply a probability method. Before beeing able to use quantitative date for such an approach, we try to study each king of accident with a "reasonable amount of pessimism" taking account of the whole characteristics of the reactor. Our safety criteria are not officially listed excepting for the operational radioactive releases.

There are some difficulties to apply a "reasonable pessimistic approach" for fuel testing reactors and specially for fast testing reactors, which present a risk for compaction accident. We cannot say how a compaction accident could happen and what should be the compaction rates to be chosen. Some people think we could rely on the many counter measures we take against that type of accident. However, we designed Rapsodie with a very severe compaction hypothesis, with a primary containment able to hold the explosion, and with a leak tight pressure resisting building. That approach may seem very similar to a "maximal hypothetical accident" approach, but we want to speak only of "reference accident" or "typical accident", it is in our mind more exact and allows for later refinements in assumptions.

② Evaluation of reference accident for JEFR.

For Rapsodie, we chosen a Bethe and Tait hypothesis, assuming in a one dimensional calculation, that some unknown phenomenon is able to suspend the upper half part of the core during central meltdown, and to drop it at the right time to get the maximum reactivity ramp at prompt criticality. We obtained 230 \$/second.

PNC proposes a meltdown accident with progressive radial meltdown and without suspension of above half part of the core. Such hypothesis are much more realistic; the pessimism is limited to the causes of the accident: voiding of sodium without scram. The reactivity ramp is only 29 \$/second.

We think that such assumptions are good and may be taken for a reference accident, to be held by containment means.

Concerning the energy released, we compared your results with our parametric work and found correct agreement. It may be noted that for 30 $/second the maximum fuel temperature is about the same for a Doppler constant value of 0,175 % (JEFR) or 0.5 % (Phenix) and is much higher for 0 % (Rapsodie). The result is that the mechanical energy release by UO_2 expansion is lower for JEFR (100 MW) than for Rapsodie (24 MW) or Phenix (600 MW). We think that UO_2 expansion may give for JEFR a calculated mechanical energy release in the range of 50 MJ and that residual pressure would be about 20 bars for the 6.3 m3 free volume for expansion.

Cooling of UO_2 by sodium may produce sodium vapor expansion. Our general results in that dield are lower than Hicks and Menziès one's, because we chosen a lower specific heat for UO_2. We agree that such cooling could give an energy release in the range of 100 MJ, with pessimistic assumptions in mixing ratios, we think however that final sodium vapor pressure would be in the range of 30-50 bars instead of 20 bars (PNC data). We think that relaying of UO_2 pressure by sodium pressure may be progressive, during UO_2 expansion, and that the valley between two pressure peaks, presented by PNC, is doubtful.

In the whole explosion we may distinguish three phases instead of two like PNC (shock wave and blast pressure)

- the initial phase, with very quick decrease of bubble pressure and emission of shock wave,
- the intermediate phase, with moderate pressures and hydrodynamic expansion (the bubble may oscillate in pressure exchanges with the gas cover),
- the pseudo-equilibrium and pressure decay by cooling.

The reflection of shock waves and the hydrodynamic oscillations may give overshoots in the gas cover pressure.

In conclusion, our parametric studies allow us to accept, for the given meltdown accident conditions, a total mechanical energy release in the range of 150 MJ, but we stress the fact that final quasi-static pressure could be in the range of 50 bars instead of 20 bars.

We recommend performing the accident calculations with the most probable physical constants (Doppler effect, for example) examining however by a parametric study the effects of incertainties for the final result.

③ - Simulation by TNT

The effects of the initial phase of explosion are related to the maximum bubble pressure in the very vicinity of the core; in the whole of the vessel, the effects are essentially dependent on the mechanical released energy. In the same way, for the hydrodynamic phase, the pressure peaks in oscillations and the general kinetic effects are related to energy. The quasi-static pressure, before vapor condensing, depends on the released energy and the adiabatic behaviour furing expansion. At that stage there are main differences in the effects of different working fluids.

A 50 kg TNT charge, into water, releasing 200 MJ, seems suitable to simulate the dynamic effects of the reference accident; however, it would give in 6.3 m3 a quasi static pressure of about 12 bars at 250°C and 7 bars after cooling in water. It is much less than the 30-50 bars range we evaluated for sodium vapor. We think that the experiments may be performed with TNT simulation but that leaktighness after accident is to be studied with higher static pressure.

④ - Primary containment

We think, after our Rapsodie scaled tests, that the primary containment is strong enough around the vessel, but that plug anchorages are not strong enough to avoid jumping for reference accident. We recommend and we think possible that plugs would be fixed in such a way that no significant leak could escape during such an accident. Before scaled tests which are needed as confirmation, we may recommand improvements like the following ones:

- fixing a strong flange above the periphery of the large rotating plug, with screw jacks to hold down the plug during reactor operation;
- fixing in the same manner the small rotating plug to the big one,
- suppressing the movability system for the upper core plate,
- mounting all devices lke control rod mechanisms in such a way that it would be evicent at first look that they are locked and cannot jump. For instance, the mechanisms could be held by a baionet system before beeing bolted,
- increasing argon height,
- providing some space for vessel radial expansion, on the upper part above the sodium pipes (we think however that expansion would be small),
- mounting inside the vessel and specially on plug's bottom some absorbing energy devices.

⑤ - Containment building

It is very important to take account of siting. We admit that a primary containment may be provided against missiles or significant primary sodium projections (we admit that because the energy release is comparatively much lower than for Rapsodie, for which we have good mock-up results).

In a relatively remote area (for instance at about 50 km of a large town) we would recommend a containment building resisting to moderate pressure (for instance 0.1 bar), provided with an air filtering and exhaust system able to progressively release the noble gas after a core large accident.

In a less remote area, we would ask for some means able to contain gas releases: leak proof pressure resisting building or pressure suppression sustem (gasometer, large volume exhaust channel...)

Concerning the concrete shielding around the steel building with concrete, we found for Rapsodie that irradiation injuries to workers around the building, in the case of large accident, were small provided that correct alarm could be given; we did not install such a shielding.

In conclusion, we think that the containment building proposed for JEFR (steel pressure resisting building, shielded by a concrete external wall) is a good one for a not remote site.

⑥ - General conclusion

We may approve generally the containment system proposed for JEFR, however, we ask for a stronger plug anchorage and would admit a less resistant containment building in a relatively remote site.

(ii) Main differences between rapsodie reactor and its mock-up

- In the reactor, the cooling gas circuit of the rotating plugs is simplified: there is only one cooling gas layer in the upper part of the plug, when there were two cooling layers in the mock-up.

This modification results from a decrease of the thermal output which must be removed from the plug due to a thermal sources calculation improvement.

- Two thermal shields plates are used in the sodium inlet of the reactor, although there were 4 thermal shields plates in the mock-up. : the thermal shocks envisaged on the inlet of the reactor were smaller than these which have been imposed to the mock up due to improvement of dynamic calculations.

- The insulations located between the reactor vessel and the safety vessel, on the mock up, have been dropped on the reactor. But, special sodium tanks were added to assure core cooling, in case of both reactor vessel and double wall failure.

- The lower part of the reactor vessel is simplified in the reactor, because melt down pan was dropped when use of oxyde fuel was decided for reactor.

- The actual reactor vessel is hung up, but it was supported by temperature compensated columns, for the reactor mock up: earthquake problems have justified this change.
- The instrumentation (thermocouples, straingages) is simplified in the reactor. On the reactor vessel, there are no straingages. There are no thermocouples in sodium, except the core thermocouples located in sockets.

(iii) Study of the blast resisting structure of JEFR

Compared to the Rapsodie vessel, the JEFR vessel is similar within a factor of 1,7 :

	Rapsodie	JEFR	Ratio
Diameter	2350 mm	4000 mm	1,7
Thickness	15	25	1,66
Jacket's thickness	8	15	1,88
Distance between vessel and jacket	28	60	2,6

The JEFR vessel is relatively longer than Rapsodie's one, by its extension below the core. Concerning the explosion resistance we may point out:

① JEFR vessel is surrounded by graphite blocks, with a small gap. In the case of Rapsodie, the vessel is in a large free volume.

② The JEFR argon cover thickness is only 500 mm instead of 800 mm for Rapsodie.

③ A anti-explosion plate is fitted, above the rotating plugs of Rapsodie: with tie rods able to practically absorb the kinetic energy of plugs: the hooks provided for JEFR seem less deformable.

④ The explosive mechanical energy release taken into account for JEFR is in the range of 100-160 MJ. For Rapsodie, the adopted figure was (for design and testing) 164 MJ. The mechanical effects of Rapsodie energy release could be compared with a JEFR accident in the range of

$164 \times (1,7)^3 = 800$ MJ. In fact, the JEFR calculated energy release is not so strong.

⑤ The 3/10 Rapsodie mock-up represents approximately JEFR at scale 3/17 : a 160 MJ JEFR explosion would correspond to 0.88 MJ in the mock-up. We performed tests with 1.045 MJ TNT explosion. The vessel is broken if the wrapper tubes of the subassemblies are empty or if the cover gas is not present; the plug is jumping by 11 cm, what corresponds to 18 100 Ps (pascal-seconds) as surface impulse. Taking into account that JEFR design would probably correspond to increased impulse, we may suppose 20 000 Ps for a simulated 160 MJ accident in a 3/17 JEFR mock-up and 20 000 x 17/3 = 113 000 Ps in actual vessel. The plug's section beeing 12.6 m2, the whole TNT impulse would be I = 1 420 000 Ns (Newton - seconds). The plugs mass being M= 170 t, the kinetic energy would be $I^2/2$ M = 5,9 MJ. The free jump would be 3.5 m.

The energy may be absorbed by deformable anchorage devices: with a steel elongation 20% under 50 bar stress, the absorbed energy is 100 MJ/m3; the JEFR accident would need 58.4 dm3 in form of tie rods (or other devices to resist against jumping).

⑥ - An other evaluation of JEFR explosion is 106 MJ of which 8 MJ would be furnished at high pressure by UO^2 expansion and 98 MJ at low pressure (peak 21 bars) by sodium vapor expansion. We give the adiabats of a mixing UO^2-sodium after Hicks and Menzies, ANL 7120, p. 666. The free volume in the JEFR vessel is 6.3 m3 and the oxide mass to be taken into account seem 545 kg; it gives 0,0416 m3/kg: the pressure is 46 bars; the energy release is 142 kJ/kg or 75 MJ (by later CEA results we found for a 98 MJ sodium vapor expansion a somewhat lower pressure, in the range 40 bars).

For 46 bars static pressure the JEFR vessel seems to resist even without being applied against its jaket, but the plugs anchorages seem too weak.

Conclusion: In light of the Rapsodie mock-up test results, the calculated JEFR explosive accident could likely be contained in the primary containment, but some improvements seem necessary in plug fixations.

(iv) General views on explosions in liquids

Relating to description of explosion effects, we do not distingish only a "stock wave" and a quasi-static "blast pressure". We may describe three phases:

a) Let us suppose a very high pressure initial gas bubble. A certain amount of energy is radiated to surrounding liquid in a very short time, in form of pressure and speed, in a spherical limited zone. This zone expands in form of a shock wave, carrying concentrated energy in a relatively thin layer; This radiation is governed by compressibility properties of the medium, either with large perturbations in the vicinity of the charge, or only according to acoustics at longer distance.

The amount of energy radiated by such compressibility effects is related to the ratio between compressibility coefficient in the surrounding medium, versus peak pressure in the bubble, The relationship seems not known.

In infinite medium, this amount of energy is carried until infinite distance with damping and without reflection to the charge. In actual structures, the wave may be refracted or reflected.

b) Another part of energy is transmitted to liquid with small compressibility effects, according to hydrodynamics. In spherical geometry and infinite medium, the flow through any sphere is the same and depends only on time. There are exchanges between potential pneumatic energy in the bubble (relatively to equilibrium pressure) and kinetic energy in the liquid, the bubble diameter oscillates with damping. In containment structure with gas cover, there are energy exchanges between bubble and gas cover, with large pressure peaks.

It seems that in contained geometries the shock wave effects are hard to be separated, theoretically or practically from hydrodynamics effects. From one case to another, it is easier to interpret the whole effects by shock wave reflection or by pneunatic-hydrodynamic oscillations.

c) The dynamic effects are progressively damped and pressures become static; however, due to cooling of hot gases or vapors, the "pseudo-static" pressure may progressively decrease.

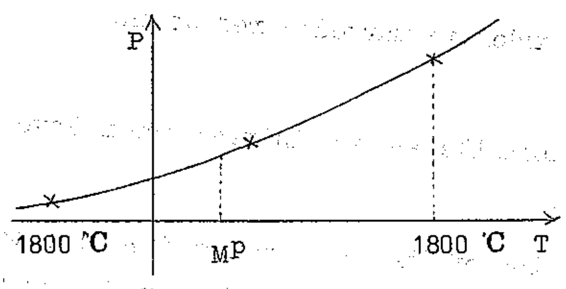
Structure response analysis

The above general description of explosion energy dissipation leads to not rely on a vessel ability to resist against a statically computed "blast pressure". The dynamic effects of an explosion must be studied on typical mock-ups in order to find crude evaluations of impulse, maximum. pressure and residual pressure; mechanical correlations must be established between the above parameter and the effects on typical structures; in case of doubts for a given complex structure, direct mock-up tests must be performed.

2) Questions and answers

(i) Q. What do you use for the equation of state? How do you consider the reliability of it?

A. the values used are the following:



$T_c = 7800\ ^{o}K$
$P_c = 2100$ bars
$Z_c = 0,32$

this set of values is well in the range found by most evaluators.

(ii) Q. Did you carry out any tests to obtain the equation of state experimentally?

To explain us the outlines if you did or have any plans.

A. Experimental studies: we are trying to measure the UO_2 vapor pressure near the melting point and if possible at higher temperatures, by the boiling point method. Heating is of induction type. Temperature is measured by means of pyrometers.

(iii) Q. Do you suppose the instantaneous melt-down? Or how do you take into account the method which introduces the propagation of the melt down to calculation?

A. No, we do not. Due to the actual temperature distribution inside the core, the fuel melting of the whole core cannot be simultaneous.

フランスで考えたmelt dounを起したときの反応度装入速度は 230$/see である。これは燃料や材料のgravityに基ずいて計算した。

(iv) Q. What value do you use on the Doppler effect in the analysis? How do you suppose of its reliability? To show us the example on the Rapsodie.

A. For Rapsodie, we did not take into account the Doppler effect (too small). For larger reactors, we make parametric studies with varying Doppler coefficients. It turns out that it suffices that the Doppler effect be larger than a given "threshold" value, in order to markedly reduce the magnitude of the explosion.

(v) Q. What value is reasonable as the initial power level on this calculation?

A. A precise definition of the accident giving rise to a generalized core melting is impossible. So, we do not know what initial power to take into account. Our calculations are made on a parametric basis.

(vi) Q. What degree of safety factor is necessary on the design of the blast resultant structure?

A. a) For Rapsodie, we did not look at special safety factors for the design of the blast resistant structure, as related with maximum incredible accident. But, tests "after design" on mock-ups at reduced scale (3/10 and 1/10) showed to us, that the vessel could probably be ruptured but that this accident should be contained inside the primary building. It is important to get a homogeneous resistance for the primary system and to avoid any weak part.

b) It is suitable to accumulate safety factors. We think that the necessary pessimism must be applied at the level of accident assessment. The accident resisting structures may work with large deformations, to the extent where local effects (weldings, penetration) do not provide appreciable risks of premature failure. On ductile metal shells without weak areas, one could admit elongations in the range of half rupture elongation.

(vii) Q. How do you think of propriety that assuming the accident of the recritical burst, it is presumed that the pressure in the reactor vessel is the statical pressure, 20 kg/cm^2 which is one of the design standards of plugs?

A. a) Plugs are designed and fixed in order to hold the blast effect. After the accident, the plug holdings are loosened and give rise to gaz release inside the primary building, and so the pressure drops inside the safety vessel.

So even if the plugs could contain a 20 bars pressure, there is no reason that they should contain this pressure after the accident. This pressure cannot be maintained in a statical way after accident.

b) The 20 kg/cm2 figure is the JEFR computation for sodium vapor peak pressure. By crude evaluations we found higher quasi static pressure, in the 40 bars (or kg/cm2) range. However, the peak value in gas cover would be higher, due to dynamic oscillations between core bubble and gas cover.

We think that pressure transient due to UO_2 expansion and sodium vapor expansion may be more intricated than showed by JEFR calculation, because of non spherical expansion and premature mixing of hot devices with surrounding sodium.

We would recommend the following design criteria:

(a) The reactor vessel or the primary containment vessel must withstand the total energy release in dynamic form, computed for final volume.

(b) After damping, the relevant vessel must hold the residual pressure, with limited leaks compatible with whole containment system.

(viii) Q. On the TNT simulation for the meltdown accident (see appendix).

It is an usual way to simulate the energy release due to a nuclear excursion by TNT explosion.

However, there exist some doubts in respect to the selection between pressure loads and structural response.

The most important defference is that the TNT explosion yields two kinds of pressure load; the initial shock wave and the blast pressure due to the gas babble expansion, while the latter may not result in case of the nuclear accident, because the vaporized fuel materials would easily be condensed by cold sodium arround the core, and will not leave any gas babble.

Another difference sometimes pointed out in hazards analysis of fast reactors is that the estimated pressure profiles in case of reactor excursion accident will not be like a shock pressure pulse but rather similar to a blast pressure.

On the other hands we know from references and out test results that the predominant load for estimating the missile jump and the resulting sodium ejection is this blast pressure - quasi-static pressure, not a shock impulse, while it is reverse for radial energy absorbing structures.

In these respects, our questions are as follows;

① What is your treatment of the blast pressure in your analysis for Rapsodie?

② As E.P. Hicks pointed out (ANL-7120) it could be assumed that the secondary blast pressure would result by the expansion process of quick vapor babble formation of sodium at the close vicinity of the vaporised core, when the fuel vapor comes contact with cold sodium after the excursion. If this be a probable assumption, the TNT simulation rationale seems to be consistent with the hypothised accident pressure profiles. What is your opinion?

A.

① For Rapsodie, the simulation of a nuclear explosion by TNT is probably not too bad; energy release and maximum pressure that could be got, seem to be of the same order that one is able to get with chemical explosives.

Correlation can be done, from the experiments made; this is what we did for Rapsodie.

② Generals: see the text "General views on explosion in liquids"

(a-1) What is your treatment of the blast pressure in your analysis for Rapsodie?

At design stage, the Rapsodie reference thermal energy release was 656 MJ in 80 microseconds, and the assumed mechanical efficiency was 25%. The assumed mechanical release time was the same, 80 micro-seconds; it corresponds to high explosive. The main tests were performed on 3/10 scaled mock ups, with water. The reference mechanical energy release (164 MJ) was scaled to $164 \times (0,3)^3 = 4,4$ MJ. High explosive "hexogen" charges were used, of following total energies: 1.045 MJ, 4.18 MJ, 9.4 MJ, 16.72 MJ. Except for the first values, these charges break the reactor vessel. The vessel's plug was strongly loaded with lead; the jump height measurement gave the value of the impulse.

Following these tests, it was considered that plugs and mechanisms anchorages were not sufficient to avoid rupture and jump of some pieces; the reactor was provided with an anti-explosion plate fixed to the upper slab of the plugs by long ductile tie rods. The amount of ejected sodium was evaluated according to measured water ejections.

The impulses received by the plug were largely increased in comparison with the impulses which would be received in infinite medium, at the same distance of the charge. The multiplicating factor is about 4 for the weakest explosion, with cover gas, and 4.6 without cover gas. That factor decreases when explosion energy increases; this fact is certainly related to rupture and deformation of reactor vessel.

Some test were performed on 1/10 scaled crude Rapsodie mock-ups, designed for sodium filling around the hexogen charges. Due to the chosen geometry, that test did not permit results for Rapsodie. The result was that the multiplication factor for impulse on plug, versus impulse in infinite water, is slightly above one, but was not measured exactly. That tests allowed to study a correlation between explosion effects in water and sodium, for the types of deformation in concern. The more "rigid" sodium gives increased effects, corresponding to water tests with charges increased by a factor 1.4.

Using chemical explosive, it is suitable to assess the effects due to residual gases. For the 4.18 MJ explosion, using a 784 g hexogen charge, the free volume being 0,08 m3 (with anchored and leaktight plug), the residual pressure would be in the 8-10 bars range. For a Hickes and Menzies explosion of 5 kg oxyde (scaled Rapsodie core) one would obtain 32 bars and 0,77 MJ due to sodium vapor.

The reference accident was later changed. Due to some doubts about actuality of large vapor production by sodium-duel thermal transfer, such production is not to day taken into account; major effort should be devoted to analyzing basic phenomena. However, in Rapsodie design, we do not rely on leaktightness of plugs after accident.

(a-2) Reference to Hickes and Menzies paper, ANL 7120.

We saw that gas release by sodium vapor expansion may be higher, with pessimistic assumptions, that gas release by equivalent chemical explosive. Our theoretical results for sodium vapor expansion gives results some what lower than former ones, but maximal gas release remains higher than for chemicals. Due to our doubts on practical possibility of vapor large production, we make the provisory assumption that TNT simulation is correct for meltdown accident. We have a test program about transient heat transfers in such explosions.

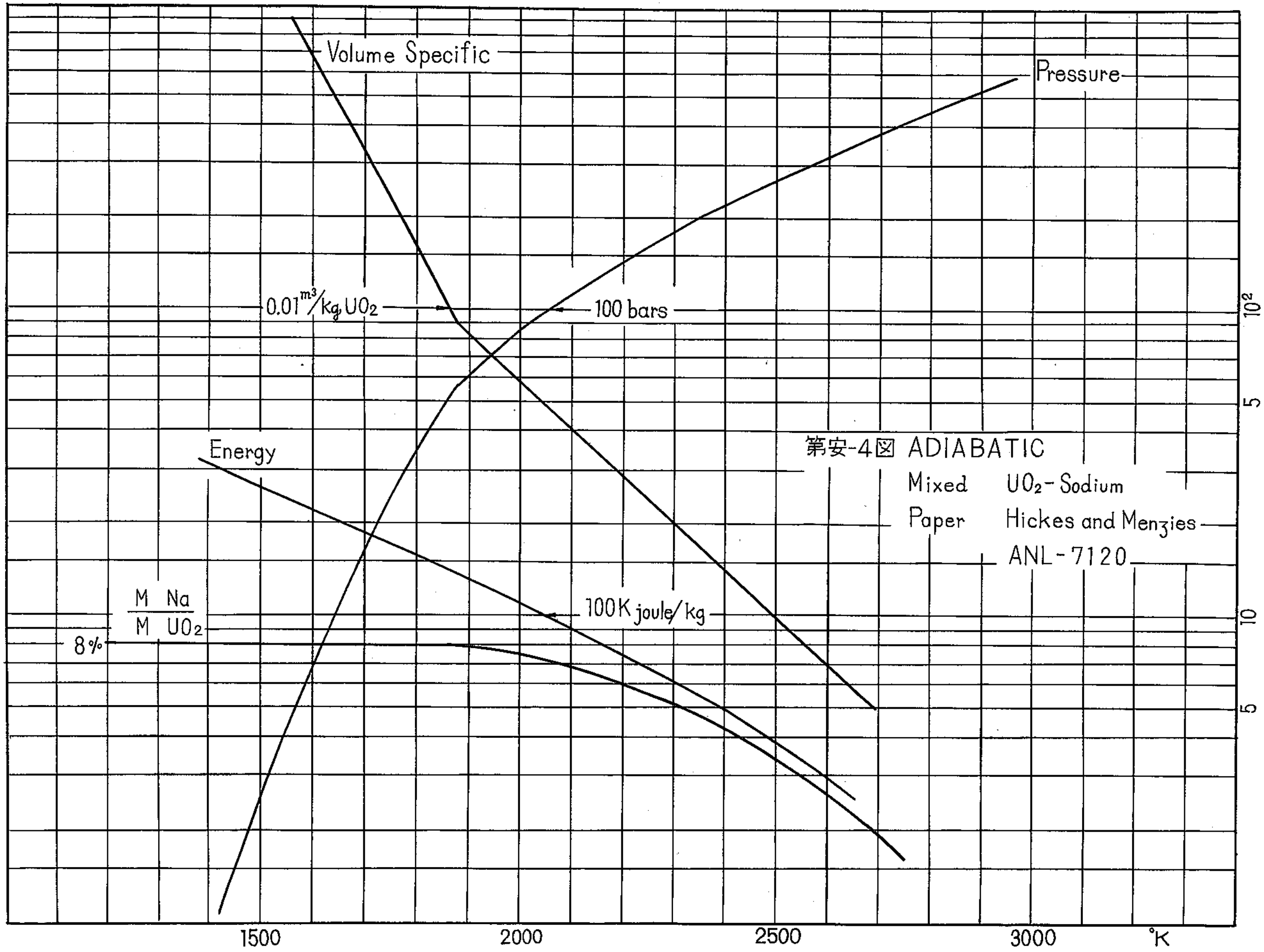
(b) On the similarity test result with sodium and water.

See above (a-1) answer on 1/10 mock-ups. The difference between water and sodium results may be due to the fact that, for a given radiated energy, pressure and impulse are higher in sodium; the directional effects are then stronger: the explosive work seems higher (more breaking). For large deformation of ductile structures, with damping, both liquides must be equivalent. The over factor for sodium depends on the type of containment structures. For structures similar to Rapsodie ones, we recommend.

(c) On the reactor vessel design considerations.

It is not only by weakening the vessel that impulse on plugs could be reduced or avoided. One may increase the cover gas volume, or/and provide absorbing structures at the plugs' bottom or around the core.

We think that an effort could be applied to plugs fixations. According to our Rapsodie tests, we think that plug could be anchored in order to resist without jumping and without large leak. We recommend a permanent anchorage, without removing it for rotating the plugs or carrying the fuel machine, like in Rapsodie. A first proviso would be to calculate the anchorage in order to resist at the inflating pressure of reactor vessel. It would be necessary however to verify the design by some mock-up tests moderately scaled (1/10 for instance) and sufficiently representative about core surrounding and bottom structure of plugs.



第安-4図 ADIABATIC Mixed UO_2-Sodium Paper Hickes and Menzies ANL-7120

(ix) Q. On the similarity test result with sodium and water.

It appeared in the paper "comparison of pressure loading produced by contained explosion in water and sodium, by G.A.V. Drevon et al." (ANL-7120) that the pressure loading in sodium is greater than in water. How should this result be applied for the analysis? For instance, should we take the twice amount of TNT energy release? While this hasn't been taken account into the analysis of the Enrico Fermi and SEFOR.

A. See next section 6.

(x) Q. On the reactor vessel design considerations.

As in case of Phenix primary containment concept, it may be possible to design the reactor vessel relatively weak in order to reduce the pressure loading for the top plug. What should the vessel design criteria be?

A. See MM. COSTES et FALGAYRETTES

(xi) Q. In JFER, the guillotine rupture of pipings is thought as the design basis accident, and in such a case the leak jacket is thought to be ruptured at the same time.

May we have your comments on the possibility and the counterplan of such accidents and also the possibility of the succeeding core melt-down, comparing with the design philosophy of Rapsodie?

A. The rupture of a primary pipe would only lower the sodium level under the outlets of the sodium holes inside the vessel (question 4, 2) (xv) and (xvi). An emergency cooling loop (with nitrogen inside the double wall of the vessel would allow to extract the residual power, but we are not sure that the temperatures inside the core sould not reach very high levels.

(xii) Q. Please show us the design criteria for pressure and temperature and cooling method in the container in the case of the hypothetical accident in Rapsodie.

N.B.: In JFER, assuming 670 kg of Na from the reactor as jet out, the pressure of container become to 2.5 kg/cm2 and the temperature become to 150°C. We are now considering that the container is cooled by the air of the annular part. There are some reactor being cooled by water, par exemple on DFR. How about on Rapsodie?

A. It has been assumed that in case of a sodium fire (pool fire) inside of the containment building, all the oxygen present in the inner atmosphere is reacting with sodium. The overpressure will become 2.4 kg/cm2 and the maximum temperature of the shell 160°C. There is no cooling of the shell by water.

Rapsodie の contaiment buildingは一重でJEFRのような annulus部の冷却はない。

仮想事故時における格納容器内の条件：Rapsodieの場合は仮想事故時はReactov Vesselが破断するとしているので、Primary tank内の圧力は大きくはならない。このときの Container 自身の最大温度はJFERの場合と同様150℃（このときのContainer内のガス温度は数百度に達している）に達するが、Containerの冷却設備は保有していない。

(xiii) Q. We would like to know your counterplans to the hypothetical accidents of the IHX and the secondary loop on Rapsodie.

N.B.: On the conceptual design of the JFER, we had not considered that the tubes of the IHX would be broken and the radioactive nuclides would go into the secondary loop in the case of the hypothetical accidents. But, we are now discussing about this accidents by considering the several plans as follows:

① The design has to be done so that the tubes of the IHX are not broken by the blast pressure after the hypothetical accident,

② The tubes of the IHX have to be designed by taking the blast pressure to the design one,

③ Considering the rupture of the tubes of the IHX, one isolation valve is prepared on the one side of the penetration of the container for the secondary loop,

④ Considering the rupture of the tubes of the IHX, two isolation valves are prepared on the both sides of the penetration of the container for the secondary loop,

⑤ We would approve that the primary sodium would go into the secondary loop after the rupture of the tubes of the IHX by the blast pressure, in this case, we should design so that the secondary loop should not be broken and examine the leakage from the secondary loop regularly as often as the container.

(The acceptable pressure of the JFER is 2.5 kg/cm2 considering the sodium fire, but the design pressure is 5 kg/cm2 for the secondary loop).

A. IHX および二次系の事故時の対策：C.E.A側より次の質問があつた。

「仮想事故の場合にJFERでは何故Vesselが破壊するとしないのか不思議に思う。この理由は何か？もしVesselが破壊するような設計基準でやるならブラスト圧（20kg／cm²g）は大きくなく、15番の諸対策も不必要となろう。」

実際RapsodieではVessel厚をうすくして、仮想事故時にはVesselがまず破壊して安全容器の方で保持することになつている。又中間熱交換器では2次側の方が1次側より通常運転中は圧力を少し高くしているので仮りにleakがあつたとしても2次側へ1次系が流れ込むようなことはない。仮想事故が生じた場合の圧力上昇は0.1sec位で、回転プラグのhop upで殆んどの圧力は減少し、中間熱交の方に及ぼすことはない。これはTNTによる実験で確めた。JFERの場合にはReactor VesselとIHXとの間の距離がRapsodieの場合よりも大きいので、Vesselが破壊するとすれば心配ないと考える。

(xiv) Q. Would you explain your view on the decay heat removal system after the gross core melt-down accident, and also your preventive measures on the melt-through accident?

A.

(xv) Q. How do you estimate the blast pressure of vessel bottom at the recritical accident and the choice of design precaution on vessel support. Would you tell us the calculation method of energy absorption by reactor vessel at the recritical accident.

A.

(xvi) Q. Meltdown accident analysis

① Equation of state

(a) We have adopted the fuel melting point (melten condition) as the zero point of energy and 1,900 jouls/g as the Threshold Energy. Would you explain what did you use as the zero point of energy in the equation of state?

In the meltdown accident analysis, we have adopted 1.65 as reduced density , because we have assumed that core will start to melt from the condition of no sodium in the core. Would you explain the value of the reduced density in your equation of state?

(b) We have calculated the UO_2 critical constants from the experimental data (ANL-5482) by Ackermann, that is, we have adjusted the critical constants such that the saturated vapor pressure which is calculated by the principle of corresponding state agrees with the experimental data. Would you explain how to calculate the critical constants?

(c) Would you tell us the value of Q^*, γ and (initial) in your equation of state?

(d) If you have made an experiment on equation of state, we would like to know about it.

(2) Reactivity insersion rate

Do you assume that sodium will exist in the core when prompt criticality occurred?

If the prompt criticality occurred by β only, we suppose that the reactivity insertion rate of Rapsodie is a little larger.

How do you think of it?

(3) Result of Rapsodies's analysis

Would you tell us the followings?

a) Reactivity insertion rate
b) Total released energy
c) Effective destructive energy
d) Calculation method of effective destructive energy
e) Calculation method covering effective destructive energy to the equivalent amount of TNT.

A. (1) (a) We do not perform the calculations in terms of energy but in terms of temperature.

This is possible because the specific heat is assumed constant over the range of temperatures. The value taken is 0.3 joules/g of UO_2. Thus there is no need to take into account an arbitrary zero point energy. The answer concerning the threshold temperature is given under (b).

(b) The critical constants are given elsewhere. They are repeated here and we try to explain how they are estimated. It is important to say that these following values are badly known.

They are:

$$P_c = 2100 \text{ bars}$$
$$T_c = 7800 \ ^{o}K$$
$$Z_c = 0.32$$

We used for the evaluations, the law of corresponding states, and the reduced equation of state of Roccard (Thermodynamique, Masson et Compagnie). It is found that the best value for the critical compressibility Z_c would be 0.32.

We tried to compute the critical temperature in two ways:

(1) with the law of rectilinear diameter. Knowing the Christensen data density of liquid at the melting point and the coefficient of linear expansion of liquid at the melting point, knowing also that the critical density is 3.1 times lower than the liquid density at the melting point, we would find a critical temperature of 6400 ^{o}C.

(2) by extrapolation of the vapor pressure versus temperature curve: with the law of Maxwell and the equation of Roccard we found the vapor pressure:

$$\log_e \frac{P}{P_c} = A \left(i - \frac{T_c}{T} \right)$$

and we got an A value = 6.4.

The latent heat of vaporization at low pressure is then L = RA Tc which agrees with the Ackerman data if Tc has a value between 8000 to 10.000 ^{o}K depending on the A value.

$$(6 < A < 8).$$

These two crude estimates check rather well with the Roccard equation giving 7800 ^{o}K. Clearly it follows that the critical pressure is estimated at 2100 bars.

There is a need to compute now the threshold temperature. At the critical point:

$$\left(\frac{\partial P}{\partial T}\right)_r = \left(\frac{d P}{d T}\right)_c = A \frac{Pc}{Tc}$$

Extrapolation to zero pressure from the critical point gives the threshold temperature $\simeq$ 6700 ^{o}K. But the threshold temperature is density-dependent. Extrapolating the Bethe Equation (*) it is assumed that:

$$\left(\frac{\partial P}{\partial T}\right)_r = A \frac{Pc}{Tc} \frac{\rho}{\rho c}$$

(c) being the fuel average density in the core corresponding to a temperature T_1. This temperature T_1 can be estimated on the boundary curve which separates liquid from vapor in a (P, V) diagram. T_1 is assumed either 7000 ^{o}K or T_c depending on whether structural material has left the core or not. Extrapolation to zero pressure gives in both cases a threshold of 6700 ^{o}K, so that we use always the same threshold temperature. The energy between fusion and threshold is thus (6800-3100) x 0.3 = 1100 joules/g (smaller than the Japanese value of 1900 j/g).

This solves the questions (a), (b) and (c) of the meltdown accident.

(d) We are working on the evaluation of the equation of state. Results are not yet available. We try to perform steady state measurements at low temperature (near the melting point). We measure pressure and temperature at the boiling point.

Heating is either by induction at high frequency or by solar furnace.

(3) Reactivity insertion rate

- We assume also that sodium has left the core, since this is pessimistic
- We do not understand the 2nd part of the question.

(xvii) Q. How do you estimate the behavior of shock wave through the cooling pipe at the recritical accident?

That is, the effects by the bends, size (diameter) and length of pipes.

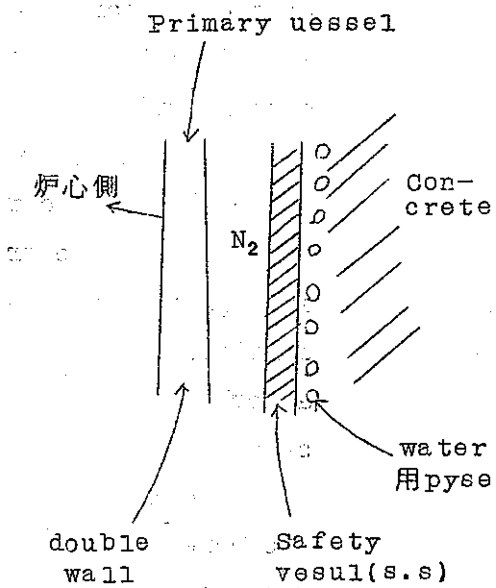
(*) see APDA 125: $\left(\frac{\partial R}{\partial T}\right)^2 \frac{\alpha T}{\rho C_r} = \alpha \rho c^2 - \left(\frac{\partial P}{\partial T}\right)_r$

A.

(xviii) Q.：再臨界事故後のdecay heatのremoveはどのようにするか。

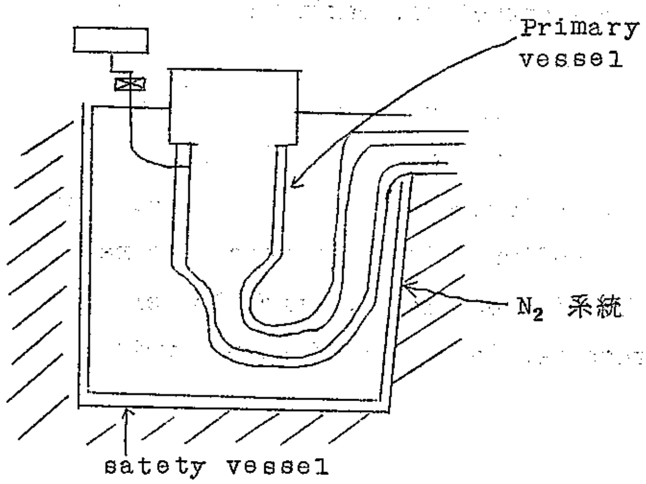
A.： Phenixの場合、右図の通りSafety vesselの外側にwaterのpipeが循環しており、これにより再臨界事故後のdecay heatがremoveできる。

但し、waterはaccidentのときにだけ流す。



少なくとも2つのwallsをNa容器と水のpipeとの間に置くというphilosophyをもっている。

Rapsedieの場合accidentの後、Safety vesselをNaで充満させることができる。



その周辺を左図の様なN_2 系が循環する。

reactor vessel周辺のN_2 系のcoolirg capacityは350KWであり、これでdecay heatはとれる。

同様に、satety vesselの方にもN_2 系のcooling systemが設置されている。

Appendix (5, 2), Q of viii)

1) Preliminary analysis of the primary blast resistant containment structure

(i) TNT rational and response of structures

It is assumed that 50 kg TNT is detonated at the center of the reactor core within the reactor vessel filled with liquid sodium. The impuls would be transmitted through the coolant medium to the reactor vessel wall, and the rapid vessel expansion would result. Since the reactor vessel is surrounded by the radial graphite shield, the impulsive load would further be transmitted to the massive graphite blocks and eject them outward with a velosity and kinetic energy consistent with their mass. This kinetic energy is to be absorbed by the energy absorbing potential of the blast shield.

The impuls would also arrive at the sodium level at above the core, and the resulting sodium spray hits the bottom of the bottom of the top shielding plug, thus giving the plug an upward velocity and kinetic energy consistent with its mass. The shielding plug is designed to be anchored to the reinforced concrete slab, and these anchor bolts absorb the energy with their plastic strain.

It is recognised that TNT explosive yields the explosion product gas babble within liquid, and with this gas babble expanding, a quasistatic pressure would result. This would be the second loading for the structure, particularly for the shield plug.

The order of magnitude is much less than the primary shock impulse due to the fact that the energy yielded at the explosion would be lost in destructing the structures and in heat loss to the coolant medium and/or the reactor core materials. However this blast pressure last much longer than the initial shock pressure; tens of miliseconds while the shock pressure lasts only less than hundred micro-seconds. And it is large enough to propel any possible missiles, the control rods or any other components which penetrate the shield plug are likely to become missiles if not anchored.

These missiles would be given appreciable velocity and kinetic energy that will be cancelled by the potential energy corresponding to the missile jump height.

(ii) Numerical calculations

① Radial energy absorption

Peak shock pressure at the vessel wall is given by Cole's empirical formula in free water for pentolite,

$$Pm = 2.25 \times 10^4 \left(\frac{W^{\frac{1}{3}}}{R} \right)^{1.13} \qquad (1)$$

where W : Charge weigh (110 lb.)

R : Vessel radium (6.7 ft.)

$$Pm = 1.74 \times 10^4 \ \text{psi} = 1.22 \times 10^3 \ \text{Kg/cm}^2$$

It is much greater than the pressure corresponding to the yield stress of the vessel material, so that the vessel yields and expands in a short period.

This impuls at the vessel wall is also given by Cole' formula,

$$I = 2.18 \, W^{\frac{1}{3}} \left(\frac{W^{\frac{1}{3}}}{R} \right)^{1.05} \quad (\text{lb-sec/ft}^2) \qquad (2)$$

$$\therefore \quad I = 7.7 \ \text{lb-sec/ft}^2 = 0.55 \ \text{Kg-sec/cm}^2$$

From the conservation law of momentam, the graphite blocks with a unit mass m will be given a velocity V_o,

$$V_o = \frac{I}{m} \qquad (3)$$

m : mass of graphite blocks per unit area,

$$= \frac{1.60 \times 120}{g} = 1.95 \times 10^{-4} \ \text{Kg/sec}^2/\text{cm}^3$$

$$\therefore V_o = 2.8 \times 10^3 \ \text{cm/sec} = 28 \ \text{m/sec.}$$

The given kinetic energy E_k is,

$$E_k = \frac{1}{2} m \; V_o^{\,2} = 7.6 \text{ Kgm/cm}^2 \qquad (4)$$

The graphite blocks with a velocity V_o finally hit the blast shield with a unit mass m' giving a velocity V_1, and again from the conservation of momentum,

$$m \; V_o = (m + m') \; V_1 \qquad (5)$$

$$V_1 = \frac{m}{m + m'} V_o$$

The blast shield is composed of a serpentine concrete and steel liners with total thickness of 30 cm surrounded by steel hoops.

Assuming the average density of the blast shield throughout the thickness is 2.4 g/cm^3, the unit mass of it m' is,

$$m' = \frac{2.4x30}{g} = 7.3x10^{-5} \text{ Kg sec}^2/\text{cm}^3$$

$$\therefore \quad V_1 = 20.4 \text{ m/sec.}$$

$$\therefore \quad E_k' = \frac{1}{2} m' \quad V_1^{\,2} = 1.5 \text{ Kg.m/cm}^2$$

Should this kinetic energy be absorbed by the plastic strain energy absorption potential of hoops,

$$E_k' = \sigma_y \cdot \xi \cdot t \qquad (6)$$

where, σ_y : dynamic yield stress of steel,

: hoop strain

t : effective thickness of hoops

Assuming σ_y is 30 Kg/mm^2 and is 5 %,

$$t = \frac{1.5}{3{,}000x0.05} = 0.01 \text{ m} = 10 \text{ mm}$$

It is equivalent to the square bar 32 ϕ at every 10 cm.

② Plug hold down constraints

The shock peak pressure and impulse at the bottom of the top shield plug may also be estimated by the Cole's free water formulae, equations (1) and (2).

$$Pm = 2.25x10^4 \left(\frac{110^{\frac{1}{3}}}{15}\right)^{1.13} = 6.2x10^3 psi = 4.4x10^2 Kg/cm^2$$

$$I = 2.18x110^{\frac{1}{3}}\left(\frac{110^{\frac{1}{3}}}{15}\right)^{1.05} = 3.15\ lb\text{-}sec/ft^2 = 0.22\ Kg\text{-}sec/m^2$$

The kinetic energy E_k given to the plug by this impulsive load is,

$$E_k = \frac{(IA)^2}{2M} \qquad (7)$$

where, M : Total mass of the plug = $\frac{150\ ton}{g}$

A : Bottom surface area = 12.56 m^2

$$\therefore \quad E_k = \frac{(0.22x1.256x10^5)^2}{2x150x10^3 x\frac{1}{980}} = 2.55x10^6\ Kg.cm = 2.55x10^4\ Kg\ m$$

The energy absorbing capacity Ea of the hold down bolts must at least be equivalent to E_k,

$$Ea = \sigma_y \cdot A_B \cdot \ell \cdot \epsilon \geq E_k \qquad (8)$$

where, σ_y : dynamic yield stress of bolts

A_B : effective crossection of bolts

ϵ : bolt strain

Assuming that σ_y = 30 Kg/mm^2, ℓ = 50 cm, ϵ = 10 %,

necessary cross section of bolts A_B is,

$$A_B = \frac{Ek}{\sigma_y \cdot \ell \cdot \epsilon} = \frac{2.55x10^6}{3,000x50x0.1} = 1.7x10^2\ cm^2$$

Since the plug is lifted up by upward pressure in excess of 1.2 Kg/cm^2, the successive internal blast pressure will act as a lifting power if not anchered, and the hold down bolts must also be resistant to this upward force.

Here arises a question; how to estimate the internal blast pressure, since it depends on the volume to which the gas can expand and on the energy losses in disrupting the core internals and in loosing the heat to any heat sinks.

One solution is to assume the pressure as an upper bound that corresponds to the ultimate strength of the vessel material. This maximum pressure P is,

$$P = \sigma_{uv} \quad t/R \tag{9}$$

where, σ_{uv} : ultimate dynamic tensile strength of vessel material

t : thickness of vessel wall

R : vessel radius

On the other hand, the necessary cross section of hold down bolts A_B is,

$$A_B = \frac{P.A}{\sigma_y} \tag{10}$$

substituting (9),

$$\therefore \quad A_B = \frac{\sigma_{uv}}{\sigma_y} \cdot \frac{t}{R} \cdot A \tag{11}$$

For this particular case, $t = 2.5$ cm, $R = 200$ cm, $A = 1,256 \times 10^5$ cm^2.

$$A_B = 1.57 \times 10^3 \frac{\sigma_{uv}}{\sigma_y} \qquad (cm^2)$$

If we use the same material for vessel and hold down bolts, the strain rate effect will be cancelled.

Assuming the rate $\sigma_{uv}/\sigma_y = 1.5$,

$$A_B = 2,350 \ m^2 \qquad (30\text{-}100 \ \varphi)$$

It is much large than that required for the initial shock energy absorption, in another words, the blast pressure is the predominant loading for the bolt design.

It may be a reasonable choice to use the high tensile strength stell for the bolt material.

Assuming $\sigma_{uv} = 60\ Kg/mm^2$ (304 stainless steel) and $\sigma_y = 100\ Kg/mm^3$ (high tensile strength steel),

$$\frac{\sigma_{uv}}{\sigma_y} = 0.6 \quad \therefore A_B = 950\ cm^2. \quad (20\text{-}80\ \varphi\ )$$

More generally, small plugs or penetrations through the shielding plug such as the control rods or fuel exchange apparatus can also be restrained by hold down bolts with A_B that is required from equation (11).

6. Sodium - air reaction and containment

1) Comment

No comment

2) Questions and answers

(i) Q. Is the calculational model being set up the pool burning in France?

Isn't it necessary to consider as like as in the JFER that the control rod drive mechanism, etc. are blown up, then the sodium is ejected from these holes on the state of the mist and the sodiumair reaction happens, in a moment in case of the core melt down accident?

A.

フランスでは大きなejectは考えなかつた。最初の evaluation では数tonのNaがejectしたとして計算した(この量はoxideがなくなる量に相当する)。しかし今日ではこの見積りはpessimisticであり大き過ぎると考え100ℓのオーダーのNaがholeを通つてejectすると考えている。

したがつてオーダーの問題で問題にならない。これは計算からcheckしたのではなく数年前に1/10スケールのモデルで実験した結果からであるこのときはrotating plugは飛び上らなかつた。Naがejectしたのはholeを通してのみであつた。したがつてNaのもれ量が大きくなることはない。この実験をやるときにはすでに設計ができ上つていたので変更しなかつたがこれ程大きなものは必要ないと考えている。

(ii) Q. It is noted in your report on Rapsodie at the Aix-en-province conference that the amount of sodium ejected from the top plug at the excursion would not exceed several hundred Kg at most even with no plug constraint. What are the assumed conditions for this estimation?

A

① Vesselが破損しNaがejetしてもsafety vesselの中に十分保有できると考える。

② rotativy plugは飛び上らない。実験的に確かめている。

(iii) Q. Does the sudden-shock-pressure happen when the large scale sodium fire occurs?

A. 起らない。何故ならNaのejectは大きくならないからである。

(iv) Q. What degree of the safety factor is necessary on the container design?

A. Rapsodie の設計圧力は 2.4 kg/cm²、この値は材料の code について材料屋が計算した。テストも設計圧力で行つた。

Naがejectし圧力が上昇してもこれ以上にはならないと思う。

(v) Q. What degree of leak rate do you estimate on the container design? What degrees were there really in Rapsodie?

A. 設計 1 % of the volume of the container/ocg at2.4kg/cm²

テスト 5 ～ 8 % 〃 〃

これは ventilation tube などのリークを完全に防げなかつたからだと思う。

(vi) Q. Is the cooling necessary for the container in case of the accidents? How reliable is this cooling system? To show us that if you examined.

A. 必要ないと思う。

(vii) Q. Is it necessary to consider the shocks from the outside (for example, the airplane accidents, etc.) on the container design? Or are any means taken to prevent them?

A.

(viii) Q. What are your back data for estimating the containment pressure build up at a major sodium fire? Is any experimental work performed?

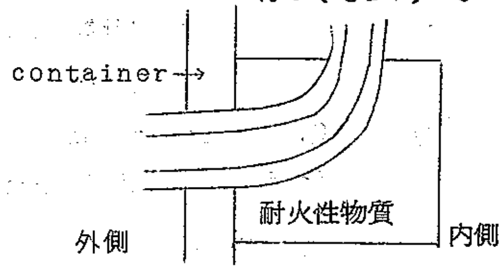
A.

(ix) Q. We decided the maximum allowable leakage rate of the secondary containment vessel as 1 %/day at 2.5 Kg/cm^2, taking account the presumable damages of gaskets or seals of penetrations due to the high temperature at a sodium fire accident. However it is not possible to test this leakage rate at the assumed conditions. What the way to verify this integrity?

付3（その1）－5

A, Rapsodieのpenetrationは右図のとおりである。

テストは圧力を上昇し1日～2日放置する方式をとつている。太陽やその他により補正する必要はある。



(x) Q. Would you show us the following items on sodium-fire accident?

① Sodium-Air reaction rate of pool burning
② preventive measures

A.

7. Hazard evaluation

1) Comment

No comment

2) Questions and answers

(i) Q. How do you decide what percentages of the radioactive fission products produced in pellets are reduced to the gas reservoir?

A. We consider that gazeous fission products are given off to the gas reservoir after a mean delay of 3 days. When released to the argon cover gas, in case of leakage, the most noticeable FP are: Xe 135, 133 m and 133 - Kr 85 has a small activity. A special calculation shows that the number of atoms of FP produced per operating day compared to the number of atoms remaining after 3 days shut down is for

Xe_{135} $2x10^2$

X_{133m} $2x10^2$

X_{133} $1,5x10^2$

(ii) Q. It is considered that the halogens would be trapped well in the sodium coolant. What percentage of the halogens released from the ruptured fuel pins reach to the over-gas region? And what is that base?

A. The experiments we have performed shows that a very small fraction of the halogen reach the cover gas region.

(iii) Q. Would NaI be decomposed when the sodium which contain NaI burn? To show us the conditions of the temperature, etc. if NaI would be decomposed.

A. Generally, it is assumed that NaI is easier produced when the sodium is under gaseous or aerosol form.

In case of a sodium fire, we do not exactly know the behaviour of NaI.

Ixperiments are under way on this item in Cadarache.

(iv) Q. What percentage of the halogens released is at the organic state? What is that reason?

A. NaI is considered to be very stable. Consequently, the percentage of halogens released at the organic state is assumed to be negligible.

(v) Q. How many degrees do you estimate the fission products being released to the reactor room in case of the maximum credible accidents and the hypothetical accident?

What value of the decay constant of the plate-out is introduced in France? What is that reason?

A. We are performing experiments in order to estimate the fraction of fission products which can be released to the reactor room and plate out.

PIRANA の実験値を示そう

（仮定）① one subassemblyのmelt down

② reachor room中へのNa200kgのadiabatic burvr → 0.25 bars

③ F.P.gasの放出経路

fuel → argon → reaceor buildivg

④ one subassemblyからのすべてのF.P.がtotal primary sodiwm 13.5ton中にhomogeneousにmixして、そのmixしたNaが200kg reactov room中に放出するとする。F.P.はsolidおよびuolatileのFission gasを意味する。

⑤ Puの100gがreactor buildivgにreleaseするとする。 1 subassembhy中のPuは1～2kgである。

これらの仮定は今日まだ決定されたものではなく、最初の提案であつて、現在議論されている。

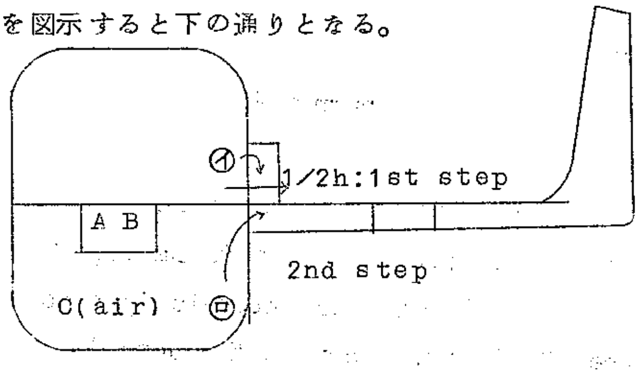
Q.　plate outとはどんな意味か

A.　reactor buildingからのleakage に次の2つのstepsをおく。

1. st step ：　0.25 bar　1/2 h 続き、この間 no plate out, no filteration とする。1 %／dayのlealsageとする。

2. nd step: containmentからの leakage, filteration, no plate ont の状態が many hours 続き、 stuck を利用する。Pu に対しては 0.5 %の release とする。

この状態を図示すると下の通りとなる。



通常運転中は①の line を通り、accident のときは回の line のみを stuckにぬくようにする。

I (iodine)は Na 中に trap されて Na I となり gas 中には出ない。Na I の melting temp. は 650℃、 boiling temp. は、1300℃である。

(vi)　Q.　Is the waste gas stored in the decay tank in the case of the normal operation in France?

A.　Decay tank is used for storage before release of active gas.

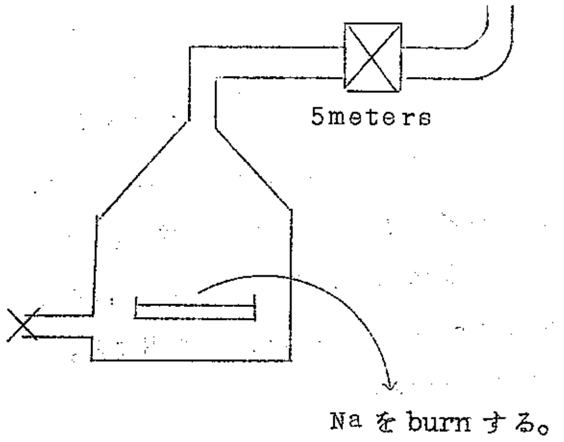
(vii)　Q.　What kind of filter is used to collect the plutonium?

feel uneasy that the efficiency of collection is kept actually well at the state which includes the aerosol of the sodium.　What is your opinion?

A.　No special filters are used to collect the Pu.　The special asbestos filters for sodium aerosols will also retain Pu. Experiments have shown that 30% only of sodium aerosol reach the filters.

Q ：上記の３０％とは何か

A ：下図のような実験から得た値である（ＦＡＮＡＬの実験）



Na を burn した結果、 filter に達する sodium aerosol(CO_3Na_2)は３０％であつた。これが Rapsodie の設計値となつている。

(viii) Q. JEFR containment system needs the particular iodine removal filter, resistant to Na vapor or NaOH.

We find that the Rapsodie is also furnished with this type of filter. Show us the detail if possible.

A. The experiments have shown that on the filters, only sodium carbonate are found (carbonates are more stable than oxides). We do not yet know where and when these carbonates are produced.

In these experiments, we consider the degree of humidity of air and the rate of CO_2. We do not well understand the usefulness of iodure filters in the case of a sodium cooled reactor.

(ix) Q. We adopted the aerosol model for the Pu hazards analysis as proposed in SEFOR and NA-2. What are the official opinions of CEA for this assumption?

(x) Q. Our allowable Pu exposure limiting value is taken as 0.04 Ci, the maximum body burdon for employee proposed by ICRP recommendation. What is yours?

A. We use the CMA worker and not the QMA (maximum body burden). We admit 2 CMA.

(xi) Q. To show us the matters which are useful and are different on the plutonium hazard protection between for the reactor establishments, and for the reprocessing establishments from your experiences in France.

(xii) Q. Please show us the design criteria for dose of radioactive exposure and limiting time for staying in container of Rapsodie

N.B. : JEFR : 0.6 mrem/hr or 1 mrem/hr

In JEFR design we can usually enter into the reactor room except special regions (where contain streaming on the control rod).

A. The safety thresholds for radioactive exposure are the following in the reactor building :

alarm : (evacuation of personnal) 25 mrad/h

reactor normal shutdown : 125 mrad/h

scram : 250 mrad/h

(xiii) Q. Insertion of foreign bodies into core region

Considering the core arrangement, it is very difficult to assume that foreign bodies will be inserted into core region. Would you explain what kind of materials have you expected in Rapsodie? Would you explain you preventive measures and detection method on the channel blockage accident ?

A.

(xiv) Q. Fission products release rate

Would you tell us the release rate of each fission product from fuel pellet to plenum?

And if you have classified, in your hazard analysis, all fission products into rare gas, halogen gas, volatile solid and other solid, would you tell us the each release rate and its reasons?

A.

8. Experiments on the safety analysis

1) Comment

No comment

2) Question and answer

(i) Q. To show us the outlines of the methods, the results, etc. of the experiments which were necessary to confirm the assumptions and the results of the safety analyses, by dividing to the following items; the accidents analyses, the analysis of core melt down accident, the analysis of sodium-air reaction, the hazard evaluation etc.

A. For hazard evaluation, the rate of FP release is evaluated after experiments. The behaviour of FP in the primary coolant is analysed, as well as in the containment building and in free atmosphere. The diffusion in atmosphere is studied on the site and concentration is determined in relation with releasing flow. The characteristics of the containment are fixed in consideration of the results.

9. Calculation codes in France

1) Comment

No comment.

2) Question and Answer

(i) Q. To show us the table which is classified by the items on the outlines of the calculation codes being used for the safety analyses in France.

A. a) CODE PERMETTANT LE CALCUL DES EXCURSIONS DE PUISSANCE -

by I. PUIG

I - CE CODE CALCULE ESSENTIELLEMENT -

1 La température moyenne du combustible du coeur en fin d'excursion de puissance ;

2 L'énergie mécanique produite lors de la détente isentropique du combustible liquide - vapeur ;

3 La pression dans la cuve après la détente.

II - DEUX VERSIONS EXISTENT -

Dans la première, les pressions sont calculées à l'aide d'une équation d'état à seuil.

Dans la deuxième, les pressions sont calculées par une relation simple entre la tension de vapeur du combustible et la température.

III - GEOMETRIE -

Le coeur est un cylindre droit, à deux zones de combustible coaxiales, d'enrichissement différent.

Les distributions de la puissance et des coefficients de danger sont maintenues constantes au cours de l'excursion et identiques aux distributions initiales (entrée dans le code sous forme polynominale).

IV - LE TAUX D'ADDITION DE REACTIVITE EST CONSTANT -

La densité est constante et uniforme dans tout le coeur.

L'effet Doppler dépendant de la température est de la forme

$$T \frac{dK}{dT} = \text{constante.}$$

Il est affecté à la température moyenne du combustible.

Les neutrons retardés sont pris en compte en un seul groupe.

b. CODE DYNAMIQUE -

by J. LADET

Il permet de représenter le réacteur et deux boucles de refroidissement. Chaque boucle comporte un circuit primaire, un circuit secondaire, un échangeur intermédiaire et un échangeur sodium-air.

Le réacteur peut être découpé radialement en 5 dérivations et chaque dérivation peut être découpée axialement en 50 zones. Pour chaque zone élémentaire on calcule 2 températures combustibles et une température sodium; on suppose que la gaine est à la température du sodium.

Dans la représentation de l'échangeur intermédiaire on suppose que la paroi est moitié à la température sodium primaire et moitié à la température sodium secondaire.

L'échangeur sodium-air est traité comme un échangeur à contre-courant.

Le canal chaud est traité par un sous-programme spécial.

Les grandeurs de commande sont

- la réactivité extérieure ou la puissance
- le débit primaire ou la vitesse de la pompe primaire
- le débit secondaire ou
- le débit d'air
- la température d'entrée de l'air.

10. New item

1) Comment

No comment.

2) Question and Answer

Q. Items which are to be analysed newly

Please show us the items which are not analysed on the JEFR and which are necessary on the safety analyses, and to give us the methods of these analyses.

A. JFERについてはあと下記の2点について考慮する必要があろう。

a) oilが一次系に入つた場合の核的 behavior

b) 可能性のありそうな材料が炉心を通過した場合

11. Standard of safety examination in France

1) Comment

No comment.

2) Questions and Answers

(i) Q. ① On what philosophies are the maximum credible accidents and the hypothetical accident classified?

② Are there the general design philosophies (for example, the 70 items of USAEC) for the permission of the reactor constructions? To show us them if there are. And are they applied also to the fast reactor?

③ What are the principles or these principal values on the selection of the site concerning of the distance to the residence area of the public, the direction of the wind, the speed of the wind, the foundation etc.?

A.

① Maximum credible accidentは考えていない。

② ない

③ きまつた方式はない

①②③を通していえることはフランスは実際に直面した問題に重点を置いて考えているということである。

(ii) Q. Would you tell us your view on safety criteria of your Phenix ?

12. Request of the analysis

1) Comment

No comment.

2) Question and Answer

(i) Q. We would like to request for you to analyse on the items of safety analyses which are the questions especially on the JEFR.

We will discuss this subject with you on the details after the first talks.

A. Here are a few indications on which you could judge on the opportunity to perform those calculations.

1 - Comments on the consequences of these incidents -

a) - Jamming of the primary pump -

The primary cause of concern is the temperature of cladding and of the sodium. The maximum is reached in less than 5 seconds, thus the sodium inlet temperature stays constant and there is only a reduction in the flow rate through the core. Hence, the calculations can be performed by the existing Japanese code.

b) - Coast - down of the secondary pump -

This incident causes an action of the safety system (decrease of the power production and stoppage of the corresponding blower). It has no harmful effect on the plant.

c) - Stoppage of the blower -

No harmful effect on the plant. For Rapsodie, it is even considered to suppress any safety action for this incident.

The first incident can be analyzed with the existing Japanese code. As for the two other ones, it is not compulsory to analyze them. Anyway, the Japanese code can yield some useful information on these incidents (the results obtained for Rapsodie can be qualitatively applied to JEFR); our calculated and experimental results can be made available to the Japanese.

2 - Comments on the importance of the job -

The adaptation of our code to JEFR (formulation, detail the constants, exact configuration) would be very lengthy and difficult, particularly because of the distance between the Japanese specialists and us. The most efficient way would be to send a Cadarache specialist to Japan for about a week.

3 - Engineering time and machine time -

We estimate the required effort at 3 engineer-months and 3 or 4 computer hours. We cannot give precise estimates on the dates.

3) Reply from PNC to CEA

The PNC gave to the CEA the list of the constants which is necessary to analyse the dynamics of the JEFR on the 29th January 1969 at cadarach center.

第 3 章　　開発試験の部

1. 開発試験概説

In most cases, the tests which were made before Rapsodie construction were specific of a peculiar component and extrapolation for not similars components is difficult. Moreover, specifications and tests conditions themselves narrowly resulted from the operating conditions which were envisaged. A typical case concerns the prototypes of control and safety rod mechanisms carried out for Rapsodie : for these mechanisms the tests which were performed reproduced as truly as possible the conditions presumed in the reactor. For example, in the experimental testing sodium pot, we reproduced thermal gradients similar to these which might be in the actual rotating plug of the reactor. Finally the incidents themselves which appear during tests are also typical of a peculiar component : bellow failure in one of the two prototypes carried out by private companies jamming problems in the other one. Meanwhile the same general specifications had been prescribed to the two makers.

Therefore we shall discuss here after, only some general items which seem the most important in this matter, thinking that detailed remarks are too bound to a peculiar design of component.

2. - GENERAL VIEW OF TESTS PERFORMED FOR RAPSODIE :

(i) Preliminary tests before construction :

The principal tests performed by makers concerned the following items :

1. the choice of basic materials for the Rapsodie vessels. Specially, creeps tests were performed for the type of steel which was agreed (a special 316).
2. development of welding methods and choice of electrodes compatible with this basic material.
3. preliminary tests for the liquid metal seal.
4. hydraulic tests using water and concerning flow inside the reactor primary vessel : shape of the reactor inlet, influence of the core upper mechanism on the sodium free level, position of the outlet pipes which regard to this level, etc. ....

(ii) Mock-up tests :

On the mock-up of Rapsodie reactor vessel at full'scale, the principal tests which were run are the following ones :

1. loading of dummy subassemblies and checking of their positions on the support plates.
2. preheating by hot nitrogen and sodium filling of the primary vessel mock-up.
3. reliability tests with sodium flow and temperatures corresponding to the reactor nominal conditions.
4. thermal shock tests.
5. operation test of the fuel transfer machine by loading and unloading core and blanket subassemblies.
6. decay heat removal by nitrogen circulation around the

reactor vessel, using an electrical heating core.

7. cooling of the rotating plugs.

The tests which were performed on the two mock-up sodium loops (1 and 10 MWth) were essentially aimed to development of principal sodium components (pumps, heat exchangers) and instrumentation.

All these tests were completed by :

8. hydraulic tests using water to determine the flow distribution system in the primary vessel and subassemblies : annular diaphragm in the vessel and diaphragms inside the different subassemblies.

9. very achieved tests concerning the control and safety rod mechanisms, which were performed in a special facility including static sodium tanks.

(iii) Controls and tests during reactor construction :

Apart the conventional controls which were made by the makers, we can essentially mention :

1. large component assemblings in the factories to check dimensional controls : these assemblings concerned pieces which might ulterialy fit together in the reactor : for example, pre-assembling of thermal shielding shell of the reactor vessel, pre-assembling of the subassembly, core support plates on its support flange in the reactor vessel, etc. ....

2. tightness tests made for the main components :

   a) helium test for reactor vessel, pumps and intermediate exchanger, in which vacuum test was used.

   b) helium test for the rotating plugs, where a small helium over pressure was used.

   c) final tightness test on the site, with a over pressure was used, with ammonia detection for the primary sodium pipes and the reactor vessel.

3. hydraulic tests using water to check resistance of the pumps, shells and plugs, and the heat exchangers shells and plugs.

4. sodium acceptance tests under responsibility of the makers, for the main pumps.

5. Tests during the start of Rapsodie :

   The reader is referred to the report CEA R-3406.

3. - USEFULNESS OF THE TESTS WHICH WERE PERFORMED FOR RAPSODIE :

(i) The test for development of components in sodium and in actual temperature conditions were absolutely necessary. We can mention, for example, the cases of the control and safety rod mechanisms, for which many adjustments were essential and the fuel transfer machine for which very long preliminary tests, before the final tests on the reactor itself, allowed to avoid many incidents.

(ii) The tests run on reactor mock-up have brought out the following informations :

(1) the general conception of vessel and internals structures was good, particularly in regard to the tolerance and the gaps between the subassemblies.

(2) It was necessary to improve the adjustment devices which were intended to get :

a) first, a good alignment between the control and safety rod mechanisms on the one hand, the control rods themselves on the other hand.

b) second a good positionning of the core upper mechanisms above the subassemblies : therefore, in the reactor, the adjustment of the plates which supports the core thermocouples was made possible in the three directions.

c) the main problems concerning the movement of the rotating plugs were the deposit of sodium-vapor in the gaps between the plugs : once they jammed, and it was necessary to be able to heat the plugs.

d) the problems in connection with thermal shocks were certainly over estimated : in any case, it appeared important to reduce the parasitic sodium flows between thermal shields, these flows cause thermal dissymetries.

e) these tests have also allowed to bring out design simplifications. For example, one simplification was afforded to the upper part of the control and safety rod mechanisms : one tight casing only contains all the mechanisms of the reactor. On the reactor mock-up, there was one casing for every mechanism.

f) another simplification seemed possible for the surroundings of the reaction vessel : on the mock-up, there are boxes filled up by insulating materials, between the reactor primary vessel and the safety vessel. Difficulties for the carrying out and assembling of these boxes appeared. They contributed to drop these boxes, for the reactor.

3 Different procedures were developed on the reactor vessel mock-up, which were afterwards used for the reactor itself. They concerned nitrogen preheating, nitrogen emergency cooling, liquid metal seal heating, etc. ....

4 The most important information due to tests performed on the mock-up sodium loops (1 MWth and 10 MWth) concerned :

a) modification of the lateral inlet pipe location of the primary sodium pumps and the design of a simplified plug for the intermediate heat exchanger.

b) the conception of a sodium-air exchanger with several elements to make easier the repairing in case of a sodium leak.

c) development of purification and preheating procedures and development of the instrumentation : flow meters, sodium level indicators, valves, etc. .....

4. - GENERAL COMMENTS ON THE DEVELOPMENT PROGRAMME OF JEFR :

Hereafter, we shall discuss the main criteria which must be considered for the definition of an experimental program :

(i) Construction of important reactor vessel or sodium loops mock-ups :

It seems to us absolutly indispensable to operate at least one large sodium test loop having an industrial scale, before the building of the first sodium cooled reactor. This experimental loops must include the main components of the definitive loop and operate with nominal reactor temperatures. The experience shows that the main results of such a loop concern more technological developments, determination of the operating conditions and training of the reactor operation people than checking of heat transfer coefficients.

It seems to us that, for JEFR, a sodium loop with a thermal power in the range of 10 and 20 MWth al least, could be envisaged : this thermal power would be a good compromise between representativity and cost of the loops. Meanwhile, sodium test of the main pumps at full scale must be performed if the pump is constructed by a maker non competent in sodium technology.

On the other hand, the building a full scale mock-up of the reactor vessel does not seem to us indispensable, thouth it is useful.

Indeed the most important advantage of such a facility is to make a global checking test, more or less representative of the reactor, because some ineluctable differences will exist between the reactor and its mock-up (see comparison made for Rapsodie in the appendix). With the global test it is possible to be sure that there are no important mistakes in the general reactor conception. But, in the other hand, the usefulness to performe tests of vessels internal structures and whole plugs is debatable because :

1. the problems concerning sodium flow in the vessel are better solved in water mock-ups.

2. the thermal shock tests are not necessary, because it is difficult to carry out quantitative informations from these tests and they require an expensive instrumentation (thermocouples in sodium, straingages ...)

3. the problems concerning the rotating plugs are specific (liquid metal seal, sodium deposits ...) and can be tested on special experimental apparatus : so it is with the control and safety rod mechanisms.

4. the problems concerning the fuel handling system, the positionning of the subassemblies can be tested in reduced mock-ups. The test cost must be an important factor for the choice. Otherwise, it is necessary to take into account that the flexibility of the test programme is smaller, if all the test are performed on the same facility.

(ii) Sodium tests for the main components :

It is very important to run the components tests as representative of the definite components as possible : the test conditions must be as close to the actual reactor conditions as possible. However, exception can be made for the heat-exhangers. For these, it is possible to performe tests with bundles which contain a limited number of tubes.

To illustrate the above discussion, we shall indicate the two following incidents of Rapsodie :

1. déplacements of the primary vessel bottom was observed on the reactor, but could not be observed on the reactor mock-up. A difference in the primary vessel support between mock-up and reactor explains this fact (see appendix).

② deformations of pumps shells was seen on the definitive primary loops, but was not apparent during the preliminary tests, because there were some differences in the laying out between the test loops and the definitive loops.

(iii) Hydraulics tests with water :

It is indispensable to develop as much as possible water tests because they are more workable than sodium tests and measurements can be more complete in water.

5. ANNEXE AU PARAGRAPHE 3 :

PRINCIPALES DIFFERENCES ENTRE MAQUETTE RAPSODIE ET REACTEUR ;

(i) Sur le réacteur le circuit de refroidissement du bouchon tournant est simplifié : une seule nappe de refroidissement existe alors qu'il y en a 2 sur la maquette (cette modification est intervenue après réajustement de la puissance calorifique à évacuer).

(ii) Le nombre de baffles thermiques dans la tuyauterie d'entrée du réacteur est de 2, alors qu'il y en avait 4 sur la maquette (les chocs thermiques envisagés sur l'entrée du réacteur étaient nettement plus faibles que ceux imposés à la maquette).

(iii) Les caissons de calorifuge, situés entre cuve de sécurité et cuve d'étanchéité, ont été supprimés sur le réacteur (mais du point de vue sûreté on a prévu en remplacement des réservoirs de noyage).

(iv) Le bas de la cuve d'étanchéité de forme conique, sur le réacteur est simplifié par suite de la suppression du ballon (évolution des questions de sûreté, passage du coeur métallique au coeur oxyde).

(v) La cuve d'étanchéité du réacteur est suspendue, alors qu'elle était posée sur des piliers compensés dans la maquette (prise en compte de l'hypothèse de séisme).

(vi) L'instrumentation en ce qui concerne thermocouples et jauges de contraintes a été simplifiée sur le réacteur : pas de jauges de contraintes sur la cuve d'étanchéité, pas de thermocouples situés à l'intérieur de la cuve d'étanchéité.

6. Questions

Answer for all questions

It is not possible to answer to part A. - Fabrication "Main Items of Testing" of your note concerning the development test.

These questions are part of our know - how in the field of manufacturing.

(i) Please show us the tests which were performed by makers before the mock-up tests of the reactor and its acceptance tests.

An example for reply

Mechanism	Item	Condition	Result	Purpose
1 Rotating plug	1. air-tight test of the through tube	500°C, $2Kg/cm^2g$	leak: 0.5 l/day	to check the air-tightness in high temperature
2 Reactor vessel				
3 Core support plate				
4 Core fuel assembly (fuel, blanket)				
5 Structure neighbouring the core (contain reflectors and thermal shield)				
6 Core upper structure				
7 Control and safety rods				
8 Cask (contain the valve on the rotating plug)				

(ii) Please show us the acceptance test which was performed before the composition test by mock-up was performed.

An example for reply

Mechanism	Items of acceptance test	Parts to be repaired by makers as the results of the acceptance test
(the same as item 1)		

(iii) Please tabulate the test plan and its performance for the components in the primary system of the reactor, concerning the tests which were performed in the air (at low temperature), in the water, in the air (at high temperature), and in the sodium, respectively as the mechanism test, and kindly show us your opinions about the relation to the design and to the operation of the reactor. An example for reply is shown in the next page.

(iv) Please show us the structure, the flow sheet (planning), the test, and the test condition of the test rig which was used.

(v) Please show us the ability, the position for use, and the structure about the detector used in the test rig mentioned above.

(vi) Please show us the method and the criteria for the mock-up tests which you had performed

An example for reply

Control and safety rods

1. the functional tests of gripper
   The unbalance between a fuel and the axis of a gripper was made 3 mm. One drop test per 10 min. was performed and it was repeated 500 times.
   The gripper operation was good, so it's O.K.

② the tests of the moving components.

. . . .

(vii) Please show us your opinions about the relation between the guarantee of the ability by makes and the mock-up tests by CEA. (For example, on the following).

An example for reply

Mechanism	1 In air (at low temperature)	2 In water	3 In air (at high temperature)	4 In argon (at high temperature)	5 In sodium	Relation to design	Relation to operation
Rotating plug (the same as item 1)					(heat removal test) (seal test)	1 the cooling formula in the inner part of plug was changed from the results of the test of 1-3-1. 2 from the test of 1-5-1, the leak of sodium vapour was detected, and reconstruction was performed 3 from the test of 1-5-2, the oxidation prevention for the seal material was improved	1 from the test test of 1-5-2, 10 min. should be taken for heating and cooling of the liquid metal seal

What kind of components are not guaranteed concerning its ability by makers at the step of the end of the mock-up tests ?

What kind of components are guaranteed concerning its ability by makers over a few months of the operation of Rapsodie ?

Were there any troubles which actually happened between makers and CEA? etc.)

(viii) Please show us the relation between the conditions in the mock-up tests (temperature, preasure, flow, and frequency of the tests) and the conditions in the design (not the conditions in the operation), concerning every items of the tests.

(ix) What is the differences between the estimated values from the data which were obtained in the mock-up tests and the phenomena in the reactor ?

(x) What parts of the structure are different from each other between the installation for the full mock-up test and the reactor itself ? How did this difference affect the mechanism operation and the reactor operation in the reactor itself ? In this connection, why couldn't you defend the distortion of the reactor itself ?

(xi) Please show us your opinions about the usefulness of the thermal shock test from the results obtained in the installation for the full mock-up test.

(xii) It seems that you didn't perform the full core mock-up test in water in the case of Rapsodie. Please show us how the vibration of the mechanism in the core and the other liquidity were estimated.

(xiii) We are planning to measure the mixing of the sodium by the flux of the heater pins having actual size in the mock-up test. Please show us the problems about the measurement of the sodium mixing.

(xiv) In connection with the hot spot factor, how did you estimate the distribution of the sodium flow in the channel between each fuel pin.

(xv) Please show us the process of selecting the spiral fin instead of spiral wire in the fuel assembly, and the process of setting the springwise support on the venturi nozzle in the lower part of the fuel assembly.

(xvi) What is the purpose of the decay heat removal test with a heater in the installation for the full mock-up test ? And, why didn't you use the other equipment ?

(xvii) What kind of the results which were obtained in the mock-up test couldn't be put to a good use for the design and the operation of Rapsodie ?

(xviii) What was the problem in the case of the combination of the components which were O.K. in unity into the installation for the full mock-up test or the plant of Rapsodie ?

(xix) What is your procedure of washing the components which are used in the radioactive sodium ?

(xx) Please show us your opinions about the curriculum for the training of the reactor operators using the installation for the full mock-up test, and its problems.

(xxi) Are you planning to perform a special mock-up test for Phenix ? If it is so, what degrees is it ? If not, how about the fundamental test in this connection ?

(xxii) What kind of the detections which were planned couldn't be performed in your full mock-up test ? What is that reason ?

(xxiii) What kind of items of the full mock-up test bring the effective feed back to the design ? And what kind of items are not usefull ?

(xxiv) The others

① Please show us the method of repairing the following components, and the structure of the instrument used in repairing these components.

Primary main pump

Intermediate heat exchanger

Cold trap

Valve

Fuel exchanger, fuel transfer mechanism

Control rod moving mechanism

② On the isolation valve, please show us its manual and its structure in the case of Reactor Plant actually. In addition, have you a special mechanism in order to shut the valve at once in the case of accidents ?

③ Please show us the structure and the design criteria of the sodium tube penetration of the reactor containment vessel.

④ Please show us the schedule of checking the reactor for the ordinary conservation in the case of Rapsodie, and its actual results.

⑤ Please show us your security regulation.

7. 議事録抜粋

1) 燃料棒に関する flow test は2種類以上の流速に関して行なう必要がある。流速を変えれば圧力降下は非比例的に変わる。

Rapsodie の流速は3.5 m／sec である。

2) 燃料取替中の冷却材流量は Rapsodie では約20％である(JEFR5％)。

3) 制御棒の seal は Rapsodie ではベローを使つている。

4) モツクアツプテストは下記のように3段階でやるのが好しい。

第1段階　small scale test

第2段階　sinople な full scall test …… 水使用など

第3段階　full scale mock up test

5) mock up について

Rapsodie の mock up test に約3年かゝつた、JEFRの schedule は少し短過ぎるように思う。

6) Development Test のやり方について

(i) 進め方(そのI)

air，→Hot gas (Ar or N_2)→Na と Test を進めているが、Water→Na の2 steps がよいように思われる。Air，Hot gas 中を Na 中では Material の Bebavior が異なり、Air or Hot gas 中で悪くても Na 中で悪いとは限らない。これはある条件では非常にシビアなテストとなつてしまう。

(ii) 進め方(そのII)

Pump，IHX などの開発を独自で行なう場合、Principle はよくても細かいところに大きな問題が出ることを覚悟しなければならない。

これらについてはメーカが多くの経験を持つことが重要である。

(iii) その他

Rapsodie の Development Test には200人の人が参加した。期間は1960年ごろから始めている。これらの人は今 PHEXIX のテストをしているので現在もここに居る。

8. Rapsodie および Phenix の開発試験見学記

1) Control rod drive mechanism

(i) 部分的テスト

Seal, gripping nechanism, Dashpot などについて部分的テストを行なつて後良いものを組合せて full mock up test を行なつた。

Seal に対しては、2 group に大別され各々3種類ぐらいづつ test を行なつた。

grip についても3 type について test を行なつた。すなわち、①吸収体を軸から離さない方式　②機械的に離す方式および　③電気的に離す方式である。

(ii) Control rod の seal について

Bellow と bellow を使わない2方式があり、PHENIX は bellow を使わない方式になるかも知れないが、designer でないのでわからない。

JEFR（東芝案）の Stainless－stainless はよくない。stainless－iron の方がよいだろう。詳細はわからない。

(iii) gripping mechanism

東芝案についてNa 上の問題はないが、高温と高放射能にさらされているので Spring action はどれだけもつかわからない。少し簡単すぎるようにも見える。

2) 試験現物の説明その他

(i) Rapsodie については全て full mock up を行なつたので、テスト項目を1つづつ書くわけにはいかない。

(ii) 回転プラグ用ベアリング、ギアー、ゴムシール

ギアの径は約6mである。ボールベアリングも、これに合う大きさ。ボールの径は、約40㎜レースはない。ミゾは円形。ベアリングは、とくに問題にならない。

シールは、Rapsodie においては、液体金属シールがほとんど問題がないので JEFR の選択はよいと思う。

PHENIXで、ラバーシールを行なうように予定しているが、何故変更せねばならないのかよくわからない。

(iii) PHENIX の流れ分布テスト

プラスチック1/10 モデルが2つある。1つは、1台のポンプと2台のIHXの関係で調べるもの、他の1つは、原子炉タンク内の流れを調べるものであり、測定は高温の水を流して、熱電対で測定している。

(iv) PHENIXの燃料集合体流動テスト（実物大ダミー燃料）約75℃の水で、流動抵抗の測定を行なつている。振動は、別に行なう予定。PHENIXの集合体はJEFR と同じような流入方式であり、流入孔による流量のError は、ほとんど認められない。しかしここで orifficing を行なうので、この部分のみのテストループを建設中である。

(v) 燃料集合体内の混合の測定

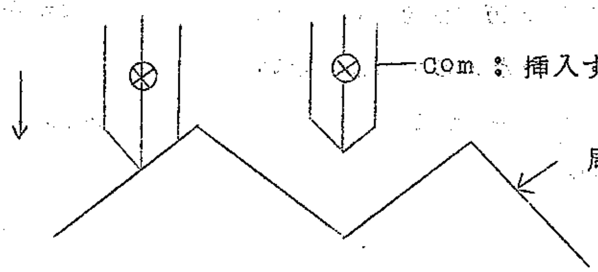
3倍モデルでテスト予定 (iii)と同様高温水の Injection により、測定する。モデルは、プラスチック製を使用する。

(vi) 燃料集合体について、その他のテスト

サポートプレートとのかん合部のリークについては、上方は零、下方は少しある。穴から流出する流れによるキャビテーションなども調べる予定である。

(vii) 燃料集合体の集合と挿入

91本の集合体を、サポートプレート上に並べてクリアランスが適当かどうかを調べる。このサポートプレートとダミー集合体を見学した。また、燃料にはカムがあり、これでうまく入るようになつている。



(viii) Control rod の dash pot のテスト

水の中でテストを行なう装置を作つている。

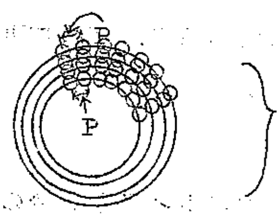
(Na 装 置)

(ix) Rapsodie の full mock up　　Na 試験装置

Vessel の Suppoct 方式が Rapsodie と異り、Rapsodie の主 Vessel の曲りを発見できなかつたが、その他炉体内の全ての機器は、これで試験したとのことである。

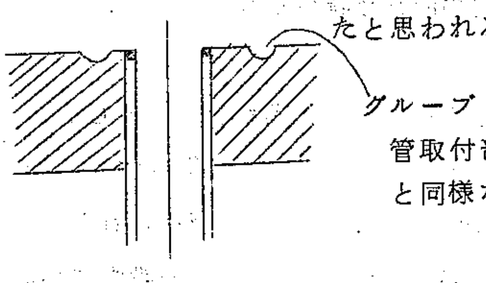
(x) IHX およびポンプ

Rapsodie に使用した IHX と2次系のポンプが置かれてあつた。IHX は、直径60cm程で、チューブの配列、端部溶接部は、下図の通りである。



すなわち内外部共全て同じピッチである。

見えなかつたが、一次側にバツクルがなく、平行流を採用しているから、このような配列になり、コンパクトにできたと思われる。



管取付部は、グループを切つて端部を溶接する。通常と同様な方法と見受けられた。

(xi)　PHENIX 用 NEW Na 装置

7 ton の Na が入つている高さ約10 mの巨大な装置である。タンクが3つあつて、制御棒駆動装置燃料交換機、燃料取出機のテストができるようになつている。

PHENIX では、RAPSODIEのように全ての mock up を作るようなことはせずに、各 Component のテストだけを行なう予定。ここには7本の燃料集合体がダミーサポートプレートに挿入されており、燃料交換機がこれをつかんでいて現在組立調整中であつた。

(xii)　CAPHE : PHENIX 用燃料集合体ナトリウム試験

ヒータ容量1 MWで、①熱衝撃テスト　②寿命テストを行なう予定。ループは2つのテストセクションと機械式ポンプ、タンク、熱交換器などより成り、ほとんど完成していた。

(xiii)　COPACABONA : Ar 中の Na ベーパ分離装置

Ar ガス中に Na を吹込んで、ベーパを作り、これをベーパ分離装置に導いてその性能をテストする装置であり、タンク類を検討中。

(xiv)　その他

(1)　NANET : Sodium Cleaning test , Rapsodie 追放時代のもの

(2)　Rapsodie 用　10 MWループ : 取こわし中

第Ⅱ部　研究所および民間会社見学記

1. Saclay 研究所

1）核計装機器展示会見学および小検討報告書

日　時　　11月28日　15：00～17：00

場　所　　Saclay 研究所、Departement Electronique General 建屋内

見学者　　井上、尾尻

案内者　　Duchēne　DEG／SER（SER：Service d'Electronique des Reacteurs）

小検討での面会者　　Bacconet　　DEG／SER

Plaige　　〃

2）議　事　録

(i) フランスで使用しているDetectorの特性は、報告第6号付2〝Detecteurs Nucleaires a Gaz〟に記載されている。

(ii) フランスでは目下高温におけるfission chamberを開発中である。問題となつているのは特にcableである。

フランスで試作しているものは600℃に耐えるものですでに500℃での経験をもつておりJEFRの設計をしている300℃、no cooling のものは問題ではないと思う。

(iii) Campbell法についてはフランスでも目下開発研究中である。今回の展示会には試作品がでていたが、これは500℃で使用可能であり6 decades にわたつて測定可能だがやはり上端、下端の精度は悪い。500℃における誤差は15％位ある。

Phenix にできれば途中から試用する予定である。

(iv) Atlantic の製品についてのカタログを後日もらうことにした。

(v) 計測機器の値段（detector）

これはCEAの値段でCEA以外に売る場合はウランの値段その他が追加されるだろう。下記のものはいずれも300℃以上に使用できるものである。

CFU－3	6,300F（約47万円）
CFU－65 （ユネルタ含む）	5,700F（約43万円）
CFU－M （cable 含む）	4,000F（約30万円） （予想値）

2. Cadarache 研究所

1) Rapsodie 見学記

日　時：1969年1月28日（火）　10時より

場　所：二次系建屋、制御室、Reactor Room

説明者：Gajac

Wustner

Abdon

Spöri

Paziaut

Valintin

Thevenot

見学者：石川、川口、阿部、井上、尾尻

見学、討論の内容：

(i) 2次系ナトリウム・ループ建屋

① 1次2次ポンプの回転制御機　：　ワードレオナード方式

63 KW　0～360 V　0～175 A　各2台づつ

② N_2 gas Tank, Ar gas Tank　：　建物より10mぐらい離れた露天にあり。

N_2 gas tank　径約1.5m　高さ約3m

Ar gas tank　径約2m

③ Ar ボンベは90㎥　196 Bar のものが用意されている。

(ii) 電源室および原子炉容器周辺

① Diesel Generator は1,000KVA（380V　1,250A）のものが2基ある。これらは compressed air で start する。

② Emergency 用の reservoir tank は原子炉容器周辺に設置され、原子炉容器への容器への Na 挿入は Ar gas 加圧により行なう。このため Na reservoir tank のわきに36本の Ar gas ボンベが用意されている。

(iii) Reactor Room

① 廻転プラグ上部では、Ar Cover gas 中の F.P. の量を FFD detector を使用して測定中であつた。

② CIC 用の plug の activity を decay させるための穴が原子炉室に設置されている。

③ Primary pump および IHX 周辺の雰囲気は N_2 系となつており、それらの上部は空気で生体遮蔽がある。

④ 炉室床面下一階には下記の主要施設が設置されている。

a) Pump

b) IHX

c) Ar 系統のValue 2ケ

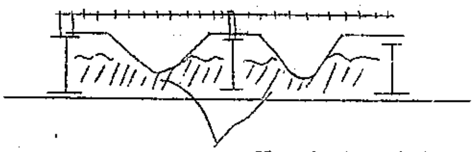
d) Na を drain する Value 12ケ(これは制御室パネル板に電光指示されるようになつている)

e) 1次系Na を Sampling するセル

f) Na purification 用の value 20(Saint Chamond Granat 製)

⑤ Reactor floor は Na fire protection 用の機構となつており、component となつて部分的に取りはずしが可能である。

床構造

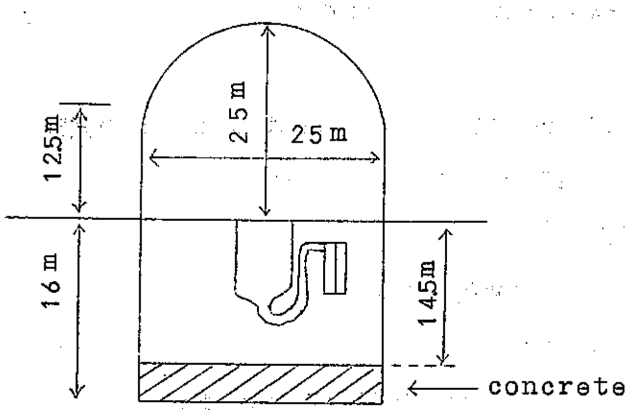


Na が入つた場合 air との接触面積が非常に少くなる。

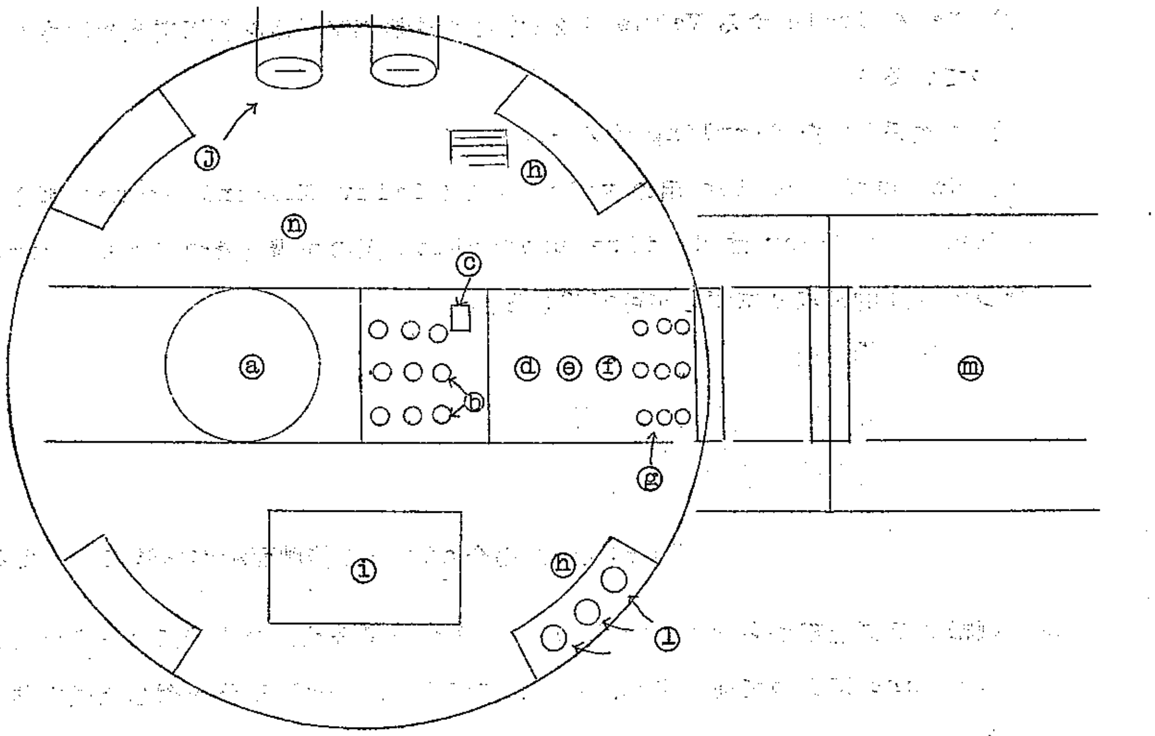
⑥ 廻転プラグ上部から Ar gas leak があつた場合は、炉室壁に設置されている detectors によつて検出され、人が炉室に入り、leak 部分を探した後修理作業が行なわれる。

⑦ N_2 雰囲気となつている1次の遮蔽系統には沢山の detetors が設置されている。

⑧ Containment Vessel の概略寸法は下記の通りである。



⑨ Reactor Room 床面の主要機器の配置は下記の通りである。



ⓐ 炉心上部

ⓑ irradiated fuel の storage

ⓒ clad failure がある場合の試験孔（Ar gas を５００℃までheating する施設あり）

ⓓ Xray および visual による irradiated fuel の試験

ⓔ irradiated fuel 運搬用 pots 保持孔

ⓕ new fuel 挿入孔

ⓖ 予備用 pots（Ar）貯蔵孔　１１ケ

ⓗ New fuel machine

ⓘ Control rod および fuel に対する special machine

ⓙ personal air lock（２つの通路があり、一方通行）

ⓚ 炉室床面から地下へ降りるための階段

ⓛ manual loading rods ２〜３本および peliscope の貯蔵位置

ⓜ 新燃料および照射済燃料出入れ用通路

ⓝ 床に全面グリッドで、Na flow だめとなつている。

⑩ 燃料交換系統は、将来はReflector region にある１２ケの storage に一担貯蔵し、decay させた（約２５日後に４００Wとなる）後取出す。このための燃料取換機は下記の通りである。

a) Hiepano ― Suiza 社製の oil machine・・・・将来は使用しない。理由は半年間の運転維持費で1台の machine が製作出来る。

b) A.C.B 社製の new machine ・・・・ 照射した燃料を vessel 内の storage へ transfer するためのもの。

c) 近い将来製作される新しい machine（A.C.B 社になるであろう・・・未決定）・・・ storage から照射済燃料を取出すためのもの。

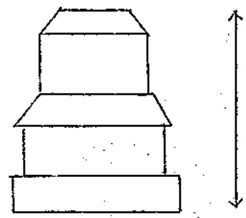
⑪ 現在の Rapsodie は次の2つの計画を実施中である。

a) 〃Aurore〃計画・・・・ Phenix 用燃料を照射中

b) 〃Orphee〃計画・・・・ Graphite 試験用集合体を照射中

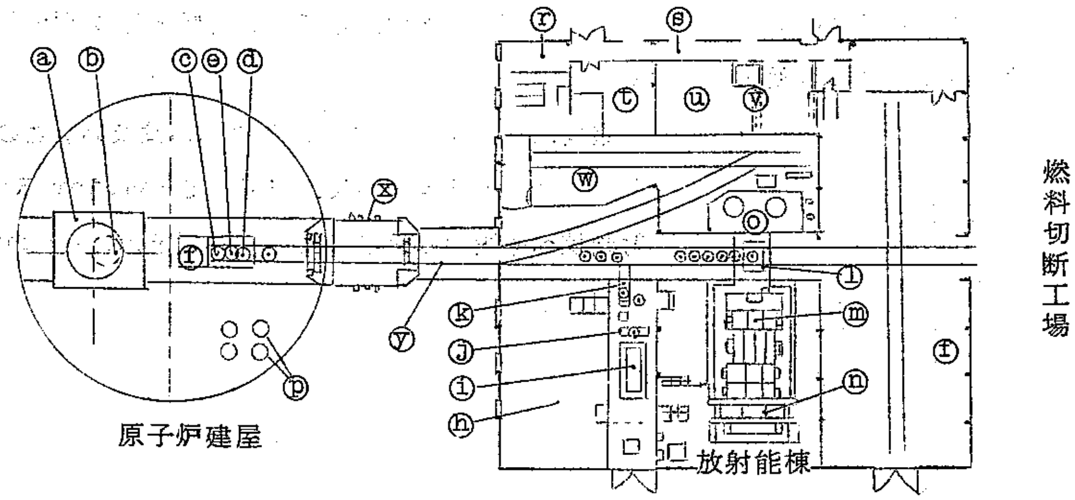
⑫ 燃料運搬用 cask car は燃料貯蔵池より l'ADAC へ運搬するための coffin で大体右図程度の大きさである。

重量は16トンである。



(Ⅳ) 放射能棟（燃料貯蔵室および大型機器修理室）

① 放射能の概要は下図の通りである。



ⓐ 一次燃料取扱装置

ⓑ 原子炉からの燃料取抜位置

ⓒ 照射済燃料孔

ⓓ 新燃料孔

ⓔ X線試験孔

ⓕ 集合体用貯蔵孔兼被覆管破損検査孔

ⓖ 二次燃料取扱装置および特殊大型機器運搬用車の輸送レール

ⓗ 新ブランケット燃料貯蔵およびポット

ⓘ 新炉心用燃料集合体貯蔵

ⓙ 燃料要素の操作

ⓚ 一連の調節

ⓛ プールへの挿入位置

ⓜ 照射済集合体貯蔵用プール

ⓝ プール用橋

ⓞ 洗浄用および除集用孔

ⓟ 放射能を減衰させるための孔

ⓠ 特殊大型機器運搬用装置貯蔵

ⓡ 更　衣　室

ⓢ 廊　下

ⓣ 倉　庫

ⓤ 小型機器除染

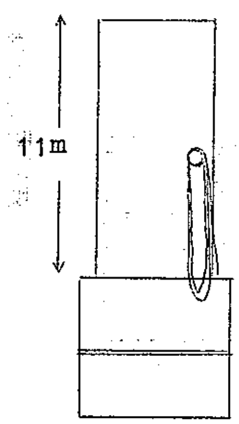
ⓥ 蒸気発生器

ⓦ 除染用広間

ⓧ 閘　室

② activity をもつ大型機器運搬機

Pomp , IHX など activity のある Naが付着している大型機器の運搬は下図のような高さ11mの装置を利用する。

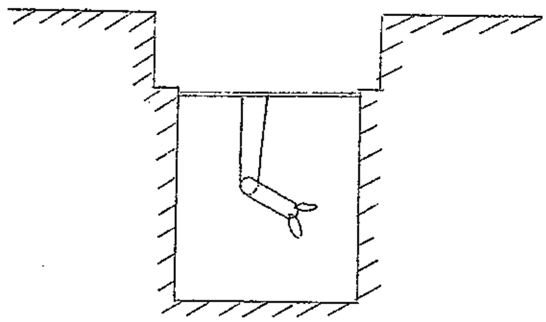


重量は11 ton であり、しやへいはない。この装置中にPomp 又はIHXを挿入、密閉してReactor Room から放射能棟へ運搬する通路には30トン級クレーンの施設です。

③ BF_3 , CIC などの取り出し機を作つたが、実際には使つていない。activity は非常に低く、no shield で作業が行いうる。

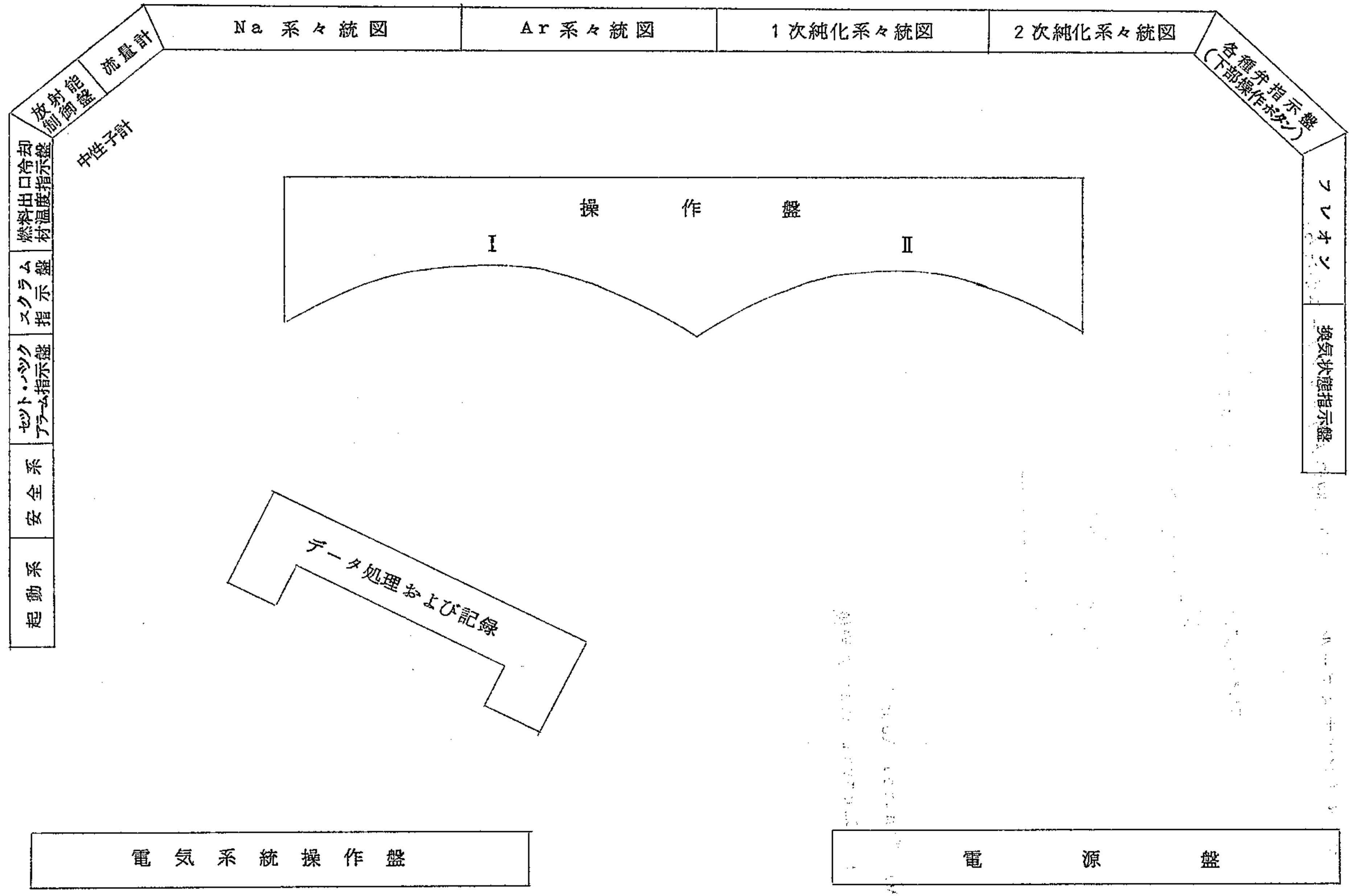
④ 照射済核分裂性燃料貯蔵池は、各集合体中心間隙を250㎜に保ち、集合体頭部から上に約7m厚さの水を保持するようになつている。

⑤ メインテナスプール　　　１００kgのメカニカルアームがある。



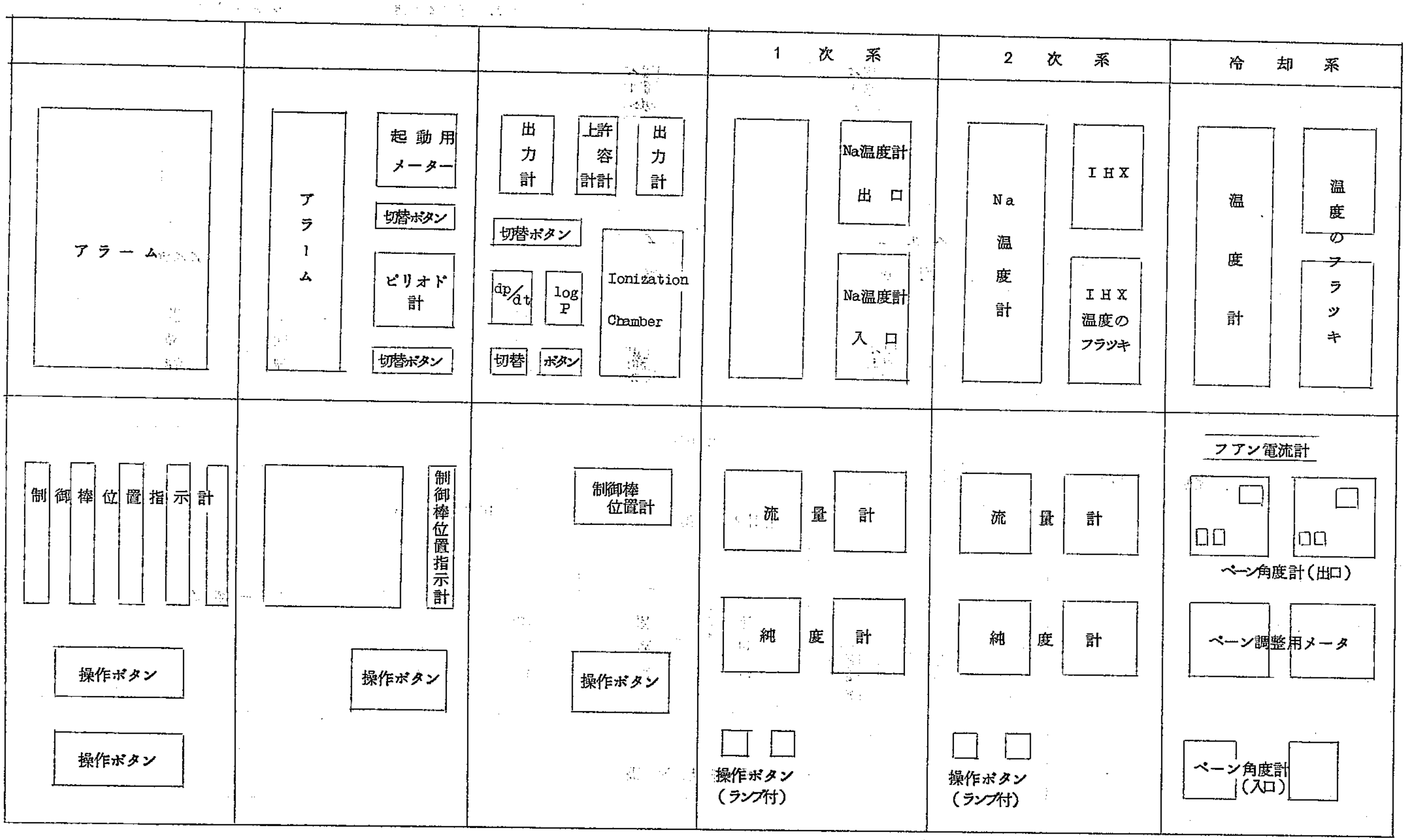
(V) Control Room

① Control Room の配置

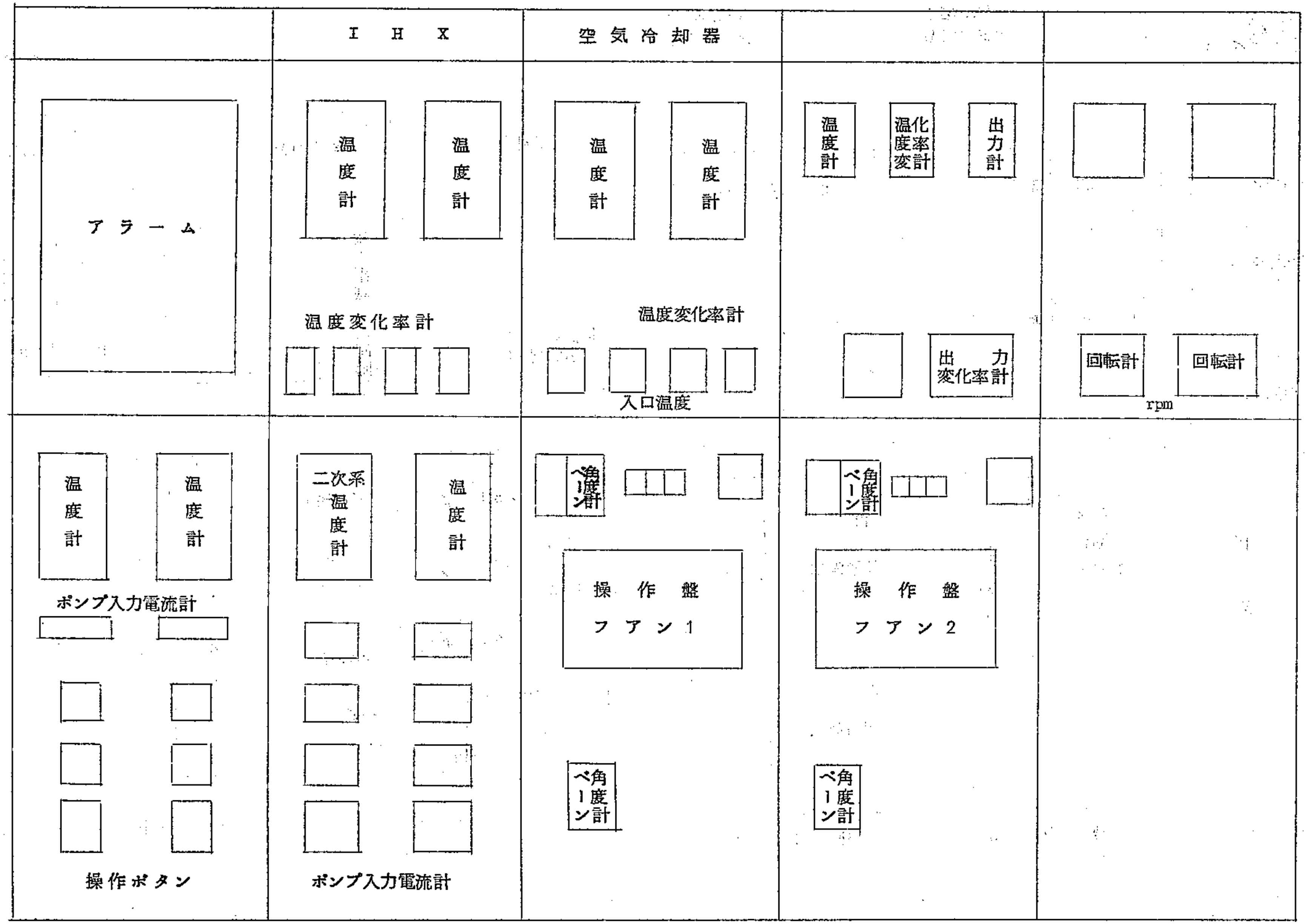


第II－1図　Rapsodie 制御室

第Ⅱ−2図　Rapsodie　操　作　盤　I　（主に通常運転用）



第Ⅱ－3図　Rapsodie　操作盤Ⅱ（主に起動用）

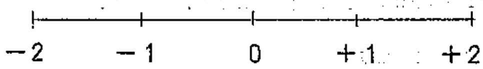


② 主要な計器について（通常運転中監視される重要なもの）

a) Reactivity meter

Merlin − Gérin 社（Grenoble）製の meter を使用しており、CIC からの情報を取り入れている。sensitivity および range は下図の通りとなつている。

o メーター指示



（20 pcm に相当する）

o メーター指示の信頼度 ： ±0.1〜0.2 pcm

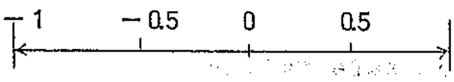
o range

0 − 5 KW

5 KW − 200 KW

200 KW − 20 MW

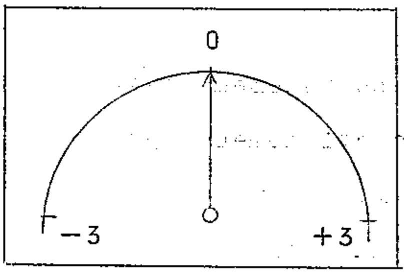
b) 出 力 計（p）



仮りに 0 を 24 MW とすると、1 は 25 MW、−1 は 23 MW に相当する。

24 MW 運転時の出力の振動は、実際の記録上では 0.1〜0.2 MW である。

c) dp/dt 計



計器の値は MW／min. である。

Range 切換器には HP（stop），0.3 MW／min.，3 MW／min. とがある。

d) 原子炉入口、出口温度

e) 燃料集合体出口温度

f) 1次系、2次系回路上の12ケの thermal resisters

g) Primary Na circuit の flow meter

permanent magnet flow meter と EMF の2基がそれぞれの loop に設置されている。実際には permanent magnet flow meter のみが使用され、EMF の方は予備であるが、2年間の運転で drift はない。

③ Alarm, Set back, Scram の指示盤

a) Alarm 用指示 I (Disc. Mes.)

- o Neutronique 1 Disc. Mes.
- o Nouveau Defaut
- o Radioprotection 1 Disc. Mes.
- o Nouveau Defaut
- o Thermique 1 Disc. Mes.
- o Nouveau Defaut
- o TCMC
- o Nouveau Defaut
- o Discordance Alim. Electros (Alim. = supply)
- o Panne T.I.F.
- o Panne Circuit AU
- o Panne Mesure
- o Panne Non Sure AU 1 (Non safe failure)
- o Panne Non Sure AU 2
- o Test Arrete
- o T.I.D. en Service
- o Panne Sure AU 1
- o Panne Sure AU 2
- o Arret D'urgence Fonctionement 1/2
- o Arret D'urgence Fonctionement 2/2

b) Alarm 用指示 II (Disc. Seuil)

- o Neutronique 1 Disc. Seuil
- o Radioprotection 1 Disc. Seuil
- o Radisprotection 1 Disc
- o Nouveau Defaut
- o Thermique 1 Disc. Seuil
- o Nouveau Defaut
- o Grippage 1 Disc. Seuil
- o Nouveau Defaut
- o M.F. C.H. Therm. Sec.
- o Reservé
- o 〃

c) Set back 用指示

- o Debit/Puissance
- o TCMS 2 discordances (Computor による two out of three)
- o Ucs_2 (Core outlet temp.)
- o TCMS Oe Oi (Computor, 燃料出口温度)
- o Activité γ
- o − dUis 1/dt (IHX outlet temp. 1次側)
- o − dUis 2/dt (〃 〃 2 〃)
- o + dUis 1/dt
- o + dUis 2/dt
- o + dVfs 1/dt (最終熱交 Na 出口温度)
- o + dVfs 2/dt
- o Disj. PPna 200 (Primary pomp №2 の stop)
- o Disj. PPna 600 (Secondary pomp №2 の stop)
- o Disj. VLai 602 (Ventilator №2 の stop)
- o Disj. PPna 500 (Secondary pomp №1 の stop)
- o Disj. PPna 100 (Primary pomp №1 の stop)
- o Disj. VLai 502 (Ventilator №1 の stop)
- o Beton − Bouchono (Concrete と rotating plug 間循環 N_2 Temp.)
- o Opérateur
- o BF Chaines Lin, Sec (linear, safety 回路)

d) Scram 用指示

- o 220 DB (220 Vがなくなり、set back が出来ないとき)
- o Opérateur Pupitre SDC (Control room)
- o Activité γ
- o Opérateur Enceine étanch (Reactor room)
- o Sism. gén.
- o Sism. loc.
- o Ucs_1
- o TCMS Oi (Computor による Temp. level)
- o Ucs_2
- o Débit/Puissance
- o P (power)
- o Tp (period)

o log P

o Tn

o log N

o N_2

o N_1 (neutron count の leuel) (start up 用)

o N_0

o Vfs 1mini. (Ventilator の stop)

o Vfs 2mini. (〃)

(VI) Operation Group の Organization

① 通常運転時は 6 persons × 6 groups で 12 h/day の労働時間、日本流の勤務態様と比較すると4直2交代に近い system を採用している。

臨界後の試験期間中は1 groupe 8名であつたが、1年後には7名となり、現在6名を採用している。

② 燃料交換時には、10名の group 編成となる。

③ duty の charge は4 shifts であり、10 weeks 後に 2 shifts が service から離れ、新しい 2 shifts が service につく。period は30 weeks となる。

④ 1人の人からみると、5 weeks が maintenance (9 h/day) 10 weeks が normal duty となる。

⑤ Rapsodie 運転室内には、engineer が15名、engineer でない人が103名で、合計118名である。

⑥ 上記⑤の他に2名の heslth physics の人がいる。この2名は Rapsodie, Harmo-nie, Masrca をも兼務する。

⑦ Operation mamial にない Test をする場合、それ程重大でないことは local safety group. で討議する。

Vandryes 氏の Caduache における代表者であるDenielou氏が重大であると思うことはAECで審査される。

⑧ 運転室には次の5つの manuals がある。

a) Operation

b) Fuel handling

c) Repaire

d) Incidential procedure

e) Periodical test

⑨ 運転所要時間

a) cold point (150℃) → hot point (400℃): 15 h no nuclear beating で2次系の1MWの heat capacity（500KW／loop）により加熱される。

b) hot point からの起動開始 → full power（24MW）：2h

（Critical point → high power ：45 min）

⑩ 1 group 内各人の任務

- o chief ・・・・・・・・・・・・・・・1人
- o chief 補佐・・・・・・・・・・・・1人
- o 電　気　系・・・・・・・・・・・・1人（数秒で control room に来られるところにいること）
- o 運転操作盤前の運転員 ・・・・ 1人で充足
- o 巡　回　員 ・・・・・・・・・・・・2人

計6名／group

⑪ 燃料交換所要時間

400℃から150℃への温度低下 ・・・・・・・・ 12h

照射燃料取出、新燃料挿入 ・・・・・・・・・・・・・・・ 2.5h

2) L′ADAC 見学記

（燃料切断工場）

日　時　：　1969年1月29日　8時30分～11時

場　所　：　Cadarache 研究所　ADAC

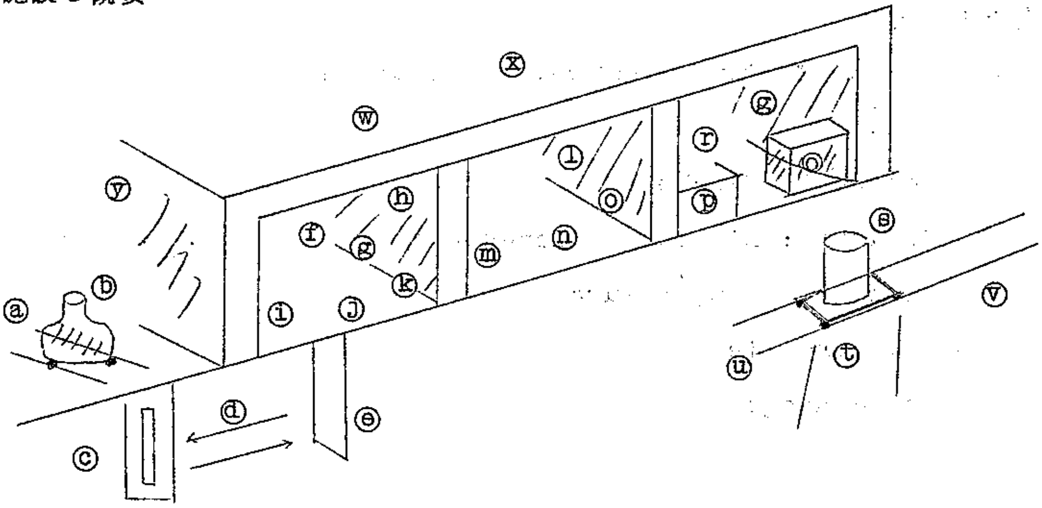
説明者　：　MAS

見学者　：　石川、川口、阿部、井上、尾尻

見　学　記

L′ADAC（燃料切断工場）は Rapsodie 原子炉に隣接しており、Rapsodie 炉で照射された燃料集合体および燃料ピンの非破壊検査を行なうための3つの cells を所有する。

(i) 施設の概要



ⓐ Hall de L' ADAC

ⓑ Hotte secondaire

ⓒ Sas d' introduction

ⓓ Chariot support de four

ⓔ éléateur

ⓕ Cellule N° 1

ⓖ Général Mills 150

ⓗ Pont de 1 tonne

ⓘ Basculeur

ⓙ Scie Alternative

ⓚ Porte $\alpha \beta \gamma$

ⓔ Cellule N° 2

ⓜ Fraiseuse Longitudinale

ⓝ Banc de démontage

ⓞ Stockage

ⓟ Tunnel & Chariot de transfert

ⓠ Cellule N° 3

ⓡ Press a Conditionneur

ⓢ Chateau de Plomb

ⓣ Manipulateur CRL Modele A

ⓤ Periscope

ⓥ Télévision

ⓦ Mécanisme des Portes Levantes

ⓧ Mécanisme Translation des Poutres

ⓨ Arriere cellule

(ii) 各セルの機能

① Cellule 1 : Traitment des assemblages

- Pesée
- Mesure de Fléche
- Mesures ; Diamétres, Longueurs
- Mesure de température
- Radiographie
- Tronconnage

Sous Cellule 1 :

Réception de L'assemblage

— Lavage

— Radiographie

② Cellule 2

— Fraisage des gaines Hexagonales

— Extraction des Faisceaux

— Démantellement des Faisceaux

— Nettoyage des aiguilles

— Pesée

— Mesures : Fléches, Longueurs, Diamétres

— Spectrographie

— Test Fuites

— Gas de Fission : Mesure Pression, Prélévemeut

— Neutrographie

Sous cellule 2 :

— Radiographie ; couleur

— Spectrographie

③ Cellule 3 : Conditionnement des Aiguilles

— Présentation des Containers

— Soudure des boites a aiguilles

Sous cellule 3 :

Evacuation des Bocts a aiguilles

Présentation des Chateaux de Ploml

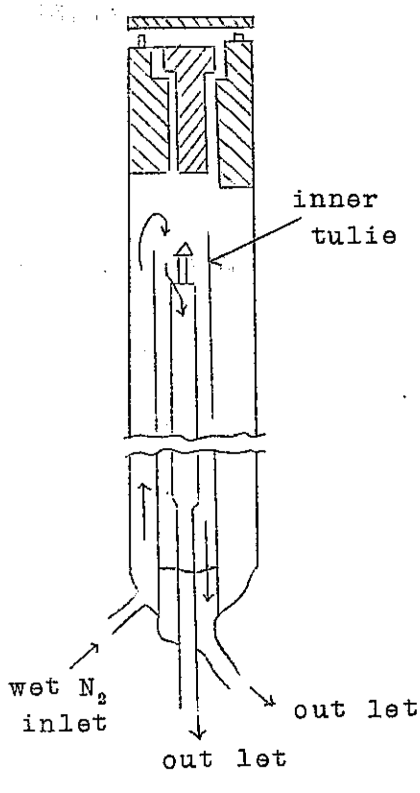
(ii) 付 属 説 明

① Cleaning

Steam は使用せず、wet nitrogen を使用する。

№1のセルの furnace 中には induction heating の施設がある。

クリーニング装置



集合体外形寸法測定装置

高さ2m、巾60cm程度のフレームにダイアルゲージがついており、ラパ管の外側寸法、曲り、パッドの間隔などを測定する。（パッドの間隔は歪んでいるのがある）

② X − ray device

装置は2つあり、1つは燃料集合体を切断する前に10本の燃料ピンを同時に test 出来、他の1つはセル№2内で1本づつの pin について test できる。

③ Diameter measurement

Capacity method（静電容量）により±2 μ の精度で測定できる。

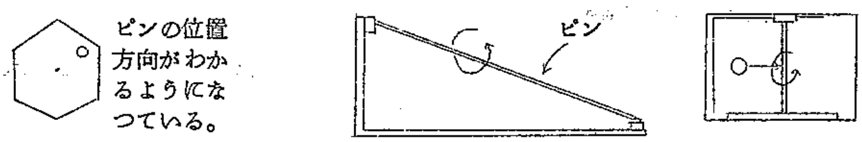
④ Bowing

mechanical micrometer により行なう。

ピン Bowing 測定

集合体内の位置と曲りの方向との関連がわかるようになつている。

ダイアルゲージで測定するが、針の接触力は小さいので、ピン表面にキズをつけたり、測定量に誤差が出るようなことはない。



⑤ γ spectrography

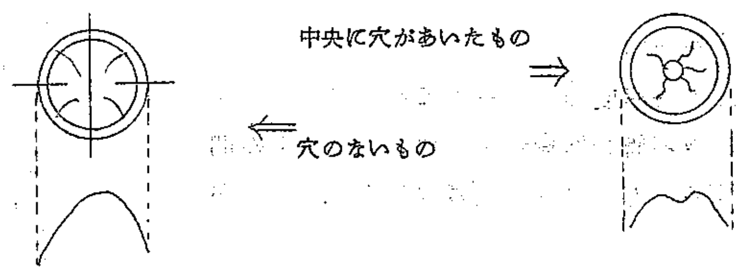
axial および radial 両方向について測定可能で、燃料ピン中心から半径方向に分布する Fission Products の蓄積量について立体的な分布が測定される。

γ spectrography

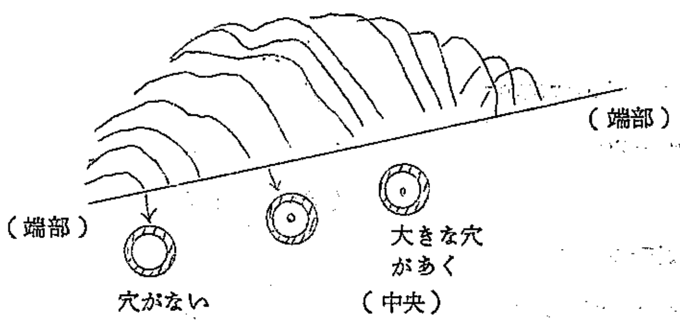
比較的キレイなピン長手方向スペクトログラフイ

F.P. の movement や Fissile mateual の movement があるもののスペクトル

半方向のスペクトログラフイ



ピン1本についてγスペクトログラフィーの結果を表すと図のようになる。

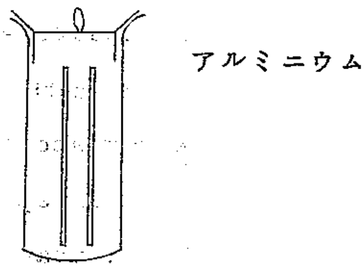


⑥ Neutron radiography

Sb−Be source を使用する。全部の燃料ピンには適用せず、Sampling により行なう。今のところ″not quite good″ である。

⑦ 切断試験後の燃料ピンの運搬

右図のようなピン挿入箱へ111本の燃料ピン(3本の燃料集合体に相当する)を contain し、200トンの cold weld work (Hydranlic pres)により密閉する。



⑧ · 1′ADAC での処理能力

全部の燃料ピンが 1′ADAC の facility で test されるが、いくつかの test は省略することは可能である。現状では1集合体について test するのに必要な日数は約10日間である来月からは automatic な処理機能を使用して一週間に2本の燃料集合体を処理出来るようにする。

⑨ 各セルの裏側の機能

雰囲気は N_2 で、repaire のための部屋となつている。gamma activity が大きくないときは、glove box を利用して手で取扱いできる。TVカメラの施設がある。

⑩ 各セルの地階の機能

Na を cleaning する operation place となつている。洗浄用の pot があり inlet が1ケ、outlet が2ケある。

⑪ カラー試験

1本の燃料集合体中、1本あるいは2本の燃料ピンについてのみ行なう。

⑫ 各試験に必要な時間

(相対的な時間の比について以下に示す)

a) 燃料集合体1本について

Visual experiment	:	1
Metallurgy	:	8

Temp. measurement	:	5
Weight	:	1
X − ray radiography	:	5
Pressure drop	:	2
	(計)	22

b) 燃料ピン1本について

Visual experiment	:	3
Weight	:	10
Metallurgy	:	20
Leak test	:	5
F.P. gas pressure	:	20
γ spectroscopy	:	30
Cross section spectroscopy	:	30
X ray spectroscopy	:	20
Neutron spectroscopy	:	30
Temp. measurement	:	10
	(計)	178
	総 計	200

3) Waste Disposal

日 時 : 1969年1月29日

場 所 : ADAC (Cadarache)

出席者 : Mas, Chapelet, 川口

議 事 録

ADAC見学後次のようなWasteについて討議をした。

(1) ADACでFuel S/A cleaningのときでてくるcontaminated Naはdecayさせる。

① ～50ℓ/S/A　　水とmix

② 2～3 m　　dilute

③ 特殊ローリーにポンプで移す。

④ Health Physics Centerへ輸送

Chemical Agentを加えた後concreteとmix

例　40,000 MWD/T　　S/A

Liquid (50ℓ)	β		1.75	c/m³
	α		4.25×10^{-6}	〃
	γ	Co^{58}	11.7	〃
		Mn^{54}	2.1	〃
		Fe^{59}	1.85×10^{-2}	
		Cr^{51}	2.45×10^{-2}	

N_2は Decay 後 stack へ

(ii) Transportation

Surface dose rate　　200 mr/hr

at 1 m　　10 mr/hr

これを満足するために水で稀釈する。

(iii) Solid Waste

1968年　25 m³

大部分は low activity であつた。

その他

15 lead container　1 r/hr at surface.　10〜20ℓ/container

(iv) Cold trap からの Na

Cs–137
Na–22　in cold trap

将来 cold trap の△pが大きくなつた場合には uriat として外して Health Physics Center へもつてゆく。

(v) Gaseous Waste

Saclay でも述べた通り Papsodir では Fuel Fasture Detection のため、〜1 m³/hr Ar gas を抽出している。(将来は sampling 後再び reactor へ返すつもりである。

Total Gas	$<10^{-2}$ c/m³ Ar	max allowed
(mainly Xe^{133})	10^{-6} c/m³ Ar	at present
		(1〜2 pins failure)
Xe^{133}	3 c/m³ Ar	max up to now
Xe^{135}	1 c/m³ Ar	
Ar^{41}	0.45 c/m³	normal operation
Kr^{88}	negligible	
I	not detected	

(vi) Liquid Waste

pump, ・・・・ の cleaning が source

汚染水はＡＤＡＣで貯蔵後 Health Physics Center へ輸送

4) Ｎａ－水反応に関する見学

日　時 ： １９６９年１月３０日　9.００ ～ １1.００

場　所 ： Cadarache 研究所

出席者 ： Lions，石川，川口，阿部，井上，尾尻

議事録 ：

Cardarache において行なつているＮａ－水反応に関する実験の現状について討議および実験施設の見学をした。

(i) Little Test Loop

５０ＫＷ、５～１０㎜φ tube、１３０ bars、max ５８０℃

漏洩は水（水蒸気ではない）についてのみ行なつた。

試験の目的は

① 試算方法の開発

② 漏洩している tube に隣接する tubc への影響を調べること。

(ii) Single Linear Model of Steam Generator

４ｍ（６ｍ？）long × １００㎜φ　shell

7 tubes

水圧　１６０ bars　３５０℃

Na 中の圧力測定　漏洩ケ所と同じ lovel で行つた。

管の１部をうすくして短時間に漏洩を起させたが reproducibility は極めて良好であつた。

過渡状態での圧力は計算値の方が測定値よりも２０％高い値を与えた。現在 Ｎａ の compressibility を考慮した解析方法を開発中である。

温度は Na 中に0.２５㎜のＴ.Ｃ. を入れて測定、H_2 bulble の平均温度は最初の１００ msec で７００℃

漏洩孔が大きいときは周辺の tube への影響はほとんどない。

(iii) Small Leak

Westage が問題で現在実験中である。対象は Pheuix の SG の tube で材料は３ＨＫ５Ｓ（２¼ Cr）

3. Grenoble 研究所見学記

日　時　：　１２月１７日　９：００　〜　１３：００

場　所　：　Ｎa ループ実験室（ corrosion, 自然循環熱伝達， boiling ）
材料試験炉

訪問者　：　Wustner, Levandowski, 井上，尾尻

面会者　：　Balligand（研究所副所長）

Konovaltschikoff, Champeix(Saclay), Baqué(Saclay)

（以上３名 corrosion test 用 Na ループ）

Millies（材料試験炉）

Martin（自然循環熱伝達試験用 Na ループ）

Costa（ boiling 試験用 Na ループ）

１）研究所概要

基礎的な研究を主に行ない、大学と密接な関係をもつている。従業員は約2,０００名。課は物理関係化学各５を持つている。

２）corrsion test 用 Na ループ

(i) ループの概要

① １度に行える test pieces の数　　約５００

② Na 中の O_2　　１０〜１５ ppm

③ 系統流量（ＥＭＰ容量）　　９ m³/h（ at ３００℃）

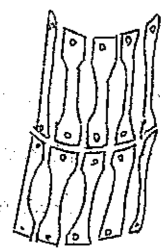
④ 加　熱　器　　電気加熱　　１５０ＫＷ

(ii) テスト方法

化学的なテストのみで機械的なテストはやつていない。

test piece は７０mm×１０mm×３mmで右図のように多数ビスで固定する。

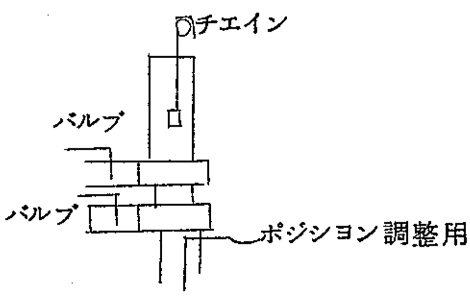
test piece の種類は純粋の Stainless Steel ばかりでなく表面処理をしたものについて行なつている。



(iii) 試料の取出し装入

運転中の取出し装入が可能である。

装置は右図のように簡単なもので chin でつり下げ Ar ガスを２バルブで control しながら装入、取出しを行なう。



３）自然循環熱伝達試験用ループ

非常に簡単な装置でループにはポンプをもたずテスト部は２ｍ×３０cmで系統は電気加熱である。テスト部の最大温度勾配は 1,０００℃／ｍ

である。

詳細は受領資料の方を参照されたし。

4） boling 試験用 Na ループ

2実験装置をもつている。1つは事故時の boiling 実験用で1ループを構成している。このテスト部の最高温度は900℃まで上昇可能、流速は0～7 m/sec で可変。

他の1つは boiling の現象の実験でテスト部だけ他のループに取付けテストを行う。

void 率の測定はX線法を使用している。

5） 材料試験炉

(i) 概　要　　（詳細は受領資料を参照されたし）

① 出　力　　30MWt

② 型　式　　スイミングプール型

③ 燃料濃縮度　　93％

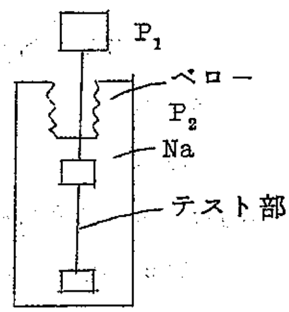
④ instrument の数　　45

⑤ 初期反応度余裕　　9,000 pcm（500 pcm＝1日）

⑥ 中性子束　　2.4×10^{14} n/㎠ sec（但し1 MeV 以上）

(ii) stress をかけた状態での照射……（特に興味があつたもののみ記述）

テスト部への stress は右図のようにベローを使い内外圧の差を120㎏/㎠にすることにより引張力を与える（図面は受領資料中にあり）。



ベローはフランス製、ベローの径は10㎜以下、大きなものはまだ未開発。

(iii) 燃料の単位長さ当りの出力について

最高2,000W/cm位までいけると考えている。

現在は600W/cm位が最高。

4. Grand Quevilly 研究所

日　時　：　1968年11月6日（水）　10：30　～　12：30

場　所　：　Rouen

面会者　：　Johnsson　　Grand Quevilly
　　　　　　Lebigot　　Grand Quevilly
　　　　　　Birault　　S.E.M.T.R/D.R.P at Saclay

訪問者　：　小杉、井上、尾尻

1）装置の概要　　　詳細はカタログ参照

(i) 主目的　　1重および2重管式蒸気発生器の開発試験

(ii) 定格出力　　5MW

(iii) 建設および運転

① 建設費　　5,000,000F　全額CEA出資

② 建設者　　Stein－et－Roubaix

③ 運転費　　2,000,000F／年　全額CEA出資

④ 運　転　　22名で昼夜連続運転

(iv) 経　過

① 1958年建設開始

② 1964年6月、2重管式蒸気発生器の運転開始、交換熱量のチエツク、蒸気発生器の特性、運転の安定性などの各種テストが行われた。

③ 1966年、Phenix用－Ⅰ型プロトタイプの蒸気発生器（寸法やゝ小）を取付け運転開始。現在までの運転時間約3,000時間。

④ 今後、今後も hydro dynamic stability のテストを行い、これらのデータをもとに Phenix用蒸気発生器の仕様決定を行う予定。

1969年からは Phenix 用－Ⅱ型蒸気発生器の開発試験を行う予定。

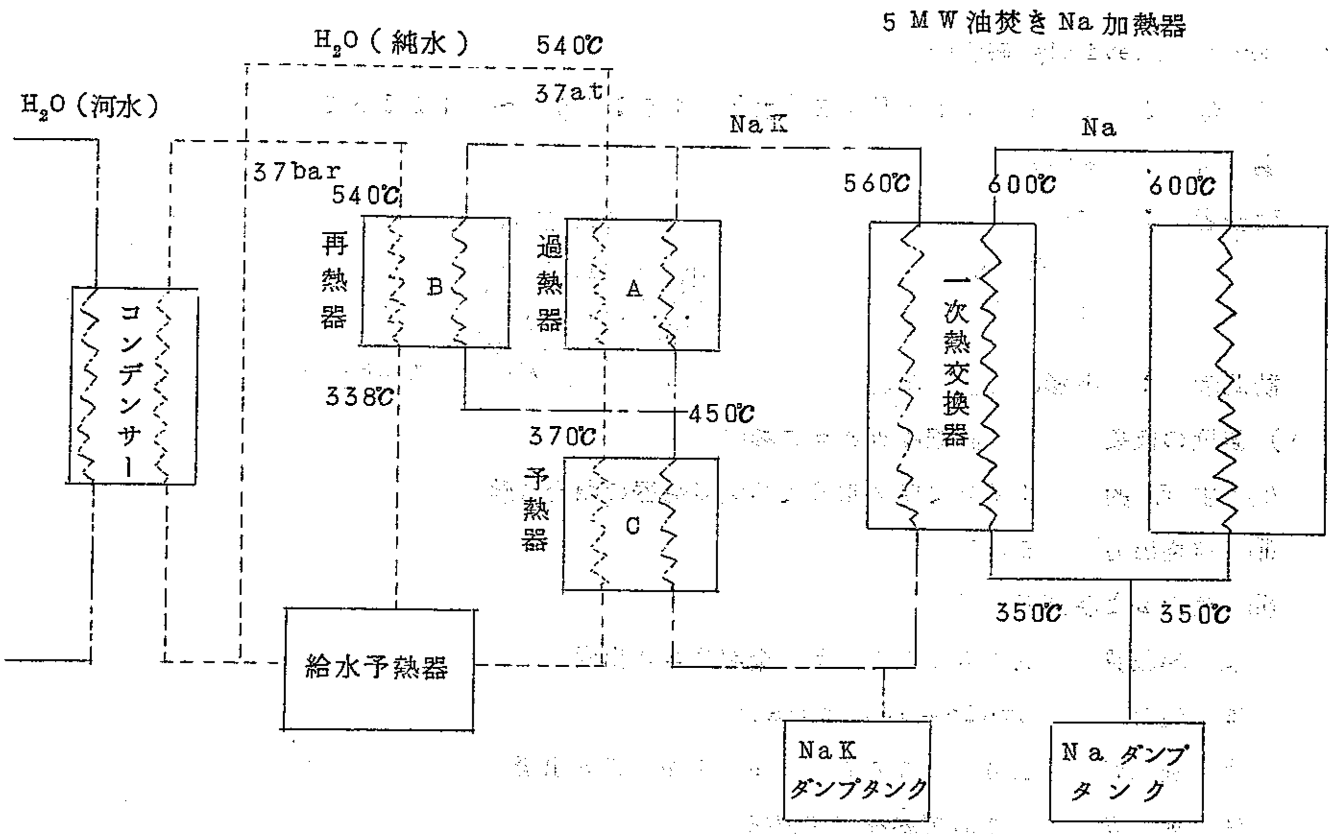
なお、Phenix 用蒸気発生器試験用として Les Renardieres に50MW の新ループを建設中である。50MWループについてはGAAA社訪問記参照。

(v) フローシート

実験装置のフローシートを下記に示す。

各部の値は運転状態、蒸気発生器の種類・大きさによつて異なるが、現在実験中の定常運転状態における値を示す。

図中A、B、Cがテスト部でCEA提案の Phenix用蒸気発生器（採用しない予定）をこゝへ置きテストを行う予定である。



2) 各機器の主な仕様

(i) 1次系ポンプ

① 流 量 56.8 t/h

② 回転数 300〜1,500 rpm 交流モーターの周波数交換による回転数制御

(ii) 2次系ポンプ

① 流 量 73 t/h

② 回転数 〜3,090 rpm 直流モーターのSCR制御による回転数制御

(iii) 中間熱交換器

Uチューブ横形

(iv) コールドトラップ

拡散型、空冷式

(v) 流 量 計

電 磁 型

(vi) 配 管

1次系 4″の2重管で内側オーステナイトステンレス鋼、外側フエライト鋼

2次系 4″の単管

(vii) 予熱

1次系保温材の外側に巻いたフイルムによる誘導加熱方式
2次系, 管外面に取付けた外熱ヒータ

(viii) 蒸気発生器

① 2重壁式 試験完了 管材質 321SUS

② Phenix 用－Ⅰ型　　試験中　　管材質　加熱部、再熱部　３２１ＳＵＳ

蒸発部　2 ¼ Cr, 1 MO, 1 Nb, 0.08 C フェライト鋼

蒸気条件　５４２℃　１６７ bar

③ Phenix 用－Ⅱ型　１９６９年中に試験開始。詳細後述

3）蒸気発生器の安全対策

(i) 水の leak を検出した場合、隔離弁により直ちに隔離する。

(ii) 隔離と同時に N_2 ガスを蒸気管側より注入する。

(iii) Na－水反応による Na 管の圧力上昇のため排出装置を設けてある。

4）事　故　歴

大事故はなかつたが、下記のような小事故は経験している。

(i) ポンプケーシングの下部に小孔が開き Na リークが起つた。鋳造品はピンホールが発生し易いのでこれ以後鋳造ケーシングの使用はやめた。

(ii) 加熱器のリングヘツダーのハンガー取付溶接部に crack が発生し Na が leak した。

5）その他質疑応答を行つた主な事項

(i) NaK を２次系に使用したのは実験を容易にするためである。Na に比較し NaK の方が特に取扱い困難ということはない。大事故の経験はないが小 leak 時は煙が出るのでかえつて leak の発見が容易であつた。

(ii) leak 検出器には電気的検出法を採用している。

配管における検出器の間隔は約１０ｍ、各バルブには１個ずつ取付けてある。今までの経験では上記のようにポンプ、バルブからのリークはあつたが配管からのリークは経験していない。

検出器の信頼度（誤動作がない割合）は９５％位である。

(iii) 修理の場合は Na および NaK を全てダンプタンクへ落してから行う。

(iv) Ar ガスの供給は自動で行つているが、圧力調整は非常にうまくいつている。

(v) １次系のコールドトラツプ温度は１３０～１５０℃に調整しているが ppm コントロールは行つていない。

(vi) Ｎａ Ｋ 系には圧力上昇の異常時自動的にダンプする装置を設けてある。

(vii) Ｎａ Ｋ 系のリザーバータンクは Na 系より大きい。これは充塡容量自体が大きいのと膨張率が高いためである。

(viii) 系に使用している最大配管の内径は１０８㎜

6）corrosion test loop について

(i) 目下１２本の test loop を建設中、test 開始は１９６８年１２月

(ii) 実験は Stein－et－Roubaix が行う。ＣＥＡは実験装置の供給および保証を行うだけである。

(iii) 各試験片について、3,000～10,000時間の dynamic corrosion test を行う予定。

7) Phenix 用－II型蒸気発生器(設計中)について

(i) 5MWの経験にもとずき Phenin 用としてCEA提案による蒸気発生器(蒸気と Na が接する部分は welding をやらない)と Stein et Ronbaix 提案によるもの(welding はあるが12ユニットに分れていて分離可能)を検討中であつたが事故があつた場合容易に取替えられるのと製作費がやゝ安いということより Stein et Ronbain 提案のものに決定した。

(ii) specification は今年末までに決定する予定である。

(iii) Stein et Ronbaix 製の蒸気発生器は目下EDFが Renardieres 研究所に建設中の50MWループを使用してテストを行う予定である。

5. Groupment Atomique Alsacienne Atlantique

日　時　：　1968年12月6日　10：00　～　16：30

場　所　：　午前　GAAA　Siege　Social

午後　Centre　Pierre － Herreng

面会者　：　Ertaud　Directeur Technique　GAAA

Haffnerlehner, Chaminade, Escrive, Clayer, Migaud

訪問者　：　Wustner，井上，尾尻

1）GAAAの概要　　詳細はカタログ参照

(i) 従業員約500名他にEDF、CEAに約100名出向。

General Manager は M. Besse

(ii) 組織は4部および1テストセクション部門から出来ている。

① General Research Department　　部員約25名

この部は neutron physics, shield, radition measurement hert transfer, cost 計算など主に基礎的な物理、数学関係を扱つている。計算機もこの部に属し1台しかないが（それ程大きくない）、Paris の大きな Self － service を使つている。

② Nuclear Applications Department　　部員約150名

この部は nuclear assemblies の設計、原子炉や functional subassemblies の計画などをやつており、engineering, mechaniss, thermal thechnology, testing, material strength, corrosion などの techniques を扱つている。

また fluid metal cooling technique もこの部で扱つている。ループは3～4ループが持つている。

③ Industrial Design Department　　部員約200名

この部は原子炉プラントの設計建設で主に下記の事項を取扱つている。原子炉、制御棒、燃料、循環系（ガス、重軽水、液体金属）、loading machine, cover plate, thermocouple, cable dust など electromechanical および electrical な equipment などを取扱つている。

なおこの部のうち30名はCadarache に出向している。

④ General Engineering Department　　部員約100名

この部はプラントの industrial architecture および general engimeerimg 即ち、設計から最後の据付までの建設の supervise をやつている。

⑤ Test Department

上記の各部でテストが必要と判断された場合

(iii) 受注は、Rapsodie, Phenix はもちろん、EDFのガス冷却炉の燃料取扱機その他、

high flux 炉、ＥＤＦの５０MWt Na loopなどの実験設備を受注しかなり手広くやっている。

(iv) 外国メーカーとの関係は、Interatom（独）、Montecatini ― Edison（伊）と契約関係にある（程度不明）。

2）設計機器の概要　　　詳細はカタログ参照

(i) 電磁ポンプ

６００℃まで使用可能の標準品（５０ in^3／min）と８００℃まで使用可能の高温用（３ in^3／min）を開発している。

(ii) レベル計

Phenix, Rapsodie 用のレベル計を担当、連続、非連続共開発している。

(iii) プラギング計

cold プラギンプ hot プラギング共開発している。

3）C.G.V.S.（ＥＤＦの５０MWt Na ループ）

(i) スケジュール

１９６９年中頃　　完成　Na 注入　直ちにテストに入る

１９６９年１０月末　　テスト完了

(ii) ５０ＭＷループの主要仕様

最高出力	５０MWt	通常出力	４０MWt	最低出力	１０MWt

最高温度	１次系	６５０℃
	２〃	５６７℃
温度変化率	１次系最高	－１５０℃／10sec　＋５０℃／sec
	２次　〃	＋３０℃／10sec
回路数	１ループ	
最大 Na 流量	1,５００t／h	
ポンプ台数	２台	
ポンプ１台当り最大流量	1,０００㎥／h（1,５００rpm に対し）	
圧力	１０kg／㎠（〃）	
蒸気圧力	１６７kg／㎠	
〃温度	５６５℃	

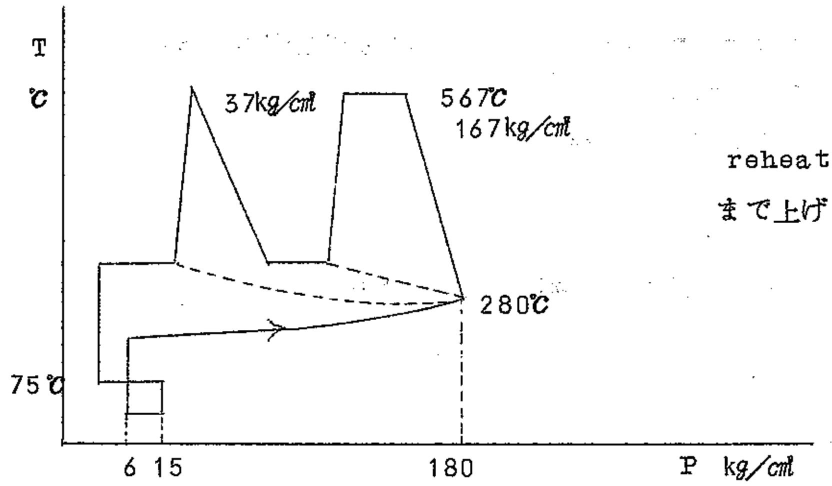
(iii) 系統図

第　図参照

配管材質は３１６＋Ti がほとんどであるが、３０4,３２１も使用。

最大管径は３５０mm

(iv) 2次系のT－P線図



reheatは37 kg/cm²だが50kg/cm²まで上げられる。

なお Steam generator の corrosion 防止のため NH_2-NH_2を注入

(v) 運転についてGAAAが行つている主な研究事項

a. 圧力、温度などのヒート・バランス

b. 単純化および自動運転

c. 材料についての thermal fatique

d. 安 全 性

e. Na－水反応

4) Rapsodie に関しGAAAが担当した主な事項

(i) 計測制御系の研究

(ii) 燃料取替系の冷却系ループ(1次系、2次系とも)

(iii) 制御棒の mechanism

(iv) 回転プラグ

(v) hold down mechanism

(vi) 燃料切断系

(vii) Na のどぶずけ実験装置

(viii) alarm system

なお Rapsodie の前に担当した主な事項は

(i) HRI(Na 関係の建屋)の試験回路

(ii) 建屋・・・・active, primary, secondary, 燃料切断，ベンチレーション

5) Phenix に関しGAAAが担当している主な事項

E.D.F. Babcock Atlantic, Stein et Roubaix, Alcatel と協力して、下記の事項の研究開発を担当している。

(i) EDF 50MWt Naループ

(ii) Naのどぶづけ実験装置B

(iii) fuel loadig － unloading machine

(iv) control rod

(v) E.M. Pump

6）その他の炉で担当している事項

(i) Graphite 炉の loading machine ・・・・ これにはかなり手こずつている。

(ii) 重軽水炉の reactor development

7）Rapsodie, Phenix の制御棒についての情報

(i) Rapsodie の長さ　　450㎜

Phenix　　〃　　1,000〃

(ii) Rapsodie の垂直度に対する誤差は3～4㎜以下になるように設計した。

8）Pierre ― Herreng 研究所

(i) ナトリウムループ

① 1MW機器試験ループ

a）製　作　　1957年設計　　1960年運転開始

b）使用目的　　当初はCEA装置の運転員訓練のために製作

現在はポンプ、流量計、プラギングメータ、液面計、コールドトラップなどのナトリウム機器の試験

c）主要仕様

1次系および2次系をもち、1次系は油焚き加熱器、2次系に空気冷却器およびE.M.ポンプテストベッドを有する。

最大流量　　10㎥／h

設計温度　　800℃（しかし断熱材の厚さ不足で600℃以上上昇不可能）

② E.M. ポンプ試験ループ

a）目　的　　linear induction pump の試験

b）主要仕様

最大流量　　500㎥／h

最高使用温度　　600℃

③ E.M. ポンプ耐久試験ループ

a）目　的　　corrosion, leak その他の試験

b）主要仕様

最大流量　　5㎥／h

最高使用温度　　800℃

リークの検出は自動式で警報ができようになつている。

④ 流量計較正ループ

a）装置概要

タンクを2つもちその間にNaを流し、タンクの重量を測定して流量を算出する。精

度は±2%位。

流量の大きい場合は、ベンチュリ型の流量計で較正するが、水でやると精度が悪いのでできればNa でやる必要がある。

b) 主要仕様

較正できる最大流量	12㎥/h
最高使用温度	600℃
(通常	400℃)

(ii) その他の主な実験

① 高速炉用制御棒駆動装置、燃料交換機器の開発試験

② ガス冷却炉用　〃

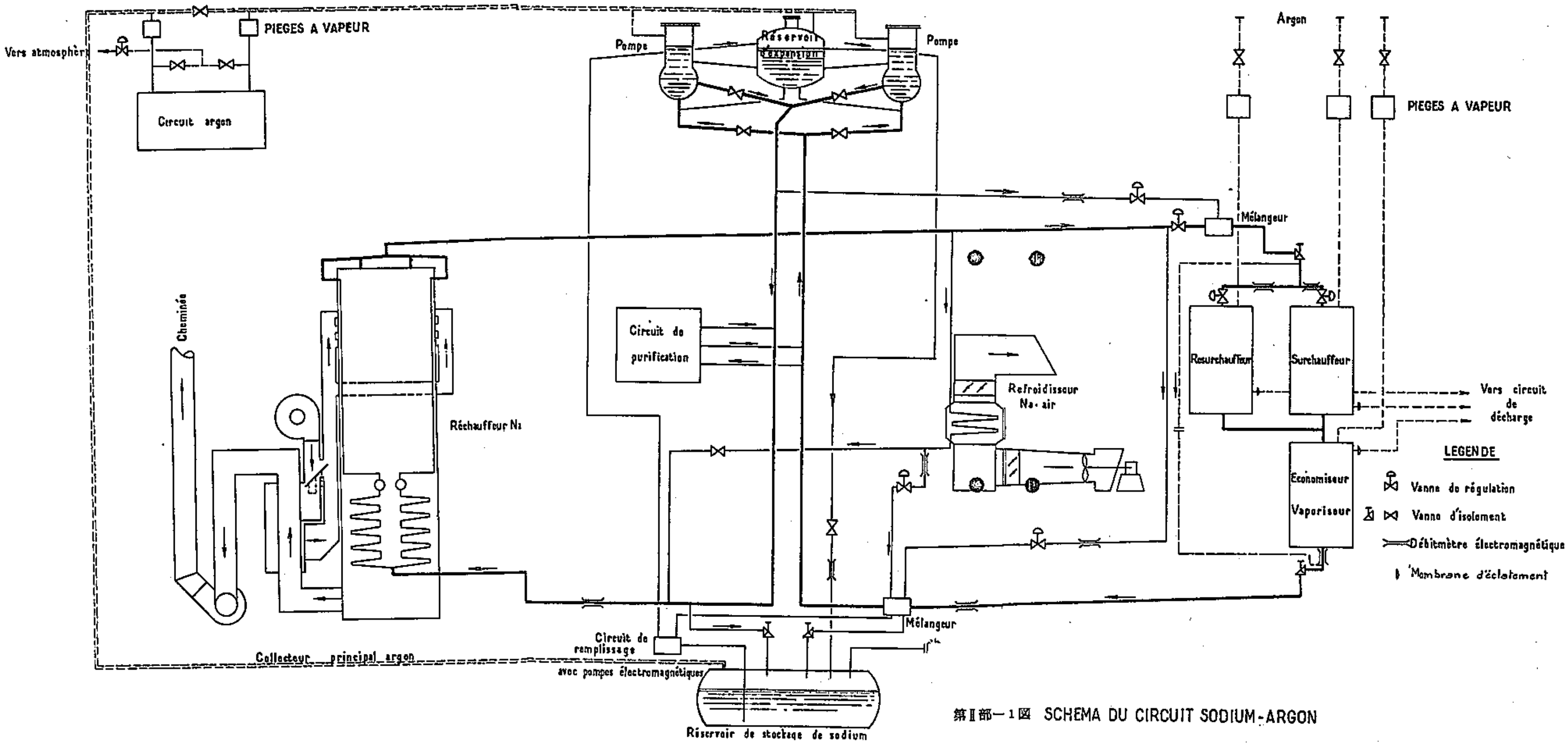
③ HFR、炉容器の応力試験

1/5プラスチックモデルで試験中、ヘッド、ノズル開孔部附近の応力をストレンゲージおよび光弾性で測定

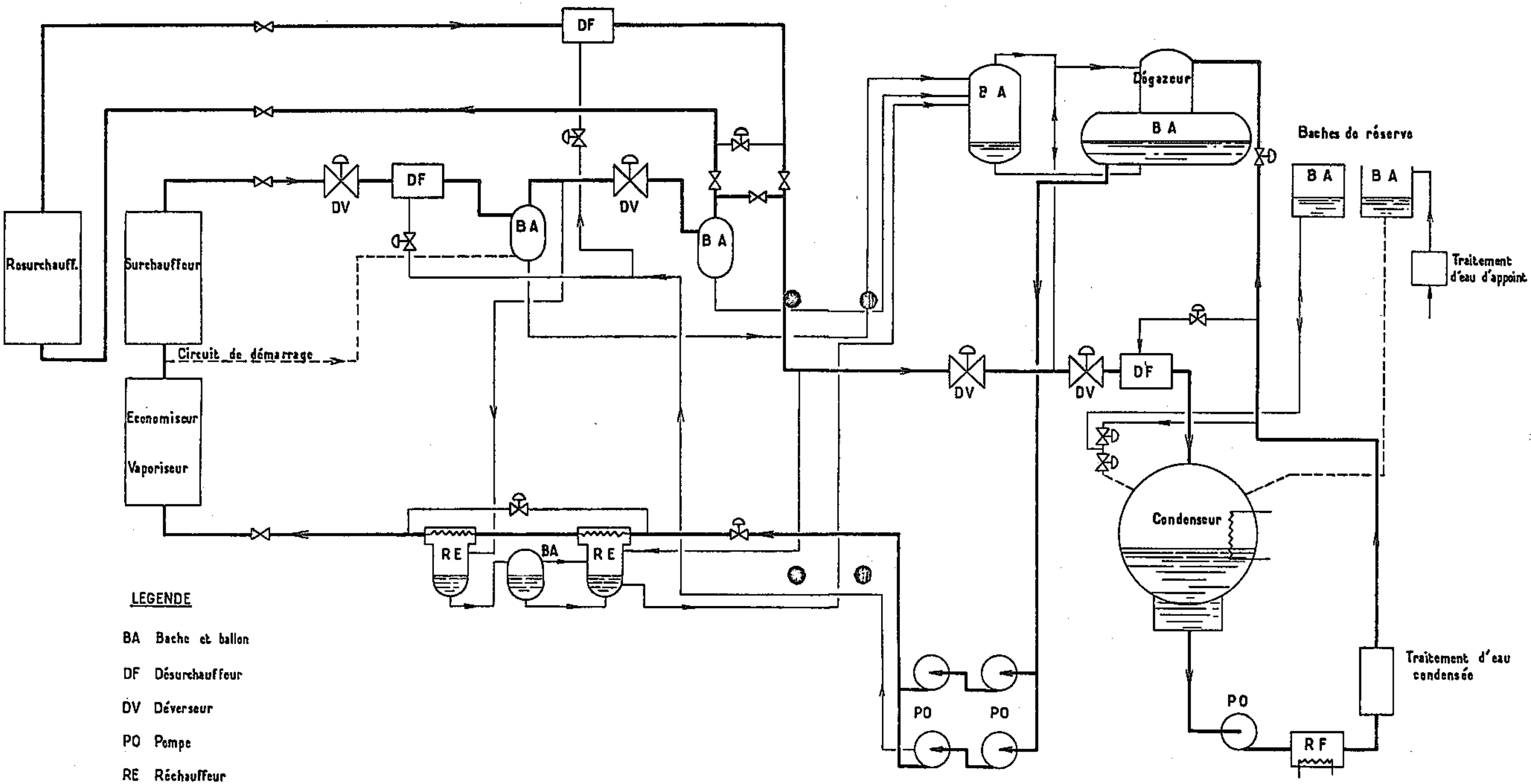
④ シール試験

Al、Zn等を波状の板の間にはさんで塑性変形させるタイプのものを開発しサーマル試験、リーク測定などを実施中。

ナトリウム炉用材質としてはステンレスを考えている。



第Ⅱ部—1図 SCHEMA DU CIRCUIT SODIUM-ARGON



LEGENDE

BA Bache et ballon

DF Désurchauffeur

DV Déverseur

PO Pompe

RE Réchauffeur

Vanne d'isolement

Vanne de régulatic

第Ⅱ部—2図 SCHEMA DU CIRCUIT EAU-VAPEUR

日　時：1969年1月31日

場　所：Saclay 研究所

来訪者：ARSENE MIGAUP, CLAUDE RAMADIER

面会者：BAUMIER，井上，阿部

1) GAAAの高速炉の役割は次の通りである

Architect Engineer of Rapsodie

Architect Engineer of PHENIX

Control rod of Rapsodie

Test control rod for PHENIX

Fuell Handling Mechanism for PHENIX

Sodium loop.　purification loop

EMP

Level indicator

EDF　50MW　テスト装置

2) Control rod drwe mechauism

シールはベローとしている。ゴム製Oリングのバツクアプシールをもうけている。

ベローは、316ステンレス製である。

ベローシールと他の方式が考えられるが、ベロー方式の方が安全（more safe）であると考えている。

ベロー形の制御棒駆動装置は、現在までに27,000回のスクラムテストを行つて、ベローのラプチヤはなかつた。

現在の Rapsodie の駆動装置は、プラグの下部貫通部で径が100㎜であるが、2nd type を現在製作中で、これはメカニズムを上方に持つて行き、プラグの貫通部はこれより細くなつている。JEFRにも使用できるかも知れないと考える。

この 2nd type は Fortishimo計画に使われるだろうと考えている。これを再設計した理由は、前の駆動装置の Spec が、2年の運転経験よりみて厳し過ぎたので、これをゆるくしたことによる。

上部の潤滑はモリコートを使用している。

制御棒駆動装置開発途上に生じた問題点は、グリッパ部およびガイドローラのフリクションの問題と軸のステイックの問題である。

ガイドローラは変更した（2nd Design かどうか不明）。

芯合せについては次の値が許容限界である。

① 吸収体と軸とを結合する場合は5㎜まで

② スクラムは２０㎜まで

3）その他

ナトリウムポンプ、フローメータなどを作つている。

フローメータは 14″でパーマネントマグネットで作ることができる。是非注文を受けたい。

Control rod は Rapsodie type で６本４００,０００＄で売ることができる。

6. UGINE

日　時　　1968年12月17日　15：30〜17：00

場　所　　Grenoble 工場

面会者　　Brun, Jush, Graff, Besson（以上UGINE）

Baqué, Champeix（以上CEA）

訪問者　　Wustner，井上・尾尻

1）会社概要　　詳細はカタログ参照

(i) この会社は各種の化学製品を生産している化学会社である。Naは直接生産していないが、NaClを生産し、関連会社でNaを生産している。Naに関する研究としては腐蝕試験を行なつており、Naループを2つもつている。したがつて、UGINEではNaループの見学のみにした。

(ii) 従業員数　　約1,100人　　しかし近日中に他社と合併し

約30,000人となる。

2）腐蝕試験用 Na ループ

Grenoble 研究所見学記に示すようにCEA自身で腐蝕試験をやつているがテストの量が多いので下記の2ループを製作しCEAと協力して腐蝕試験を行なつている。2ループ共腐蝕試験用である。

Aループ	1 m^3／h	加熱器熱出力	15 KW／h		
Bループ	2 〃	〃	30 〃	入口温度	300℃
				出口温度	700℃

テスト内容としてもループとしても特に目新しいものはなかつたが、プラギングメータ（MECI製）の較正を化学的にやつていた位である。この化学的較正方法についてはCEA M. Champeix の文献参照

3）ステンレス鋼についての腐蝕試験の結果について

CEAおよびUGINEで現在まで行なつてきたステンレス鋼についての腐蝕試験の総合的な結果より順位をつけるとすれば下記のようになる。

① 316L

② 304L

③ 318

④ 347

⑤ 321

7. GACHOT

日　時　　1968年12月30日　10：00〜12：30

場　所　　Aruentenil

面会者　　Peyron　director
　　　　　Pascand, Renandai, Neun

訪問者　　Wustner, Baumier, 井上・尾尻

1）会社概要

大手メーカーではないが、Na用ベローバルブなど特殊バルブの製作を行なつているバルブメーカーである。輸出もかなり行なつている。小規模ながら生産管理に計算機（IBM）を使用しかなり自動化している。

(i) 従業員　約450名

(ii) 生産高　ball valve　20,000コ／month
　　　　　gate 〃　5,000 ／ 〃
　　　　　その他

(iii) 売上げ高　約40,000,000F／年

(iv) 日本の取引先　三菱、伊藤

2）Na用ベローバルブ

(i) バルブの種類

最大350㎜φ（14 in）まで種々の大きさの経験をもつている。特に今回は50MWthループ用20㎜φの製品を見学した。

(ii) 20〜25㎜φバルブの主要仕様（もちろん注文により異る）

①	最高使用圧力	90Kg／㎠
②	〃　温度	400℃（60Kg／㎠において）
③	材　質	316L（304も目下Grenobleの Corrosion loopで test 中）
④	ベローのdetect方式	スパーク・プラグ方式
⑤	可動長さ	（200㎜φバルブ‥‥ベロー全長300㎜に対し）約60㎜
⑥	溶接個所のチエック	1ケ所づつtestする方式をとつている。（溶接個所は5〜6ケ所ある）
⑦	規　準	UKEA standard に合わしている。設計はシグル40

(iii) その他の高温用バルブ

最高750℃まで使用可能バルブの製作経験をもつている。

(iv) 値　段

a. 15㎜φ　650F／1コ
b. 20 〃　900F／1コ
位である。

8. Robinetterie A.M.C

日　時　　12月31日　11：00〜12：00

場　所　　Saclay 研究所内駐在員居室（A.M.C. 社員来訪）

来訪者　　Orio

面会者　　Baumier，井上・尾尻

1）実績と計画

(i) 実績…… バルブメーカーでNa用としてはRapsodie，PER，50MWt Naループ（オランダ）のベロータイプバルブ（1部はまだ製作中）の製作している Rapsodie の実績では15,000時間の運転経験をもつている。

(ii) 計画…… 英国で計画中の Civil Fast Reactor（約1,000MWe）の試作試験用として納入する予定。これはE.E.CとA.M.Cの特別契約によるもので実際の使用バルブはEECが製作する。

2）バルブの種類

(i) シール方式　　a. ベロー型　　b. フリーズシール型の2型式

(ii) 弁開閉方式　　a. ゲート式　　b. バタフライ式　　c. ボール式の3型式

ボール式は目下まだ開発中である。このボールバルブの利点は高さを非常に低く出来ることである（約1/3）。

(iii) 大きさ　　10mmφ〜900mmφ（36 in）の各種

3）値　段

値段は圧力、温度、型式、大きさ、使用期間などにより異るので簡単に出せないとのことであつたが、いくつかの例について示すよう要求してある。

1 例

4 in　ベローバルブ、最高使用温度600℃最高使用圧力10Kg／cm²で約3,700 F

4）その他

(i) 製作しているバルブのうち最高使用圧力20Kg／cm²、最高使用温度600〜700℃位のものが多い。

(ii) バルブの概略の大きさを知つておく必要があると思い、いくつかの例について概略寸法を示すよう要求してある。

9. Hispano — Suiza

日　時　　1月6日　　10：00〜13：00

場　所　　Bois — Colombes

訪問者　　Baumier，Wustner，井上・尾尻

面会者　　Raczynski，Fouré

1）会社の概要　　詳細はカタログ参照

(i) 主な製品　① 航空機用エンジン、特にジエツトエンジン…… これが主力

② ガスタービン、コンプレッサー等

③ 原子炉用機器、特に機械ポンプ

④ その他

(ii) 従業員数　約20,000名

(iii) ポンプについての他社との関係　インペラ、シヤフトなど内部機構は自社で製作するがケーシングなどは他社に改めて発注している。

しかし、ポンプの設計、研究、計算、組立、水中試験は自社でやつている。

(iv) 日本との関係　帝国酸素が日本側の窓口となつている。しかし現在三菱原子力工業がJEFR2次系用ポンプを設計しそのチエックを依頼したり、他社との関係も深めたい意向。

2）会社の実績（原子力関係）

(i) EDF−1の loading machine

(ii) heavy gas compressors

(iii) Rapsodie の handling cage と liquid metal pumps

(iv) Pegase（実験炉）のループと Compressors

(v) その他種々の原子炉およびテストループの liquid metal pumps，compressors sealed blowers，control rod

3）液体金属ポンプに関する製作実績

(i) 11種類のポンプを計21個製作した。

(ii) 上記の7種14ポンプに関しては計12万時間の運転実績がある。

(iii) 最長運転実績のポンプは Rapsodie のポンプで1万6千時間。

4）原子炉用ポンプについて

(i) Phenix 用ポンプの概略大きさ（設計した中ではこれが最大）

① インペラー径　　900㎜

② インペラ用流路ケーシング外径　　1,200㎜

③ 軸　　長　　6,300㎜

④ 入　　力　　1,200KW（950rpm）

⑤ 流　　量　　4,500㎥/h（〃　〃）

(ii) Rapsodie 用ポンプ（カタログ参照）

① ポンプの流れは横から入り下へ流出する型でＪＥＦＲとは全く異る。

② ポンプ内のNa レベルは最大流量時と最小流量時では約１ｍの差がある。

③ ポンプに関する特許は mechanical seal 部のみもつている。

5）ＪＥＦＲ用ポンプ（三菱設計案）について

(i) 三菱案を採用するなら添付図面のように改良した方がよい。

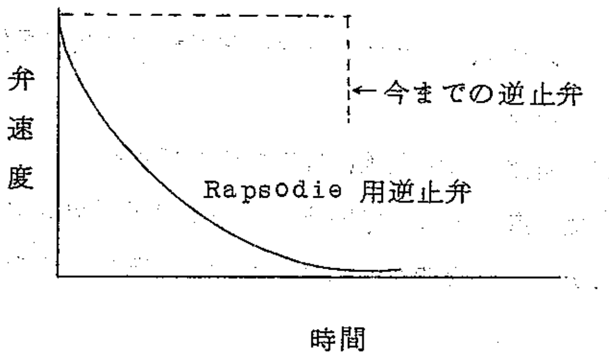
(ii) liquid bearing の Na 供給方式は特によくない。遠心型にした方がよい。

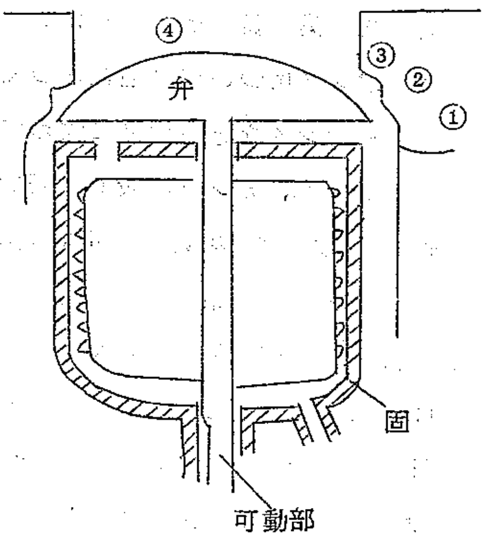
(iii) もし当社が提案するなら Rapsodie 型を提案する。

6）Rapsodie 用逆止弁について

逆止弁の構造は右図のとおり。

弁特性は下図のとおりで今までの逆止弁とは異る。





これは弁が①から可動し始め、②の位置にきたとき流量の逆流がほゞ停止する。それに伴い、④の圧力は静圧のみになるので、逆の圧力が作用することにより弁の速度が減少する。

7）Phenix 用燃料交換機および出入機について

(i) 交換機の長さは（全長）約１６ｍ、出入機は約７ｍ

(ii) 交換機の上下可動距離は約３ｍ、回転方式である。

(iii) 交換機の上部機構は取替可能。それより下部には複雑な機構は採用していない。

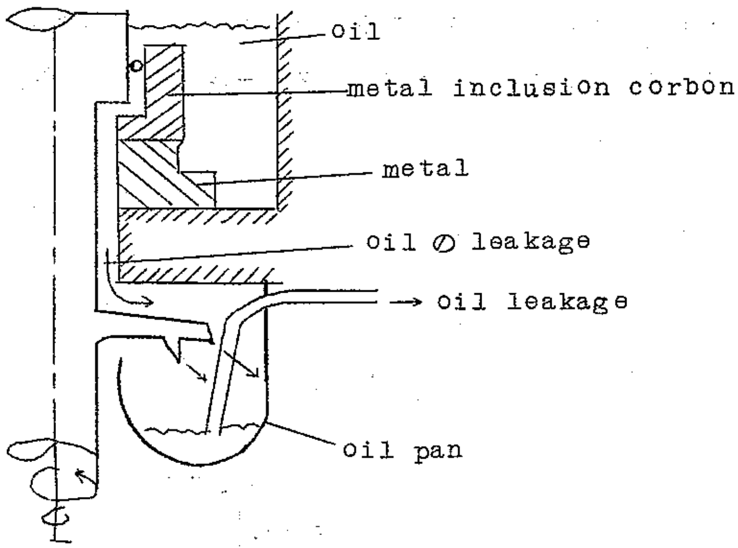
(iv) 燃料つかみ機構の材質は Ni 合金である。

日　時　　１９６９年１月１３日
場　所　　Saclay 研究所
来訪者　　WLADYSLAW　RACZYNSKI
面会者　　阿　部・井　上

1）ポンプについて

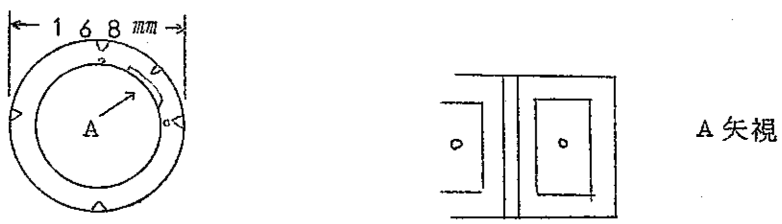
シールはメカニカルシールでOILを満している。Na ベーパによつてOILが汚されることはない。Ar ガスを循環していないからである。

メカニカルシールのコンタクト面材質は、metalic giaphite, metal inclusion corlon である。



軸シール部は大体上図のような構造で、oil は外部で closed circuite を形成しており、外側のリザーバタンクの液位が下ると oil pan に油がたまつたことを示すので、ポンプでくみ出す。

ハイドロスタテイツクベアリング



（あまり正確でない）

材料は１次ポンプステライト／ステライト　２次ポンプ　コルモノイ／コルモノイで各々材質を少しづつ変えている。

チエツクバルブ

１００％流量で圧損は、２ｍ Na である。自然循環のときはわからない。流量が少いので問題でない。自然循環のときのポンプの抵抗も問題にしなかつた。

シ　ャ　フ　ト

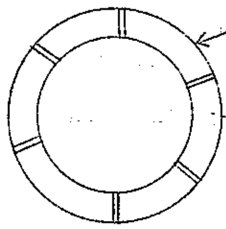
材料は、オーステナイトとマルテンサイトの mixture で１,０００℃以下の熱処理をしている。弾性限が６０㎏／mm^2程度である。オーステナイトステンレスではこのように強くない。

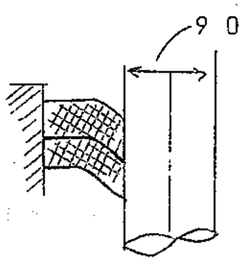
運　転　経　験　：　１６,０００ hr の運転経験がある。

２）燃料交換機のシール

３１６ステンレスのシヤフトに cast iron のスクラバーをつける。

グラフアイトアスベストのリツプシールをつける。軸はクロムメツキ





（この辺は説明者により差があり本当のことはわからない。）

10. E.D.F−4.5（ガス冷却炉）

日　時　1月8日　10：00〜16：00

場　所　Loir−et−cher

面会者　Rudeau Girard

訪問者　Baumier，Wustner，井上・尾尻

1）フランスの原子力計画

(i) フランスの電力供給計画

現在はまだ水力が主力であるが、今後水力の増加はあまり期待できず、火力、原子力による供給を増加していく予定である。1980年頃からは主に原子力を主力にしていく予定である。

1985年における予想電力供給割合は下記のとおりである。

水力　20％　石炭・ガス　15％　石油（重油）　25％

原子力　40％

(ii) 原子力開発状況

CO_2ガス冷却炉を主に水冷却炉（266MWe）なども建設しているが、EDFとしては今後もガス炉中心に開発を進める計画である。

ガス冷却炉の建設状況は下記のとおり。

①	EDF−1（chinon−1）	電気出力	70MWe	臨界	63年	6月
②	〃 −2（ 〃 −2）	〃	200 〃	〃	65〃	2〃
③	〃 −3（ 〃 −3）	〃	480 〃	〃	67〃	10〃
④	〃 −4（St.Laurent−1）	〃	480 〃	〃	69〃	1〃
⑤	〃 −5（ 〃 −2）	〃	515 〃	〃	71〃	始予定
⑥	〃 −6（Fessenheim−1）	〃	750 〃	〃	74〃	計画中
⑦	〃 −7（ 〃 −2）	〃	750 〃	〃		

2）St.Laurent−1.2の特徴

(i) 同一地点に同一型式、同一大きさ（出力は異る）のものを建設するので建設費（2号炉）が下り、建設期間（2号炉）も短縮できる。

(ii) インテグラルタイプを始めて採用した。………… 原子炉と蒸気発生器を同一Vessel内に入れる型式

(iii) Vessel にPrestressed Concreate を採用したので、生体遮蔽、安全容器を省略できた。

(iv) 完全に計算機制御にした。

3）St. Laurent 1.2の主な仕様の比較　詳細カタログ参照

	St.Laurent 1	St.Laurent 2
熱出力	1,667MWt	1,750MWt

電気出力（通常）	480MWt	515MWe
効　率	30.26%	31.36%

但し効率は最大電気出力に対する値である。

最大電気出力	St. Laurent 1	500MWe
	〃 2	530 〃
CO_2入口温度	225℃	235℃
〃 出口温度	400℃	410℃
〃 ガス圧力	26.5 Kg／cm²g	28.5 Kg／cm²g
蒸気出口温度	390℃	400℃
〃 出口圧力	35 Kg／cm²g	36.5 Kg／cm²g

4） St. Laurent 1のその他の主な仕様（詳細はカタログ参照）

臨　界	1969年1月7日
建設単価	1,100F／kwe（約8万円／kwe）
発電単価	3.1C／kwe（約2.3円／kwe）
燃　料	天然ウラン　1%Mo
総燃料装荷量	479.5M.Ton（1% Mo含む）
減速材	Graphite
燃料交換方式	運転中取替
圧力容器	Prestressed Concreate
〃 外側高さ	165ft　外径直径　95ft
内 〃	120ft　内 〃　64ft
原子炉冷却材	CO_2
タービン発電機	2　台

5） 運転について

(i) St. Laurent 1の運転員数は1直6名。

(ii) 交代は5直3交代、但し1直は日勤することもある。

(iii) 運転員に対しては食事代位で特別の手当は出していない。

11. Stein — et — Roufaix

日 時 ： 1月13日 14：30 〜 17：00

場 所 ： 本社 24，Rue Erlanger, Paris 16e

面会者 ： Poudéroux

訪問者 ： Baumier，井上，尾尻

1）会社の概要　　詳細はカタログ参照

(i) 主な製作品

a）火力用ボイラーなど　各種ボイラー

b）ゴミ燃焼処理施設

c）原子力関係機器及び配管

d）火力用石炭粉砕機

e）浴鉱炉など各種 furnace

f）空気およびガス乾燥機

g）そ の 他

上記のうちでもボイラーと furnace が主力である。

原子力関係に於いてはガス炉の冷却系をかなり受注している。高速炉に於いても早くからNa の研究を開始し Rapsodie の冷却系を担当し Phenix においても熱交換器およびループを受注しようとしている。

フランス国内では原子力特に高速炉に関しては大手メーカーの1つである。

(ii) 従 業 員　　約3,000名

2）高速炉（Na ループ関係）に関する実績

(i) Fantenay－anx－Rose 600 KWt 蒸気発生器試験ループ 1955〜1956 start

蒸気発生器蒸気条件　　30kg／cm² 400℃

(ii) Grand Quevilly 5,000 KWt 蒸気発生器試験ループ

蒸気発生器蒸気条件　設計30kg／cm²，545℃

（その後改造している）・・・・ Grand Quevilly 見学報告書参照

(iii) Cadarache 10 MWt ループ

(iv) Rapsodie の1，2次系 Na 純化系、IHX、Air Cooler

(v) Phenix 用蒸気発生器とIHXを設計し受注を待つている。蒸気発生器はほぼ stein — et — Roubaix に発注される模様

(vi) その他 Renardier 50MWtループを Alcatel, GAAA と共同で建設中（11月完成予定）。

Stein — et — Ronbaix の担当は Na の加熱器純化系、ループの設計

3) Rapsodie および Phenix のＩＨＸの概略大きさ

(i) Rapsodie　出力１０MWt　全長5,４９０㎜（細管部長さ3,２００㎜）、外径　９００㎜φ

(ii) Phenix　出力　１００MWt　全長１2,３００㎜（細管部長さ5,４００㎜）、外径　1,１７０㎜φ

4). Phenix 用ＩＨＸについて

(i) 全般には Rapsodie のＩＨＸに似ている。

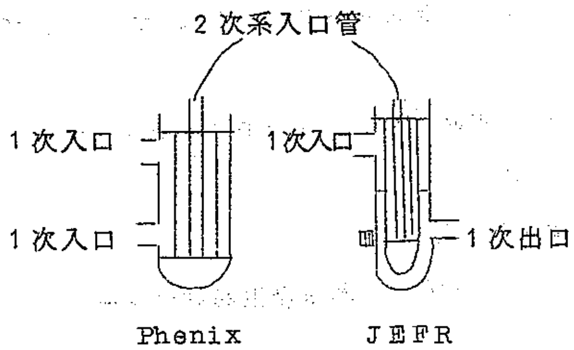
(ii) １次系の入口、出口はＪＥＦＲと同じ横から入り横へ出る形だが入口は cylindrical に入り cylindrical valve（タンクタイプの為必要）を設けてある。

(iii) ２次系の流入、流出方法は Rapsodie と同じだが入口管のまわりには N_2 部があり常時充填されている。

(iv) ２次系細管は Rapsodie，ＪＥＦＲのような N_2 による加熱部はない。

(v) Ｎa の中にある為ＪＥＦＲのような N_2 による加熱部はない。

(vi) 容器はＪＥＦＲや Rapsodie のように２重になつておらず１次系Ｎa は上から入り下に出るだけで非常に simple な為１重である。

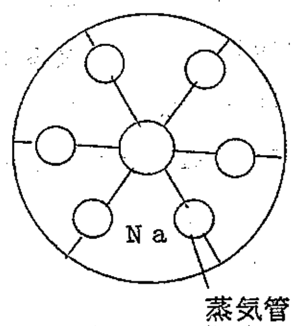


(vii) ＪＥＦＲのようなバツフアプレートはない。

5) Phenix 用蒸気発生器について

(i) 流速　Ｎa 側　２～３ m/sec　蒸気側（出口）　約３０ m/sec

(ii) 蒸気管とＮa 管は蒸気管の出入口で直接溶接によりＮa 管に固定し途中は右図のような支持プレートを入れ thermal expansion を避けられるようにしてある。



(iii) Ｎa の mixing をよくする為の所々にじやま扱のついた ring を直接外周６本の蒸気管の外側にまきつけてある。

6）ＩＨＸの価格について

(i) ＪＥＦＲのＩＨＸを発注するとすれば正確に計算しなければ言えないが５００,０００ ～ １,０００,０００Ｆ（３,６００～７,３００万円）の間に受注できると思う。

(ii) Chech and Reviewを依頼されゝば１００,０００Ｆ（７３０万円）位でチエックできると思う。

7）２次Ｎａ － Air 熱交換器について

(i) ＪＥＦＲの設計は非常にRapsodieに似ている。

(ii) １unitの容量は２０ＭＷｔもたせられるからＪＥＦＲは５～６ケで十分であると思う。

日　時 ： １９６９年１月３１日

場　所 ： Saclay研究所

来訪者 ： Pouderoux

面会者 ： Baumier，阿部，井上

1）Rapsodie　建設に果した主な役割は次の通り。

ＩＨＸ、Air － Na主冷却器、１次および２次の純化系統（コールドトラップ、バルブプラギングメータ、コールドトラップ用ＩＨＸなど）

2）Rapsodie　建設前の経験は次の通り。

キャダラシユの１０ＭＷループ ： ポンプを除く全て

すなわち、ヒータ、Ｎａ貯蔵、その他全て

グランケビの５ＭＷ Steam Generator

3）現　在

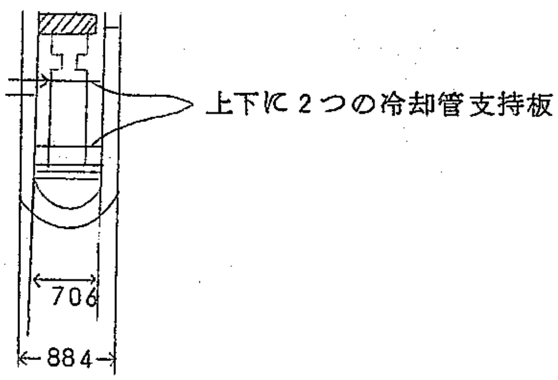
ＥＤＦの５０ＭＷループ　ル レナデールに建設

4）ＰＨＥＮＩＸ関係

ＩＨＸ を Contiact　二次系

5）中間熱交換器について

Rapsodieは３１６ステレンス製で、大略下図の如し。



この熱交換器は１０ＭＷで設計したが、２０ＭＷにすることが出来る。

ＴＥＭＡ　Standard はNa熱交には適していない。別な考え方を採る必要がある。（ＴＥＭＡ Standard ではチューブサポートの規定がある。）

水流動試験を行つたが、これは次の２つが主目的である。

(i)　流入口とチューブバンドル間の流れ分布を良くするため

(ii)　チューブの震動を調べるため

(iii)　流動抵抗を調べるため

Rapsodie の水テストの主な結果は次の通りである。

(i)　流動抵抗は acceptable であつた

(ii)　流入口に変更を加えた

(iii)　振動は、設計値と試験値が一致した。

中間熱交換器については１００ＭＷ級まで水テストをやつており５０ＭＷまでなら(ＪＥＦＲ対象)水テストをやらないで製作できる。

平行流で管外の流速は１m/sec 程度としている。Rapsodie は約0.5 m/sec　ぐらいにとつている。

RapsodieのＩＨＸは中央にドレン管と熱電対が入つている。

パツキングは、ゴムパツキングである。温度に注意する必要がある。放射線レベルは低く問題にならなかつた。

放射線しやへい体は iron ball である。これは施工が容易である。

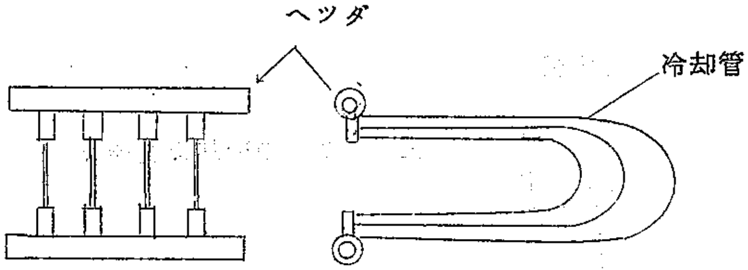
ＪＥＦＲ第二次概説のＩＨＸに対しては、①　もつと小さくできるだろう。②　流入口を大きくする必要があろう（見た感じでは）。　③　平行流がよい。

6）空気冷却器

リークして困つた経験がある。Rapsodieはこのリークの問題より、結局モジユラーした。

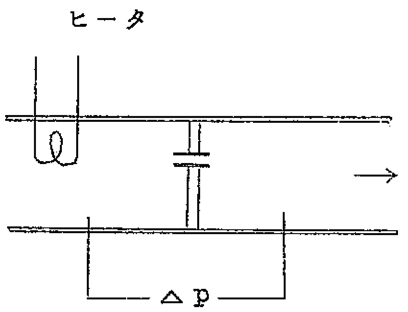
空気の流速が大きいので振動の問題もある。また熱膨張の問題もある。

空気調節用ドアが高温で問題が出た。入口（低温）、出口（高温）にドアをつけたが高温の方だけ問題が生じた。



7) プラギングインジケータ

パーシヤルプラギングを利用した。連続測定器を開発し、安定に運転している。この原理は特許をとつている。



流量　　k φ

0＜k＜1

すなわち、十分温度が高くプラギングがない場合の最高流量をφとすると、運転時は流量がkφ（0＜k＜1）になるようにヒータをコントロールする。△pは常に一定にしておく。

このようにして流れるナトリウムの温度を測り、プラギング温度を求める。kは実際には0.8近傍としている。

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3) TELEMANIPULATEUR IKN.

4) Automatismes et traitement de l'information.

5) DISPOSITIFS D'IRRADIATION POUR REACTEURS EXPERIMENTAUX.

6) Introducing alcatel.

7) Groupes de contrôle d'etanchéité á l'hélium SERIE ASM4.

8) Microdéteteur á l'hélium ASM7.

第Ⅳ部　会議，面会者，交換文書一覧

1. 会議および見学一覧表

月　日	行　　　事	場　所	出席者または面会者
10/8 〜 10/18	第Ⅰ回派遣団による説明会	Saclay	省　略 川島・能沢・井上・尾尻・斉藤
10/21 〜 10/23	第Ⅰ回派遣団 Cadarache 訪門	Cadarache	省　略 川島・能沢・井上・尾尻・斉藤
11/4	第1回定例会議	Saclay	Vautrey, Wustner, Leuandowsky, Chalot 小杉・井上・尾尻
11/6	Grand Quevilly 研究所見学		Birault, Johnsson, Lebigot 小杉・井上・尾尻
11/12	第2回定例会議	Saclay	Vautrey, Wustner, Leuandowsky, Chalot, Costes 井上・尾尻
11/16	第3回定例会議	Saclay	Vautrey, Wustner, Chalot, Costes 井上・尾尻
11/22	第1回臨時会議 （核設計について）	Saclay	Clauzon Wustner 井上・尾尻
11/25	第4回定例会議	Saclay	Vautrey, Wustner, Lewandowsky, Chalot, Abdon 井上・尾尻
11/28	核計装機器展示会見学	Saclay	Duchene 井上・尾尻
11/28	第2回臨時会議	Saclay	Bacconet, Plaige 井上・尾尻

月 日	行 事	場 所	出席者または面会者
12/2	第3回臨時会議 （安全性について）	Cadarache	Storrer, Puig, Clauzon, Ladet, Jallacle, Wustner 井上・尾尻
12/3	第4回臨時会議 （核設計・安全性について）	Cadarache	Storrer, Puig, Clauzon, Chaumont, Lacapelle, Jallade, Barré, Wustner 井上・尾尻
12/4	第5回臨時会議 （炉物理実験と計測制御について）	Cadarache	Ratier 井上・尾尻
12/6	第6回臨時会議 （第Ⅱ次派遣団の日程について）	Saclay	Vautrey, Wustner 井上・尾尻
12/6	GAAA訪問	Paris Aarny	Ertaud, Haffnerlehner, Chaminade, Escrive, Clayer, Migaud, Wustner 井上・尾尻
12/9	第5回定例会議	Saclay	MM. Leduc, Wustner, Chalot, 井上・尾尻
12/16	第6回定例会議	Saclay	MM. Leduc, Wustner, Chalot, 井上，尾尻
12/17	CEA grenoble 研究所見学（Na 施設）	Grenoble	Balligand, Konoualtschckoff, Champeix, Baqué, Millies, Martin, Costa Wustner, Leuandowski, 井上・尾尻
12/17	UGINE見学	Grenoble 工場	Brun, Jush, Graff, Besson Baqué, Champeix, Wustner 井上・尾尻

月日	行事	場所	出席者または面会者
12/18	第7回臨時会議 (原子炉本体について)	Cadarache	Leduc, Wustner, Chalot, Marmonnier 井上・尾尻
12/18	第8回臨時会議 (燃料交換系について)	Cadarache	Leduc, Benoist, Wustner, Chalot 井上・尾尻
12/19	第9回臨時会議 (冷却系について)	Cadarache	Delisle, Wustner, Chalot 井上・尾尻
12/19	第10回臨時会議 (安全性について)	Cadarache	Puig, Chalot, Wustner 井上・尾尻
12/23	第7回定例会議	Saclay	Slicewicz(午前), Mas, Leduc, (以上午後), Wustner 井上・尾尻
12/30	GACHOT (Ualve maker) 見学	Paris	Peyron, Pascaud, Renaudin, Neun Baumier, Wustner, 井上・尾尻
12/30	第8回定例会議	Saclay	Wustner 井上・尾尻
12/31	Robinetterie A.M.C. 社員来訪説明会	Saclay	Orio Baumier 井上・尾尻
1/6	Hispano-Suiza Pump maker 訪問	Paris	Raczynsky, Fouré, Baumier, Wustner, 井上・尾尻
1/6	第10回臨時会議	Saclay	Wustner, Chalot, 井上・尾尻
	第9回定例会議	Saclay	Wustner, Chalot, 井上・尾尻
1/8	E.D.F-45(ガス冷却炉) 見学	Saint Laurent	Rudeau, Girard Baumier, Wustner, 井上・尾尻

月 日	行 事	場 所	出席者または面会者
1/13	第10回定例会議	Saclay	Vautrey, Wustner, Chalot, Abdon, Daziaud 井上・尾尻
1/13	Stein et Roubaix 訪問	Paris	Poudéroux, Baumier, 井上・尾尻
1/14	第11回臨時会議 (遮蔽設計について)	Saclay	Culambourg, Wustner 井上・尾尻
1/15	第12回臨時会議 (機械設計について)	Saclay	Leduc, Wustner, 井上・尾尻
1/20	第Ⅱ次派遣団による会議(第1日) 炉心設計、冷却系統設計、計装 について	Saclay	Vautrey, Wustner, Costes, Leduc, Baumier 石川・川口・阿部・井上・尾尻
1/21	第Ⅱ次派遣団による会議(第2日) 燃料使用中検査施設、燃料交換系、原子炉機器設計 について	Saclay	Vautrey, Wustner, Mas, Leduc, Thevenot, Boulinier 石川・川口・阿部・井上・尾尻
1/22	第Ⅱ次派遣団による会議(第3日) 冷却系統設計、開発試験 について	Saclay	Vautrey, Wustner, Leduc, 石川・川口・阿部・井上・尾尻
1/23	第Ⅱ次派遣団による会議(第4日) ○計測制御設計について	Saclay	Abdon, Paziaud, Wustner, Leuandowski 石川・川口・阿部・井上・尾尻
	○遮蔽設計について		Culambourg, Wustner, Leuandowski, Clauzon, 石川・川口・阿部・井上・尾尻
	○核設計について		Clauzon, Wustner, 石川・川口・阿部・井上・尾尻

月 日	行 事	場 所	出席者または面会者
	○燃料交換系について		Boulnier, Dubois, Wustner, 石川・川口・阿部・井上・尾尻
1/24	第Ⅱ次派遣団による会議(第5日) ○核設計について	Saclay	Vautrey, Valintin, Clauzon, Wustner, 石川・川口・阿部・井上・尾尻
	○放射性排棄物処理系について		Slicewicz, Le Quinio, Wustner, Vautrey, Valintin 石川・川口・阿部・井上・尾尻
	○材料について		Vautrey, Wustner, Champaix, André, 石川・川口・阿部・井上・尾尻
	○冷却系統機器について		Baumier, Vautrey, Wustner, 石川・川口・阿部・井上・尾尻
	○一般問題討議	Saclay	Vandryès, Vautrey, Wustner, Lemoine, Baumier, Valintin, 石川・川口・阿部・井上・尾尻
1/27	第Ⅱ次派遣団による会議(第6日) ○安全性について	Saclay	Bourgeois, Slicewicz, Vautrey, Rozenholc, Puig, Costces, Leduc, Wustner, De, Vathaire, 石川・川口・阿部・井上・尾尻
1/28	第Ⅱ次派遣団 Cadarache 訪問(第1日)	Cadarache	Gajac, Wustner, Abdon, Spöri, Theucnot, Valintin, Dazimet 石川・川口・阿部・井上・尾尻
1/29	第Ⅱ次派遣団 Cadarache 訪問(第2日) I'ADAC 訪問 排棄物処理系 について 炉心設計	Cadarache	Mas, Ratier, Wustner, Ginier, 石川・川口・阿部・井上・尾尻

月 日	行　　　　事	場　所	出席者または面会者
	核設計 動特性　について 安全性		Clauzon, Lacapelle, Puig, Paziand, Wustner, 石川・川口・阿部・井上・尾尻
	HR－1，4施設見学		Spori, Wustner, 石川・川口・阿部・井上・尾尻
1/30	第Ⅱ次派遣団 Cadarache 訪問（第3日） Na－水反応	Cadarache	Lions 石川・川口・阿部・井上・尾尻
1/31	民間会社との技術討論	Saclay	
	Stein et Roubaix		Poudéroux Wustner, Baumier, 阿部・井上
	GAAA		Ramadier, Migaud, Palomo Wustner, Baumier, 阿部・井上
	Hispano－Suiza		Raczynski, Wustner, Baumier, 阿部・井上

2. 面会者一覧表

氏　　名	所　　　属	JEFR に関する担当
Abdon	D.R.P. ingénieur	Instrumentation and control
André	D.M.	Materials
Bacconet	D.E.G./S.T.R.	
Balligand	D.M/CEN Grenoble	
Baqué	Adjoint de directeur du centre/CEN Grenoble	
Barré	DRP/SETR	Nuclear design
Baumier	D.R.P. ingénieur	Guide to Fench makers for P.N.C. representatives
Benoist	S.E.M.T.R/D.R.P	Fuel handling system and fuel monitoring facility
Besson	UGINE	
Birault	S.E.M.T.R./D.R.P.	
Boulinier	D.C.P.	Fuel handling system and fuel monitoring facility
Bourgeois	Chef du D.E.P./G.T.S.P.	Safety
Brun	UGINE	
Chalot	D.R.P. détaché dans l'équipe de réalisation de Phénix	General
Chaminade	G.A.A.A.	
Champeix	D.M.	Materials
Chaumont	D.R.P./SETR	Nuclear design
Clauzon	DRP/SETR	Safety analysis and hazard evaluation, Nuclear design. Appendix.
Clayer	G.A.A.A.	
Costes	G.T.S.P.	Safety analysis and hazard evaluation

氏　　名	所　　属	JEFR に関する担当
Culambourg	Ingénieur, S.E.P.P. D.E.P., D.P.A.	Shield design
Delisle	D.R.P./S.E.M.T.R.	Cooling system
Deniélou	Adjoint au Chef du D.R.P	
de Vathaire	D.E.P./G.T.S.P.	
Dubois		
Duchêne	D.E.G., service d'Electronique des Reacteurs	Instrumentation and Control
Ertaud	G.A.A.A. directeur technique	
Escrive	G.A.A.A.	
Fouré	Hispano - Suiza	
Gajàc	D.R.P./S.C.R.	
Ginier	D.R.P./S.E.M.T.R.	Thermal design
Girard	E.D.F.	
Graff	UGINE	
Haffnerlehner	G.A.A.A.	
Jallade	D.R.P./S.E.T.R.	Dynamic analysis
Johnsson	Grand Quevilly	
Jush		
Konovaltschikoff	D.M./CEN Grenoble	
Lacapelle	Ingénieur D.R.P. S.E.T.R., CEN Caderache	Nuclear design
Ladet	D.R.P./S.E.T.R.	Safety
Lebigot	Grand Quevilly	

氏　　名	所　　属	JEFR に関する担当
Leduc	D.R.P. détaché l'équipe de realisation	Core design, Design of the reactor components, cooling system, Fuel handling system and fuel monitoring facility
Lemoine	Department des Relations Industrilles	
le Quinio	CEN Cadarache	
Levandowsky	D.R.P. Ingénieur	General
Lions	S.E.M.T.R.	Instrumentation and control
Marmonnier	D.R.P./S.E.M.T.R.	Design of the reactor components
Martin	DM/CEN Grenoble	
Mas	ADAC-Post irradiation examination Rapsodie	Fuel monitoring facility
Migaud	G.A.A.A.	
Millies	CEN Grenoble	
Neun	GACHOT	
Orio	AMC	
Palmo	G.A.A.A.	
Pascaud	GACHOT	
Paziaud	D.R.P./S.E.T.R	Instrumentation and control
Peyron	GACHOT	
Plaige	D.E.G., S.E.R.	
Poudéroux	Stein et Roubaix	
Puig	D.R.P./SETR	Safety
Ramadier	G.A.A.A.	
Ratier	SECPER, STAPu	Core design

氏　　名	所　　属	JEFR　に関する担当
Raczynsky	Hispano-Suiza	
Renaudin	GACHOT	
Rudeau	E.D.F.	
Slicewicz	S.E.S.R.	Safety analysis and hazard evaluation, the others
Spöri	D.R.P.	
Storrer	D.R.P./SETR	
Thevenot	D.C.P.	
Valantin	Service deConduite de Rapsodie/Cadarache	Cooling system
Vandryés	Chef du D.R.P.	General
Vautrey	Adjoint au Chef du D.R.P.	General
Wustner	D.R.P. ingénieur attaché à la direction du département	General

3. PNCよりCEAへの提出書類一覧表

Mon/Date	No.	Title
10/8		The Questions for the Rapsodie's Safety Rod
10/8		The Questions for the Vacuum Breaker
10/8		The Questions for the Structure of the Fuel Elements and the Fuel Assemblies
10/30	1	The Request on the Check and review and the Comments on the JEFR Design.
11/4	2	The Additional Questionnaire on the Mechanical Design of JEFR.
11/18	3	The Additional Questionnaire on the Mechanical Design of JEFR.
11/22	4	The Demand of "Fiches d'Essai" and "Comptes Rendus d'Essais".
		The Demand of the List of Codes on Safety Analysis and Dynamic Analysis.
11/25	5	The List of the Detail Planning for Usage Concerning the Instrumentation on JEFR.
11/25	6	The Additional Questions on the Design of The Fuel Element.
11/25	7	The Answers to Some Questions Asked from CEA Experts on the Meetings during the First Two Weeks at Saclay.
11/28	8	The Demand on the Decay Heat.
11/29	9	The Request on the Schedule from December 9th to 23rd.
12/2~4		The Order of the Discussion at Cadarache Center.
12/3	10	The Request on the Check Calculation on JEFR Nuclear Design.
12/10	11	The Answer on the Questions Asked From CEA Experts.
12/10	12	The Answer on the Aseismic Design Philosophy for JEFR.
12/10	13	The Answers on the Philosophy for Vessel Rupture and on the Ventilation.

Mon/Date	No.	Title
12/12	14	The Answers on the Master Schedule for Fast Breeder Reactor Project in Japan.
12/12	15	Reply to your Proposal on Nuclear Check Calculation.
12/12	16	The Questionnaire on the Slumping Effect of the Fuel and the Temperature Coefficient.
12/12	17	The Additional Questionnaire on Sampling and Analysing of Sodium and Cover Gas at Rapsodie, and on the Containment.
12/16	18	The Discussion Items of the 6th Regular Meeting On the 16th December.
12/16	19	The Proposal on the Core Upper Plugs by Japanese Company.
12/30	20	The Additional Questionnaire on the Power Calibration and the Containment Vessel of Rapsodie Plant.
1/6	21	The Answers on the Vibrational Method and the Orientation of Double Rotating Plug.
1/6	22	The Answers on the Modification of the JEFR Design.
1/6	23	The Additional Questionnaire on the Drawings JEFR 6056-1/6, C-0156 and C-1442.
1/6	24	Reply to the Preliminary Remarks in the Mechanical Design.
1/6	25	The Research and the Development Tests of JEFR.
1/6	26	The Request on the Philosophy for Vessel Rupture at the Hypothetical Accident.
1/14	27	The Answer on the Meaning of "Integrity of the Fuel Assembly".
1/14	28	The Additional Questions on the Control Rod of JEFR.
1/14	29	The Request on the Reports of "Fiche d'Essai" and "Comptes - rendus d'Essais".
1/14	30	The Request on the Organization for the Operation of Rapsodie.
1/14	31	The Answers on the Questions asked from CEA Experts.
1/20	32	The Additional Questions on the Mechanical design, Control and Instrumentation, Shield design and Safety.

4. CEA より PNC への提出書類一覧表

Mon./Date	No.	Title	
11/8	7056	Reply to Your Note No. 1	answer
1/20	7170	General Philosophy of JEFR and General Comments about Design, Construction and Operation.	comment
1/17	o	Design Philosophy of JEFR.	comment
1/17	o	Design Modifications.	comment
12/10	7101	Mechanical Design of J.E.F.R. and It's Cooling System.	comment
1/17	7163	Core Design of J.E.F.R.	answer
1/17	7165	Reactor Components,	comment
1/17	7157	Mechanical Design of the Reactor Components.	answer
2/24	o	Mechanical design.	answer
1/17	o	Answers to Note PNC No. 24	answer
1/17	o	Answers to Note PNC No. 23	answer
1/17	o	Answers to Note PNC No. 19	answer
1/17	o	Answers to Note PNC No. 11	answer
1/17	o	Answers to Note PNC No. 7	answer
1/15	7156	Cooling System	comment
"	o	" （仏文）	answer
2/24	o	" （英文）	"
1/17	7160	Development Tests （仏文）	comment
2/24		" （英文）	"
1/24	7178	Maximum Accident and Containment Philosophy	comment
1/17	7159	Relating to Dynamics and Safety	answer

Mon./Date	No.	Title	
1/17	7164	Blast Resistant Structure of JEFR and Core Meltdown and Containment.	comment answer
1/22	7176	Hazard Evaluation	answer
1/20	o	Answer to Not No. 6	answer
1/9	7132	Fuel Handling System	comment
1/17		Fuel Handling System and Fuel Monitoring Facility （仏文）	answer
2/24		〃 （英文）	〃
1/9	7158	Dynamics Calculations	comment
12/11	7106	Fuel Monitoring Facility of JEFR.	comment
1/17	o	Consulation for Each Part of Design	answer
2/24	o	Failed fuel detection system	comment
1/9	7140	Nuclear Design of JEFR.	comment answer
12/11	o	Rapsodie 照射期間中の反応度変化に関する検討	
12/11	o	核設計の出力係数および温度系数の計算値一覧表	
12/11	o	安全解析上必要なコード	
12/11		Hot spot factor	comment answer.
11/25	o	Instrumentation and Control	comment
1/9	7134	Instrumentation and Control （仏文）	answer
2/24	o	〃 （英文）	〃

Mon./Date	No.	Title	
1/17	o	Remarks on the Instrumentation and Control	comment
1/17	o	Instrumentation and Control Shielding.	answer
12/13	7112	Shielding Design	answer
1/30	o	Answers to Note No. 33 (Control)	answer
1/17	7067	Mechanical Design of JEFR	answer
1/21	7172	Hexagonal Wrapper Tubes of the Sub-assemblies.	comment
1/27	7180	Wigner Effect in Borated Graphite and Domage Detection in Shielding Materials.	comment
1/17	o	Answer to Note No. 33 (Safety)	answer
2/24	o	Rapsodie reactor and its mock-up	comment

あ と が き

高速実験炉のフランス原子力公社による検討評価に際し、日本原子力研究所動力炉開発管理室の能沢正雄氏をはじめ高速炉設計班の諸氏により寄せられた技術的協力に対し厚く感謝の意を表する。またフランス原子力公社との交渉およびその後の推進および連絡に当られた倉本昌昭計画管理部長および同部の元田謙氏に感謝する。さらに高速炉開発本部の大山技術相談役、市野部長はじめ各研究開発グループの諸氏の御指導援助に対し感謝する。