UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 13, 1979

Mr. Edward J. Hanrahan
Director, Nuclear Alternative
Systems Assessment
U.S. Department of Energy
Washington, D.C. 20545

Dear Mr. Hanrahan:

Staff members within the NRC licensing and research offices have completed their review of the NASAP reports that you submitted to us. The results of their review are summarized in the attachments -- first in terms of General Comments, then in terms of selected Specific Comments and finally in terms of detailed comments on a docu ent-by-document basis.

If, after going over our comments, you o your staff need to discuss any points further with the NRC individuals who prepared the review, please let me know.

Sincerely,

Norman M. Haller

Norman M. Haller, Director
Office of Management and
Program Analysis

Enclosure as stated

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527 167

NRC Staff Member[1/] Comments on

DOE NASAP Concepts

Enclosed are comments and questions on the NRC Staff's review of the seven NASAP Preliminary Safety and Environmental Information Documents (PSEID's), as well as a number of backup documents, submitted to NRC by DOE. (See Exhibit 1) Meetings have been held between the NRC Staff and DOE contractors on the HWR (11/6/78); the LWR-variants (11/7/78); and the GCFR (2/26/79); the HTGR (2/27/79); and the LWBR (3/20/79). (We decided to delay a meeting on the LMFBR until we received the carbide fuels amendment.) Our comments and questions are then based on our review of the PSEID's and some of their references, the information provided to the staff at the DOE contractor meetings, and the staff's expertise and previous experience accumulated over the past several years in reviewing Fort St. Vrain (FSV), the large HTGR plans (including GASSAR), the GCFR conceptual design, CRBRP and FFTF, Shippingport and the CE System 80 LWR design.

Specific NRC comments on the seven PSEID volumes identified in Exhibit 1 are provided in Attachment 1. Also in Attachment 1, integrated with the comments on the PSEID's, are NRC comments on supporting documentation identified as Reports 3, 4, 5, and 6 in Exhibit 1. Attachment 1 was produced by NRC's Office of Nuclear Reactor Regulation (NRR) with inputs from the Office of Nuclear Reactor Research (RES) and the Office of Nuclear Material Safety and Safeguards (NMSS). NRC comments on the safeguards aspects of the reactor and fuel cycle concepts discussed in the PSEID's are provided in Attachment 2. These comments were developed by NMSS's Division of Safeguards. Attachment 3 contains comments on the document, Nuclear Energy System Characterization Data (Report 2 in Exhibit 1), prepared by NMSS. Attachment 4 contains comments on the DOE document, Safeguards Systems for NASAP Alternative Fuel Cycles (Report 7 in Exhibit 1), which was prepared by the Division of Safeguards. Finally, in Attachment 5, the Division of Fuel Cycle and Material Safety of NMSS has provided comments on ORNL-5388, Interim Assessment on the Denatured ^{233}U Fuel Cycle: Feasibility and Nonproliferation Characteristics (Report 8 in Exhibit 1).

Our comments on DOE NASAP documents have been delayed as a result of NRC's increased Three Mile Island responsibilities, and future NRC resources and schedules for NASAP are likely to be adversely affected in the aftermath of the TMI accident.

527 168

General Comments

It should be pointed out that the information provided in the PSEID submittals is at best uneven; there are large differences in the details of reactor descriptions in comparing one conceptual design to another. For example,

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1/ This review was conducted by individual staff members in NRC's licensing and research offices. The review does not carry the weight of full office coordination or full NRC staff concurrence, and it has not been provided to the NRC Commissioners.

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there are considerable detailed descriptions in the supporting documents of the Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) for the HWR and LWR variants but virtually nothing but a core description for the variety of LMFBR conceptual designs. This disparity will make any comparative evaluations extremely difficult, to say the least.

Also, the lack of information and discussions in key areas such as GDC's[2/], the adequacy of containment systems, ECCS, and decay heat removal systems, for some of the NASAP concepts makes it impossible to arrive at judgments and make comparative evaluations. We, of course, realize the difficulty that DOE faces in submitting consistent packages when the development status of these reactor concepts varies dramatically. The lack of information should be noted as a problem, however. Some of our questions are attempting to deal with this "uneven" type of situation.

It should be noted that, as part of LMFBR (Volume VI) comments, we have included a section entitled "The Basis for Containment Design in LMFBR's" that has been put together by the licensing staff in order to describe staff positions regarding the three classes of accidents that have in the past been considered in the containment evaluations of CRBRP, FFTF, and FNP.

Finally, we would like to point out that because of the large number of variations of the six reactor concepts (both fuel cycle and design variations) we are becoming somewhat overwhelmed. It would be helpful, for example, if DOE could provide some guidance as to which variations are favored (from DOE's overall point of view) over others. For example: (1) GCFR upflow vs downflow; (2) HTGR direct cycle (gas turbine) vs steam cycle; (3) LWBR as a breeder or high efficiency converter; (4) LWBR prebreeder backfit into existing LWR or the first phase of a separate LWBR design; or (5) one of the present LMFBR variations relative to the other 15 or more variations.

Specific Comments

Detailed specific comments are provided in the attachments, as indicated above. Some of the major comments are highlighted below; however, the complete list of

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2/ Conformance to the spirit of the General Design Criteria (GDC) is key in considering the licensing of any new concept. Information was lacking in the following areas: (a) [illegible] of with which of the GDC for Nuclear Power Plants provided in 10 CFR 50, Appendix A, you intend to comply; (b) discussion [illegible] fuel type and reactor design meets the spirit and int[illegible] and (c) justification for those GDC that you feel eith[illegible] or only apply partially, including substitution of criteria wh[illegible] y[illegible] consider are more appropriate for each specific reactor type.

527 169

POOR ORIGINAL

detailed comments should be examined to provide the full context for these comments.

. In overview, for the spiked recycle LWR concept (PSEID Volume 1), the benefits obtained by this concept should be balanced with the economic, environmental, social, etc., costs that are incurred and this should be compared with similar C/B analyses for other alternative fuel cycle concepts.

. With respect to HWRs, the assurance of long-term reliability of the primary pressure boundary depends on a number of aspects about which the staff has significant concerns. Among them are: the leak-before-break arguments based on linear elastic fracture mechanics; the requirements for extremely reliable in-service inspection and leak detection; and the evaluation of the probability of failure propagation due to pipe whip and missiles.

. For the high-temperature gas-cooled reactor, what additional features of the plant protection system or engineered safety features may be needed to cope with failure modes of the gray control rods, the turbine-compressor unit, primary system valve, the recuperator, hot duct, and the precooler? Responses to this question will require identification of or reference to failure mode studies, postulation of a spectrum of accidents, predicted responses of the existing plant protection system and engineered safety features, and information on potential system interactions. NRC anticipates that it may not be possible for DOE to supply definitive responses to this question in the near future. Nevertheless, we expect that you should be able to supply preliminary and conceptual responses together with a discussion of the status of related accident studies and any estimate of when this question can be finally answered.

• While the PSEID concerning fuel cycle facilities (Volume VII) identifies systems and principal issues related to nonproliferation alternatives, it does not assess the proposed systems or facilities in sufficient depth to permit definition of appropriate licensing criteria or the potential difficulty of meeting such criteria.

• The fuel cycle facilities PSEID does not make a quantitative comparison of safety or environmental trade offs in areas such as occupational exposure, regional exposure, accident risk or environmental impacts. NRC believes that such comparisons will have to be developed as part of the NASAP final evaluations, even though they are not given in this draft document.

• With regard to reprocessing, Sections 5.10, 5.11, and 5.12 of Volume VII present fuel cycle variations that were not considered in Volumes I-VI covering reactor concepts. Accordingly, NRC believes that these variations should be either related to a specific reactor concept or deleted.

- NRC believes that the volume, chemical and physical form, and isotopic activities in low level wastes from each operation (excluding mining) of all fuel cycles should be estimated in Volume VII to permit an evaluation of the overall impacts of low level waste disposal. (Data on low level waste from reactors using the particular fuel cycle should also be given in Volume VII.)

- The PSEIDs do not adequately address safeguards concepts, systems design, operations, or issues.

- A major portion of the document, Nuclear Energy System Characterization Data, is devoted to presentation of numerical data. NRC comments are directed generally to the rather limited amount of analysis presented in the document and the lack of consideration of the objectives of improving proliferation resistance and making use of available resources.

- Since the objective of the NASAP work is to identify ways of improving proliferation resistance, it would appear that some of the cases should have been designed to minimize stockpiles of fissile materials and thus be responsive to this proliferation resistance concern. There has been apparently no effort in this regard and this appears to be a key weakness of the Nuclear Energy System Characterization Data document.

- With regard to the concept of adding radioactivity to new reactor fuel as specified in the document, Safeguards Systems for NASAP Alterative Fuel Cycles, DOE does not acknowledge or address the increased difficulties in the routine handling of such material or the increased potential for a public health hazard in the event of a transporation accident with the increased radioactivity. A cost-benefit analysis should be performed that accounts for these costs.

- Regarding safeguards, the final version of NRC's proposed physical protection upgrade rule for Category I material is awaiting final Commission review and consideration. This proposed rule is moving closer to being published in effective form and, together with existing regulations, should provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base could be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

Detailed specific comments are provided in the following attachments.

~~527 172~~

EXHIBIT 1

NASAP REPORTS
REVIEWED BY NRC

1. Preliminary Safety and Environmental Information Documents

   Volume I (Revision 1) - Light-Water Reactors
   Volume II (Revision 1) - Heavy-Water Reactor
   Volume III (Revision 1) - Light-Water Breeder Reactors
   Volume IV (Revision 1) - High-Temperature Gas-Cooled Reactors
   Volume V (Revision 1) - Gas-Cooled Fast Reactor
   Volume VI (Revision 1) - Liquid-Metal Fast Breeder Reactors
   Volume VII (Revision 1) - Fuel Cycle Facilities

2. Nuclear Energy System Characterization Data

3. Summary Report - Preconceptual Study of 1000-MWe Carbide-Fueled LMFBR Designs

4. Homogeneous Carbide Fueled Cores for the Proliferation Resistant LMFBR Core Design Study

5. Preconceptual Design Study of Proliferation-Resistant Homogenous Oxide LMFBR Cores

6. Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core Final Report, Volumes 1 and 2

7. Safeguards Systems for NASAP Alternative Fuel Cycles

8. ORNL-5388 - Interim Assessment of the Denatured ^{233}U Fuel Cycle: Feasibility and Nonproliferation Characteristics

12

ATTACHMENT 1

SPECIFIC NRC STAFF COMMENTS

NASAP PRELIMINARY SAFETY
AND ENVIRONMENTAL INFORMATION DOCUMENTS

527 173

NASAP PSEID - VOLUME 1
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF NASAP LWR-VARIANTS

A. EXTENDED BURNUP

1. When there is a request for a license to permit extended burnup to 50,000 MWD/MT, the applicant will, of course, have to satisfy the criteria established in the Standard Review Plan, in particular, for Fuel System Design. A considerable portion of the Standard Review Plan is concerned with the analyses and assessment of transients and accidents. In the LWR PSEID (Vol. 1) and supporting documentation we see little evidence of a comprehensive and systematic program to address these areas. As we understand it, the bulk of the experimental effort in the area of extended burnup is in the area of "normal operation" irradiation of lead assembles to 50,000 MWD/MT, while little, if any, is in the area of transient behavior. To what extent does the DOE R&D plan for extended burnup include transient testing of high burnup fuel pins? Include in your discussion the type of testing planned, the schedule, and the facilities to be used (e.g., PBF).

2. The PSEID, Vol. I, presented little specific information on the various design changes necessary to accommodate the increased fission gas inventory for the high burnup option (to 50,000 MWD/MT). At the meeting with CE on 11/7/78, various possibilities were presented including change in fuel rod length, and/or change in fuel column length for solid, hollow, and duplex pellets. Has DOE been able to narrow down these possibilities and arrived at a best option for accommodating the fission gas pressure problem?

3. In addition to the NASAP program at CE, there are other reactor manufacturers who have extended burnup studies in place (e.g., B&W, W, EXXON and GE). How do these other programs complement, if at all, the CE program directed to

527 174

better fuel utilization? Are there unique features of any of these programs that should be taken into account in an overall licensing/safety evaluation of extended burn up cores?

4. Provide a complete list of the non-saturating fission products produced for a 50,000 MWD/MT burnup and their activities and compare to a typical 30,000 MWD/MT cycle. Also provide a decay heat curve for extended burnup.

5. Provide any analyses or assessments of power peaking due to the increased U-235 enrichment necessary for the LWR extended burnup option.

6. The present enrichment limit for fuel handling and storage at PWR plants is [illegible]. What approach does DOE intend to take in these areas in light of the increased enrichment (4.3%) for extended burnup cores.

7. Provide any analyses or assessments of the change in shutdown margin in going from 30,000 to 50,000 MWD/MT.

8. (Page 2-50) Consideration could be given to listing the 17 unresolved safety issues -- as an aid in identifying safety issues for the various concepts.

9. (Page 2-51) This section could be updated to include the seismic structural and mechanical research categories in RSR.

B. <u>SPIKED RECYCLE</u>

The flow sheet for the fuel cycle of a PWR using 3-5% LEU with Pu recycle and Cobalt-60 spiking shows two Purex reprocessing operations. In this arrangement fresh uranium fuel is reprocessed in Purex 1 and mixed oxide fuel is reprocessed in Purex 2. This plan is difficult to understand and leads to the following questions:

527 175

1. From a proliferaton standpoint, why is it acceptable to recover about 40% of the plutonium as pure plutonium, while the remainder is recovered as coprocessed 2% in uranium?

2. What is the purpose or intent of the two Purex operations? Are they designed to optimize recycle of uranium? Do the two Purex operations represent the same solvent extraction line, with fuel being compaigned, or are the operations carried out in physically distinct equipment?

   Some discussion of the purpose of these two Purex lines is required, together with an indication of the incremental reprocessing costs relative to a single Purex operation.

   Does the flow sheet for both PWR's and BWR's? If not, what is the plan for BWR units?

3. The use of cobalt-60 represents the addition of a spikant to the presently conceived recycle flow sheets. In developing a generic environmental assessment of a fuel cycle, major impacts of producing and disposing of all the fuel cycle material should be included. In the case of the Co-60 feed material, the assessment should include all of the operations involved to produce the radioactive cobalt, including the reactors and cobalt processing facilities.

527 176

A cobalt-60 balance across each of the fuel cycle operations should be given (i.e., input, amount to waste, amount release). In addition, the behavior of the soluble cobalt in the recycle fuel fabrication operations (preparation of oxide, sintering) should be given.

Further, the use of Co-60 should be analyzed in light of its effects on operations such as fuel transportation, fuel fabrication, reactor operations, reprocessing, etc.

In addition, the level of occupational exposure in the overall handling and use of recycle fuel cycle should be assessed and the potential effects on population exposures should be considered.

4. In overview, the benefits obtained by this concept should balance with the economic, environmental, social, etc. costs that are incurred and it should be compared with similar C/B analyses for other alternative fuel cycles.

C. DENATURED URANIUM RECYCLE

1. Chapter 5 outlines the concept of a PWR using uranium fuel enriched with 12% U-233 and mixed with thorium oxide. The flow sheet for this denatured U-233/thorium cycle (PWR DU(3)-Th recycle DU(3)) does not appear to be a self-standing or independent one. The flow sheet and Reactor Charge Data show that U-233 are required for sustained operations. The source of the U-233 supply is not mentioned or described.

527 177

The fuel cycle shown requires at least two "Secure" centers - one for 50% U-233 which is denatured to 12% U-233 during fuel fabrication, and another for storage of spiked plutonium which is recovered from this fuel cycle.

Substantial additional information on the flow sheet is required for its assessment, such as:

a. What is the source of U-233 supply? [illegible] must be supplied on t[illegible] reactor cycle that produces U-233 so that environmental, safety and safeguards impacts of that production can be given.

b. What are the definitions of "secure" storage center for U-233 and a "secure" storage center for plutonium? What are the fuel fabrication and reprocessing facilities not considered to require "secure" status?

c. Additionally, data on the cobalt-60 spike must be given. What is the plan for sale of Pu? Who is the customer and what fuel cycle is it to be employed in? How is the problem of the relatively short half life of Co-60 (7 years) handled? What is the form of plutonium in storage?

d. Is the flow sheet valid for BWR's as well as PWR's?

e. Detailed information on gaseous effluents from Thorex fuel reprocessing must be provided.

f. What are the fuel cycle economics? How many reactors are required to justify reprocessing for this cycle? How many reactors must be used to produce the U-233 and Co-60 used in this cycle?

2. It appears that the first licensing issues to be addressed may be those concerned with the known physical and chemical property differences between

thorium and uranium, and the physics behavior of U-233 as opposed to that of U-235. Any modifications in behavior or component design introduced as a result of these initial considerations must then be examined for any effects they might have on the previusly licensed reactor and plant features. The initial evaluation would be assisted by an expanded discussion of the following questions.

a. Fuel Qualification. A comprehensive picture of fuel behavior, growth, densification, fission product migration, transient fuel damage limits, and other safety-related fuel performance information (Section 5.5, final paragraph) is required for qualification in a large, high performance reactor. To what extend can this information be obtained in Shippingport or other scheduled tests? Are other fuel development programs visualized? Will there be transient experiments or simulated accident conditions to examine the range of capabilities of this fuel?

b. How extensive in nature is the physics verification program projected to be as a requirement for licensing?

c. According to information presented at the November 7 meeting at Windsor, CT., the composition of the core supplying the U-233 may undergo considerable variation over its lifetime. If this is correct, it would present a problem in the licensing of a fairly wide range of core compositions. What ranges of core compositions (chemical and isotopic) are anticipated for the various prebreeding options and what arrangements would be undertaken for core qualification over these ranges?

D. General Comment for all LWR-variants

Noting pages 3-26, 4-21 and 5-18, some cost and time-magnitude data needs to be identified for each on a consistent basis to provide the comparability required.

NASAP PSEID - VOLUME 2
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF NASAP HWR

We have reviewed the HWR PSEID and have formulated the following questions. We also wish to call to your attention the attached letter from Brookhaven National Laboratory which was prepared following a preliminary review of the HWR PSEID and which provides amplification of the concerns expressed in the first nine items.

1. Natural Convection. Although the CE design may not be sufficiently detailed to assess its potential for natural circulation decay heat removal, are there specific design steps that could be taken to augment natural circulation? In view of the possibility of steam bubbles in the horizontal pressure tubes, are there reasons (experiments) to believe that natural circulation would not be inherently ineffective in this type of reactor?

2. Primary heat transport system. We share the concern expressed by BNL that the two primary loops are connected at a common pressurizer, though they are otherwise independent. Because of the reliance placed on isolation of these loops to maintain a loss-of-coolant reactivity less than one dollar, discussion along the lines suggested by BNL would be useful. How reliable are the pressurizer isolation valves against improper activitation?

3. Moderator cooling. BNL's reference to the PWR plena as heat sinks is well taken although the calandria vessel is certainly much larger and cooler. We also note in Nuclear Engineering International, January 1979, the article by J. T. Rogers describing calculated heat transfer to the moderator by use of the codes IMPECC and CONCYL. Are you aware of any experimental verifications of these codes, or do you think such verification would be feasible? Would it be a suitable subject for future research?

527 181

4. A Loss-of-Heat Sink Scenario. What analysis or experiments cover a two-phase flow situation in the horizontal tube geometry? What would be the effect of bubbles on natural convection circulation?

5. Common Mode Heat Removal Failure at Headers. Are there break locations such as that suggested by BNL where the :S would fail to cool a substantial part of the core? How successfully does flow reversal work if the ECCS must be switched to the outlet header?

6. Outstanding Questions to the Applicant. We agree that discussions of the questions raised by BNL would provide useful input to the licensability evaluation.

7. Temperature and Void Coefficients. BNL's calculation of temperature coefficients is admittedly very preliminary. Nevertheless, the indicated trends should be pursued further.

The temperature coefficients apparently become more positive or less negat as burnup proceeds. The moderator temperature coefficient appears to be ghtly positive at equilibrium burnup. We are told that the trend of these coefficients with burnup corroborates Canadian calculations. What is the effect of these positive coefficients on kinetic behavior at power? Are there any instabilities?

Although we do not necessarily endorse the view of BNL that "there appears to be entirely too much reactivity associated with voiding one loop ..." we are interested to know if alternatives to the two-loop design have been considered from the physics point of view. The designers have rejected the feasibility of dividing the core into more than two independent loops on the basis of capital cost. What are the maximum period limitations to offset the capital costs and how much cost is involved?

8. Xenon Oscillations. It appears that this problem is being addressed in the CE design. Are allowances being made for abnormal behavior and for the increased complexity of the larger system?

9. Neutron Behavior Associated with LOCA. Has neutron streaming in voided or partially voided horizontal tubes been estimated? If the upper 1/3 to 1/2 of a tube is voided, does this provide a direct path for well thermalized neutrons to reach the center fuel rods which normally do not see as much thermal flux? What effects would this have on reactivity coefficients?

In addition, the NRC staff has focused on a review of the materials and inspection requirements for the primary system boundary. Questions on this topic plus several general questions about the PSEID comprise the following group.

Pages 2-9, 2-10 Section 2.2.2

Pages 2-46, 2-47 Section 2.4.2.1

527 183

1. References 15 and 18 are 7 years old. Is more data available regarding irradiated properties of Zirconium 2 1/2 Niobium? (Fracture toughness)

2. Since the pressure tube is part of the reactor coolant boundary it would be desirable to present both the material and tube joint as an approved ASME Code Case for a licensed reactor. Assuming that the only data available is that already in the public domain, what is your estimate of the time and resources required to get a code case ruling on the material and the tube joint?

3. Discuss why you believe linear elastic fracture mechanics is an appropriate tool for assessment of the leak-before-break possibility, which may be dominated by an agressive environment such as stress corrosion or radiation damage.

4. Postulating that the leak-before-break hypothesis can be satisfactorily demonstrated, and that sufficient time exists for leak detection and reactor shutdown before a self-propagating crack develops, discuss whether the leak detection system should be considered a safety grade system, Seismic I, Single failure, etc.

5. Can reasonable assurances be given, a priori, that the risk associated with rupture of the primary coolant boundary of the HWR is comparable to the risk of pressure vessel failure for an LWR? What in-service inspection procedures, parallel materials research studies, and engineered safety features are included in the program to assure that the risk of pressure tube failure plus failure propagation in the HWR is as low as that of pressure vessel failure in the LWR?

We believe that it would be helpful in making a judgment of licensability if the following subjects were addressed:

2.4.2.2 Expand to give scope of in-service inspection program and materials research. We are not persuaded that questions of failure propagation due to pipe whip and missiles have been satisfactorily resolved, especially at the embrittled material conditions at end of life.

2.2.2 Can the entire pressure tube be inspected for cracks without unloading the fuel or just the region near the rolled joint? If the moisture detection of leaks-before-breaks is determined to be not sufficiently reliable, what frequency of direct UT or acoustic inspection for cracks would be necessary as a supplement? Is it feasible to perform such inspections with this frequency?

6. Have experiments or analysis been performed with respect to jet impingement or tube whip against the callandria tube and if so what conclusions were reached?

7. If collectively the pressure tubes are to have an equivalent reliability as a BWR or PWR Vessel then even greater reliability of the individual pressure tubes is required in the Heavy Water Reactor. What is this estimated greater reliability and is it demonstrable?

8. Discuss how comparability with Appendix K would be demonstrated with equivalent margins of safety. What R&D may be needed to show comparable safety with respect to blowdown, metal water reaction, reflood and PAHR?

9. 2.2.2 P. 2-9 An amplified discussion of the means of protection against failure of the automatic control systems is required.

10. 2.2.3 P. 2-11. The statement is made that "A serious fault in the process system is defined as one that would, in the absence of safety systems, result in a substantial release of radioactive material to the environment." Later on the same page, under item 2, the statement is made that "a serious fa. re is one that in the absence of protective action would lead to serious fuel failure."

Are these statements in conflict? In connection with items 5 and 6 on the same page, please note the US licensing will require conformance to applicable sections of US Code and Regulatory Guides in regard to acceptable levels of effluents from normal operation and accidents.

11. P. 2-16. The NRC would be inclined to continue the use of the source term defined in Reg. Guide 1.3 unless inherent differences between LWRs and HWRs provide a substantial basis for expecting considerably different accident behavior in the HWR. In this connection we would consider the burnup, gap pressure, clad design, ECC temperatures and any other notable differences between the reactor systems. If these can be shown to effect considerable reduction in the source term with a high degree of assurance, consideration would be given to appropriate modifications of the source.

    The Canadian practice, as described here, appears to be similar to the more realistic calculations of the source term, as done in WASH-1400, rather than the conservative calculations that the US licensing procedure uses. It would be inconsistent with our review of LWRs to calculate the HWR source term in this way without first showing major differences in the scenarios.

    Please submit any such discussion of major scenario differences that you believe to be relevant.

12. 2.2.4 Is it justifiable to assume that the safety analyses, if carried out, would lead to results comparable to light water reactors? In what areas do you foresee major differences?

527 186

13. 2.3.6.1 Operation with the CANDU fuel design except for higher enrichment with higher burnup is suggested to lead to higher rates of fuel failure than the Canadians have experienced in the past. What test data are available on failure rates at high burnup? What steps will be taken in design, fabrication and operation to keep this failure rate acceptable? What level of contamination of the primary coolant system is expected from various failure rates and how is this controlled?

As you know, the currently used General Design Criteria rule out designs that include fuel failure as a normal occurrence. Reconsideration of this position would be expedited by any information you might develop regarding the effect of routine fuel failures on subsequent accident consequences, such as might occur by way of containmination of primary coolant. Even if this criterion were reconsidered, it would seem reasonable to require that a predicted failure rate be low enough so that one damaged fuel element could be expected to be removed before the problem was compounded by additional failures.

14. 2.3.6.4 Why is the volume of housekeeping-type low level waste expected to be so much smaller for the HWR than for the LWR?

15. 2.4.1 We are not prepared at this stage to agree or disagree with the statement "... in recognition of the fact that the CANDU reactor is considered to be at least as safe as the LWR ...."

16. 2.4.3 Should not the monitoring and control of hydrogen be regarded as a subject to be included with Safety System Research? If not, please expand Section 1.2.8.8 to provide details of description and capacity of equipment and sensors.

17. 1.1 P. 1-4 Is there diversity in the in-core sensors for the "two diverse reactor shutdown systems?"

18. The potential for a small LOCA due to on-line refueling malfunction, particularly resulting from a seismic event, should be addressed.

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April 13, 1979

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Dear Jim:

As per your request on April 4, 1979, the BNL staff has prepared a statement of its comments and concerns in relation to potential licensability issues for the NASAP Heavy Water Reactor (HWR). These comments are based mostly on the information contained in Chapters 4 and 5 of the preliminary design document that you transmitted to me, Volume II of the NASAP PSEID, and selected documents on the CANDU reactor. We have also utilized information obtained during our meeting at Combustion Engineering in November 1978 and during our follow-on telephone conversation with Mr. Fred Jesick of Combustion Engineering in March 1979.

As requested by you, our comments focus mainly on plant systems dynamics (in particular, decay heat removal capability) and reactor physics (in particular, reactivity coefficients and transient stability). However, some additional comments in related areas are provided as well.

1) Natural Circulation

In the event that all forced shutdown cooling capability is lost in the HWR, it is claimed (p. 59 and p. 266 of the preliminary design report) that heat removal from the primary system via natural circulation will suffice. Although the steam generators are positioned above the core, the fact that the pressure tubes are horizontal raises some obvious questions. Thermal buoyancy effects originating within the tubes will be in a direction orthogonal to the desired direction of the flow. Any steam generated within the tubes during a transient will tend to flow upward toward the top of the tubes and may stagnate there due to the lack of sufficient flow to overcome two-phase frictional resistance. On the other hand, in a pressurized light water reactor with a vertical core

527 189

arrangement, the thermal buoyancy of the steam will tend to promote its removal from the core region. One must therefore conclude that with respect to this circumstance, the potential for dryout is greater in the HWR than it is in the PWR. Further remarks on the behavior of bubbles in the primary system are discussed below.

It would, of course, be of interest to learn of Canadian experience with respect to natural circulation in the CANDU reactors. We spoke to C-E about this and apparently the Canadians claim that the CANDU reactor has a natural circulation capability. However, we were not able to receive information on documentation which would substantiate this apparent claim. Fred Jesick (of C-E) also noted to us that the NASAP HWR design is not sufficiently detailed to assess its potential for natural circulation decay heat removal capability.

2) <u>Primary Heat Transport System</u>

The primary heat transport system is a two-loop design with two pumps, two steam generators, two inlet headers and two outlet headers on each loop. According to Fig. 5.1.3-3 of the preliminary design report and our conversation with Fred Jesick, the two loops are connected at a location common to two outlet headers (one from each loop) and the coolant system pressure is controlled by one pressurizer which is also common to the two coolant loops at this location.

If a LOCA occurs on one loop, then it is possible, via valves provided at the common location to isolate the damaged loop from the intact loop such that the pressurizer is connected only to the intact loop (or isolated entirely) and the damaged loop is then valved to ECCS operation. Isolation of the pressurizer from the intact loop would affect system pressure control in that loop and, therefore, would not be recommended by us.

If a loss-of-heat-sink event occurred in the secondary coolant system such that the initially intact primary heat transport system became effectively adiabatic and system temperature and pressure began to rise (in both loops), then it is expected that the pressurizer relief valve would open and the pressure would be relieved. If this relief valve failed to re-close after the primary system pressure was reduced to a safe level, then the accident becomes a LOCA via the opened pressurizer valve. An obviously important distinction between the course of events for this hypothetical accident and the accident which occurred recently for a pressurized light water reactor is that the loss of coolant in the HWR is associated with a positive void reactivity feedback coefficient.

527 190

A LOCA at the pressurizer is particularly important since the complete loss of coolant from both loops (but not from a single loop) results in a reactivity insertion greater than one dollar.

Because of the presence of the loop isolation valves, certain variations of the above scenario become possible. For example, if, due to the observation of this LOCA, it is decided that (for whatever reason) one of the loops should be isolated from the pressurizer and the other loop, then, due to the continued presence of the loss-of-heat-sink condition, the isolated loop could overpressurize and be breached in a manner which would compromise coolability via that loop.

If a loss-of-flow event occurs in one primary loop, it should be noted that, since the loops are in common only at two outlet headers, the potential for providing forced circulation via the other primary loop appears to be small.

3) Moderator Cooling

The moderator cooling system of the HWR provides a heat sink which is not available in the pressurized light water reactor. Even if flow is not available in the moderator system, it may function as a passive heat sink following a loss-of-heat-sink accident as described above. However, the efficacy of the moderator system as a heat removal path under a spectrum of conditions cannot be evaluated by us at this time due to a lack of sufficient design information. A comparison of the HWR and PWR in this regard should recognize the existence of upper and lower plena in the PWR as additional heat sinks not available in the HWR. The process of uncovering the core via steam production is quite different for the two designs and the analysis of available heat sinks must be analyzed with care.

4) A Loss of Heat Sink Scenario

By considering failure in the secondary coolant system similar to that which occurred at the Three Mile Island plant on March 28, 1979, the possible situation that may exist in the primary loop of the NASAP HWR is discussed as follows. The discussion is based on the information included in Chapter 5 of the HWR preliminary design report.

If the primary loops are overheated and intensive boiling causes bubbles to form in the pressure tubes, these bubbl[illegible] cannot be removed from the core (pressure tube) as easily as in the [illegible] system where bubbles are carried upward by thermal buoyancy. The bubbles would either stay in or flow through pressure tubes which may aggravate the heat transfer from the cladding and enhance the temperature increase. Because of the structure and the layout of the inlet and the outlet headers, it is not likely that large bubbles would be formed there. Bubbles entering the outlet header/inlet header would be expected to enter into the loop/-pressure tubes through the hot leg/cold leg of the steam generator. It is also not expected that bubbles downstream of the outlet headers would enter the pressurizer (and be released) any easier than in the PWR system.

It is believed that, in the HWR, if there are intensive bubbles formed, these bubbles would mostly circulate along the loops through headers, steam generators, main pumps, and pressure tubes. This may not only enhance the temperature increase in pressure tubes but also cause pumps to cavitate. Without forced circulation, the potential for natural circulation could be reduced or even could be blocked by the existence of a large number of bubbles. The reduced flow would cause more overheating and/or damage of fuel elements and pressure tubes.

As discussed above, the moderator inventory is an alternate heat sink in the HWR system. However, this heat sink is not in direct contact with the coolant and the pressure tubes (there is a He gas filled space between the pressure tube and the guide tube) and thus may not provide a sink which would respond quickly enough to preclude bubble formation.

If a meltdown occurs, then the potential interaction of the moderator system with the core debris would require investigation.

5) Common Mode Heat Removal Failure at Headers

In the HWR, there is a low pressure injection system (LPIS) and there is a high pressure injection system (HPIS) to protect the core during LOCA. These systems provide borated water to both inlet and outlet headers. There are hundreds of tubes connecting each header to the pressure tubes via welds. Based on our limited information on the design, we note that if a LOCA is initiated by a failure of an inlet header, then it is possible that this failure may also prevent enough emergency cooling water from entering the cooling channels connected to the failed inlet header. The potential for this common source for losing cooling ability should be investigated further.

6) Outstanding Questions to the Applicant

In a telephone conversation with Fred Jesick of C-E on March 6, 1979, several questions were asked by BNL (R. A. Bari and Y. H. Sun) on

527 192

t.. overall design and functionability of the shutdown heat removal system. This information is needed before a detailed quantitative analysis of the system reliability can be performed.

As of April 5, 1979, BNL is awaiting responses to the following questions.

Q1 - What is the heat removal capability of the system if less than four steam generators are operational?

Q2 - Can the main heat transport system be used during cold shutdown, if the shutdown cooling system (the analog of the residual heat removal system in the PWR) fails?

Q3 - If the main feedwater pumps fail, can an adequate heat sink be provided by the condensate pumps and the safety valves in connection with the steam generators? How many safety valves (out of five) are required to open?

Q4 - If both the main feedwater pumps and the condensate pumps fail, how many safety valves together with atmospheric relief valves in conjunction with the emergency feedwater system (the analog of the auxiliary feedwater system in the PWR) a.. required to open if all other heat sinks are not available?

Q5 - How is the electrical system (both AC and DC) connected to the various safety loads and control systems? It was agreed that this question could be answered by C-E providing us with a better diagram (than contained in the preliminary design report) of the electrical system.

527 193

7) Temperature and Void Coefficients

The (preliminary) temperature and void coefficients computed at BNL for the C-E design 1.2 wt % ^{235}U PHW fuel bundles are shown in the table below (in units of 10^{-5} $\Delta\rho$).

	Burnup Cycle			
	Start	Middle	End	Reactor Average
Fuel (Doppler) Coefficient (per deg. C)	-1.0	-0.6	+0.2	-1.4
Coolant Temperature Coefficient (per deg. C)	+2.0	+3.5	+5.7	3.7
Moderator Temperature Coefficient (per deg. C)	-5.8	+0.14	+11.0	+1.8
Coolant Void Coefficient (100% Void)	+1100.0	+850.0	+850.0	~850.

The fuel (Doppler) coefficient is negative at the start and middle of the cycle and slightly positive at the end of the cycle. The reactor average value of the Doppler coefficient (averaging all fresh and high burnup bundles in the reactor) is negative. The coolant temperature coefficient is positive at all times in the burnup cycle. The moderator temperature coefficient is negative for fresh bundles but positive for bundles that have achieved more than half their design burnup. In the equilibrium cycle of continuous refueling, the moderator temperature coefficient is positive. The at-power coolant void coefficient (at 100% void) represents a reactivity of ~\$1.30.

The large mean neutron generation time $\sim 10^{-3}$ sec in the PHWR mitigates the effect of the positive temperature coefficients by providing time for the control or safety systems to respond to small changes in temperature.

527 194

The following (preliminary) table illustrates the effect of coolant or moderator temperature increase.

Coolant Temp. Increase	$\Delta\rho$	Prompt Power Increase	Reactor Stable Period
1°C	3.7	0.6%	2250 sec
10°C	37	6.0%	214 sec
50°C	185	40%	32 sec

Moderator Temp. Increase	$\Delta\rho$	Prompt Power Increase	Stable Period
1°C	1.8	0.3%	4650 sec
10°C	18	2.8%	454 sec
50°C	90	16.1%	80.4 sec

The loss of moderator cooling will have a positive reactivity effect, but there appears to be sufficient time to sense the moderator temperature change and shut down the reactor.

Temperature increases in the coolant would be accompanied by temperature increases in the fuel, resulting in a reactivity increase of about half that shown above for the coolant temperature increase. For slow increases in coolant temperature there appears to be sufficient time to control the power.

In a LOCA where both coolant loops are voided the PHWR will be prompt critical with a reactor period ~0.5 seconds. In one second the power would increase by a factor of 10. If only one loop lost coolant, the reactivity insertion would be approximately $0.65. This would cause a 65% increase in power within one second of voiding one coolant leg and increasing the power by about a factor of 10 within 7 seconds unless the reactivity transient is stopped by the safety systems. There appears to be entirely too much reactivity associated with voiding one loop of the

527 195

two loop design PHW. As a comparison, voiding a single loop in a four loop system would double the power in about 7 seconds, providing more time for safety systems to respond.

8) Production/Discharge Data (Preliminary)

The following table compares the annual discharge of HWR fuel and LWR fuel, based on thermal power of 4029 MW. Although the data given here is preliminary, the estimates are approximately within 10% of the C-E values.

		HWR	LWR
Burnup MWD/MT		19,750	29,789
Total discharge kg		56,516	37,025
^{235}U	(kg)	74.3	328.2
^{236}U	"	87.3	129.3
^{238}U	"	54920.	35144.
^{239}Pu	"	170.1	197.5
^{240}Pu	"	115.2	76.1
^{241}Pu	"	29.6	42.5
^{242}Pu	"	21.5	15.8
Total Pu	"	336.5	331.9

The HWR discharges 1-1/2 times more burned fuel by weight than the LWR, thereby increasing the volume of waste to handle. The total amount of plutonium produced is about the same in the HWR as the LWR. In the HWR, 59% of the discharge Pu is fissile, while 72% of the discharge Pu of the LWR is fissile material. The relatively larger amounts of ^{240}Pu and ^{242}Pu in the HWR fuel make it less suitable for recycle or weapons purposes than LWR discharge fuel.

527 196

9) Xenon Oscillations

In this section, the control problem associated with the Xe instability in C-E HWR is summarized. More detailed information can be provided if necessary.

The neutronic dimension of the CANDU reactor is about 4 times larger than the pressurized light water reactor and the oscillation of power distribution, due to Xe concentration build-up and decay, becomes a serious problem for reactor operation. In the C-E HWR, the total electric output is 1250 MWe which is about twice the output of the current CANDU reactor. Therefore, the physical size of the core would be twice the size of the CANDU reactor. Furthermore, the enrichment of ^{235}U in the fuel rods is 1.2% instead of natural uranium in CANDU fuel. This results in a migration area about 6% smaller than in the CANDU reactor. Thus, the neutronic dimension of the C-E HWR is more than twice that of the CANDU reactor and thus the higher harmonic Xenon oscillations will be excited. In order to control these Xe oscillations, a control mechanism such as the water compartment used in the CANDU reactor should be used. The number of control zones (water compartments) will be increased from 14 in the current CANDU to 32 in the C-E HWR. As the number of control zones increase, the self-powered in-core detectors such as Vanadium and Platinum detectors will be increased from 100 and 28, respectively, to 230 and 64, respectively. The size of the computer which controls the flux distribution should be increased in approximate proportion to the neutronic size of this reactor.

10) Neutronic Behavior Associated with the Loss of Coolant Accident

The current light water reactors have vertical coolant channels but in the heavy water reactor of CANDU type, the fuel rods are oriented horizontally. In addition to the limited heat transfer data (available in the open literature) for rods having horizontal flow, the flow patterns in horizontal tubes are significantly different from the vertical flow patterns.

The void coefficient of the reactivity change is a very important quantity to analyze for the neutronic behavior in the case of loss-of-coolant accident. The stratification of voids inside tubes will affect the neutron transport inside the core. Furthermore, the neutron streaming effect, due to void stratification, will change the void coefficient which usually is calculated under the assumption of homogeneous void.

527 107

If you need any further information on the subject matters discussed in this letter or on related matters, please do not hesitate to contact me.

Warm regards,

Bob

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527 198

NASAP PSEID - VOLUME 3
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF NASAP LWBR

In the previous preliminary staff comments on the LWBR (NUREG-0364 and the 9/25/78, Haller to Hanrahan letter) some of the areas where more information and detail evaluations would be needed were identified as: (a) Nuclear stability, (b) power and temperature coefficients, (c) adequacy of the control system, (d) provisions for accident prevention, (e) potential for recriticality during a meltdown accident and (f) core thermal margins.

The main sources of information up to this time, have been the PSEID and a meeting held March 20, 1979 between the staff and DNR and its contractors. Some information on four Prebreeder/Breeder pair conceptual designs was provided in the PSEID, while at the March 20, 1979 meeting DNR provided some details of the developmental effort on the Shippingport core and related supporting documentation. As a result of this information additional questions have been raised regarding: (a) the stability of the LWBR core with duplex pellets due to the delayed heating of the pellet core (which contributes the major part of the Doppler feedback) and (b) the potential for separation of ThO_2 and UO_2 in a duplex pellet under transient conditions due to the difference of about 800°F in their melting points.

Of the questions raised in the first paragraph, the one on nuclear stability has been answered as far as Xe and Iodine Isotopes are concerned. Likewise the question of power and temperature coefficients has been answered because there is a reasonable indication that the commercial size LWBR will not have different flux distribution and specific power than Shippingport and hence will not have different power and temperature coefficients. The question of recriticality has been answered par*ly for the homogeneous binary pin where, however, questions relating to configurations reflected by water or hydrogenous concrete remain. Also, the recriticality question remains open or is couped to the behavior of the duplex pin (i.e., potential separation of ThO_2 and UO_2 under transient conditions).

The general contention of the March 20th meeting and in the information provided in the PSEID is that, except for core related changes, the LWBR is basically not different than any other conventional PWR. However, the lower power density in the Thorium prebreeder and breeder cores give rise to lower ΔTs across the core and then higher flows (for similar power levels); this necessitates the use of larger pumps, larger safety equipment and even a larger pressure vessel. For such a design it is not clear that the core or the plant will behave in a manner similar to a previously analyzed PWR, neither it is clear that the type and/or design as well as response of the engineered safety features will be the same. The design will have to be reviewed in the light of the applicable regulatory guides, the provisions of the standard review plan and existing staff positions. The following areas are listed as examples of regulatory policy to which the LWBR must conform with respect to the equipment changes and their impact on safety. From the Standard Review Plan:

3.9.4	Control Rod Drive Systems
4.2	Fuel System Design
4.4	Thermal and Hydraulic Design
4.5.2	Reactor Internals Materials
4.6	Functional Design of Reactivity Control Systems

527 200

5.2.1.4 Compliance witn 10 CFR §50.55a

6.3 Emergency Core Cooling System

*15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a subcritical or Low Power Startup Condition

*15.4.2 Uncontrolled Control Rod Withdrawal at Power

*15.4.8 Spectrum of Rod Ejection Accidents

The LWBR fuel cycle shown in Volume III (Figure 2-6, page 2-17) shows a (mechanical) disassembly step carable of separating the duplex fuel (shown on Figure 2-5) into two streams - a U-235 stream and a U-233 stream. We did not see any flow sheet in Volume VII where a mechanical head end treatment was not followed by a chemical separations step. The LWBR fuel cycle flow sheet in Volume III should be consistent with the applicable flow sheet in Volume VII, and vice versa. In addition, we believe that if the concept of the duplex fuel is important to the viability of the breeder concept, additional information must be provided in Volume III or Volume VII concerning the development status of the disassembly operation shown in Figure 2-6.

Taken as a whole, the LWBR fuel design concept is extremely complex, as compared with the LWR. Differing enrichments, duplex fuel pellets, stationary (blanket) vs movable (see) components, thorium fingers, tertiary oxides, differing grid materials, taken with the various permutations and combinations afforded by th various breeder-prebreeder options pose potential problems with respect to the development and verification of adequate design bases, design limits and acceptance criteria.

*Not directly applicable in the LWBR but they should afford guidance as for the intent of the regulatory requirements.

There is probably an over-reliance on extrapolation of Shippingport Technology in regard to LWBR fuel design licensability. Although we have not previously identified any major fuel design problems during our Shippingport review, we did not perform a typical LWR-type review of the Shippingport design. We would expect to conduct a fully comprehensive review of a future LWBR fuel design and would place considerable emphasis on reviewing the supporting information for the proposed design limits and acceptance criteria.

Two potential fuel damage issues that would be expected to receive particular scrutiny would be (a) pellet cladding interaction (PCI) and (b) rod bowing. The bases for the proposed limits on power changes (power history and fatigue usage factor) would be examined closely. We would expect to see considerable analytical and/or experimental support for the Life Equivalency Parameter (LEP) vs lifetime recommendations.

Based on the preliminary review, some additional outstanding issues for the LWBR are:

- Zircalloy core support Grid (now designed of stainless steel)
- Potential consequences of molten ThO_2 (e.g., autocatalytic behavior)
- Control element performance with respect to its hydraulic support system
- Potential effect of oversized coolant handling systems in a backfit prebreeder (e.g., oversized pumps, oversized safety injection)
- Potential effect of the unique radioactive materials contained in the reactor on siting criteria (e.g., U-233)

- Analysis of the potential effect of the design basis accident and the low probability accident
- Adequacy of the proposed Thorium finger control system
- Fuel reprocessing and remote refabrication
- Validity of assuming the extrapolability of the Shippingport technology and safety implications
- Required level of effort to address the above issues.

NASAP PSEID - VOLUME 4
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF NASAP HTGR-GT

1. As the gas turbine (GT) concept for the HTGR has been formally adopted for development it will be necessary to revise the PSEID to reflect this design. This revision should include removal of material extraneous to the concept so that no ambiguity remains concerning the status of the design features being developed for the HTGR-GT. Material from the HTGR-Steam Cycle (SC) report should be retained however where useful comparative information exist (see question 18). All questions or comments given below should be responded to in the context of the GT design or its design status, with reference to earlier HTGR designs only when this information is fully generic or appropriate. This revised documentation may include responses to our questions by direct incorporation into the text or by separate paragraphs, as appropriate. In order to complete our review of the HTGR-GT in accordance with the objectives of the NASAP study we will need this information no later than August 15, 1979.

2. It will be necessary to establish explicit licensing criteria for the HTGR-GT as a portion of its construction permit review. Many of the criteria will of course be based on HTGR criteria used in past licensing actions. However, it will be necessary to review and re-establish the use of these criteria in terms of current requirements and to develop additional criteria as may be needed to meet the unique aspects of the gas turbine design. The objective of these criteria will be to assure that at least a comparable level of safety is achieved in comparison with other commercial reactors. Means for establishing such criteria, in decending order of desirability, are (a) direct adoption of existing criteria, (e.g., IEEE

527 204

criteria and many Regulatory Guides), adaption of existing criteria where necessary discrepancies can be justified, and the development of new criteria to meet the unique aspects of the design. Preliminary criteria development during the pre-application review phase is desirable in order to guide the conceptual and preliminary design activities and to anticipate areas which will need to receive increased attention during the construction permit review stage. We appreciate that General Atomic has been active in HTGR criteria development in the past and is presently active in developing criteria for structural graphite and inservice inspection.

One aspect that has not yet been explored is the contribution to criteria development by the Federal Republic of Germany under its cooperative agreement for the development of the HTGR-GT. We are generally aware of some of the differences in criteria between the Federal Republic of Germany and the U.S., but have not considered how such difference might be manifested in either the design of the HTGR-GT or in its licensing criteria. We are interested in the potential effect of these differences with particular regard to inservice inspection and testing, seismic design, and requirements for redundancy and diversity of engineered safety features. Please discuss how you expect these criteria differences to influence the design and licensing criteria of the HTGR-GT in the United States. If there are other criteria differences you believe are significatly different please discuss these also (e.g., design basis accidents, containment system design bases, and primary system integrity).

527 205

3. From our meeting with General Atomic on February 27, 1979 we understand alternatives to the reference design for the HTGR-GT presented at this meeting are being considered. Please identify the nature of these alternate concepts with emphasis on those design features most likely to effect the finality of our safety and licensing review of the reference design. If possible, indicate the degree of "firmness" that can be attached to the current reference design and estimate when decisions will be final on the incorporation or exclusion of significant alternatives.

4. Additional general and detailed information on the HTGR-GT research, development and testing program would be desirable if available. In particular, we would like identification of the research responsibilities of the various program participants, including foreign participants, a discussion of the relationship of these responsibilities to the HTGR-GT design described by General Atomic, identification of the roles to be assigned to Fort St. Vrain and operating HTGs in the F.R.G., a description of available and projected test facilities for the development of the turbine-compressor unit, a description of the research program pertaining to the replaceable hot duct and identification of the critical items for which research data must be available before pacing design decisions can be made.

5. What is the ground acceleration value deemed a practical maximum for the HTGR-GT design. What physcially limits the HTGR-GT to this value?

6. There are no explicit criteria directly applicable to the design construction and inspection of the turbine-compressor unit that we are presently aware of. Indicate to what extent existing codes may be adopted, such as the ASME Boiler and Pressure Vessel Code, and comments on the applicability of NRC documents that may afford guidance. A list of NRC documentation which may be useful in this regard follows:

   (1) Standard Review Plan 5.4.1.1, "Pump Flywheel Integrity (PWR)

   (2) Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity"

   (3) Standard Review Plan 4.4, "Thermal and Hydraulic Design (material pertaining to flow oscillations, loose parts, vibrations, load following maneuvers, part loop operation)

   (4) Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors"

   (5) Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles"

   (6) General Design Criterion No. 4

   (7) Standard Review Plan 3.5.13 - "Turbine Missiles"

   (8) Standard Review Plan 3.5.3 - "Barrier Design Procedures"

   (9) Standard Review Plan 10.2 - "Turbine Generator"

   (10) Standard Review Plan 10.2.3 - "Turbine Disc Integrity"

7. Tabulate the thermal and mechanical limits established or being considered for normal, transient and accident plant conditions for the fuel, control rods, structural graphites, ceramic materials, metals, and any other

527 207

component of the core, the primary system, or the primary system boundary deemed safety related. Identify which of these limits have been established by past HTGR licensing actions, which limits are to be established during HTGR-GT licensing reviews or topical report reviews, and which limits and confirmation by research and testing programs.

8. What additional features of the plant protection system or engineered safety features may be needed to cope with failure modes of the grey control rods, the turbine-compressor unit, primary system valve, the recuperator, hot duct, and the precooler. Responses to this question will require identification of or ref[illegible]nce to failure mode studies, postulation of a spectrum of accidents, predicted responses of the existing plant protection system and engineered safety features, and information on potential system interactions. We anticipate that it may not be possible for you to supply definitive responses to this question in the near future. Nevertheless, we expect that you should be able to supply preliminary and conceptual responses together with a discussion of the status of related accident studies together with an estimate of when this question can be finally answered.

9. The discussion of certain low probability accidents in the PSEID should be amplified beyond the use of the results of the AIPA study. In particular, describe the hypothetical consequence of a control rod ejection accident, consequences from a spectrum of failures in the core support structure, and the consequences of water injection from a failed pre-cooler simultaneously with rapid depressurization of the reactor.

10. The low probability accident customarily used for siting studies is an adiabatic core heat up caused by the sustained loss of forced convection cooling. Discuss the potentials for mitigation of this accident by designing for emergency heat removal by natural convection. What are the helium pressure requirements for emergency cooling by natural convection and how would these requirements vary with time after the accident? What role might the containment vessel and containment back pressure provide in natural convection cooling.

11. Substantially more information should be supplied with respect to internal pressure equilibration accidents in comparison to rapid depressurization accidents. Describe design criteria and design changes that might be needed to cope with the larger differential pressure forces experienced by thermal barriers, flow diffusers and other primary system components and boundary surfaces. Are any of the needed design changes sufficiently beyond the state-of-the-art that development programs will be necessary.

12. The direct cycle concept offers the potential advantage that water and other oxidant materials could be totally eliminated from the primary system by using a non-oxidant fluid in the precooler. Discuss the practicalities of this suggestion.

13. The information provided in the PESID on inservice inspection and testing was too generalized for our needs. Further, while you maintain that state-of-the-art equipment and practices are adaptable to current ASME Code requirements, we point out that Division 2 of Section XI has not yet been adopted by either the ASME or the NRC. Please revise your response with emphasis on the needs and means for inservice inspection, with special consideration of the following: (1) base and lateral core support structures, (2) the thermal barrier, (3) the PCRV liner, (4) the restraint mechanisms that preclude control rod ejection. As equipment designs relative to the GT plant develop in more detail we will expect more information than presented on February 27th pertaining to the needs and means for inspection of these developing designs.

14. Based on past licensing reviews for HTGRs it is likely that seismic design requirements will restrict siting choices to locations of relatively low ground accelerations in comparison with those acceptable for LWRs. Discuss this siting flexibility limitation from the standpoint of environmental and cost-benefit considerations in comparison to the other NASAP reactors.

15. What additional information with respect to occupational exposure can be made available beyond that provided in the PSEID relative to LWRs and the other NASAP reactor designs. Consider normal operation, refueling, inspection and decommissioning requirements.

16. Past experience and Sections 2.4 and 2.5 of the PSEID illustrate the point that the more we know about a reactor's conceptual design the more issues we are able to define for resolution. The HTGR-GT concept will likely undergo significant evolution before design details become firm. By that time, more detailed safety programs may also be defined. In spite of these difficulties, costs in time and dollars should be estimated for the resolution of design and safety issues. In responding to this question we recommend that tables of a format similar to Table 2-17 be included with expansions that compares R&D requirements, costs and schedules of the reference HTGR-GT, promising alternates, and a base case for the 900 MW(e) steam cycle plant.

NASAP PSEID - VOLUME 5
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF NASAP GCFR

1. In the event that the upflow core cooling design is adopted for the GCFR, it will be necessary for DOE to re-describe the GCFR's principal design features and provide an assessment of its safety characteristics in the prevention and mitigation of postulated accidents. We will need this information no later than August 15, 1979 in order to complete our review of the GCFR in accordance with the objectives of the NASAP study. This documentation should address all thirteen of the safety considerations given on Page 2-211 of the PSEID, provide discussion of any additional safety considerations DOE considers necessary and appropriate, and address all of the comments and questions contained herein in the context of the upflow design. We foresee that our general conclusion given in the 1974 Pre-application Safety Evaluation Report would not be adversely affected by the upflow design and that some of our conditions and reservations regarding the adequacy of the present emergency core cooling provisions might be positively addressed. Criter a related to the adequacy of thermal margins and fuel damage in the case of natural convection cooling would have to be developed in connection with the assessment of the adequacy of the use of natural convection for emergency core cooling.

2. It will be necessary to establish explicit licensing criteria for the GCFR as a portion of its construction permit review. The objective of these criteria will be to assure that at least comparable level of safety is achieved in comparison with other commercial reactors. Means for establishing such criteria, in decending order of desirability, are (a) direct adoption of existing criteria, (e.g., IEEE criteria and many

Regulatory Guides), adaption of existing criteria where necessary discrepancies can be justified, and the development of new criteria to meet the unique aspects of the design. Preliminary criteria development during the pre-application review phase is desirable in order to guide the conceptual and preliminary design activities and to anticipate areas which will need to receive close attention during the construction permit review stage. We appreciate that General Atomic has been active in this area in the recent past.

One aspect that has not yet been explored is the contribution to criteria development available from the several European governments cooperating in the development of the GCFR. We are generally aware of some of the differences in criteria between the Federal Republic of Germany and the U.S., but have not considered how such difference might be manifested in either the design of the GCFR or in its licensing criteria. We are interested in a discussion of the potential effect of these differences with particular regard to inservice inspection and testing, seismic design, and requirements for redundancy and diversity of engineered safety features. Please discuss how you expect these criteria differences to influence the design and licensing criteria of the GCFR in the United States. If there are other criteria differences you believe are significantly different please discuss these also (e.g., design basis accidents, containment system design bases, and primary system integrity).

527 213

3. At the February 26, 1979 meeting the main body of the information provided was for the 300 MW(e) design although the conceptual characteristics of the 1200 MW(e) plant were outlined. Please resolve from the standpoint of DOE's desired approach to the safety review of the 1200 MW(e) plant which course you will follow to provide our needs for additional information: (a) establish the scale-up feasibility of the 300 MW(e) design to the 1200 MW(e) size, (b) provide information in greater depth for the 1200 MW(e) size with reference to features of the 300 MW(e) plant that demonstrate feasibility of the 1200 MW(e) design, or (c) identify some alternate plan that will satisfy the NASAP objectives.

4. What additional to that provided in the PSEID can be said about occupational doses for the GCFR relative to LWR's and the other NASAP reactor designs. Consider normal operation, refueling, inservice inspection, and decommissioning plans.

5. What equilibrium fraction of noble gases, iodines and other volatile fission products is resident in the GCFR fuel rods in comparison to non-vented fuel rods? Also, how is the comparative decay heat level of the reactor altered by continuous venting of the volatile fission products? Do accident studies consider this lower inventory of fission products in the GCFR core?

527 214

6. What are the specific criteria and requirements for inservice inspection and how will these be integrated into the preliminary design? What role will the ASME Section XI committee play in ISI decisions?

7. How will development programs for the GCFR's primary system components be affected if the development of HTGR technology is not carried in the United States substantially beyond the Fort St. Vrain reactor.

8. We understand from the February 26th meeting that GA is now considering core disruptive accidents and core melting as containment design bases, and has patterened its reactor siting source term and its containment configuration after the Clinch River design safety approach articulated by the staff in a May 6, 1976 letter to ERDA. Please provide the following:
   (a) Documentation confirming or correcting relevant material presented to the NRC on February 26, 1976.

   (b) A discussion of why the Clinch River containment design and siting source term are considered appropriate to the GCFR.

   (c) A description of experimental research programs planned to confirm assumptions used in the CDA analysis and the containment system design.

9. For your information we have included as Enclosure 7 the document "Basis for Containment Design in Large LMFBRs." Although LMFBR terminology is used throughout we consider this document also applicable to the GCFR.

527 215

NASAP PSEID - VOLUME 6
COMMENTS AND ADDITIONAL INFORMATION NEEDS
NRC REVIEW OF LMFBR-VARIANTS

1. Since the 15 variations submitted as part of the NASAP PSEID package are variations on core design and fuel cycles only, it will be difficult, if not impossible, to perform a comparative evaluation of "integral" reactor systems (i.e., NSSSs and BOPs). We believe that it will be unfair to the LMFBR assessment for the staff to assume and use an extrapolation of the CRBR design for making these judgments. The CRBRP, being a loop design vintage early 70's, does not reflect recent design innovations/improvements. Also a number of key safety issues associated with the CRBRP remained outstanding at the time of the suspension of the safety review (Spring '77). (For a summary of these safety issues see letter from Gammill, NRC, to Caffey, CRBRP dated 11/9/78.)

It is important for DOE to recognize that any one of these LMFBR conceptual designs must be consistent with and conform to the spirit and intent of the staff licensing positions as reflected in the regulatory guides, criteria in the Standard Review Plan, the General Design Criteria, and other licensing regulations. Some of the key areas that must be addressed include fuel system design, inservice inspection, control system diversity and independence, decay heat removal system diversity and independence, and finally containment system design. Due to the importance of containment system design, we have included a recapitulation of recent licensing staff positions on containment design in a separate enclosure for your information. Before we can proceed with the LMFBR portion of our NASAP review, we need to know: your basic safety approach; to what extent this approach conforms to accepted practice; how and when you will decide on specific design concepts (e.g. loop vs pot); and the level and direction of R&D effort for reactor safety. It is important that DOE be reminded that, in the past, the staff and DOE differed in basic safety approach and implementation for both the CRBR and the FFTF reactors. These differences have been documented in great detail for the CRBR and FFTF in correspondence between the staff and projects, and in the CRBR SSR and FES and in the FFTF SER. It is imperative that DOE recognize and understand these differences, and that DOE factor them into their overall planning, in particular into their formulation of and commitment to a much-needed safety R&D program. Anything short of this could have serious implications for licensing.

2. It appears that the only substantive reference regarding LMFBR Core Disruptive Accidents (CDAs) for these alternative fuel types and core designs is INFCE/5-TM-5, H. K. Fauske, "Safety Implications of Alternative Fuel Types." Several general comments and questions are in order:

   a. NRR does not necessarily agree with some of the basic conclusions, methodology and basis for design features presented in this report, e.g.

      (1) the methodology of using the "first principles," listed for example on page 26, to draw global conclusions on the relative merits of oxide vs carbide or metal fuels.

      (2) the conclusion that metal fuels are inherently safer fuels than carbides, drawn from application of these first principles.

(3) the conclusion drawn (page 29) that, for a loss-of-heat sink accident fuel melting is initiated only if the coolant level drops below the core.

(4) the conclusion that sodium bonding of metal or carbide fuel has only safety advantages in CDA sequences.

(5) Based on some of the above conclusions, the author recommends certain design options, such as: upper structure removed from lead subassemblies (S/As); perforation of S/A ducts; and sodium-bonding for carbide and metal fuels.

b. This report has an outline for "experimental resolution of key issues" for all three fuel types. To what extent will DOE rely on the definition of problems and resolution approaches as outlined in this report. (It is important for DOE to recognize that the technical judgments and opinions in this report are not necessarily those of the technical community either within NRC or without. Thus DOE should proceed with caution in implementing the research programs described therein.) More generally, what would be the DOE experimental and analytical program to resolve key safety issues if, say, metal-fueled LMFBRs are a major part of the U. S. LMFBR program?

3. Can DOE supply any analysis in the area of CDAs for the design options including large homogeneous vs heterogeneous cores; carbide and metal vs oxide; and ThO_2 blankets vs UO_2 blankets?

4. Does DOE have a position on the homogeneous vs heterogeneous core? And, if so why? Provide analysis including CDA transition phase analysis.

5. In a number of reports supplied to NRC, there is a design constraint that the positive sodium void coefficient be less than $3.00. Provide the basis for this constraint and its effect on consideration of the LMFBR-variants in the NASAP study.

527 219

BASES FOR CONTAINMENT DESIGN IN LARGE LMFBRs

In the past the NRC staff took the position that an LMFBR containment system should be able to withstand not only design basis events such as sodium fires, but also the consequences of low probability or Class 9 accidents (see e.g., CRBR FES, NUREG-0139, dated February 1977; the CRBR Site Suitability Report, dated March 4. 1977; FFTF SER, NUREG-0358, dated August 1978.) Specifically, for the case of CRBR, the staff took the position that the containment system should be protected from the effects of low probability accidents (commonly referred to as core disruptive accidents in LMFBRs or CDAs) such that, comparability to the inherent protection of LWR containment systems to core melt events is achieved. This resulted in the 24 hour containment integrity requirement for CRBR which can be found in the above given references(recent NRC letter to DOE on the 24 hour requirement is attached to this enclosure). Since the termination of the CRBR review in April 1977, the staff completed the FFTF review and also completed a comparative study between land-based and offshore sited floating nuclear plants of the radiological consequences of core meltdown events. On the basis of this study (see NUREG-0440, dated February 1978), the staff recommended the issuance of a manufacturing license for barge mounted plants subject to the condition that "the applicant shall replace the concrete pad beneath the reactor vessel with a pad constructed of magnesium oxide (MgO) or other equivalent refractory material, that will provide increased resistance to melt-through by the reactor core in the event of a highly unlikely core-melt accident and which will not react with core-debris to form a large volume of gases ...." (see NUREG-0502, December 1978).

For the case of FFTF, the staff analysis indicated that overpressurization and the generation of hydrogen resulting from sodium and core debris interaction with concrete are the principal challenges to containment. The quantity of hydrogen generated that could create potentially explosive or highly energetic flammable mixture in the FFTF containment building atmosphere, or portions of the building, preceeded in time the point of threatening containment integrity by overpressurization. Even though the staff in the FFTF SER, NUREG-0359, dated August 1978, considered various means to alleviate the buildup of pressure and hydrogen in the containment building following postulated core meltdown events, some of the recommended steps to deal with the problem were probably not appropriate in view of the facility being essentially constructed. For example, even though refractory materials (e.g., similar to the MgO recommended for the FNP design) which are highly resistant to molten core debris and do not generate hydrogen could have been used in the reactor cavity and in the containment subcavity of the FFTF, its use would have been difficult, expensive, and maybe detrimental from an overall safety viewpoint since the cavity and subcavity were already built and sealed.

For future large fast reactor designs, the approach should be to integrate in to the containment system design from the start the necessary features and designs, so that the containment will be able to withstand and mitigate not only the consequences of design basis events, but also the consequences of lower probability, higher consequence accidents. Accordingly, three (3) broad classes of accidents, which are summarized below should be taken into consideration in the designs of large fast reactor containments.

The three classes of accidents are: (1) those postulated accidents considered in the design basis of plants (i.e., 10 CFR 50), (2) hazards not exceeded by those from any accidents considered credible (i.e., 10 CFR 100 of Site Suitability Source Term) and (3) low probability or Class 9 accidents. Since the information provided in the LMFBR PSEID relates primarily to the core (i.e., various fuel cycles) and it has not been integrated into a system design, the following staff comments on these three classes are somewhat generic in nature and are primarily based on the staff's experience with previous reviews of LMFBRs and LWRs, as well as the recent staff position mentioned earlier regarding floating nuclear plants.

1. DESIGN BASIS ACCIDENTS (10 CFR 50)

In an LMFBR, the accidents which represent the principal challenges to containment are sodium fires coupled with potential sodium-concrete reactions which result from failure and subsequent release of sodium from pipes and vessels containing sodium. Following sodium release, combustion with oxygen (even for those areas which are inerted) will result in increasing pressures and temperatures. The specific initiating events, as well as consequences will be very system dependent. Based on the staff's review of CRBR and the FFTF, the sodium releases were based on considering a spectrum of postulated component and piping failures of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated sodium fire accidents is covered. Some of the specific challenges to the containment presented by sodium release accidents that should be considered in a containment design are as follows:

1. Mechanical - The deterioration of concrete by sodium can weaken structures, cause cracking, enlarge leak paths. Therefore, means should be used to prevent or reduce the likelihood of direct contact between sodium and concrete. For FFTF and CRBRP, cell liners were used to accomplish this.

2. Thermal - The chemical heat of sodium reactions with oxygen or concrete can build up pressures within inerted cells or the containment building which must be included as part of the containment design basis.

3. Explosive - The generation of hydrogen from reactions between sodium and water (or concrete) can lead to explosive mixtures in the air atmospheres of the Reactor Containment Building. Therefore, water should be kept to a minimum in buildings containing large amounts of sodium. Hydrogen recombiners are provided in LWRs to control hydrogen. For LMFBRs (FFTF and CRBRP), the applicants have claimed in the past that the presence of sodium oxide has a catalytic effect in promoting recombination of hydrogen and oxygen and keeping the hydrogen concentration below the explosive limit. Based on the available information, the staff has in the past been unwilling to accept the view that hydrogen can be depended upon to burn benignly under the natural processes associated with these accidents.

4. Non-radiological toxicity - If released from containment or the steam generator building, large quantities of non-radioactive sodium could be an inhalation and environmental hazard. Effective methods can be used to suppress or extinguish sodium fires, as well as isolation can prevent the release of the hazardous smoke.

5. Filters - The dense smoke from sodium fires can rapidly plug ventilation filters. Scrubbers or prefilters are generally required to eliminate this problem.

In recognition of the above, the NRR staff during the review of the CRBRP issued general safety design criteria for the CRBRP, including Criterion 41 "Containment Design Basis," which stated in part ... "the reactor containment structure, including access openings and penetrations, and if necessary, in conjunction with additional post-accident heat removal systems including ex-vessel systems, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences and any of the postulated accidents."

2. <u>SITE SUITABILITY SOURCE TERM (10 CFR 100)</u>

The site suitability source term (SSST) is non-mechanistic, and its use is intended to represent an assumed radiological release from the core whose consequences would result in potential hazards not exceeded by those from any accident considered credible (see footnote 1 to 10 CFR 100.11 (a)). A primary objective of the staff's safety review is to assure that no other accider equences within the design envelope result in the release of fission products to the environment greater than those postulated for the SSST. As part of this review, the staff has in the past examined very carefully such factors as core physical and geometrical configuration including the type and quantity of fissionable material, control system(s), decay heat removal system(s), and amount of redundancy and diversity in important safety systems. Also, the manner in which the as designed plant responds and interacts to a spectrum of accidents, including very severe ones has also been considered.

Without a particular detailed design description, such as presented in a PSAR, it is not clear from the PSEIDs that the consequences of <u>all</u> credible events would be enveloped by a SSST, nor is it apparent that "generic" attenuation mechanisms would apply in all scenarios. At present, both the design concepts for large fast reactors and the analytical methods for examining accidents are in a state of development.

We would, therefore, recommend that the containment design be based on sufficiently conservative source terms which encompass all the uncertainties in presently available data, analyses and design concepts. As an example, the staff reviewed the bases provided by the CRBR project for its source term and concluded that insufficient information had been furnished to establish that it met the requirements of 10 CFR 100.11 (including footnote 1). As a result, a more conservative radiological source term was adopted (see CRBRP Site Suitability Report, pg. III-14).

Additional materials not included in a SSST for LWRs, or even the CRBR, might have to be included for the NASAP concepts to account for the introduction of alternative materials (e.g., U-233, U-232, ...).

Additional design requirements imposed on containment systems, such as filtration, fission product removal and containment heat removal systems will have to be considered very carefully. In these areas, additional R&D and proof testing will almost certainly be needed.

3. CORE MELT AND DISRUPTIVE ACCIDENTS

In an LMFBR, the low probability accidents which represent the principal challenge to containment are associated with core melt and disruption with the potential for concurrent energy release. The energy release is a result of either core vaporization (direct core disassembly and/or recriticality), or sodium vaporization from the transfer of heat from the molten core to the sodium coolant. Energetics could lead to early (order of minutes) containment failure if the containment system is not designed to accommodate the generated loads; on a longer time scale failure of the containment would occur from the evolution of the meltdown accident progression. This latter evolution could involve chemical reaction products and/or sodium vapor resulting from the inadequate post-accident decay heat removal of a molten and/or disrupted core and could lead to hydrogen explosions, or overpressurization and/or thermal/structureal degradations, either one or a combination being able to cause containment failure. Without a particular design description as presented in a PSAR, it is not possible to evaluate either the potential evolution of an accident scenario and its consequences or whether it will/can be mitigated and/or contained. Based on the staff's past involvement and experience with the safety analyses and reviews of LMFBRs, containments should be designed to mitigate or significantly reduce the consequences of core melt and disruptive accidents. From the viewpoint of the two major accident sequences (i.e., early accident energetics and longer time meltdown consequences) that can threaten containment integrity, the following should be considered:

3.A ACCIDENT ENERGETICS (DIRECT DISASSEMBLY, RECRITICALITY AND FUEL COOLANT INTERACTIONS)

In the past some LMFBR designers have relied on the Primary Heat Transport System (PHTS) to accommodate the potential energetics; this was especially true for the CRBRP. At the time of suspension of the CRBRP licensing review, the staff and applicant had not resolved the question of whether the design was adequate to accommodate the value of the energetics described in NUREG-0122. Other designers (e.g., the UK in the case of the Commercial Fast Reactor (CFR) design) have considered pre-stressed concrete vessels with inherent capability to accommodate large energetics. The choice of a particular vessel/containment system would of course depend, among other things, on the requirements derived from a specific design. Some of the key considerations (note NUREG-0122) that influenced the selection of the level of energetics for the CRBRP were:

1. The potential for large work-energy release during the "initiating phase" (direct disassembly) due to the autocatalytic, positive-sodium-void effect without the presence of the mitigating effect of timely and substantial fuel dispersion.

2. The potential for large work energy release during the "transition phase" (recriticality).

3. The many uncertainties and unknowns associated with CDA phenomena including: the potentail for sodium as a working fluid; fuel pin failure dynamics; freezing, plugging and remelting of molten fuel and fuel/steel mixes; and molten pool boiling dynamics.

Areas and parameters that will influence accident scenarios and consequences for the design(s)/fuel cycle(s) considered in the NASAP study are:

1. the effect of a heterogeneous core (compared to a homogeneous core like CRBR) on accident progressions,

2. the effect of core size,

3. the effect of fuel type such as carbides and metals vs. mixed oxides (e.g., on Doppler Coefficient). In the area of Fuel-Coolant Interactions (FCIs), the effect may be major for both carbide and metal fuels because the potential for sodium becoming a working fluid is considerably enhanced.

4. the effect of various bondings for metal and carbide fuels (either helium or sodium)

5. the effect of fuel cycle types such as Pu/Th with Th blankets vs. Pu/U with Uranium blankets.

6. the effect of a pot design vs. a CRBR-type loop design

7. the effect of design specifics such as: upper fission gas plena vs. lower plena; perforated subassembly ducts; and temperature profiles across subassemblies.

An aggressive and comprehensive experimental and analytical research and development program will be necessary in order to understand the above effects and their relevance to the safety of a particular LMFBR variant. We need to understand DOE's policy and planning, time frame and resource commitment, for these safety-related areas.

3. B <u>CORE MELTDOWN</u>

As was previously mentioned a benign (i.e., non-energetic) core meltdown can result in hydrogen explosions, overpressurization due to sodium vapor and non-condensible gas generation, and thermal/structural degradation. All of these effects can lead separately or contribute jointly to containment failure. For example, for the FFTF containment failure was predicted to occur either from hydrogen explosions in the time interval of 10 to 20 hours, or from over-pressurization in the interval of 30 to 60 hours.

Evaluations performed by the staff for the CRBRP and FFTF, as well as the FNP indicate that containment integrity can be extended substantially or even indefinitely with the additiona of refractory sacrifical materials and/or cooling systems in the lower reactor cavity area. In other areas outside the reactivity cavity, steel liners constructed as engineered safety features can be used to protect the concrete from sodium attack. For both cases, the objective is to reduce or eliminate the potential for the buildup of hydrogen and other non-condensible gases, as well as sodium vapor, that can threaten the containment integrity. Areas of work that should be pursued within the framework of future large LMFBR design(s) are:

1. Examination of refractory sacrificial materials that are highly resistant to core melt debris and do not interact to form a large volume of gases;

2. Examination of cooling systems, both active and passive, to prevent sodium from evaporating following a core meltdown and to remove decay heat from the outer extremities of the refractory material, such that containment of molten core debris can be assured;

3. Investigate methods to monitor and control the hydrogen concentration in the containment building following postulated core meltdown events; and

4. Examine means to further reduce radiological releases from containment following postulated core meltdown events, such as the addition of sand and gravel filters.

In summary, the licensing staff believes that positive and clearly identifiable actions should be taken in large fast reactor designs to mitigate significantly the consequences of core melt and disruptive accidents.

MAR 8 1979

Docket No. 50-537

Mr. Lochlin W. Caffey, Director
Clinch River Breeder Reactor
Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Dear Mr. Caffey:

SUBJECT: USE OF WASH-1400 IN CLINCH RIVER BREEDER REACTOR (CRBR) LICENSING REVIEW

On May 6, 1976 the Staff issued its position on the overall design safety approach and criteria for the Clinch River Breeder Reactor. One of these criteria was that "containment integrity be provided for at least 24 hours following a core disruptive accident." As was discussed then with the CRBR Project and the ACRS, the selection of the 24 hour criterion was partially based on the then available WASH-1400 analyses of the times to containment failure from a spectrum of core melt scenarios.

In light of the Risk Assessment Review Group's conclusions and recommendations (NUREG-CR-0400) and the recent Commission Policy Statement on this matter, the 24 hour criterion will be reconsidered in any reinitiation of the CRBR review.

Sincerely,

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

527 231

NASAP PSEID Volume VII

Fuel Cycle Facilities

Preliminary Comments

The stated purpose of this document is: to highlight safety and environmental issues, identify licensing issues, and ascertain whether concepts being considered in connection with the Nonproliferation Alternative Systems Assessment Program (NASAP) are fundamentally licensable.

While this document identifies systems and principal issues related to nonproliferation alternatives, it does not assess the proposed systems or facilities in sufficient depth to permit definition of appropriate licensing criteria or the potential difficulty of meeting such criteria. It is assumed that the systems or proposed facilities would meet the applicable local, state and Federal criteria at the time they are proposed. However, it is quite likely that some of these systems may never reach a demonstration phase since they may not have enough promise to warrant appropriate funding. Nevertheless, to the extent of present definition, the proposed facilities all appear to be fundamentally licensable on a qualitative basis.

General

1. In general, this document touches on the principal safety and environmental and licensing issues that would be associated with various alternative fuel cycle systems. In addition, it also suggests some areas where more information will have to be developed through further studies or future research. It does not make a quantitative comparison of safety or environmental trade offs in areas such as occupational exposure, regional exposure, accident risk or environmental impacts. NRC believes that such comparisons will have to be developed as part of the NASAP final evaluations, even though they are not given in this document.

2. Some chapters of the document contain statements concerning R&D requirements that are generic to the nuclear industry rather than specific to a certain operation or system (i.e., "determine the relationship, whether proportional or threshold, of total radiation exposure to health"). These requirements should be moved to a general section of the document.

3. The EPA regulation 40 CFR Part 190 limits the allowable exposure of an individual from normal effluents from most uranium fuel cycle facilities. The requirements to meet this regulation should be

set forth early in the document, since most facilities will be required to comply with it. In addition, the probable existence of a comparable regulation (with perhaps different limits) for thorium fuel cycle facilities should be noted.

4. The document should be reviewed for consistency within itself and with other NASAP documents (e.g., p. 6-48 states the waste repository design was based on a nuclear power growth to 480 GWe by 2000. This level is not consistent with the growth scenarios in other NASAP documents). In particular, since the fuel cycle facilities must support the several reactor concepts set forth in the PSEID Volumes I-VI, a review of those PSEID's and the fuel cycle facilities PSEID should be made to determine that all fuel cycle facilities required (either explicitly or implicitly) by the reactors discussed in Volumes I-VI are discussed in Volume 7. (Volume I, Light Water Reactors, contains a flow sheet that implies a requirement for ^{233}U storage, plutonium storage, and thorium storage.)

Chapter 1. Mining and Milling

1. Section 1.1, Uranium, contains data on radon releases from uranium mining taken from a DOE document dated 1975. NRC has presen data on radon releases both during mine operation and after shutdown to its licensing boards in 1978 and is in the process of updating the radon value in 10 CFR 51 - Table S-3. We suggest that the latest NRC material be reviewed to determine whether or not these later data should be used in this PSEID.

2. Section 1.1.2, Safety Considerations, contains a statement (p. 1-3, paragraph 5) that risks of flooding resulting from failure of tailings dams are decreased by the "standard practice" of requiring a 5-foot minimum free board be maintained. NRC notes that a 5-foot free board is often not sufficient to meet the criteria of its Regulatory Guide 3.11. NRC therefore suggests that the last three lines of the paragraph be written as:

   > "the pond area. New NRC guidance for the design and construction of tailings dams contains guidance on determining acceptable free board and require that tailings dams be designed to prevent failure due to a probable maximum flood."

3. Section 1.1.3, Environmental Considerations (3rd paragraph of section), may, by the statement "overburden...is used to backfill the mined-out areas...", convey the impression that backfilling of mines is a requirement. NRC is unable to verify that backfilling of mine areas is required in all states.

527 235

4. Section 1.1.4, Licensing Status and Considerations (paragraph 1), contains the statement that the Federal Government has no licensing authority over uranium mines. NRC recommends that the validity of the statement be verified to determine the responsibilities of the Forest Service, Bureau of Land Management, and Bureau of Indian Affairs concerning any obligations they may have to prepare EIS'.

The second sentence of the same paragraph states that NRC reviews the safety and environmental impacts of mines that are a part of a mine and mill complex. The sentence should be revised to state that NRC assesses the environmental impact of radiological emissions from a mine that is part of a mine-mill complex. Since NRC does not assess all mines, we recommend that the sentence be stricken.

5. Although Section 1.2, Thorium, contains some data on thorium reserves, no data are provided on by-product thorium stockpiles. By-product thorium might be used for an extended period to fuel the early cores of the thorium fueled reactors. NRC believes that a brief statement be added to Section 1.2 describing the amount of thorium that is projected to be available from by-product thorium.

Chapter 2. Uranium Hexafluoride Conversion

1. NRC's Safety Evaluation Reports and Environmental Impact Appraisals for the Allied Chemical Corporation UF_6 plant and the Kerr-McGee Nuclear Corporation UF_6 plants are available and should be cited in Section 2.1.

527 237

Chapter 3. Enrichment

1. The most recent data on Technetium-99 that NRC has been able to develop from DOE data show lower technetium releases to the environment than those given in Table 3-1. In addition, NRC's data show the following disposition of Tc-99 in the diffusion plants: about 90% reports to the solid waste stream, about 9% to the liquid effluent, and 1% to the gaseous effluent. With the dominant fraction of Tc-99 in the solid waste, we believe that some attempt be made to account for the technetium in the solid waste.

2. NRC is unable to verify that the WASH-1543 document cited in reference 3 was published by NRC.

Chapter 4. Fuel Fabrication

1. Although NMSS supports DOE's attempt to aggregate the discussion of fuel fabrication of twelve fuel forms into four classes of fuel - low gamma activity pellets; low gamma activity pellets containing plutonium; high gamma activity pellets; and HTGR fuels, we feel that much information has been lost by the use of the additional level of aggregation represented by the discussion presented in Section 4.3, Environmental Considerations. The data in Table 4-3 do not adequately represent emissions from fabrication of ^{233}U fuel, spiked plutonium fuel or partially decontaminated ^{233}U or plutonium fuels.

   We believe that Section 4.3 should be revised to provide a comparison of the environmental impacts from fabrication of the four fuel forms considered above. Special effluent treatment systems (if any are required) should be described.

2. Section 4.5, Research, Development and Demonstration, contains statements that more properly should be contained in Volumes 1-6. R, D and D work necessary to demonstrate fuel performance (p. 4-20, paragraphs 4 and 9, and p. 4-21, paragraph 8, for example) have more to do with licensability of the reactor concept than the fuel fabrication step.

527 239

3. In Section 4.3, Research, Development and Demonstration, there is no information on any required R, D and D program on spiked or partially decontaminated fuel. NRC has been no evidence that conventional fuel fabrication processes can be used with spiked or partially decontaminated fuel. Some statements should be included in Section 4.5 about the R, D and D requirements for these types of fuel.

4. Section 4.6, Decommissioning and Decontamination. NRC Regulatory Guides require that some information on decommissioning be submitted at the time the initial application for facility licensing is submitted. Step 1 of Section 4.6.2 carries the implication that decommissioning planning need not be initiated until 1-2 years before the plant operations are completed. (Page 4-27 does not make clear that planning during the design stages can facilitate decommissioning.) NRC believes that Section 4.6.2 should be revised to convey the fact that decommissioning is a licensing consideration from the time of receipt of the initial application.

Chapter 5. Reprocessing

1. Section 5.4, Purex 4 Reprocessing: Coprocessing with Pre-Irradiation, does not identify any characteristics of the "irradiation facility." No statement concerning licensability of this operation can be made without some definition of the irradiation facility.

2. Section 5.6, Thorex 1 Reprocessing: Uranium and Thorium Recovery, is based on recycling thorium after 10-20 years storage. NRC believes that some consideration should be given to potential recycling of thorium promptly with the U-233, and the potential need to dispose of thorium as a waste if excess stockpiles accumulate.

3. Section 5.7.5, Research, Development and Demonstration, cites the requirement for radiation experiments. These experiments bear more on the licensability of the reactor concepts than the licensability of the fuel reprocessing concept, and should therefore be a part of the reactor PSEID's.

4. Section 5.8, Thorex 3 Reprocessing: Partitioned U/Pu/Th Fuel, contains in Section 5.8.3, Environmental Considerations; the statement, "The presence of a plutonium stream will increase the transuranic content of the waste streams and the off gas releases." A more precise statement would appear to be "Use of denatured thorium cycles results in increased transuranic content of the waste streams."

5. Section 5.9, Thorex 4 Reprocessing: Partitioned U/Th, Pu to Waste, is a discussion of a Thorex variation that is not used in the Reactor PSEID's (Volumes I-VI). NRC believes that Section 5.9 be deleted or altered to fit one of the reactor concepts.

6. Sections 5.10, 5.11, and 5.12 also present fuel cycle variations that were not considered in Volumes I-VI covering reactor concepts. Accordingly NRC believes that these variations should be either related to a reactor concept or deleted.

527 242

Chapter 6. Waste Handling and Treatment

Section 6.2 Geologic Disposal of Spent Fuel

1. Section 6.2.4.1 reports that the "current DOE National Waste Terminal Storage (NWTS) program calls for the selection by 1979 of two sites overlying suitable salt formations, followed by the construction and startup in 1985 of one NRC-licensed repository..." Current NRC estimates for the time between submittal of a license application and licensing decision are 3-12 years.
2. Section 6.2.4.6 (paragraph 3) states that feasibility of using dismantling to decommission a geologic repository was demonstrated on a small scale in Project Salt Vault. The NRC staff does not believe that the feasibility of using dismantling to decommission a geologic repository was in any way demonstrated in Project Salt Vault as claimed. Decontamination of the mine and returning it to its owners has no relevance whatsoever to the problems involved in decommissioning a facility in which high-level wastes are to be permanently stored. The concept of "dismantling" has little meaning for decommissioning of an underground repository.

527 243

Section 6.5 Waste Disposal 5: Shallow Land Burial of Low-Actinide Wastes

1. Section 6.5.1 implies that several fuel cycles (PWR spiked recycle, LWR U-233/Th recycle, Shippingport Type I pre-breeder and breeder, HWR once through (DU 235), HTGR once through, and HTGR DU 235/DU 233-Th) will produce no low-actinide waste. The NRC staff believe that all fuel cycles will be likely to generate low level wastes. For example: low level wastes would probably be generated in significant amounts at reprocessing plants (evaporator bottoms, filter elements, combustible trash, etc.); all fuel cycles using uranium would likely generate low level wastes in UF_6 conversion plants; all fuel cycles using enriched uranium only would generate low-level wastes similar to those of the LWR once through fuel cycle.

   NRC believes that the volume, chemical and physical form, and isotopic activities in low level wastes from each operation (excluding mining) of all fuel cycles should be estimated to permit an evaluation of the overall impacts of low level waste disposal. (Data on low level waste from reactors using the particular fuel cycle should also be given in Volume VII.)

2. Section 6.5 uses the terms "low-level" and "low-actinide" wastes. It is not clear what wastes the terms are meant to include or exclude.

3. The basis for the 40% reduction in low-level waste from the improved LWR once through fuel cycle relative to the reference once through fuel cycle should be explained briefly in Volume VII.
4. The information on waste volumes from the LMFBR Homogeneous U-Pu/U Spiked Recycle Fuel Cycle discussion (Section 6.5.1.4) gives data only on the low-level actinide wastes from fabrication of the blanket fuel elements. The reference cited, WASH-1535, gives volumes of solid wastes other than high level, produced at reactors, reprocessing plants, and fuel fabrication plants. NRC believes an attempt should be made to provide the most inclusive tabulation of waste sources possible.
5. The sections in Chapter 6 on Decommissioning and Decontamination contain very little information. In the absence of specific information on decommissioning and decontaminating alternative fuel cycle facilities, NRC would recommend that a discussion be provided of D&D operations for the LWR once-through uranium fuel cycle, and that the impacts from D&D operations for other fuel cycles be compared with the LWR once through cycle.

Chapter 7. Transportation

1. Sections 7.1.1.5 and 7.1.1.10

   NRC does not permit plastic bags around fresh fuel to be sealed in order to preclude the bag's becoming filled with water. The reference sections should, therefore, specify that Shippingport pre-breeder and breeder fuel modules and GCFR fuel assemblies are shipped in open bags.

2. Section 7.3.4.1

   Some of information in Section 7.3.4.1 is outdated. To update the information, both the IF-300 and the NLI-10/24 should be shown as built, and the NAC-1 should be removed from the list of currently available legal weight truck casks and the TN9 added to the list of truck casks.

3. Section 7.3.4.8

   NRC can verify that a truck cask for the shipping of spent HTGR fuel has not been licensed by NRC.

4. Section 7.4.2.1

   a. NRC does not know whether or not the sealed steel canisters surrounding the high level waste would be designed and fabricated to act as a level of containment. We therefore recommend that the

Table 7-2. Available shipping casks for present-generation LWR spent fuel

Cask	Usual transport mode	Weight (lb)	Length (in.)	Diameter (in.)	Working length (in.)	Type	PWR	BWR	Number available in 1977
IF-300	Rail	130,000	209.5	58.5	184	Wet	7	18	4
NLI-10/24	Rail	200,000		88	204.5	Dry	10	24	~~---~~ 7.
TN-12	Rail	213,800[a]	265	98	210	Dry	12	32	~~1~~
	~~Rail~~	~~225,000[b]~~	~~283~~	~~98~~	~~210~~	~~Dry~~	~~12~~	~~32~~	
NFS-4[c]	Truck	49,000	214	50	202	Wet	1	2	6
NLI-1/2	Truck	47,500	227.25	42.5	193.25	Dry	1	2	5
TN-8	Truck	78,000	217	68	192	Dry	3	--	~~1~~
TN-9	Truck	77,600	227	68	202	Dry	--	7	1
~~ENL-1300~~	~~Truck~~	~~51,000~~	~~293.25~~	~~51~~	~~238.25~~	~~Wet~~	~~1~~	~~3~~	

[a]~~Cavity length 180 inches.~~

[b]~~Cavity length 195 inches.~~

[c]~~The NFS-5/6 cask is essentially the same as the NFS-4 cask, except that the lead shielding has been replaced with depleted-uranium shielding.~~

words "thus providing two levels of containment" be deleted from paragraph 1, line 4, of the reference section.

b. NRC believes that licensability of the ATMX is not likely in its present form.

5. Section 7.4.4.1

NRC does not "license" DOE shipping containers. Although the Government may be making shipments of transuranic wastes in licensed containers, we cannot confirm this fact. We recommend that the first paragraph on line 7-49 be revised to indicate only that the Government has shipped transuranic wastes (i.e., delete the words "in licensed containers" from the first sentence of the paragraph).

6. Table 7.2 has been updated. A marked up copy is attached.

Chapter 8. Heavy Water Production Facilities

1. This document should be coordinated with "Nuclear Energy System Characterization Data" which shows that HWR's would not be introduced in the United States until 2003. Hence, new heavy water facilities would probably not be required in the United States until 2010 or so. This date should probably be given somewhere in the discussion on page 8-1. In addition, the date of "after the mid-1990's" on page 8-6 should be changed to "about 2005 or so."

Although NRC has no current authority to license heavy water plants, it is impossible to project the licensing authority that might exist in the next century if large numbers of commercially owned heavy water plants were to be built. We recommend that the words "currently" or "at this time" be added to the Statement on page 8-9 concerning licensing status, so that the statement would read: Currently (or at this time), heavy water production plants are not subject to NRC licensing regulations.

527 249

Chapter 9. International Fuel Service Centers

1. Section 9.1

   a. The assumption (paragraph 2) that the single reprocessing plant at an IFSC is a multipurpose (Purex, Thorex) plant is a very key assumption and highly difficult to be implemented. Its basis should be discussed as noted below. Nowhere in Chapter 9 is there any discussion of the need to develop a conceptual multipurpose reprocessing plant, or any delineation of the potential problems in the operation of such a plant. If the concept of an IFSC depends strongly on the ability of a single reprocessing plant to process many different fuels, then this fact should be stated, and the program necessary to determine the plant characteristics and reprocessing costs should be delineated.

   b. Although the title of the Chapter is "International Fuel Cycle Centers," whether or not any of the off-site reactors would be foreign reactors is not made clear.

527 250

The data presented in this section include cumulative spent fuel storage requirements through the year 2025. From this presentation it appears that no Federal waste repository would be available to accept spent fuel until that year. In addition, although numerous reactor types are covered and there might be different spent fuel requirements for one or more of the types, no provision has been made for separate reprocessing and/or spent fuel storage facilities based on fuel differences. It would appear to be helpful to present such a breakdown for these types of required facilities.

Section 7. Summary Tables and Graphical Comparisons

The cumulative U_3O_8 requirements and commitments to the year 2025 in many cases seem to be considerably in excess of known and predicted economic resources for this material available to the United States. It would certainly be helpful to indicate how the analysis of the various cases compare with predicted uranium availability since it would appear that the cumulative U_3O_8 commitments for all of the high energy demand cases from once through and thermal recycle seem beyond normally accepted ranges of uranium availability.

In all cases studied either the fissile plutonium stockpiles or the U-233 stockpiles are very large by the year 2025; in fact, they amount to hundreds to thousands of tonnes. The once through fuel cycle appears to create the largest plutonium fissile material stockpiles. Although thermal recycle (of uranium only) reduces the plutonium stockpile by a factor of about 2, it does so by producing a large amount of fissile U-233 and results in even larger stockpiles. The introduction of the breeder seems to reduce the fissile material stockpiles by the greatest amount. Since stockpiles of fissile materials are one concern of proliferation resistance, consideration of this aspect of proliferation resistance would apparently tend to favor introduction of the breeder. Since the objective of the NASAP work is to identify ways of improving proliferation resistance, it would appear that some of the cases should have been designed to minimize stockpiles of fissile materials and thus be responsive to this proliferation resistance concern. There has been apparently no effort in this regard and this appears to be a key weakness of this very voluminous report.

Summaries of fuel fabrication and spent fuel requirements are noted for the year 2025. Since a number of different reactors are involved, many types of fuel will also be involved and different types of fuel fabrication plants and spent fuel storage facilities will undoubtedly be required. The needs for the different types of facilities should be delineated.

PRELIMINARY COMMENTS ON PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENTS FROM A SAFEGUARDS PERSPECTIVE

Introduction

NRC has reviewed the Preliminary Safety and Environmental Information Documents (PSEID's) from a safeguards perspective, and found that these volumes do not adequately address safeguards concepts, systems design, operations or issues.

We find the technical descriptions and assessments of the various fuel cycle processes and reactor concepts provided in the PSEID's do not provide a basis for direct, specific comments on known or suspected safeguards issues and problems that could arise from implementation of these systems on a commercial basis. Our comments concerning safeguardability of these nuclear systems are, therefore, for the most part generic in nature. We note that a companion document assessing safeguards issues will be issued as a separate report by DOE (PSEID's, Foreword, page i, paragraph 4). We believe that this document, as described, should provide a firmer basis for NRC review of specific safeguards issues. To the extent possible, we have applied the approach outlined in the "NRC Review of Safeguards Systems for NASAP Alternative Fuel Cycles," (NRC proposed revision dated April 27, 1979, of the DOE draft, same subject, dated February 13, 1979) as the basis for this review of the PSEID's. For your convenience, a copy of this document is attached (see Enclosure 1).

The following section, Generic Safeguards Issues, focuses on the major features that distinguish the NASAP candidate fuel cycles from each other and from the reference uranium-plutonium fuel cycle. The objective is to identify known or suspected safeguards issues and problems which might have an impact on licensability of these nuclear technologies, on research and development needs, costs, or significantly delay their commercial implementation. Of course, these findings are preliminary and should not be interpreted as commiting the NRC to specific positions in future licensing actions. Specific comments keyed to the PSEID's are provided in the final section of this paper.

Generic Safeguards Issues

A. Spiking

The main safeguards issue presented for NRC consideration is the relative effectiveness of spiked fuel as a safeguards measure against nuclear weapon proliferation which might evolve from exported fuels (Appendix A, Volume III). Exported unirradiated fuel which affords nuclear weapon potential would be spiked to emit radiation levels comparable to spent reactor fuel. If diverted by the importing country for use in nuclear weapon development, the fuel must first be processed remotely before such use. It is hypothesized that the difficulty and time imposed by the remote processing requirement associated with spiked fuel would be equivalent to that for obtaining weapon grade material from spent reactor fuel. This stated belief, in light of the purpose of the alternate fuel cycle studies, gives the impression that potential weapon grade material contained in spent reactor fuel or spiked fuel is protected effectively against national diversion to nuclear weapon development.

In a safeguards analysis of this spiking concept for deterrence (as opposed to spiking for purpose of detection), we suggest the foregoing impression be clarified to avoid possible misunderstandings relative to the nature and degree of protection actually gained by spiking. From the standpoint of national capabilities that could be focused upon the problem, remote processing appears to be a relatively minor chemical processing hurdle to overcome for any technologically advanced nation that chooses to do so. India, for example, is the most recent nation to accomplish this, as demonstrated by detonation of a nuclear explosive device fabricated from weapon material apparently obtained from operating reactors. Thus, from a national perspective, the remote processing of spiked nuclear fuel can logically be considered more of a nuisance to solve than a truly effective obstacle against nuclear weapon proliferation.

There is always risk that some nation possessing spent fuel will divert it to nuclear weapon development. The same level of risk would prevail in the case of exported spiked fuel. Thus, rather than providing an effective safeguard barrier against proliferation, spiking normalizes the difficulty of diversion equivalent to that characterized by spent fuel. In short, spiking cannot reduce the risk of diversion to nuclear weapon development; it merely assures that such diversion is not made easier to accomplish than it would be for the spent fuel, which many nations already possess.

NRC staff views presented in NUREG-0414, (Reference 1) "Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat," dated May, 1978, concluded that, "while spiking is technically feasible, other measures...could provide improved safeguards benefits with markedly less potential for societal impact." It must be noted that this study focused on the feasibility of safeguarding a domestic mixed oxide industry from a subnational threat, rather than national diversion, and that international considerations and nuclear proliferation were beyond the study scope. Although the Commission has not addressed this specific policy issue or approved this staff conclusion, the issues raised in NUREG-0414 and other reports (Reference 2, 3, 4) concerning the effectiveness, technical feasibility, practicality and relative advantages and disadvantages of spiking would have to be addressed and answered definitively before a decision to employ this technique could be considered.

Because of the extremely dangerous levels of radioactivity involved, spiking may be expected to have profound effects on technical safeguards and to raise a host of regulatory issues. These are discussed briefly below*.

a. The Effect of Spiking on Material Accountability

The high gamma activity of the spikants would greatly complicate the taking of samples for chemical analysis and somewhat complicate

*Much of the following discussion is drawn from or based on the Brookhaven National Laboratory Report entitled, "Safeguards for Alternative Fuel Cycles," dated February 23, 1979, Reference 2, and other referenced material.

the analysis itself. Fabrication plants would be most affected, since reprocessing plants already have to contend with highly radioactive materials for analysis. Most nondestructive assay techniques would be rendered inoperable by the high gamma levels. This is particularly true of those techniques that rely on measurement of the gamma rays from the plutonium or uranium and of techniques using gamma-sensitive neutron detectors .g., organic scintillators in certain types of coincidence counters). Isotopic assay by gamma spectrometry would be impossible. Measurement of scrap and waste, now best done by nondestructive methods, would be made much more difficult, requiring either a return to the less satisfactory sampling and analysis methods, or the development of new, gamma-insensitive, nondestructive methods. Both quality control and assay of fabricated fuel rods would be much more difficult to perform.

The incapacitation of most nondestructive assay methods would make it very difficult to perform real-time or near-real-time accountability in bulk processing plants. Physical inventory taking would be slowed considerably, both because of the unavailability of nondestructive techniques and because of the difficulty of taking samples for analysis. Operators would be reluctant to take more than the bare minimum of samples, because of the cumbersome nature of the operation.

527 256

As a result of these problems, the material balances would be expected to be less accurate than in the absence of spiking.

Inspection activities, both domestic and international, would be hampered. Nondestructive assay has been used by NRC inspectors to cut down the number of samples taken for verification of inventories. Spiking would therefore increase the sample load sent to the New Brunswick Laboratory for analysis. The high radiation levels from materials on inventory and the requirement for remote operations would make the taking of samples by inspectors more laborious and time-consuming. IAEA inspectors, likewise, depend strongly upon nondestructive assay for verification purposes. The timeliness of both NRC and IAEA verification of material balances can therefore be expected to be reduced as a result of spiking.

It has been suggested that the gamma rays from the spikant could used to aid nondestructive assay. This would require that the ratio of spikant to fissile species be accurately known (to ~½% or better) throughout the fabrication process. The feasibility of such an approach remains to be demonstrated.

b. The Choice and Maintenance of Spiking Levels

If spiking were adopted, NRC would have to determine appropriate spiking levels. These would be incorporated into the regulations governing the operation of processing facilities. The NASAP program has proposed a level adequate to produce a gamma radiation field of 100 R/hr at 1 meter from a 1 kg mass of plutonium two years after separation. This would be achieved by a combination of fission-product retention during reprocessing and the addition of Co^{60} at some point before the product stage.

The ability to provide such spiking levels and maintain them during the subsequent fabrication process has not been demonstrated. There are uncertainties in the fraction of fission-product ruthenium and zirconium that can be retained during reprocessing. The high temperatures used to sinter oxide fuels will require the spikants to be present in non-volatile forms (most cobalt compounds are non-volatile).

The relatively short half-life of the fission products Zr^{95} (64 days) and Ru^{106} (368 days) will require timely recycle of the recovered fissile material, or else the radiation levels will fall below the desired value. Again, Co^{60} would be especially useful in this regard, because of its relatively long (5.27-year) half-life. Spent fuel older than a few years (e.g., 5) will have essentially no Zr^{95} and insufficient Ru^{106} for adequate spiking. Fissile material recovered from this fuel would have to be spiked entirely with Co^{60}. An important factor in considering this spiking scheme is therefore the adequacy of the Co^{60} supply to meet the growing demands of a recycle or LMFBR industry. NASAP has not provided data on this question.

Regulations will also have to be developed for dealing with intermediate or fabricated fuel materials that are stored for long periods of time, allowing the radiation levels to drop below acceptable values. Measures would also have to be taken to ensure the maintenance of adequate spiking levels during or immediately after the purification of dirty scrap.

c. Reconciliation with "ALARA" Philosophy

NRC requires that exposures of the public and workers to radiation from activities of the licensed nuclear industry be kept "as low as reasonably achievable" (ALARA). The use of spiking may increase the exposure of workers and the public under either routine or abnormal (i.e., accident) conditions, and therefore would be difficult to reconcile with this philosophy.

d. Possible Requirement for Environmental Impact Statement

Spiking may have substantial adverse effects on the environment through the production of increased amounts of radioactivity (e.g., Co^{60}) beyond those necessary for the generation of electric power. There may be a significant increased potential for accidental release or exposure. An environmental impact statement may therefore be required. It would have to include a cost-benefit analysis and a consideration of alternatives. Since NASAP arose out of foreign policy considerations (non-proliferation), NRC would have to justify imposing the spiking requirement on the domestic industry primarily to achieve foreign policy goals. Both establishing the authority to act primarily out of these foreign policy considerations and demonstrating that the resulting benefits outweigh the risks (or costs) might be difficult.

In any event, an environmental impact proceeding probably would involve appreciable delays.

e. Physical Security Requirements for Spiked SNM

NRC's proposed upgrade rules would subject irradiated SNM, defined as SNM "not readily separable from other radioactive material and which has a total external radiation dose rate in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding," to the same physical protection requirements at fixed sites as other SNM.

SNM spiked in accordance with the NASAP criterion would presumably come under the upgrade rule. However, spiked fuel materials are not precisely the same as irradiated fuel, since the former may exist in intermediate forms (powders or pellets) which are more easily shielded and transported than whole reactor fuel elements. NRC may therefore consider whether or not there is a need to apply physical security requirements similar to those recently imposed on spent fuel shipments to shipments of spiked SNM, spikants, and high level wastes emanating from production facilities for these nuclear materials.

f. Effect of Spiking on Quality Control of Reactor Fuels

At present the quality control of reactor fuels requires stringent inspection of the fuel rods and assembled elements. This is done by "hands-on" mechanical inspection and NDA scanning for non-uniformities, "rogue" pellets, etc. Spiking the fuel before assembly into elements would require the mechanical inspection to be done remotely, while new NDA methods, not affected by the intense gamma-ray background, would have to be developed. It was noted earlier that these NDA methods are also used to assay the fuel rods and therefore are important for material accountability.

g. Possible Legal Consequences and Potential Liability

The intentional use of a spiking material or radicactive sleeves to provide lethal, or near lethal doses of radiation to a thief who steals or attempts to steal nuclear materials may subject the fuel owner and/or his agent to civil and criminal liabilities. Although this specific issue has not been addressed by the courts, the intentional use of a "dangerous device", "instrumentality", or "trap" has been viewed by courts with disapproval, when the "trap" is used to protect property by exposing the thief to death or serious injury. There may be grounds for a national defense and security exception for use of spikants in this proposed rule, but recent court cases indicate a narrow interpretation and application of this legal concept.

In summary, there are serious questions concerning the legal justification for application of spiking and resultant liability in the event of death or injury caused by nuclear materials spiked by NRC order.

527 261

B. Coprocessing

Coprocessing means the processing of mixtures of uranium and plutonium or their compounds in such a way that the plutonium is always diluted by uranium. Most often the term is used for a possible mode of operation of spent-fuel reprocessing plants in which the product consists of a mixture of uranium and plutonium oxides, co-precipitated from a mixture of nitrates in solution.

Thermal recycle fuels typically consist of mixed uranium and plutonium oxides with a plutonium concentration of 2-5%. Feed to a mixed-oxide fabrication plant would have to be somewhat higher than this to allow for blending; a mixture with 10% plutonium oxide has been suggested. Fast breeder reactors require higher plutonium concentrations; mixed-oxide feed to an FBR fuel fabrication plant would probably have a plutonium oxide concentration of about 25%.

The major safeguards advantages of coprocessing are the increased quantity of material that a diverter would have to take for the same amount of plutonium and the increase in the time and resources required to convert the mi oxide to a form suitable for use in an explosive weapon. The concentration of plutonium in mixed oxides for thermal recycle fuels would probably be too low for direct use in an explosive. This may not be true for FBR mixed oxide feed, with its much higher concentration of plutonium. In both cases the maximum allowable percentage of plutonium would have to be set by NRC regulation, and the values selected would have to be based on a consideration of both the practical needs of the fabrication plants and the explosive utility of mixed oxides as a function of plutonium concentration.

The needs of the fabrication plants for large batches (master blends) of mixed oxides with specific plutonium concentrations and fissile composition would probably requir blending at the reprocessing plant, either in the liquid nitrate or in the converted powder stage. If the former, then large nitrate storage and mixing tanks with associated pumps and piping would have to be provided and safeguarded, possibly as a separate material balance area. Identification of the accountability problems in this area would require detailed analysis.

Scrap recovery facilities processing dirty mixed oxide scrap will have to be operated in a coprocessing mode also. Accountability should be essentially the same as for facilities producing separated oxides.

C. The Use of U-233-Th Fuels

A number of the fuel cycles proposed by NASAP involve the use of U-233-Thorium fuels. Compared with plutonium, U-233 has the advantage that it can be denatured (i.e., rendered unsuitable for direct use in an explosive) with U-238; this advantage is shared by U-235, of course. The use of denatured fuels is discussed in a separate section. This section will concentrate on the general safeguards problems of U-233-Thorium fuels.

Present NRC regulations treat U-233 as similar to plutonium rather than to U-235. Thus, U-233 occuring in any enrichment is treated as strategic special nuclear material (SSNM), whereas uranium must be enriched to 20% or more in U-235 to be so treated. For physical protection, quantities of U-233 are the same as those of plutonium and two-fifths those of U-235.

There is little experience with the commercial reprocessing of highly irradiated thorium fuels. Some fabrication has been performed for the light water breeder reactor program. It is therefore difficult to say at this stage whether present NRC material accountability regulations can be met in commercial size reprocessing and fabrication plants for U-233-Th fuels. Most likely it will be necessary to operate pilot plants owned by or under contract to the Federal government for a period of time in order to gain experience with these materials.

The unique characteristic of U-233 fuels is the high radiation levels associated with the presence of even trace quantities of U-232 and its daughters. The levels are high enough to require remote fabrication. This has the advantage of limiting physical access to the material. However, it also greatly complicates the assay of U-233 by nondestructive techniques, because of the high gamma activity from U-232 and its daughters. The magnitude of this gamma background depends strongly on the age and processing histories of both the U-233 and the thorium in the fuel mixture. For a given amount of U-232 the older the U-233 (i.e., the longer the elapsed time since its last purification) and the thorium the larger is the background. For some U-232 concentrations and ages likely to be encountered in any U-233 recycle program, this background will completely swamp the gamma rays from U-233. Large backgrounds will be produced in any gamma-sensitive detector, whether or not used for gamma detection (e.g., organic scintillators used for neutron detection). Nondestructive assay techniques will therefore have to be developed for any fuel cycle using U-233. Some effort along these lines has already been made in the HTGR recycle program[5] but it was primarily of an exploratory nature. The feasibility of performing real-time accountability in U-233 fabrication plants will depend on the successful outcome of such efforts.

Accountability in reprocessing plants for U-233-Thorium fuels would be less affected by the radiation from the U-232 decay chain because most assay in plants of this type is by standard chemical analysis, and radiation levels in much of the process, due to fission-product activity, are already very high. The more difficult chemistry of thorium may cause problems for accountability because of its tendency to polymerize in solutions.

The verification activities of NRC inspectors will be hampered by the high radiation levels in U-233 fuels. As with spiked fuels (but to a lesser degree), the taking of samples will be laborious and time-consuming, and the samples will have to be sent off-site for analysis, with an attendant loss of timeliness.

Physical security for U-233 fuels should be better than for plutonium fuels because of the remote nature of the fabrication process and because of the abundant and penetrating gamma rays from the U-232 daughters (principally, those from Tl^{208}), which should result in a greatly increased sensitivity of detection by portal radiation monitors.

In summary, a great deal of development and demonstration of accountability techniques will have to be done for U-233-thorium fuels before it can be shown that NRC regulatory requirements can be met, particularly if those are extended to include real-time accountability.

D. <u>Denaturing</u>

Denaturing may be defined as the addition of a non-fissile isotope to a fissile isotope of an element in such proportions as to make the fast critical mass of the mixture impractically large for a nuclear explosive weapon.

Since all the isotopes of plutonium have appreciable fast-fission cross sections, plutonium cannot be denatured. The fast-fission cross section of U-238 is low enough, however, to allow the fissile isotopes U-233 and U-235 to be denatured by its addition.

The choice of a threshold enrichment for denaturing is important. It will be noted that the definition given above does not imply a sharp enrichment cut-off. Such a cut-off could be defined as the enrichment at which the fast critical mass becomes infinite, but this choice would limit the use of U-233 to enrichments in the neighborhood of 3% and U-235 to those in the neighborhood of 5%. NRC regulations define a threshold enrichment of 20% for U-235-bearing materials to be considered strategic special nuclear material, subject to the full requirements for physical security. This corresponds to a bare spherical critical mass of 850 kg of U. The enrichment in U-233 at the same critical mass is about 12%, which is usually assumed to be the threshold enrichment for denaturing of U-233 fuels in NASAP studies. The use of appropriate reflectors may substantially reduce the total mass of a nuclear explosive, however, and NRC may want to review the data for U-233 before selecting an enrichment limit for uranium containing this isotope.

527 266

Enrichment limits for uranium containing both U-233 and U-235 may also have to be set. Another consideration that may enter into setting threshold enrichments for uranium containing U-233 is the greater ease of separating this isotope from U-238, compared with that of separating U-235.

The effect of the decay of U-232 and its daughters on the nondestructive assay of U-233 fuels has been noted in the previous section. This effect will occur in denatured U-233 fuels as well, of course, and will subject material accountability for these fuels to all the disadvantages already noted. However, since by definition denatured fuels are not useful for nuclear explosives, the consequences of the somewhat lower accuracy of material balance and the impairment of the prospects for real-time accountability are not as serious.

In some of the fuel cycles involving denatured U-233 fuels, such as the LWR, substantial quantities of plutonium appear in the spent fuel. The fuel will therefore have to be reprocessed by a combination of the Purex and Thorex processes. Very little, if any, experience in reprocessing such fuels exists, and therefore it is very difficult to say how well NRC's accountability requirements can be met in such a reprocessing plant, at least without detailed study. Certainly the chemical analysis of such mixtures will be more difficult than that or ordinary spent LWR fuels.

The disposition of the plutonium separated from spent denatured fuels of this type is also important. It may be either stored, for eventual use in the fast breeder reactor cycle, or recycled in "secure" energy centers. In the former case, neither the form of nor the responsibility for storage has been worked out. If the Federal Government accepts responsibility for storage, NRC may not have a safeguards role. If storage is in licensed facilities, the safeguards problems will be, generally, those already considered in the GESMO proceeding. Accountability for plutonium in storage is particularly simple if it is stored in discrete containers, each containing a few kg of Pu. Surveillance devices could be incorporated to give an instantaneous alarm in case of tampering.

If the plutonium recovered from spent denatured fuel is recycled in energy centers, the safeguards technical problems are essentially the same as for the U-Pu cycle, with the modifications associated with the physical and administrative nature of energy centers. The safeguards regulatory issues involved in the operation of a multinational center are a related, but separate issue. An additional complication would arise from the occurrence of non-denatured U-233 in the blanket of a Pu-U-Th breeder, but the U-233 could be denatured during the recovery process or shortly thereafter.

To conclude, the major safeguards technical problems associated with denatured U-233 fuels are those common to any fuel using U-233, discussed in a previous section; the lack of experience with the reprocessing of mixed U-Pu-U-233 fuels and refabrication of the denatured fuel. An important regulatory issue is the threshold enrichment at which U-233 is considered to be denatured.

E. <u>The Use of Heavy Water as a Moderator</u>

a. Introduction

One of the alternative fuel cycles under consideration in the NASAP program is based upon the use of heavy-water reactors (HWR's). There are two important safeguards problems associated with the use of this type of reactor; the availability of heavy water in large quantities, and on-line refuelling.

The significance of heavy water for safeguards is that it can be used to moderate reactors fueled with natural uranium, and these can be used to produce plutonium. A substantial commitment to the heavy-water reactor fuel cycle in the U.S. would probably, therefore, require the imposition of safeguards on heavy water, not now required by NRC regulations. Safeguards would be required on the heavy water in reactors, in the concentrators for contaminated (i.e., light-water diluted) heavy water, in production facilities, and in storage. Safeguards would consist mainly of material accounting and surveillance and containment. Since heavy water cannot be used directly in an explosive and is not highly toxic, physical protection may not be required for safeguards purposes. However, the tritium content of irradiated heavy water presents a radiological safety hazard.

The safeguarding of nuclear material at heavy water power reactors is generally accepted as more difficult than at light water reactors because of the following considerations:

- High frequency of refuelling
- Large number of fuel bundles in the inventory

527 269

- Small size of individual fuel pins and bundles
- Inaccessibility of the core (and sometimes the spent fuel) for verification purposes.

Almost all heavy water power reactors are refuelled on a continuous basis without reactor shutdown. Spent fuel is removed and fresh fuel is added during the operation by means of remotely controlled refuelling systems. The storage of spent fuel in storage baskets, often stacked in close-packed three-dimensional arrays, makes fuel bundle counting and verification difficult, if not impossible. The continuous refuelling process places an inherent limitation on the use of seals for safeguards purposes. Further, the large inventory, the frequency of refuelling, and the resultant inventory turnover imposes a need for more safeguards attention than required by light water reactors.

The IAEA is concerned with the possibility of misuse of heavy water reactors for undeclared irradiation of reactor fuel, especially natural uranium. Considerable reliance is placed upon optical surveillance, item counts, and periodic checks of operating records to detect any undeclared activities. Considerable R&D is being sponsored by the IAEA to develop systems and components for upgrading safeguards capabilities for heavy water production plants, storage facilities, and reactors[6].

If NRC is to require safeguards on heavy water it must decide on the minium amount of heavy water of safeguards significance, and the threshold concentration of D_2O in water for safeguards to apply.

527 270

Since heavy water would be safeguarded solely in the interests of non-proliferation, the values of these parameters should be at least consistent with international commitments. Safeguards on heavy water are not required under the NPT-INFCIRC/153 system of the IAEA, but may be under bilateral or trilateral agreements or voluntary submissions. Historically, the IAEA has accepted the responsibility for applying safeguards to heavy water when so requested. It is not regularly safeguarded under NPT because heavy water is not included in the definition of nuclear material by the IAEA. The technology of safeguarding heavy water is not well developed, and the ease of evaporation of water is a special problem.

It should be noted that safeguards, including accountability, are required by the Department of Energy for heavy water under its control[7].

b. International Safeguards for Heavy-Water Production Facilities

For heavy-water production facilities of commercial size (at least 200 Te of D_2O per year), present accountability techniques appear to be too inaccurate to detect the diversion from the extraction process of the minimum quantity ($\sim$10 Te) of D_2O required to supply the initial inventory of a small plutonium production reactor (annual production rate 8 kg Pu). Safeguarding such a plant would therefore require improved accountability techniques or increased reliance on surveillance and containment. This conclusion is tentative, since a careful analysis of the material balance problems in such a plant has not been done.

527 271

The finishing process, because of its much smaller flows, is more amenable to material balance techniques, and it appears that present methods have sufficient sensitivity to detect the diversion of significant quantities of D_2O. Improved design of this part of the process could reduce the present uncertainties even further. Surveillance and containment techniques would have to be developed to detect undeclared feed or product.

Because of the extremely large flows through such plants, NRC inspection would be facilitated by on-line, recording, flow and assay devices for feed, product, and waste. Portable nondestructive assay instrumentation for the measurement of concentration would also be useful for inspection purposes.

It appears that applying safeguards to such a plant would involve problems different from those NRC has encountered in other types of safeguarded plants so far. Considerable development of criteria and methods for safeguarding large plants of this type would have to be undertaken if they were to become a reality in the U.S. licensed industry. Such development could profit from the experience of DOE and the Canadians in this area.

c. Safeguarding of Heavy Water in Storage Facilities

D_2O is usually stored in 55-gallon drums. A storage facility may contain hundreds or thousands of these. NRC would have to develop sampling plans and methods for verifying the content of the drums. Portable instruments for verification would greatly reduce the timeliness of detection.

Substantial amounts of D_2O may also be stored at or near reactors. The problems of accountability and verification will be similar.

In general, the technical problems of safeguarding storage facilities for D_2O would appear to be small compared with those for production facilities; however, the economic and operational impact would probably be significant.

F. Storage of Spent Fuel

A once-through fuel cycle implies the indefinite, perhaps permanent, storage of spent reactor fuel in repositories. According to the Energy Reorganization Act, NRC would have the responsibility for regulating the health, safety, and safeguards aspects of such repositories, even if operated by the Federal Government or a multinational agency. Under present IAEA regulations, there would appear to be no grounds for terminating safeguards, unless spent fuel were classified as residues. NRC would have to ensure that safeguards on such repositories were carried out in a manner consistent with IAEA requirements.

Safeguards for indefinitely stored, retrievable spent fuel would consist primarily of periodic assurance of the presence and integrity of all fuel elements. This would be accomplished by a combination of sealing and surveillance operations. The large number of such elements at a central repository would put a premium on the ability to seal off groups of elements or whole sections of the repository. If a seal were broken (as could happen accidentally) it would be necessary to re-inventory the affected area. This would imply a capability for close inspection, either visually or instrumentally (e.g., by radiation signatures). The ability to do this would depend on how the elements were stored (whether in air, on the surface or underground, etc.). Possible problems can be identified only for specific storage schemes.

Irretrievable (presumably underground) storage does not appear to pose any serious domestic safeguards problems. It is barely conceivable that a non-governmental adversary could gain access to fuel stored in this manner, and periodic inspection should detect any serious attempt. What the requirements of the IAEA would be for the safeguarding of irretrievably stored fuel are unknown at this stage, so an assessment of the problems of the NRC in assuring compliance is premature.

G. Material Control and Accountability (MCA)

Although the subject area has been addressed, to some extent, in previous sections, the following preliminary comments summarize the major NRC staff concerns with regard to potential MCA safeguards issues and problems.

a. The NASAP PSEID's assume that dynamic real-time accounting will replace periodic clean-out physical inventories. NRC has not yet determined that such a system will be a suitable replacement for the assurance intended to be provided by physical inventory requirements.

b. Research and development will be needed to develop ways of assessing the adequacy of automated MC&A systems. The NASAP reports discuss the need for new NDA instrumentation for materials not currently measured in high throughput facilities, and for remote, shielded measurements. To this should be added research and development on measurement control procedures (calibration, standards, maintenance) for such instruments.

c. Accountability problems of hold-up and clean-out in high throughput facilities, particularly in radioactively hot processes, need to be examined in detail before licensability can be determined.

d. The PSEID's do not provide sufficient safeguards data to determine whether or not proposed MCA procedures (Volume VII) would meet current MUF/LEMUF regulations for fuel fabrication and reprocessing facilities.

H. Institutional and International Considerations

This section discusses known or suspected safeguards issues and problems which could arise from implementation of U.S. laws or U.S. international commitments. Some of these instruments are in place (e.g., the Nuclear Non-Proliferation Act of 1978), some are being developed (e.g., the US/IAEA Safeguards Agreement), and still others merely contemplated (Multinational Fuel Cycle Centers). As a result, the following section is, in large part, speculative.

Implementation of an alternative fuel cycle would directly affect import and export licensing requirements and could require additional safeguards measures. Given such implementations the export licensing requirements imposed by the NNPA and, perhaps, NRC regulations (10 CFR Part 110 and other applicable Parts) altered or changed.

In the international area, ongoing and future moves to strengthen IAEA safeguards might lead to more stringent measures being emplaced by the IAEA. The selection of an alternative fuel cycle and of appropriate safeguards variations should take into account how current IAEA techniques may have to be altered to accommodate alternative nuclear materials and processes. In addition, US/IAEA interaction through the Safeguards Agreement may necessitate changes in the U.S. domestic regulatory structure.

Potential issues and problems that might arise from implementation of a Multinational Fuel Cycle Center (MFCC) are difficult to characterize in detail. There is a wide spectrum of possible institutional arrangements within which an MFCC could be implemented involving varying degrees of participation and authority by national, international, and corporate entities. The central issue is the problem of defining the organizational participation and defining the lines of authority and responsibility. It appears necessary to study and define what Congressional and agency positions would be necessary to establish an MFCC. In addition, the major issues involve the role and authority of U.S. agencies and interagency elements (if needed). Since the principal incentive for joining an MFCC is assurance of nuclear fuel, it is necessary to determine what types of fuel assurances should be given. Further, issues arise with respect to marketing commitments, privileges and immunities which may be needed, the need to transfer sensitive technology (yet adequately protect classified and proprietary information), whether sanctions should be employed for noncompliance, and, if so, their nature. Admittedly this listing is incomplete and is intended only to highlight major areas of concern.

527 276

Specific Comments

PSEID, Fuel Cycle Facilities, Volume VII

Chapter 5, Reprocessing

The need for an irradiation facility for coprocessing and pre-irradiation of spiked nuclear fuels and for special facilities for receipt, storage, and handling of spikant material are discussed in section 5.4 and 5.5, respectively. Information on the operational characteristics of these processes and facilities are not provided. Such information is needed to provide a basis for judgement concerning their potential safeguardability and licensability.

Chapter 7, Transportation

The implementation of an alternative fuel cycle using spiked fuels or mechanically attached radioactive shields would introduce large amounts of additional radioactive materials and radioactive material shipments into nuclear commerce. The data presented imply fuel shipments could increase by factors of 7 to 10 for the spiked fuel cycles as compared to the reference cycle (section 7.2). The shipping requirements to support production fabrication and distribution of spikant materials and sleeves would be superimposed on the spiked fuel shipments noted above. These shipments would present potential additional targets for sabotage. Although this chapter addresses the potential risk of radiological release that might develop from accident conditions, the potential consequences from an act of sabotage are not considered.

NRC and DOE have ongoing research programs to characterize the potential radiological source term that could arise from explosive breaching of a loaded LWR spent fuel shipment and to estimate potential health consequences. It is uncertain as to whether or not these efforts will provide sufficient information as to source term and health consequences of explosive breaching of spiked fuel shipment or a shipment to which a spikant sleeve has been mechanically attached. Additional research and development may be required in this area.

527 278

REFERENCES

(1) U.S. Nuclear Regulatory Commission, NUREG-0414, Safeguarding a Domestic Mixed Oxide Industry Against a Hypothethical Subnational Threat, May, 1978.

(2) Brookhaven National Laboratory, Draft Report, NUREG-25557, Safeguards for Alternative Fuel Cycles, by J. J. Cadwell and J. N. O'Brien, February 23, 1979.

(3) Brookhaven National Laboratory, The Spiking of Nuclear Materials as a Safeguards Measure, Volume I, E. V. Weinstock, September, 1975, (CONFIDENTIAL).

(4) Brookhaven National Laboratory, Modification of Strategic Nuclear Materials to Deter Their Theft or Unauthorized Use, November 6, 1975 (CONFIDENTIAL).

(5) ORNL-TM-4917, Conceptual Design of the Special Nuclear Material Nondestructive Assay and Accountability System for the HTGR Fuel Refabrication Pilot Plant, by J. D. Jenkins, S. R. McNeany, and J. E. Rushton, July 1975 (Oak Ridge National Laboratory).

(6) Brookhaven National Laboratory, Program for Technical Assistance to IAEA Safeguards - Heavy Water Accountability, C. Auerbach et al, revised September 20, 1978.

(7) DOE Interval Management Directive 6104-A, page 8.

ENCLOSURE

NRC REVIEW OF SAFEGUARDS SYSTEMS FOR NASAP ALTERNATIVE FUEL CYCLE MATERIALS

Background

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage and theft. NRC has considered whether strengthened physical protection may be required as a matter of prudence. Proposed upgrades to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgrade rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NASAP Safeguards Basis

The desired basis for NRC review of safeguards systems for NASAP alternate fuel cycle materials containing significant quantities of strategic special nuclear material (SSNM)*, greater than 5 formula kilograms**, during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and

* $\geq$20% ^{233}U in uranium, $\geq$12% ^{233}U in uranium, or plutonium.

** formula grams = grams contained ^{235}U + 2.5 (grams ^{233}U + grams plutonium); ref. 10 CRF 73.30.

527 280

the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, some have proposed to increase radioactivity of strategic special nuclear materials which are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh fuel material to assure that during the period after export from the U.S. and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remove reprocessing, the difficulty and time requirement to obtain material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirement imposed by the Nuclear Non-Proliferation Act of 1978 include application of IAEA material accountability requirements to nuclear related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative which could be used to provide additional safeguards protection against diversion of shipments of SSNM by sub-national groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC Review

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. The fuel cycles under consideration should encompass use of both fuel materials with no added radiation and those to which radioactivity purposely has been added. The relative effectiveness of various safeguards approaches such as upgraded physical protection, improved material control and accountancy, dilution of SNM, decreased transportation requirements, few sites handling SNM, and in-reased material handling requirements as applied to each fuel material type should be assessed. The evaluation should address, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb making purposes; the relative impacts on domestic and on international safeguards; the impact

527 282

of radioactive contaminants on MCA detection, measurement and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tamper or defeat; the increased public exposure to health and safety for acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation we request that NRC assess the differences in the licensability of the domestic facilities, transportation system to the port of entry of the importer, and other export regulations for those alternative fuel cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, on the international and national safeguards systems of typical importers in protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, should be described in detail as well as the potential advantages in detection or deterrence. A clear identification of the potential role, if any, that added radioactivity could or should play is wanted, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its use by foreign countries to obtain material for weapons purposes.

527 283

Licensability issues that must be addressed by RD&D programs also should be identified.

The following table presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import and export:

Fuel Material Type	Minimum Radiation Level During 2 Year Period (Rem/Hr @ 1 meter) (Ref. 6)	
	Mixed[1]	Mechanically Attached[2]
a. PuO_2, $HEUO_2$ powder or pellets[3]	1,000 per kgHM	10,000/kgHM
b. PuO_2-UO_2 amd $HEUO_2$-ThO_2 powder or pellets[3]	100 per kgHM	10,000/kgHM
c. LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
d. LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[1]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[2]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[3]HEU is defined as containing 20% or more ^{235}U in uranium, 12% or more of ^{233}U in uranium, or mixtures of ^{235}U and ^{233}U in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are described further in the references.

527 284

Table I

CANDIDATE METHODS AND RADIATION LEVELS FOR SPIKING FUEL MATERIALS

Fuel Material Type	Minimum 2 year Radiation Level, rem/hr at 1 meter	Process	Minimum Initial Radiation Level, rem/hr at 1 meter	Reference
a. PuO_2, $HEUO_2$ powder or pellets	1000/kgHM	. Co^{60} addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	. Co^{60} addition	130/kgHM	2, 3, 5, 6
		. fission product addition (Ru^{106})	400/kgHM	2, 3, 5, 6
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40/MWD/MT)	1000 (30/day)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/as[illegible]	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40 MWD/MT)	1000 (30 day)/ assembly	4

527 285

References

1. NUREG 0414, "Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat"

2. ORNL TM-6412, "Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence," J. E. Selle

3. ORNL TM-6483, "Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence," J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen

4. GEFR 00402, "Pre-Irradiation Concept Description and Cost Assessment," G. F. Pflasterer and N. A. Deane

5. IRT-378-R, "Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use," (Vol 2 of "The Spiking of Special Nuclear Materials as a Safeguards Measure," BNL File No. 5.9.1 for Vol 1)

6. SAI-01379-50765, "Material Radiation Criteria and Nonproliferation," E. A. Straker, January 8, 1979

Preliminary NRC Comments
NUCLEAR ENERGY SYSTEM CHARACTERIZATION DATA

NRC has reviewed the subject document, the major portion of which is devoted to presentation of numerical data. Our comments are directed generally to the rather limited amount of analysis presented in the document and the lack of consideration of the objectives of improving proliferation resistance and making use of available resources.

Section 1. Introduction

In this section three overall fuel cycles are delineated as follows:

a. The once through fuel cycle is defined and is properly identified as being limited by uranium availability and producibility. The improvement of enrichment technology is noted as a potential enhancement for the once through fuel cycle. While this may be true from a resource availability standpoint, no mention is made of potential important considerations with regard to proliferation and safeguards aspects of such enrichment improvements.

b. The second fuel cycle outlined is the uranium recycle option. While this cycle is theoretically a potential alternative, it is only being considered by those countries that plan large and aggressive breeder programs and must be associated with facilities for the long term storage of plutonium which are not mentioned. The economic and proliferation resistance implications of this aspect are not described or mentioned in the document. Another method of utilizing this fuel cycle would include the disposal of plutonium. Also implicit in this fuel cycle is the effect of the build-up of U-236. Neither of these aspects has been mentioned.

c. The third fuel cycle considered is the uranium and plutonium recycle case. All modes of recycle in this case seem to be based on two theoretical safeguards regimes. One regime assumes universal worldwide safeguards and is apparently based upon technological measures such as denaturing, spiking, dilution, etc., which are apparently considered adequate deterrents from a proliferation resistance standpoint. The second safeguards regime assumes two safeguards regions, one region with no restrictions and one region where only denatured, spiked or diluted fuel can be utilized.

   As a result, the entire concept of uranium and plutonium recycle seems to depend upon the acceptability of technological measures as deterrents to proliferation resistance. Based upon progress to date in the INFCE work, it appears that such technological measures are judged to have little potential for preventing national proliferation and are only considered appropriate for sub-national diversion. On this basis the discussion of this fuel cycle seems quite inadequate.

527 287

Section 2. Energy Demand Forecast

This section discusses numerous scenarios and seven different forecasts are noted. However, no mention is made of the forecast developed for INFCE by U.S. and other nations and how that forecast compares with the suggested NASAP forecast. It would certainly be helpful for this comparison to be included.

Section 3. Reactor Design and Fuel Management Data

The amount of data included in this section is too voluminous for any detailed review by NRC in the short time available. However, it is noted, based on a rough approximation, that the so called 15% improved LWR seems to result in about a 10% improvement in U_3O_8 consumption, and the so called 30% LWR results in about a 25% U_3O_8 improvement. If this is correct, it would appear that the terminology should be changed.

Section 4. Reactor Introduction Dates

This section is limited to one page listing projected reactor introduction dates. No explanation or rationale is provided and it appears from the one page of data that no introduction of new reactor designs is projected before early in the next century. On this basis effects on resources utilized and new facility requirements should be minimal for the next 20-25 years.

Section 5. Reactor Market Penetration and Phase Out Rates

This section is also a one page section which merely lists reactor market penetration for so called nominal or rapid deployment scenarios. No rationale is provided for this table. However, it seems quite optimistic in that it predicts for new plants a 50% penetration in 11 years and essentially 100% penetration in about 30 years. This seems overly optimistic in view of the present situation with light water reactors.

Section 6. System Analysis Results

The two-sentence introduction indicates that the tables include ore requirements; however, in actuality U_3O_8 requirements are the parameter discussed. Consistent use of terminology should be employed here. For cases A.2.2 and A.2.3 the analytical results indicate that improved LWR fuel can be introduced into all reactors, both existing and projected, within a five-year period from 1985-1990. This would appear to imply complete industry penetration within five years. In view of development, demonstration and regulatory requirements, it certainly would appear that such a fast penetration rate would be overly optimistic.

527 288

In Section 7.2 several of the graphical comparisons of cumulative resources requirements apparently show a maximum point. This is apparently in error since cumulative requirements cannot be a decreasing function under any circumstances. In general, we believe that the function of the Summary Section would be better served if an attempt were made to reduce the number of graphs (over 40) and to provide an expanded analysis of the results.

527 289

ATTACHMENT 4

NRC REVIEW OF SAFEGUARDS SYSTEMS FOR NASAP ALTERNATIVE FUEL CYCLES

In response to the request by the Department of Energy, the NRC staff has reviewed and commented on the subject document. The Office of Nuclear Material Safety and Safeguards (NMSS) staff comments have been consolidated and are presented below.

The section entitled General Comments defines the scope and approach to be employed by NRC for safeguards and licensability evaluation of the NASAP. The section entitled Specific Comments provides substantive and editorial comments on the subject document. For convenience, NRC comments are shown as annotations to the DOE proposed NRC Review of Safeguards Systems for NASAP Alternate Fuel Cycle Materials in Enclosure 1 and the NRC revised version is provided as Enclosure 2. Of the other Offices contacted, the Office of Inspection and Enforcement (IE) submitted comments which are presented as Enclosure 3, and the Office of Standards Development (SD) comments are provided as Enclosure 4.

We believe the revised Safeguards Review Basis sets forth the needs and objectives for both DOE and NRC and will be acceptable in its entirety to DOE. We are now applying these criteria in our ongoing review of NASAP. In the event DOE staff has questions concerning the revision or wishes to discuss the NRC comments, we urge them to do so at their earliest convenience.

General Comments

NRC will utilize its existing framework of regulations and proposed upgrade rules as the basis for its review of the DOE NASAP alternative fuel cycle systems and its reports to the President and cognizant Congressional Committees of findings of known or suspected licensing issues and problems associated with commercial implementation of these alternative technologies. As noted in Chairman Hendrie's letter to Secretary Schlesinger, dated June 9, 1978, the NRC review of NASAP will include a comparative evaluation of safety and environmental as well as safeguards issues and consideration of licensability aspects for each of these areas.

527 290

In addition to 10 CFR 70 and 10 CFR 73 which define requirements for material control and accounting and for physical protection of plants and SNM, respectively, the proposed physical protection upgrade rule (43 FR 35321) is currently being considered by the Commission and, upon approval, will be published in effective form. In the area of material control and accounting, a preliminary assessment of potential future issues will be made using the recommendations set forth in the NRC Report of the Material Control and Accounting Task Force, NUREG-0450, dated April, 1978. This source must be used with care, however, in that specific Task Force recommendations have not as yet been incorporated into the regulatory base and the Task Force did not address the applicability of their recommendations to new types of facilities.

Although NRC is in general agreement with the scope of the NRC review as defined by DOE in the subject document, we believe that undue emphasis has been placed in evaluation of the added radioactivity fuel materials types. To provide a more balanced approach, we have prepared two amendments to the subject document (see Enclosure 2, inserts, pages 2a and 2b) which note that safeguards measures other than added radioactivity will be evaluated during the course of the study and that a comparative evaluation of unadulterated versus added radioactive fuel types will be performed. We believe the suggested amendments are in accord with DOE intent and accurately reflect our mutual views and objectives.

The NRC review of NASAP will focus on, but not be limited to, consideration of the relative costs and benefits from application of various safeguards approaches to protection of fuel cycles utilizing unadulterated nuclear fuels, with and without mechanically attached radioactive sleeves, and nuclear fuels intermixed with radioactive contaminants in terms of;

- the degree to which added radioactive contaminants provide protection against theft for bomb making purposes,
- the relative impacts on domestic safeguards,
- the impact of added radioactivity fuel types on international safeguards,
- the impact of radioactive contaminants on MCA detection, measurement and accuracy,
- the availability and process requirements of radioactive contaminants,
- the vulnerability of radioactive sleeves to tamper or defeat,
- the increased public exposure to health and safety from acts of sabotage, and
- the increased radiation exposures to plant and transport personnel.

527 291

Specific Comments

As noted above, Enclosure 2 provides an annotated NRC Review of Safeguards Systems for NASAP Alternative Fuel Cycle Materials which indicates the nature of suggested amendments. The following provides a line by line index of NRC amendments to the subject document reflecting specific comments and questions raised by NRC staff.

page 1, lines 3, 4, 6, 8, 9, 10, and 11

These are editorial insertions or deletions as shown in Enclosure 2.

page 1, footnote 1

This definition of HEU is inconsistent with 10 CFR 73.30 and 73.50, the proposed upgrade rule, and IAEA INFCIRC 225 (Revision 1). While NRC could employ this definition as a standard for the evaluation, the applicability of findings would be based on presumed future changes to the NRC regulations and to International Physical Protection Conventions.

page 2, lines 1 and 2

Delete the typographic error.

page 2, line 3

We suggest insertion of the paragraph shown on page 2 (a) for reasons cited in the General Comments section.

page 2, line 3

Provide a new paragraph which begins, "To maintain...".

page 2, line 6

Change "add radioactivity to" to "increase radioactivity of".

page 2, line 10

Typographical error, as shown.

page 2, line 11

Suggest deletion of "non-radioactive".

page 2, line 15

Change "of" to "imposed by".

page 2, lines 16 and 17

Editorial insertions and deletions, as shown.

page 2, line 21

Change "An" to "Another proposed".

page 3, line 1, text

We suggest insertion of the paragraph shown on page 2 (b) for reasons cited in the General Comments section. This inserted paragraph incorporates the last paragraph, page 4 and page 5 which are to be deleted. The intent is to made clear that the NRC review will evaluate a spectrum of safeguards measures as they relate to both unadulterated and adulterated fuel material types.

page 3, line 2

We suggest line 1 be changed to read as follows: "As part of this evaluation we request that NRC assess the differences in the...".

page 3, line 6

Insert "U.S. domestic safeguards, on", as shown.

page 3, line 16

We suggest deletion of the sentence, "the basis for current 10 CFR 73.6(b)...".

This exemption was based on engineering judgment. The relative impact of increased levels of radioactivity as shown in the table on page 4 of the Basis will be considered during the course of this evaluation.

page 3, line 20

Editorial insertion.

page 3, lines 21 and 22 (paragraph 2)

We suggest the change as shown in Enclosure 2 for the purpose of emphasizing the study approach to evaluation of both adulterated and unadulterated fuel material types.

527 293

page 4, table, heading

The selection of two year period for maintenance of minimum radiation levels appears to be a rather short time frame. Since material can be held for four or five years and used with no health and safety problem, it is not clear how the time period was selected or on what basis it can be defended.

page 4, Table

We suggest for clarity that uncommon designations such as type b fuel and HM be defined.

page 4, footnote 3

The DOE definition of HEU restricts the inclusion of U-233 to uranium that is equal to or greater than 12% U-233. The NRC definition of formula quantity includes U-233 at any enrichment. The DOE definition does not provide physical protection for certain isotopic compositions of uranium which could be used for fabrication of a CFE. For the purposes of the NASAP review, NRC proposes to evaluate safeguards for U-233 at an enrichment level of 10% or greater.

page 4, line 3, text

Suggested editorial deletions and insertions.

page 4, lines 5 through 10 and page 5

We suggest this material be deleted since it has been incorporated in earlier changes.

527 294

ENCLOSURE 1

NRC REVIEW OF SAFEGUARDS SYSTEMS FOR NASAP ALTERNATIVE FUEL CYCLE MATERIALS

Background

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for ~~[illegible]~~ nuclear material control~~s~~ and accounting. Requirements for physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against ~~acts of~~ attack, acts of sabotage and theft. NRC has considered whether strengthened physical protection may be required as a matter of prudence. Proposed ~~rules and~~ upgrades to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). ~~A reference system was developed within these proposed revisions to 10 CFR 73.~~ A ~~The~~ reference system described in the proposed upgrade rules is ~~was~~ considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NASAP Safeguards Basis

The desired basis for NRC review of safeguards systems for NASAP alternate fuel cycle materials containing significant quantities of strategic special nuclear material (SSNM)*, greater than 5 formula kilograms**, during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed

527 295

*$\geq$20% ^{235}U in uranium, $\geq$12% ^{233}U in uranium, or plutonium.

**formula grams = grams contained ^{235}U + 2.5 (grams ^{233}U + grams plutonium); ref. 10 CFR 73.30

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~~use, transport, import, and export to the port of entry of a foreign country~~ ~~is the reference system described in the current regulations and the proposed~~

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revisions cited above. To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U. S. authority, and where diversion by national or subnational forces may occur, some have proposed to ~~add~~ increase radioactivity ~~to~~ of strategic special nuclear materials which are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh fuel material to assure that during the period after export from the U. S. and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover ~~non-radioactive~~ SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time requirement to obtain material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements ~~of~~ imposed by the Nuclear Non-Proliferation Act of 1978 ~~that apply~~ to nuclear related exports ~~including~~ include application of IAEA ~~safeguards as they relate to~~ material accountability requirements. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed ~~An~~ alternative which could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

527 296

The final version of the proposed physical protection upgrade rule for Category 1 material is awaiting final Commission review and consideration. This proposed rule is moving closer to being published in effective form and, together with existing regulations, should provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base could be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

527 297

NRC Review

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches such as upgraded physical protection, improved material control and accountancy, dilution of SNM, decreased transportation requirements, fewer sites handling SNM, and increased material handling requirements as applied to each fuel material type should be assessed. The evaluation should address, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on MCA detection, measurement and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tamper or defeat; the increased public exposure to health and safety for acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, NRC must consider the export and import of SSNM as well as its domestic use.

~~The review~~

As part of this evaluation we request that NRC assess the differences in the licensability of the domestic facilities, transportation system to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, on the international and national safeguards systems of typical importers in protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, should be described in detail as well as the potential advantages in detection or deterrence. A clear identification of the potential role, if any, that added radioactivity could or should play is wanted, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its use by foreign countries to obtain material for weapons purposes. DELETE [The basis for current 10 CFR 73.6(b) regulation that exempts certain types of special nuclear material which deliver a radiation dose rate of 100 rem/hr at 3 feet from the additional physical protection requirements in 10 CFR 73.30 etc. should be described.] Licensability issues that must be addressed by RD&D programs should also be identified. The following table presents a listing of unadultered fuel materials ~~the various types of SSNM~~ and a candidate set of associated radiation levels for each that should be evaluated in terms of ~~for~~ domestic use, import and export:

527 299

Fuel Material Type	Minimum Radiation Level During 2 Year Period (Rem/Hr @ 1 meter) (Ref. 6)	
	Mixed[1]	Mechanically Attached[2]
a. PuO_2, $HEUO_2$ powder or pellets[3]	1,000 per kgHM	10,000/kgHM
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[3]	100 per kgHM	10,000/kgHM
c. LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
d. LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[1] Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[2] Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[3] HEU is defined as containing 20% or more ^{235}U in uranium, 12% or more of ^{233}U in uranium, or mixtures of ^{235}U and ^{233}U in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. ~~Some~~ Candidate methods and radiation levels ~~and indicated in Table I and~~ are described further in the references.

In assessing the impacts of added radioactivity NRC should comment on the effectiveness of this approach to protecting SSNM as one of a spectrum of safeguarding deterrents that might be used. For example, other deterrents could include the use of heavy shipping casks and heads with and without added radiation, additional physical protection, and radiation levels other than those given specifically in the table. In particular the case of no

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~~added radiation must be considered. Finally in conducting these evaluations and assessments NRC must consider not just domestic use of SSNM but its import and export.~~

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527 301

Table 1

CANDIDATE METHODS AND RADIATION LEVELS FOR SPIKING FUEL MATERIALS

Fuel Material Type	Minimum 2 year Radiation Level, rem/hr at 1 meter	Process	Minimum Initial Radiation Level, rem/hr at 1 meter	Reference
a. PuO_2, $HEUO_2$ powder or pellets	1000/kgHM	. Co^{60} addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	. Co^{60} addition	130/kgHM	2, 3, 5, 6
		. fission product addition (Ru^{106})	400/kgHM	2, 3, 5, 6
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40/MWD/MT)	1000 (30/day)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40 MWD/MT)	1000 (30 day)/ assembly	4

527 302

References

1. NUREG 0414, "Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat"

2. ORNL TM-6412, "Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence," J. E. Selle

3. ORNL TM-6483, "Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence," J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen

4. GEFR 00402, "Pre-Irra iation Concept Description and Cost Assessment," G. F. Pflasterer and I. A. Deane

5. IRT-378-R, "Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use," (Vol 2 of "The Spiking of Special Nuclear Materials as a Safeguards Measure," BNL File No. 5.9.1 for Vol 1)

6. SAI-01379-50765, "Material Radiation Criteria and Nonproliferatio E. A. Straker, January 8, 1979

527 303

NRC REVIEW OF SAFEGUARDS SYSTEMS FOR NASAP ALTERNATIVE FUEL CYCLE MATERIALS

Background

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage and theft. NRC has considered whether strengthened physical protection may be required as a matter of prudence. Proposed upgrades to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgrade rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NASAP Safeguards Basis

The desired basis for NRC review of safeguards systems for NASAP alternate fuel cycle materials containing significant quantities of strategic special nuclear material (SSNM)*, greater than 5 formula kilograms**, during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and

* $\geq$20% ^{233}U in uranium, $\geq$12% ^{233}U in uranium, or plutonium.

** formula grams = grams contained ^{235}U + 2.5 (grams ^{233}U + grams plutonium); ref. 10 CRF 73.30.

527 304

the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, some have proposed to increase radioactivity of strategic special nuclear materials which are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh fuel material to assure that during the period after export from the U.S. and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time requirement to obtain material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirement imposed by the Nuclear Non-Proliferation Act of 1978 include application of IAEA material accountability requirements to nuclear related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative which could be used to provide additional safeguards protection against diversion of shipments of SSNM by sub-national groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC Review

It is requested that NRC perform an evaluation of a spectrum of safe-guards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches such as upgraded physical protection, improved material control and accountancy, dilution of SNM, decreased transportation requirements, few sites handling SNM, and in-reased material handling requirements as applied to each fuel material type should be assessed. The evaluation should address, but not be limited to, such issues as the degree to which added radioactive con-taminants provide protection against theft for bomb making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on MCA detection, measurement and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tamper or defeat; the increased public exposure to health and safety for acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation we request that NRC assess the differences in the licensability of the domestic facilities, transportation system to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, on the international and national safeguards systems of typical importers in protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, should be described in detail as well as the potential advantages in detection or deterrence. A clear identification of the potential role, if any, that added radioactivity could or should play is wanted, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its use by foreign countries to obtain material for weapons purposes.

Licensability issues that must be addressed by RD&D programs also should be identified.

The following table presents a listing of unadultered fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import and export:

Fuel Material Type	Minimum Radiation Level During 2 Year Period (Rem/Hr @ 1 meter) (Ref. 6)	
	Mixed[1]	Mechanically Attached[2]
a. PuO_2, $HEUO_2$ powder or pellets[3]	1,000 per kgHM	10,000/kgHM
b. PuO_2-UO_2 amd $HEUO_2$-ThO_2 powder or pellets[3]	100 per kgHM	10,000/kgHM
c. LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
d. LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[1]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[2]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[3]HEU is defined as containing 20% or more ^{235}U in uranium, 12% or more of ^{233}U in uranium, or mixtures of ^{235}U and ^{233}U in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are described further in the references.

527 308

Table 1

CANDIDATE METHODS AND RADIATION LEVELS FOR SPIKING FUEL MATERIALS

Fuel Material Type	Minimum 2 year Radiation Level, rem/hr at 1 meter	Process	Minimum Initial Radiation Level, rem/hr at 1 meter	Reference
a. PuO_2, $HEUO_2$ powder or pellets	1000/kgHM	. Co^{60} addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	. Co^{60} addition	130/kgHM	2, 3, 5, 6
		. fission product addition (Ru^{106})	400/kgHM	2, 3, 5, 6
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40/MWD/MT)	1000 (30/day)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40 MWD/MT)	1000 (30 day)/ assembly	4

527 309

References

1. NUREG 0414, "Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat"

2. ORNL TM-6412, "Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence," J. E. Selle

3. ORNL TM-6483, "Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence," J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen

4. GEFR 00402, "Pre-Irradiation Concept Description and Cost Assessment," G. F. Pflasterer and N. A. Deane

5. IRT-378-R, "Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use," (Vol 2 of "The Spiking of Special Nuclear Materials as a Safeguards Measure," BNL File No. 5.9.1 for Vol 1)

6. SAI-01379-50765, "Material Radiation Criteria and Nonproliferation," E. A. Straker, January 8, 1979

527 300

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 3

MAR 16 1979

MEMORANDUM FOR: R. A. Hartfield, Chief
Licensee Operations Evaluation Branch
Division of Technical Support
Office of Management and Program Analysis

FROM: E. M. Howard, Director
Division of Safeguards Inspection
Office of Inspection and Enforcement

SUBJECT: NASAP-SAFEGUARDS

We have reviewed the paper on NASAP-Safeguards forwarded in your memorandum of February 21, 1979. Our specific comments are noted on the attached copy. One general comment I have with regard to increasing the radioactivity of new reactor fuel is that the author (DOE presumably) does not acknowledge or address the increased difficulties in the routine handling of this material or the increased potential for a public health hazard in the case of a transportation accident. I believe that these "negative" issues need to be addressed in the overall assessment of this conceptual program for added safeguards protection.

R. G. McCormick
for E. M. Howard, Director
Division of Safeguards Inspection
Office of Inspection and Enforcement

Enclosure: As Stated Above

cc: G. W. Reinmuth, IE

CONTACT: E. W. Brach
(49-28080)

527 311

2/13/79

NRC REVIEW OF SAFEGUARDS SYSTEMS FOR NASAP ALTERNATIVE FUEL CYCLE MATERIALS

Background

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for ~~paragraph 70.58 describing funda-mental~~ nuclear material controls and accounting. Requirements for physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against acts of attack, sabotage and theft. NRC has considered whether strengthened physical protection may be required as a matter of prudence. Proposed rules and upgrades have been published for comment in the Federal Register (43 FR 35321). A reference system was developed within these proposed revisions to 10 CFR 73. The reference system was considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NASAP Safeguards Basis

The desired basis for NRC review of safeguards systems for NASAP alternate fuel cycle materials containing significant quantities of strategic special nuclear material (SSNM)*, greater than 5 formula kilograms**, during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed

NRC requirements do not specify enrichment of U-233

*≥20% ^{235}U in uranium, ≥12% ^{233}U in uranium, or plutonium.

**formula grams = grams contained ^{235}U + 2.5 (grams ^{233}U + grams plutonium); ref. 10 CFR 73.30

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527 312

use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U. S. authority, and where diversion by national or subnational forces may occur, some have proposed to ~~add~~ increase the radioactivity ~~to~~ of strategic special nuclear materials which are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh fuel material to assure that during the period after export from the U. S. and loading into the foreign reactor, remote reporcessing through the decontamination step would be necessary to recover non-radioactive SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time requirement to obtain material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements of the Nuclear Non-Proliferation Act of 1978 will apply to nuclear related exports including application of IAEA safeguards as they relate to material accountability requirements. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by IAEA prior to approval of a subsequent fuel export containing SSNM.

An alternative which could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

527 313

NRC Review

It is requested that NRC perform an evaluation of the differences in the licensability of the domestic facilities, transportation system to the port of entry of the importer, and other export regulations for those alternative fuel cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on the international and national safeguards systems of typical importers in protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as ~~more~~ limited access for inspection and degraded material accountability, should be described in detail as well as the potential advantages in detection or deterrence. A clear identification of the potential role, if any, that added radioactivity could or should play is wanted, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its use by foreign countries to obtain material for weapons purposes. The basis for current 10 CFR 73.6(b) regulation that exempts certain types of special nuclear material which deliver a radiation dose rate of 100 rem/hr at 3 feet from the additional physical protection requirements in 10 CFR 73.30 etc. should be described. Licensability issues that must be addressed by RD&D programs should be identified.

The following types of SSNM and a candidate set of associated radiation levels should be evaluated for domestic use, import and export:

527 314

Fuel Material Type	Minimum Radiation Level During 2 Year Period (Rem/Hr @ 1 meter) (Ref. 6)	
	Mixed[1]	Mechanically Attached[2]
a. PuO_2, $HEUO_2$ powder or pellets[3]	1,000 per kgHM	10,000/kgHM
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[3]	100 per kgHM	10,000/kgHM
c. LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
d. LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[1]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[2]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[3]HEU is defined as containing 20% or more ^{235}U in uranium, 12% or more of ^{233}U in uranium, or mixtures of ^{235}U and ^{233}U in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Some candidate methods and radiation levels are indicated in Table 1 and are described in the references.

In assessing the impacts of added radioactivity NRC should comment on the effectiveness of this approach to protecting SSNM as one of a spectrum of safeguarding deterrents that might be used. For example, other deterrents could include the use of heavy shipping casks and heads with and without added radiation, additional physical protection, and radiation levels other than those given specifically in the table. In particular the case of no

527 315

added radiation must be considered. Finally in conducting these evaluations and assessments NRC must consider not just domestic use of SSNM but its import and export.

Table 1

CANDIDATE METHODS AND RADIATION LEVELS FOR SPIKING FUEL MATERIALS

Fuel Material Type	Minimum 2 year Radiation Level, rem/hr at 1 meter	Process	Minimum Initial Radiation Level, rem/hr at 1 meter	Reference
a. PuO_2, $HEUO_2$ powder or pellets	1000/kgHM	. Co^{60} addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	. Co^{60} addition	130/kgHM	2, 3, 5, 6
		. fission product addition (Ru^{106})	400/kgHM	2, 3, 5, 6
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
		. fission product addition (Ru^{106})	40/assembly	2, 3, 5, 6
		. pre-irradiation (40/MWD/MT)	1000 (30/day)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/assembly	. Co^{60} addition	13/assembly	2, 3, 5, 6
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		. pre-irradiation (40 MWD/MT)	1000 (30 day)/ assembly	4

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3. ORNL TM-6483, "Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence," J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen

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6. SAI-01379-50765, "Material Radiation Criteria and Nonproliferation," E. A. Straker, January 8, 1979

527 318

ENCLOSURE 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR [illegible] 1979

MEMORANDUM FOR: R. A. Hartfield, Chief
Licensing Operations Evaluation Branch
Division of Technical Support
Office of Management & Program Analysis

FROM: Ralph J. Jones, Chief
SGSB:SD

SUBJECT: NASAP - SAFEGUARDS (YOUR MEMO TO R.M. BERNERO, SD, DATED 2/21/79)

We have read the subject document, and have no comment except regarding the proposed system of addir radioactivity to protect strategic special nuclear material to maintain safeguards protection beyond the port of entry into a country whose safeguards is not subject to U.S. authority. Our immediate reaction is that the scheme seems to be more theoretical than practical. This, however, could better be determined after a thorough technical review and a value/impact analysis. Such an analysis should cover the impact on radiation exposure to workers and the public of having additional radioactive material moving in transport. The analysis also should consider the cost of routine security protection versus the cost of adding radioactivity. This also should recognize that such radioactive material also may need protection against sabotage.

RJ Jones

Ralph J. Jones, Chief
Safeguards Standards Branch
Office of Standards Development

527 319

ATTACHMENT 5

COMMENTS ON ORNL-5388, INTERIM ASSESSMENT OF THE DENATURED 233U FUEL CYCLE: FEASIBILITY AND NONPROLIFERATION CHARACTERISTICS

General Comments

The Division of Fuel Cycle and Material Safety staff has reviewed the subject report from the point of view of the 233U Fuel Cycle. Impacts of 233U usage on reactors have not been considered. We find the document to be an exhaustive study of the denatured 233U fuel cycle. In the NASAP PSEID's that we have reviewed, DOE defined alternative nuclear fuel cycle systems having promising proliferation resistance and commercial potential. It is this stated NASAP purpose that is the focus for our comments.

Chapter 7, Overall Analysis of Denatured Fuel Systems, and Appendix C, Detailed Results from Evaluations of Various Nuclear Power Systems Utilizing Denatured Fuel, present voluminous data on the uranium, uranium/plutonium and denatured uranium fuel cycles. Data are presented on eight fuel cycle options with four cases being studied for each option. Two levels of uranium supply (high and intermediate) are used in each case and option. We have the following comments.

Nuclear Power Demand

The demand curve for nuclear power used in Chapter 7 requires that nuclear power reactors be capable of supplying 1100 GWe in 2049. Not unexpectedly, for the high cost of U_3O_8 supply runs (constrained resource availability) only those fuel cycle options that include breeders are able to meet the demand. For the intermediate-cost U_3O_8 runs, options using D_2O moderated converters are also capable of meeting the demand.

Although we recognize the need to consider the ultimate capability of any fuel cycle to produce power in terms of GWe produced/MT U_3O_8, it does not appear reasonable to us that a projection of installed nuclear power 70 years from now is a realistic decision criterion on which to base near-term (1980) decisions. We believe that these near-term decisions should include consideration of the transitional period (say 2000-2020) showing changes in the fuel cycle as the new types of reactors are built instead of LWR's. We recommend some detailed evaluations of the differences among the options over the transition period.

"Safe" Centers

With the exception of the throwaway option, all other options considered in ORNL-5388 require fuel reprocessing of LWR and other types of fuel. Further, all options except the Pu throwaway option recycle plutonium to reactors built on a safe center. Appendix C contains data on the energy support ratio (defined as the ratio of installed nuclear capacity outside an energy center to installed nuclear capacity inside an energy center).

527 320

We recognize the desirability of obtaining the maximum amount of information from the complicated systems studies, and therefore consider the data on the energy support ratio valuable. We believe, however, that data must be available to consider the nuclear power fuel cycle and reactor complex in the early years (say 2000-2020) of the next century. Options that appear to be very different in 2049 may be quite similar in the early years of the next century and very likely such near term considerations may dominate the selection of alternative fuel cycles. Hence, we recommend a discussion of the energy support ratio over the early years of the next century as well as an evaluation of what the limiting ratio should be.

An additional point should also be made. The requirement for a "safe" center is an assumption in Chapter 6 of the ORNL document. Since this assumption does not affect the results of systems studies, in terms of numbers of reactors, development costs, etc., it would appear desirable to clearly indicate that this assumption does not impact on other study results. (The energy support ratio is, of course, dependent on the assumption.)

Specific Comments

Chapter 2. Rationale for Denatured Fuel Cycles

Page 2-5, Paragraph 2

"Whereas in the plutonium cycle no denaturant comparable to ^{238}U exists,..."

There was a large amount of publicity in the last year or so of using some non-fissile plutonium isotope to denature plutonium. Has DOE determined that such denaturing is not possible or practical? If so, this should be explained and the basis provided.

Page 2-5, Paragraph 3

"Moreover, the quantity of plutonium generated via the denatured fuel cycle will be significantly less than that of the other two cycles."

The statement represents a conclusion about the amount of plutonium discharged in ^{233}U fuel. A quantitative value would probably appear to be less judgmental.

527 321

Pages 2-9, 2-10

The discussion on institutional considerations of the denatured fuel cycle vis-a-vis "energy interdependence" is not very convincing. Although nations using denatured fuel may be dependent on nations operating reprocessing plants, it appears that nations operating reprocessing plants and transmuter reactors may not be dependent on the supply of spent denatured fuel. It is possible that transmuter reactors could be designed to operate either on plutonium or enriched uranium fuel (to maintain power production).

Page 3-27, last paragraph

The subject of the section, 3.3.4, is "Potential Circumvention of the Isotopic Barrier of Denatured Fuel." National diversion of denatured $^{233}U/^{238}U$ material for upgrading into weapons grade material might be a short term program if only a few weapons are desired. The last paragraph on page 3-27 is written as if enrichment of $^{233}U/^{238}U$ would be a long term program (e.g. "...unaffected over a 20-year plant life...").

527 322