4/30/82

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
UNITED STATES DEPARTMENT OF ENERGY	)	Docket No. 50-537
PROJECT MANAGEMENT CORPORATION	)	
TENNESSEE VALLEY AUTHORITY	)	
	)	
Clinch River Breeder Reactor Plant	)	

APPLICANTS' UPDATED RESPONSE #2 TO NATURAL RESOURCES DEFENSE COUNCIL, INC. AND SIERRA CLUB INTERROGATORIES (SEVENTH, NINTH, TENTH AND THIRTEENTH SETS)

Pursuant to 10 CFR paragraph 2.740b, and in accordance with the Board's Prehearing Conference Order of February 11, 1982, the United States Department of Energy, Project Management Corporation, and the Tennessee Valley Authority (the Applicants) hereby update their responses to the Natural Resources Defense Council, Inc. and the Sierra Club Seventh, Ninth, Tenth and Thirteenth Sets of Interrogatories to the Applicants, dated April 7, 1976, May 13, 1976, August 13, 1976, and January 14, 1977, respectively.

In these updated responses the following style has been utilized: For each set of interrogatories the Preamble to Questions has been set forth. Thereafter, each interrogatory within the set has been restated and the updated answer provided. Certain of the answers are unchanged from the responses initially furnished. However, for convenience those unchanged responses also have been set forth after the appropriate interrogatories.

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The answers contained in this Updated Response #2 supercede all prior answers to the interrogatories as to which they are applicable.

In some instances, interrogatories specifically related to the previous parallel design covered in Appendix F to the PSAR. Appendix F was withdrawn from the application in 1976. Applicants have attempted in these updated answers to provide updated responses to those questions relating to Appendix F where such questions appear to Applicants to be potentially applicable to the current design. This has meant a substantial amount of additional effort by Applicants since the parallel design has not been the subject of attention by Applicants during the past five years and since the interrogatories needed to be interpreted in light of the current design. Where Applicant believes the interrogatories are related to Appendix F and the previous parallel design and are not appropriately applicable to the current design, Applicant has so noted.

SEVENTH INTERROGATORY SET

GENERAL QUESTION

Each question is instructed to be answered in six parts, as follows [Where appropriate, the parts of the question have been restated to reflect the protocol for discovery agreed to by Applicants, Staff, and Intervenors NRDC et al.]:

A) Provide the direct answer to the question.

B) Identify all documents and studies, and the particular parts thereof, relied upon by Applicants, now or in the past, which serve as the basis for the answer. In lieu thereof, at Applicants' option, a copy of such document and study may be attached to the answer.

C) Identify principal documents and studies, and the particular parts thereof, specifically examined but not cited in B). In lieu thereof, at Applicants' option, a copy of each such document and study may be attached to the answer.

D) Describe the methodology of analysis, including all assumptions, and test results of the studies identified in subparts b) and c) of each answer.

E) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicants' answer. This answer need be provided only in cases where Applicants intend to rely upon on going research not included in Section 1.5 of the PSAR at the LWA or construction permit hearing on the CRBR. Failure to provide such an answer means that Applicants do not intend to rely upon the existence of any such research at the LWA or construction permit hearing on the CRBR.

F) Identify the expert(s), if any, whom Applicants intend to have testify on the subject matter questioned. State the qualifications of each such expert. This answer need not be provided until Applicants have identified the expert(s) in question or determined that no expert(s) will testify, as long as such answer provides reasonable notice to Intervenors.

GENERAL ANSWER

The following responses are identical for all interrogatories except where supplementary information is provided in the responses below.

(A) See numbered responses below.

(B) The documents which serve as a basis for the Applicants answer are identified in the responses below and have been or will be made available for inspection and copying.

(C) The Applicants have examined and evaluated many documents pertaining to the subject matter questions, however, unless otherwise indicated in the responses below, documents and other studies pertaining to the subject matter have been examined but not relied upon by the Applicants. This does not imply that the Applicants have examined all documents in existence which could pertain to the subject matter questioned. Of the documents examined by the Applicants which might pertain to the subject matter questions, only that material relied upon by the Applicants has been retained in retrievable form by the Applicants. This material is identified in response to subpart B.

(D) The methodology of analysis, including all assumptions, and test results of the studies identified in subparts (B) and (C) of each answer is described or referenced in the document itself.

(E) Except where otherwise noted below, the Applicants' program of further research work is described in Section 1.5 of the PSAR and in Appendix A of CRBRP-3, Vol. 1 and Vol. 2.

(F) At the present time the Applicants have not determined the experts, if any, whom they intend to have testify on the subject matter.

QUESTION 1 (PREAMBLE)

Fuel Element Bowing: The elimination of in core spacer pads between fuel rod bundle ducts (now placed in the axial blanket) for the CRBR raises the question of a possible autocatalytic power transient as occurred in EBR-I core meltdown due to inward fuel bowing. The PSAR notes that a positive reactivity feedback is predicted in the CRBR, but that it will be offset by negative Doppler feedback to make the net reactivity coefficient negative.

QUESTION 1(a)

Does a negative coefficient depend on maintaining the reactor power level, coolant flow and fuel and coolant temperatures within prescribed ranges? That is, does the thermal hydraulic-mechanical model predict a net positive feedback for any combination or set of circumstances?

ANSWER 1(a)

CRBRP core restraint analyses generally indicate a negative bowing reactivity above some power to flow ratio, $(P/F)_o$, where P and F represent fractions of full power and flow respectively. $(P/F)_o$ will generally range from 0 to ~0.8 depending on the particular constraint assumptions employed. Consequently for $P/F > (P/F)_o$ the power coefficient is negative for any combination of the stated parameters, since bowing is the only component which can contribute positive reactivity.

For $P/F < (P/F)_o$ a positive bowing reactivity effect is calculated. Under these conditions a negative power coefficient would depend on the Doppler feedback, coolant flow level, temperature gradient field and the inlet temperature variation associated with a given power transition. This situation is analogous to the EBR-II response (Ref. 1) in which positive

bowing reactivity effects exhibited various degrees of importance depending on the power, flow combination.

QUESTION 1(b)

Does the PHENIX fast power reactor (France) employ spacer pads in the core, as distinguished from being located outside the core region, vix blanket (as in CRBR)? R. Carle, et al., in ANL-7520 (Part 2). p. 247, indicated that the pads are in fact in the core for Phenix?

ANSWER 1(b)

Page 247 of ANL 7520, Part 2, indicates that spacer pads are located within the core region for PHENIX. The title of the above article (page 243) is PHENIX: Status of the Design before Construction and was presented in November 1968.

Personal communication with Argonne National Laboratory personnel in 1975 indicates that the spacer pads were removed from the active core area and were placed a few inches above the active core. Because of the age of the above report, it is believed that the latter information is representative of the present design.

QUESTION 1(c)

How is PHENIX designed to cope with neutron-induced swelling, if the fuel ducts are made of stainless steel?

ANSWER 1(c)

The response to question 1(b) indicates that the present PHENIX design copes with neutron induced swelling by locating the load pads above the

core region. Spacing between ducts can accommodate limited swelling in the core region.

QUESTION 1(d)

Is the maximum fast neutron (> 1 MEV) fluence for PHENIX below the limit where swelling would cause the ducts to be stuck together at the pads?

ANSWER 1(d)

The current/neutron fluence of the peak PHENIX fuel duct is estimated to be 1.0 to 1.1 x 10^{23} n/cm^2, E > .1 MEV. The fluence at the spacer pad location (above the active core) is believed to be between 1/10 to 1/15 that of the peak fluence location. No swelling contact is expected for the fluence experienced at the load pads.

QUESTION 1(e)

Does the British FPR use spacer pads in the core?

ANSWER 1(e)

The information presented below is from an article in a Fast Reactor Power Stations Document by the British Nuclear Energy Society; Proceedings of the International Conference Organized by the British Nuclear Energy Society held on 11-14 March, 1974, at the Institution of Civil Engineers, London. The title of the article on pages 307-318 is Support of PFR Sub-Assemblies and Associated Developments by J. A. C. Holmes.

The lower pads are below the lower axial breeder and the upper pads are located at the juncture of the lower axial breeder with the bottom of the core as shown in Figure 2 and 3 of the above reference, attached.

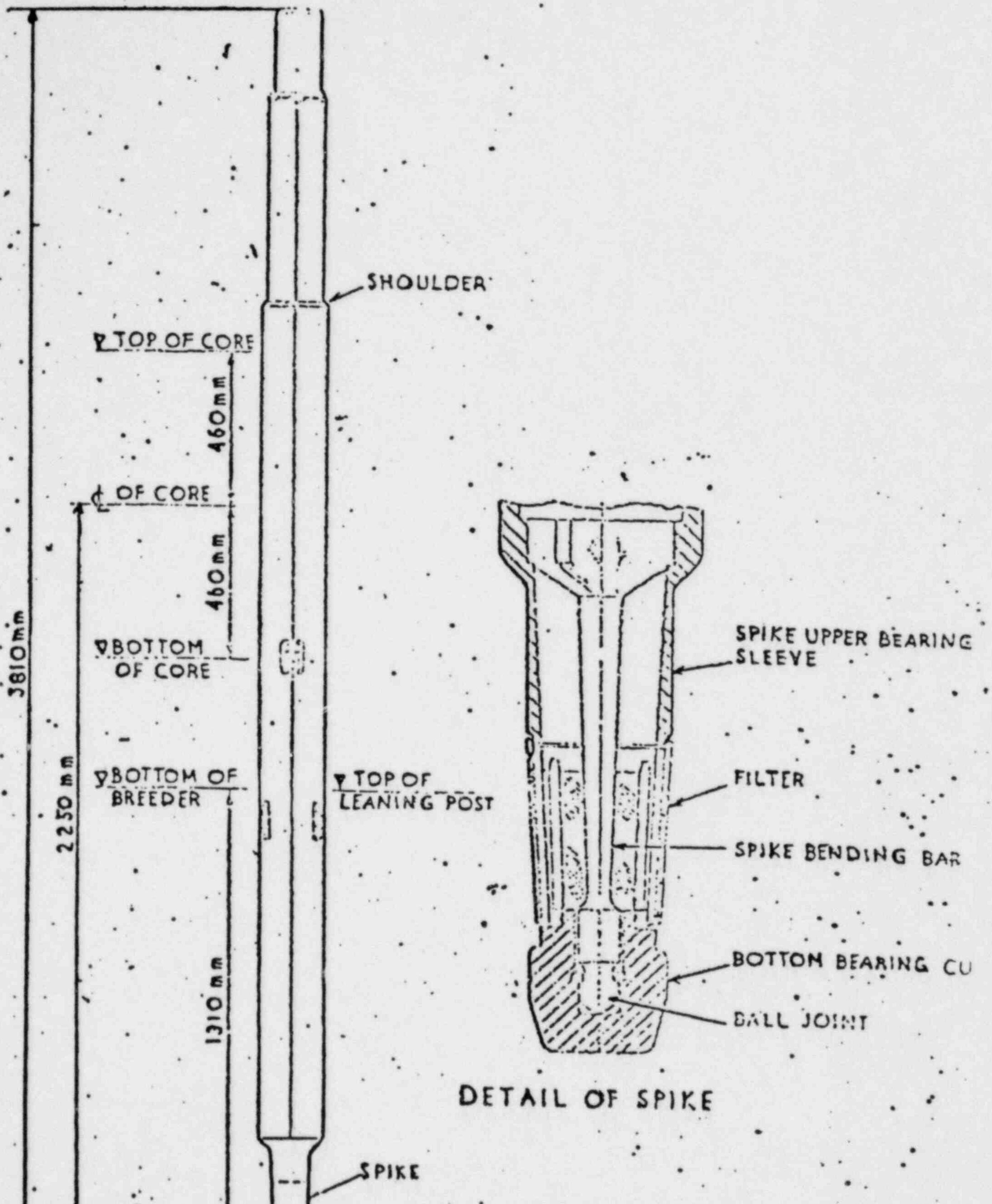


Figure 2 Sub-assembly

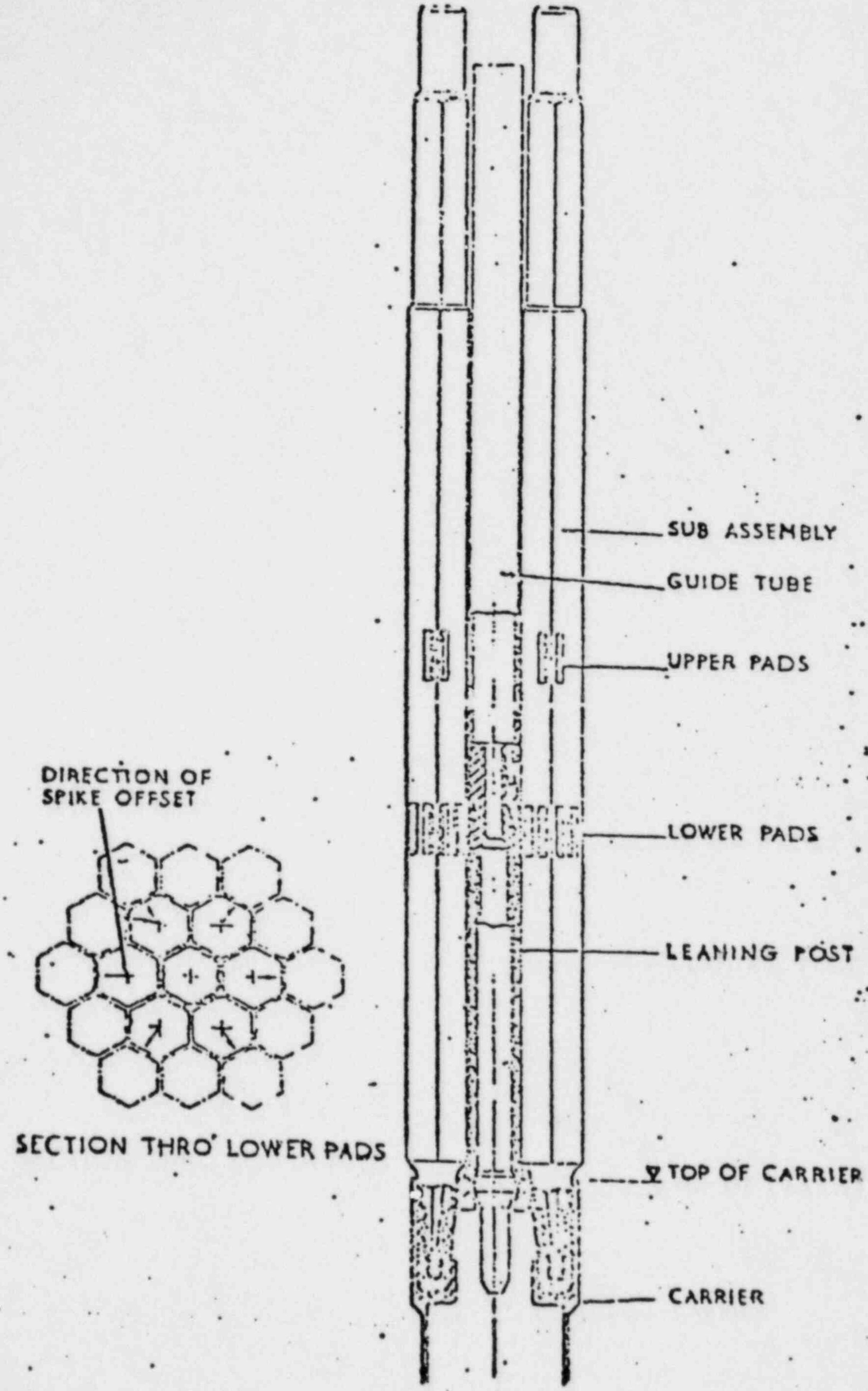
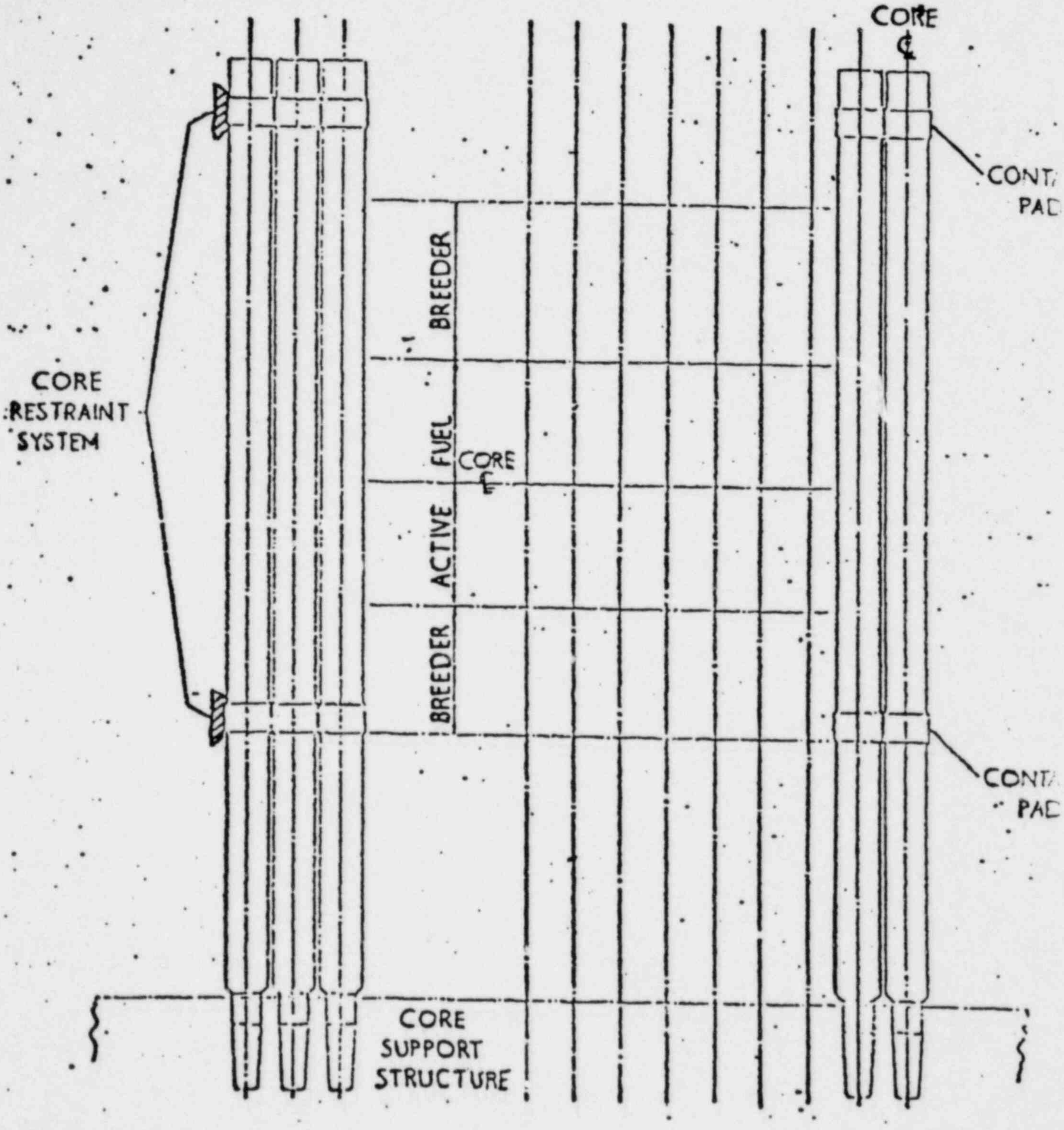


Figure 3 Sub-assembly Support



Diagrammatic Representation of Restrained Core

Figure 5

In the restrained core design the spacer pads are located above the upper breeder and below the lower breeder as shown in Figure 5 of the above Reference attached.

QUESTION 1(f)

What are the distances of the spacer pads from the core midheight level for the CRBRP, Phenix, and FPR (British)?

ANSWER 1(f)

The distance to the spacer pad centerline from the core mid height centerline is:

	71.1 cm	CRBRP
approximately	55.0 cm	PHENIX
	46.0 cm	PFR

QUESTIONS 1(g)

Please describe the detailed theoretical calculation (results, models and theory) which predict, (i) the amount of fuel duct bowing (at several elevations of the core), (ii) the initial clearances between ducts (mean, lower and upper limits due to design tolerances), and (iii) the reactivity worth per unit distance of bowing.

ANSWER 1(g)

The description of the analytical techniques and computer modeling employed in PSAR duct bowing analyses is described in Section 4.2.2.4.3. The theoretical treatment is consistent with approaches described in considerable detail in References 2 and 3. Documentation of the current analytical approaches used in CRBRP core restraint analysis together with their theoretical bases will be available for the FSAR.

ii. CRBRP Core Restraint System Gaps (At Uniform Temperatures) (all dimensions in inches).

Top Load Plane Gaps

Interassembly gaps:	0.015 ± .004
Peripheral gap:	.120 ± .012

Above Core Load Plane Gaps

Interassembly gaps:	0.015 ± .004
Peripheral gaps:	.054 ± .012

iii. The radial worth factor for several cases are presented in the PSAR: Tables 4.3-24 and 25

QUESTION 1(h)

Describe the experiments, if any, which confirm the thermal-mechanical bowing predictions discussed above.

ANSWER 1(h)

A direct comparison of bowing reactivity analytical predictions with experimentally deduced bowing reactivity effects has been performed for the EBR-II reactor (Ref. 3,4). These comparisons showed that bowing reactivity effects can be predicted with very good accuracy. These results add confidence that CRBRP core restraint analyses will be capable of predicting bowing reactivity effects in the operating reactor.

The analytical model used for CRBRP Core Restraint Design is being verified using a full core simulation in the National Core Restraint Test Facility. Tests were begun in 1977 and completed in 1981 and the verification will appear in the FSAR.

References

1. EBR-II System Design Description, Volume II Primary System, Chapter 2 Reactor.

2. ANL-8068, NUBOW: A FORTRAN-IV Program for the Static Elastic Structural Analysis of Bowed Reactor Cores, G. A. McLennan, April 1974.

3. ANL/EBR-014, BOW-V; A CDC-3600 Program to Calculate the Equilibrium Configurations of a Thermally Bowed Reactor Core, D. A. Kucera and D. Mohr, January 1970.

4. Prediction and Measurement of Reactivity Effects Due to Thermal Bowing of EBR-II Subassemblies, D. Mohr, A. Gopalakrishnan, ANS Transactions, October 1971.

QUESTION 2

(a) Discuss fully, including the theoretical and experimental basis for the stated position, the extent to which CRBR mixed-oxide fuel will densify upon heating-up to the melting temperature.

(b) Has it been established that PuO_2 - UO_2 fuel will not densify upon heating up (to the melting temperature) due to any phase changes?

ANSWER 2(a) and (b)

The density of stoichiometric 75 w/o (weight percent) UO_2 - 25 w/o PuO_2, which is representative of CRBRP fuel pellets, as a function of temperature (up to the melting point) has been derived based upon the recommended relationship for thermal expansion which has been measured.

The derivation is as follows:

The density at temperature, P_T, is the ratio of Mass, M, to Volume, V_T

$$P_T = \frac{M}{V_T} \tag{1}$$

For isotopic thermal expansion, the relationship between volumetric expansion and linear expansion is

$$\frac{V_T - V_o}{V_o} = \frac{3(L_T - L_o)}{L_o} \tag{2}$$

Combining equation (1) and (2) yields

$$P_T = \frac{M}{V_o\left(\frac{3L_T}{L_o} - 2\right)} = P_o\left(\frac{3L_T}{L_o} - 2\right)^{-1} \tag{3}$$

where P_o = theoretical density at $T = 0^oC$

$= 11.08\ g/cm^3$ for $.75\ UO_2 - .25\ PuO_2$

and $\frac{L_T}{L_o}$ = ratio of length at temperature to length at 0^oC (4)

$= [1 + a_m\ (T-O)]$

The thermal expansion, a_m, up to the melting (solidus) temperature of $UO_2 - PuO_2$ (2725^oC), is

$$a_m = 6.8 \times 10^{-6} + 2.9 \times 10^{-9}\ T \tag{5}$$

Finally, combining equations (3), (4) and (5) gives the final equation for density:

$$P_T(g/cm^3) = 11.08\ [3(1+6.8 \times 10^{-6}\ T + 2.9 \times 10^{-9}\ T^2) - 2]^{-1} \tag{6}$$

The basis for Equation (5) is the thermal expansion data of UO_2 to 2200^oC from Reference 2-1 (Section B). This data was shown by R. L. Gibbey in Reference 2-2 (Section B) to be in excellent agreement with experimentally measured values for $0.75\ UO_2 - .25\ PuO_2$ and PuO_2. Thus, minor compositional variations between CRBRP mixed-oxide fuel (0.67 w/o UO_2 - 0.33 w/o PuO_2) and the representative $0.75\ UO_2 - 0.25\ PuO_2$ fuel will not affect linear

thermal expansion. The behavior of the density with temperature is provided in Figure 2-1.

In addition to considerations of thermal expansion and phase change there is a potential for densification of as-fabricated mixed oxide fuel having lower density than theoretical. A topical report (CRBRP-ARD-0168) by Bishop 2-2a considered the impact of fuel densification on CRBRP fuel performance. The Bishop report provides a summary of existing information relative to fuel densification and the deformation accompanying it, applies this information to predict CRBRP fuel dimensional change, and assesses the deformation due to densification on CRBRP fuel performance.

Therefore, as indicated in Figure 2-1 and as supported by References 2-1 and 2-2 Section B:

(a) The extent of the densification (negative linear thermal expansion) of representative CRBRP mixed-oxide fuel upon heating up to the melting point is not significant (less than 10% change). In fact, there is no densification but rather a slight decrease in overall density.

(b) The impact of fuel densification due to unrestructured (as-fabricated) fuel densification on CRBRP fuel performance has been evaluated in the Bishop report. The effects of increased stored heat, increased heat generation rate, decreased heat transfer capability and axial gaps in the fuel column were addressed for design basis steady-state and transient operation. Any potential degradation due to these effects was found in each case to be less than that assumed in the CRBRP design process described in the PSAR. Therefore, pellet deformation due to fuel densification is expected to have negligible adverse impact on the design capability of CRBRP fuel.

There is no evidence of densification of PuO_2, UO_2 or mixed UO_2 - PuO_2 due to phase change upon heating up to the melting temperature.

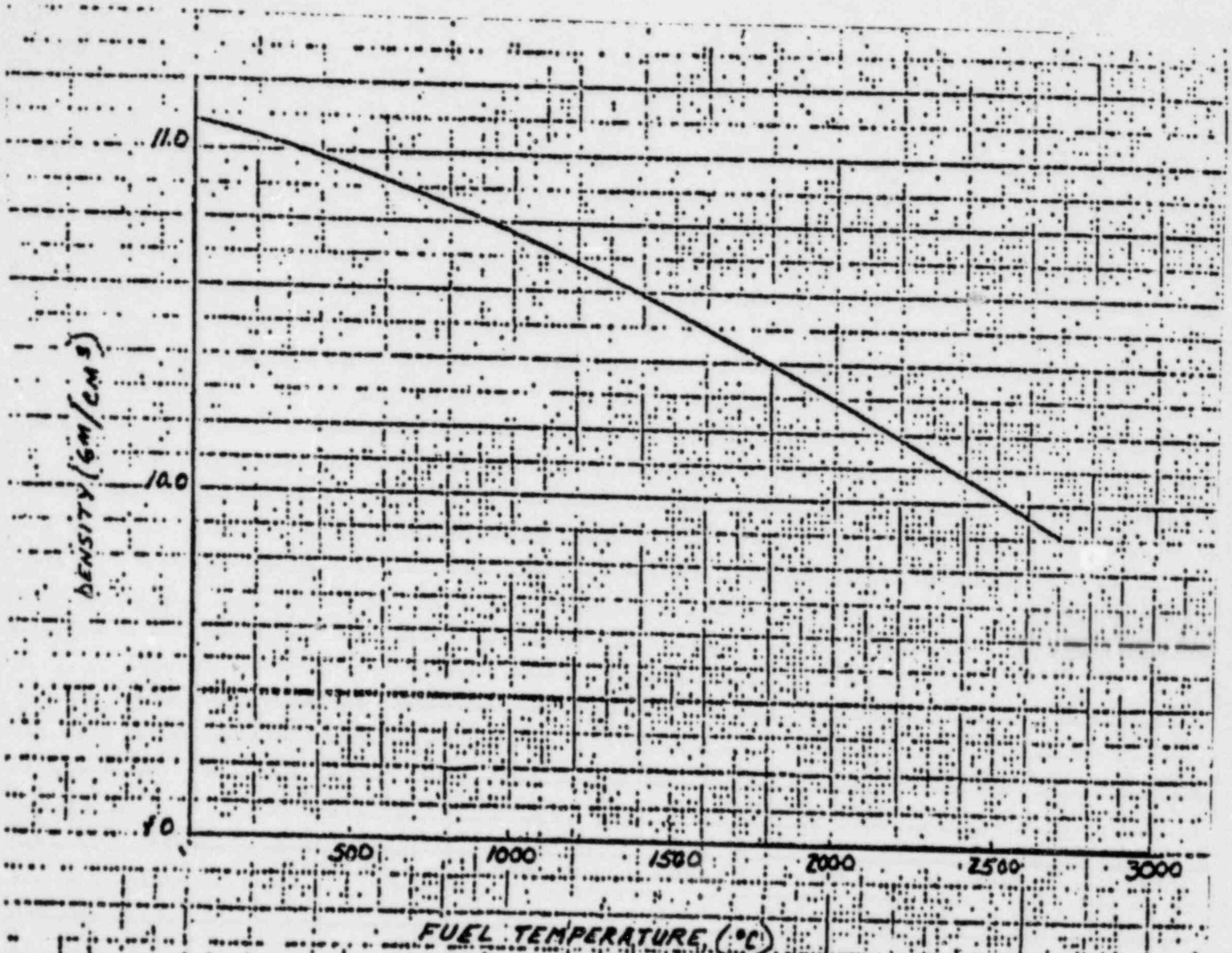


FIGURE 2-1. DENSITY OF STOICHIOMETRIC UO_2-25 w/o PuO_2

The primary references relied upon in answering this question are as follows:

Reference 2-1

Conway, J. B., R. M. Fincel, Jr., and R. A. Hein, "The Thermal Expansion and Heat Capacity of UO_2 to 2200°C:, Transactions of the American Nuclear Society Volume 6, Number 1, 1963, p. 153.

Reference 2-2

Gibby, R. L., "Thermal Expansion of Mixed-Oxide Fuel," HEDL-THE-74-3, HEDL Qua terly Technical Report, July, August, September 1974, Volume 1, Applied Research, pp. A-8 to A-10, A-15, A-22 and A-23.

Reference 2-2a

Bishop, B. A., "Impact of Fuel Densification on CRBRP Fuel Performance," CRBRP-ARD-0168, June 1977.

The secondary references which were examined but were not relied upon in answering this question are indicated below. These references provide collaborating data and/or models for thermal expansion of PuO_2 (Reference 2-3), .75 UO_2 - .25 PuO_2 (Reference 2-4 to 2-6) or .80 $UO_{1.94}$ - .20 $PuO_{1.94}$ (Reference 2-7) as indicated in the attached Figures 7 and 8 (pp. A-22 and A-23) of Reference 2-2.

Reference 2-3

Tokar, M., A. W. Nutt and T. K. Keenan, "Linear Thermal Expansion of PuO_2," Nuclear Technology, Volume 17, 1973, pp. 147-152.

Reference 2-4

Takemura, T., S. Kashima, H. Matsui and M. Koizuni, "Thermal Expansion of $(Pu-U)O_{2-x}$," Annual Report of Tokai Works, April 1, 1968 - March 31, 1969, PNCT-AR-68, Power Reactor and Nuclear Fuel Development Corporation, Toka. Japan, November 1969.

Reference 2-5

LeBlanc, J. M. and H. Andriessen, "Research on Thermal Expansion of UO_2, PuO_2 and $(U, Pu)O_2$," EURAEC 434, Translation of Blg-101, Brussels, June 1962.

Reference 2-6

Skavdahl, R. E. and E. L. Zebroski, "High Temperature Phase Studies," Sodium Cooled Reactors Fast Ceramic Reactor Development Program Twenty Eight Quarterly Report, August-October, 1968. GEAP-5700, pp. 57, 60-63.

Reference 2-7

Roth, J., M. E. Hubert, J. R. Cherry, C. S. Caldwell, "The Effects of Stoichiometry on the Thermal Expansion of 20 wt % PuO_2 - UO_2 Fast Reactor Fuel", Transactions of the American Nuclear Society, Volume 10, 1969, pages 457-458.

The test results from References 2-1 to 2-7 (other than Reference 2-2a) are shown in the attached Figures 7 and 8 of Reference 2-2. The method of deriving the experimental data and in some cases developing the associated mathematical correlations are described in the indicated pages of the respective references.

Characterization of the out-of-pile and in-reactor behavior of reference CRBRP mixed-oxide fuel (e.g., dilatometer measurements of pellet sintering behavior) is a continuing effort of LMFBR Mixed Oxide Fuels Development Program and the LMFBR Reference Fuel Steady State Irradiation Program.

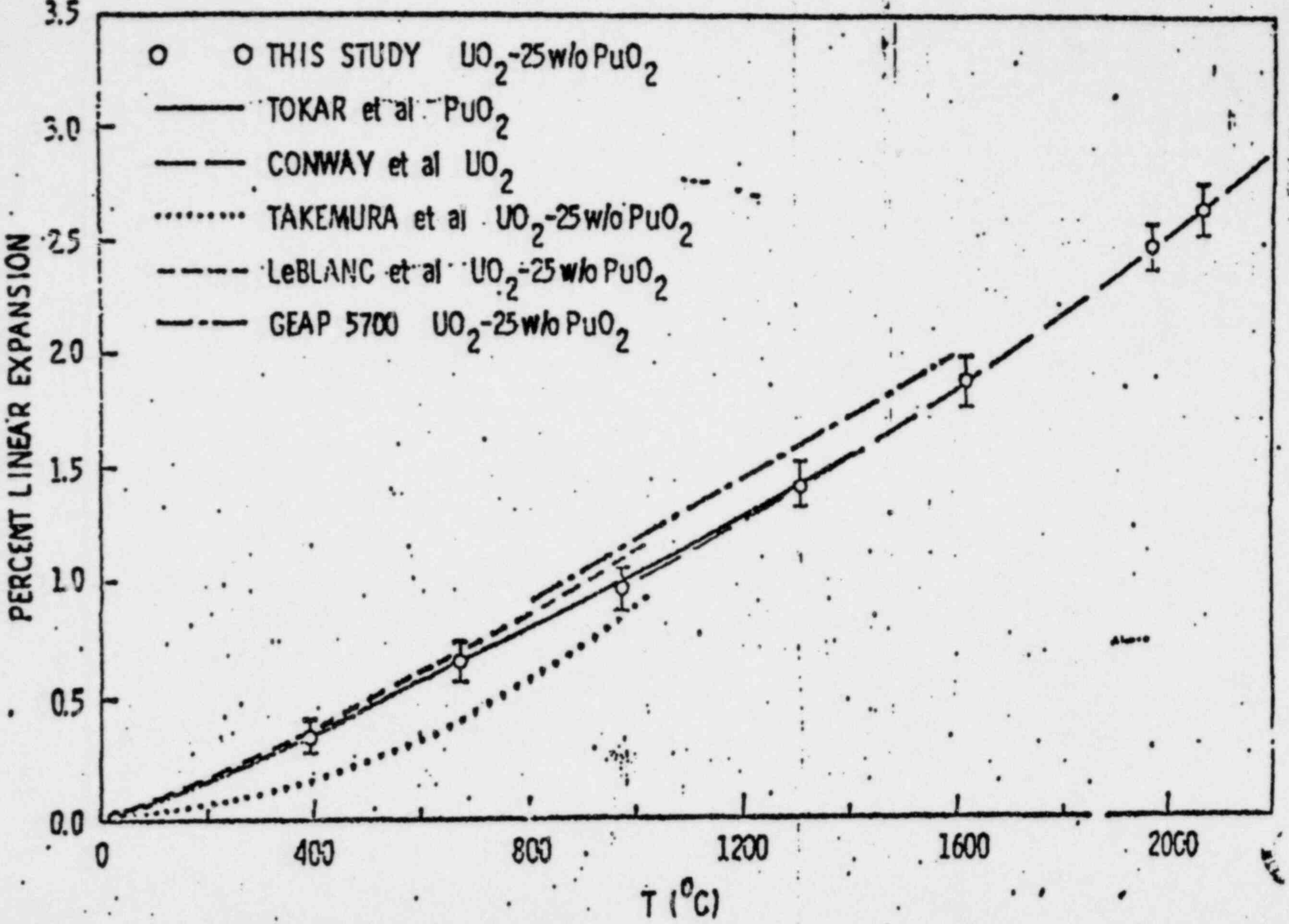


FIGURE 7. Thermal Expansion of Stoichiometric Mixed Oxide Samples.

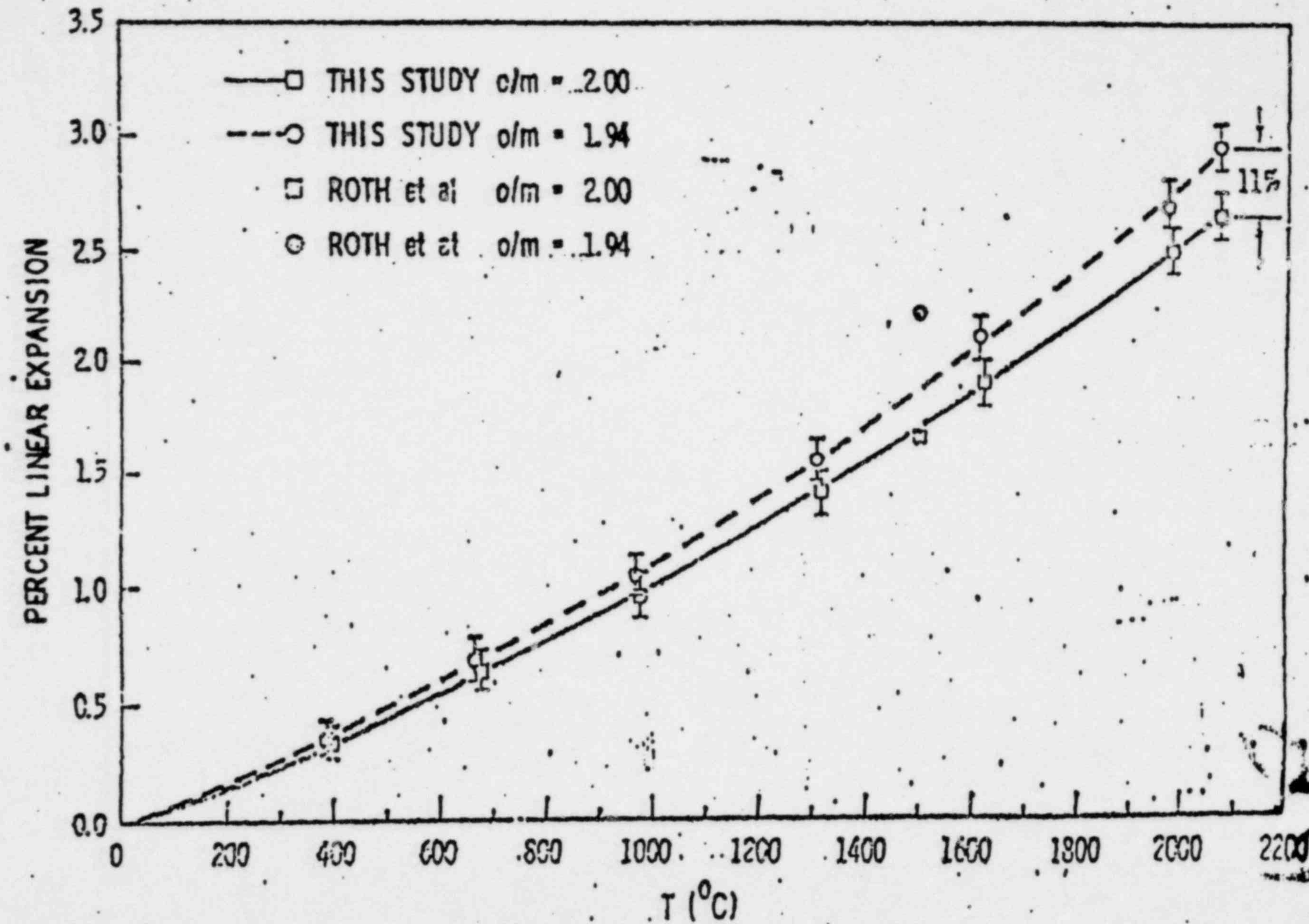


FIGURE 8. Thermal Expansion of Substoichiometric UO_2 - 25 wt % PuO_2.

QUESTION 3

G. S. Lellouche in the EBR-I Incident: A Reexamination (Nuc. Sci. & Eng., 56, 3:303-307) showed that the reactivity changes during the EBR-I power excursion (sub-prompt) were grossly different than previously believed.

(a) To what extent is the Applicant, or anyone to the Applicant's knowledge, analyzing the EBR-I incident with SAS3A (or other SAS codes) modified for metallic fuel, to see if the least-squares-deduced temperature coefficient of Lellouche can be predicted accurately?

(b) If no such analysis is being performed, why is this considered unnecessary?

ANSWER 3(a)

Neither the Applicants nor anyone known to the Applicants is analyzing the EBR-I incident with SAS3A (or other SAS codes), to see if the least-squares-deduced temperature coefficient of Lellouche can be predicted accurately.

ANSWER 3(b)

The major difference between earlier analyses of the EBR-1 incident and that of Lellouche lies in his categorization of the fast coefficient as a sum of the fuel temperature coefficient and the coolant temperature coefficient. The analysis of EBR-I by Lellouche is not relevant for CRBRP safety analysis, because the SAS3A code has terms which account for both effects. The sodium worth tables used in SAS account for the dependence of sodium worth on radial core location in that channel-dependent reactivity will be introduced as the coolant density changes. Reactivity changes caused by increases in fuel temperature are accounted for via the Doppler coefficient. Thus, during the early portion (i.e., before voiding), the

approach taken by Lellouche is implicitly used in SAS3A. Coolant density-induced reactivity changes are not nearly as great as those resulting when sodium voiding begins. Once a channel starts to void, it rapidly loses its sodium, i.e., the coolant density drops to nearly zero almost immediately.

QUESTION 4

Does there exist a document or set of documents other than References 1, 22, 19, 30 and 31 on pages F6.2-119 to F6.2-120 and ANL-8131 and ANL-7607, which systematically describe the experimental basis for the SAS3A code? If so, please identify them.

ANSWER 4

The only references which describe the experimental basis for the SAS3A and SAS3D computer codes are identified in response to question I(A)(6-7) in the Second Set of Interrogatories to Applicants. (See p. AA-5).

QUESTION 5

QUESTION 5(a)

In each of the several core disruptive accidents calculated in the PSAR (identified under (A) through (D) in Part I of NRDC, et al.'s, Fourth Set of Interrogatories to Applicant), including the most pessimistic ones, what fraction of the core remains in, or is converted to, the vapor state after the fuel expands down to one atmosphere pressure (assuming no containment as a bounding calculation)?

ANSWER 5(a)

Referring to (A) through (D) in Part I of NRDC et al.'s, Fourth Set of Interrogatories to Applicant, the fraction of the core in the vapor state

after expansion to one atmosphere is as follows for selected cases in CRBRP-GEFR-00103:

Case	Ramp Rate, ($/sec)	Vapor Fraction
(a) BOEC LOF	40	0.16
	50	0.13
	100	0.20
(b) EOEC LOF	40	0.15
(c) EOEC TOP	50 (partial Na)	0.02
	50 (full Na)	0.005
	75 (full Na)	0.02
	100 (full Na)	0.04
(d) EOEC LOF	28	0.13
Immediate reentry	37	0.13
	45	0.13
	63	0.19
Homogenized Core	30 (21 P_o)	0.18
Reentry	30 (2.1 P_o)	0.20
	30 (210 P_o)	0.10

The case in CRBRP-GEFR-00523 that required a VENUS-II calculation implied an input reactiivity ramp rate of 43$/sec and produced a vapor fraction at 1 atm of about .01.

QUESTION 5(b)

Provide the same answer as in (a) above with the assumption that the closure hold down devices failed after absorbing, say, one half of its design basis mechanical energy, due to faulty equipment.

ANSWER 5(b)

The closure assembly is designed to remain in place and essentially leak tight for the first 1000 seconds following the SMBDB loading. This design assures that sufficient time is available for substantial fuel vapor condensation and aerosol plate-out within the reactor vessel. The closure hold down device will not fail if subjected to the suggested value of one-half its design basis mechanical energy.

QUESTION 5(c)

With respect to the fuel which does not vaporize, does a dynamic theory exist which could predict the particle size of the unvaporized fuel?

ANSWER 5(c)

To the best of the Applicants' knowledge, a dynamic theory does not currently exist which predicts the particle size of the unvaporized fuel.

QUESTION 5(d)

In the worst CDA calculated in the PSAR, what fraction of the core would be expected to be made into an aerosol upon the assumption in (b) of a faulty holddown structure, and what would be the particle size distribution?

ANSWER 5(d)

Section 4.2 and Appendix D of CRBRP-3, Vol. 2, provides this information.

QUESTION 6

The PSAR's analysis of CDAs is based on the assumption that no sodium vapor explosion will occur in the event of molten fuel interacting with coolant.

(a) Is it the position of the Applicant that such explosions are (i) impossible, (ii) highly unlikely, or (iii) unlikely? Quantify your answer if possible.

(b) Please supply or identify the scientific justification for the conclusion in (a) if it is different from the response to Interrogatory II-16 in the NRDC, et al.'s., Third Set of Interrogatories to the Applicant.

(c) Are any whole core meltdown tests being planned or considered to investigate the possibility of sodium vapor explosions? If so, describe and document these fully.

ANSWERS 6(a) and 6(b)

The Applicants' position has been stated in the Applicants' response to Interrogatory II-16 in the NRDC, et al.'s, Third Set of Interrogatories to the Applicant (see p. AA-70) and is further explained in Section 8.2.6 of CRBRP-GEFR-00523.

ANSWER 6(c)

No whole core meltdown tests are being planned by the Applicants to investigate the possibility of sodium vapor explosions.

QUESTIONS 7

QUESTION 7(a)

Could a coolant pipe rupture (without a pipe sleeve) lead to a core disruptive accident or fuel melting even if a SCRAM occurred?

ANSWER 7(a)

It is possible that the most severe pipe rupture (i.e., the hypothetical double-ended rupture) at the worst location (i.e., the reactor vessel inlet nozzle) could result in cladding and fuel melting in some fuel assemblies and radial blanket assemblies. Since the core would be shutdown due to the rapid PPS scram, the initial conditions are considerably different than those of the HCDA's analyzed in CRBRP-3 for which the core is at nearly full power or increasing power. It is not known whether the consequences of the upper-limit pipe rupture at the worst location could result in full core involvement or whether flow would be reestablished with limited local damage. However, there is no reason to expect any substantial energetics in either case because the core is shut down by a large margin.

QUESTION 7(b)

Has the coolant pipe rupture accident without SCRAM been analyzed? If so, please supply the analysis.

ANSWER 7(b)

No, the coolant pipe rupture without scram has not been analyzed because the event would require a combination of a pipe rupture and failure of the redundant and diverse plant protection systems, three extremely improbable events. The probability of pipe rupture is discussed in Reference 7-1 and the probability of failure of both shutdown systems is assessed in Reference 7-2.

References:

7-1. CRBRP-ARD-0185, "CRBRP Integrity of Primary and Intermediate Heat Transport System Piping in Containment," September 1977.

7-2. WARD-D-0118, "Reliability Assessment of CRBRP Reactor Shutdown System," November 1975.

QUESTIONS 8

The PSAR notes that candidate CDA initiators were eliminated because of a determination of low probability (p. F3-1).

(a) Please identify all of the candidates that were considered and eliminated, and in each case supply the supporting basis for the determination of the low probability.

(b) Please identify those that were not eliminated, namely events (b) of Sec. F3.2.

ANSWERS 8(a) and 8(b)

HCDA initators are all of low probability for CRBRP. The CRBRP is designed to prevent the occurrence of conditions under which an HCDA might be initiated as discussed in depth in Section 2 of CRBRP-3, Volume 1.

An assessment of HCDA initiators is discussed in Section 3 of CRBRP-3, Volume 1. The process for selecting and eliminating potential initiators is explained in Section 3.2 of CRBRP-3, Volume 1.

QUESTION 9 (PREAMBLE)

Sodium Void Reactivity - The PSAR (p. F3-1) states that core disruption cannot occur unless a coolable geometry is lost.

QUESTION 9(a)

Does this mean that present analyses do not predict a superprompt-critical power excursion caused by rapid sodium voiding?

ANSWER 9(a)

No

QUESTION 9(b)

Please provide the basis for the answer in (a) above.

ANSWER 9(b)

In some hypothetical core disruptive accident scenarios analyzed (loss of flow with failure to scram), the sodium boiling in the hottest assemblies could provide a positive reactivity feedback that could result in a power excursion and possible core disruption. The Applicants have conservatively considered that coolable core geometry is lost when coolant boiling occurs. Thus, even if the fuel pins are intact at the onset of the power excursion, coolable geometry is already considered to be lost based on the Applicants' conservative application of the coolable geometry definition.

QUESTION 10

The control assembly withdrawal accident at startup was analyzed assuming 600°F inlet temperature and 1 MW initial power (p. F3-3).

(a) Were these parameters varied, such as assuming 1 watt of power?

(b) If so, please provide the detailed analysis.

ANSWER 10

No; a parametric study on initial conditions for unterminated control assembly withdrawl accidents at startup has not been done.

QUESTION 11

Please supply the analysis which supports the conclusion (p. F3-6) that the fuel will remain intact upon a SCRAM.

ANSWER 11

It is assumed this interrogatory refers to a rod drop without pump trip, an event that is not part of the CRBRP plant design bases since protective functions (flux to delayed flux trip) have been incorporated to assure overall plant and pump trip. The event produces a rapid reactor shutdown with full flow, leading to excessive cooling of the reactor, a condition in the thermally safe direction. The rapid decrease in hot leg temperature, however, adversely affects the service life of plant hot leg components. This leads to a requirement for protection system action to prevent the rapid hot leg temperature decrease, hence provision is made to assure a full plant trip, including pumps, for this event.

For small mass, thin section replaceable components such as fuel, rods, and cladding, rapid reactor shutdown with full flow is acceptable, because it results in overcooling below normal operating levels. As indicated in PSAR Section 4.1, fuel rod damage which, if excessive, leads to loss of integrity occurs primarily due to loadings at higher temperatures; i.e., normal operation and abnormal undercooling events.

QUESTION 12

What does PPS mean on p. F3-14?

ANSWER 12

PPS is an acronym for Plant Protection System which is the system which automatically shuts the reactor down neutronically if an abnormal condition exists.

QUESTION 13(a)

Have the possibilities of injection of a gas bubble and an oil slug (hydrogenous material) into the reactor and core been considered?

ANSWER 13(a)

Yes. The possibilities of these occurrences were considered and the design of the plant implemented to preclude them as summarized in Reference 13-1. Furthermore, their impact upon the plant should they occur has been considered (References 13-1 and 13-2).

References: 13-1 PSAR Section 15.2.3.2; 13-2 Response to NRC Question 001.338, Amendment 15 to the PSAR; 13-3 R. W. Hardie and W. W. Little, Jr., "PERT-V, A Two-Dimensional Perturbation Code for Fast Reactor Analysis," BNWL-1162, Pacific Northwest Laboratory (1969); 13-4 J. N. Fox et al., "FORE-II: A Computational Program for the Analysis of Steady State and Transient Reactor Performance," GEAP-5273, General Electric Company (Sept. 1966); 13-5 D. R. Vissers et al., "A Hydrogen Monitor for Detection of Leaks in LMFBR Steam Generators," Nuclear Technology, Vol. 12, p. 218, October 1971; 13-6 "Methods for the Analysis of Sodium and Cover Gas," RDT F3-40, Appendix N, June 1975; 13-7 Applied Physics Division Annual Report, ANL7-190 p. 241, January 1972.

Methodology for Gas Bubble Analysis -- This analysis is reported in Reference 13-1. The sodium void reactivity worths were calculated using PERT-V (Reference 13-3). Large bubbles are assumed to move at the average sodium velocity through the core. The transient variations in reactivity insertion were estimated from the bubble geometry, void worth and bubble

velocity. These reactivity insertions were used as input to the code FORE-II (Reference 13-4) and the transient changes in power and fuel pin cladding temperature calculated. Additional cladding temperature effects due to the insulating effect of the bubble were calculated and the temperature increases summed. These results are presented in Reference 13-1.

Small localized bubbles produce insignificant reactivity effects and only the insulating effect need be considered. If it is conservatively assumed that all the heat is absorbed in the cladding for the duration of bubble contact, bubbles of 5, 7.5 and greater than 12 inches length are required in order to raise the cladding temperature 25°F for fuel, radial blanket, and control assemblies respectively.

Methodology for Oil Slug Analysis — If oil enters the sodium in the pump tank, it will react to form carbonaceous particles and gases and some hydrogen (Reference 13-2). Since the reaction will be at or near the surface of the sodium, the gases would rise into the pump cover gas and be reprocessed by the RAPs. Any hydrogen which does not enter the cover gas would tend to go into solution. This requires some hydrogen in the cover gas in order to provide a partial pressure to maintain the hydrogen in solution. This sodium from the pump tank will mix with the sodium in the affected PHTS loop and further mix with the sodium in the two other loops in the reactor vessel inlet plenum. Hydrogen will come out of solution at the surfaces with other cover gas spaces in order to establish an appropriate partial pressure in all gas spaces. The saturation level of hydrogen in sodium at 730°F is 51 parts per billion (References 13-5 and 13-6) corresponding to 24.3 kg hydrogen distributed throughout the primary sodium. In order to attain this hydrogen level, it would require more than 2 years continuous oil leakage from one pump at 10 c.c./hour and erroneously replenishing the oil supply eight times during this period. The estimate also conservatively assumes that all hydrogen is liberated as gas, no gas enters the pump cover gas, the cold trap does not remove any hydrogen and no hydrogen enters the other cover gas spaces in contact with the primary sodium.

Using hydrogen worths based on ZPPR-2 measured values (Reference 13-7), this corresponds to a reactivity ramp rate of 0.8 x 10^{-4} cents/hour to a maximum insertion of 2.0 cents based on uniform hydrogen distribution of the hydrogen in the core and blankets.

Consideration of the relatively insoluble carbonaceous gases formed by the oil/sodium reaction (Reference 13-2) shows that they will gradually build up with time. The time required for this uniformly distributed gas, assuming no depletion, to cause a reactivity insertion of 1 cent is about 2,900 hours.

This conservatively assumes that the temperature and pressure of the gas corresponds to hot leg conditions throughout the core and blankets.

QUESTION 13(b)

What absolute assurance is there that such possibilities won't occur?

ANSWER 13(b)

The plant has been designed so that gas bubbles will not accumulate. However, even if a significant size bubble should enter the reactor vessel inlet plenum, the turbulence would result in dispersion of the bubble into small bubbles before entry to the core and large coherent bubbles entering the core are impossible. These small bubbles would have little effect of reactivity. This is substantiated by tests in support of the FFTF. (Reference J. Muraka, et al., "Gas Bubble Dispersion Test Reactor Inlet Model," HEDL-TME-71-69, May 1971).

The bearing/seal assembly concept is not required to function normally in order to keep the oil out of the primary sodium. Separation is maintained by design features which include oversized drainage reservoirs capable of holding the entire oil inventory even in the event of mechanical failure of the face seals. Moreover, due to the reaction of oil and sodium, it is

impossible for oil to reach the core. A significant buildup of hydrogen would require at least the following failures:

.. failure of the pump lubricating oil level indication systems,

.. failure to take action for a period of time depending upon the oil seal leakage rate in spite of oil level alarms,

.. erroneous replenishment of the oil supply.

.. failure of the RAPs to remove hydrogen from the cover gas.

.. failure of the cold trap system to maintain the hydrogen level in the sodium at 0.1 p.p.m.

The consequences of oil leakage, should it occur despite the aforementioned protective features, are negligible and "absolute assurance" of no leakage is not required.

QUESTION 14

What are all of the identifiable or known possibilities for ramps and steps "beyond the PPS design basis" (see p. F3-15)?

ANSWER 14

No mechanistic sources for ramps or steps beyond the PPS design basis have been identified or are known possibilities. Section 3 of CRBRP-3, Volume 1, discussed ramps and steps beyond the PPS design basis and showed that the PPS has a substantial margin for accommodating reactivity insertions beyond its design basis. Large reactivity insertions that could exceed the PPS capability could only result from material motions after coolable geometry is already lost in a postulated severe accident. Thus, very large

reactivity insertion rates (tens of dollars per second) have been considered for CRBRP in the context of the progression of accidents with postulated failure of both shutdown systems and which lead to loss of core coolable geometry.

NOTE: Question 15 pertains to a postulated fuel assembly flow blockage accident.

QUESTION 15(a)

Has this accident (with and without SCRAM) been analyzed (calculated) to determine the course it could take (see p. F3-19)

ANSWER 15(a)

This event has not been analyzed since a complete flow blockage is precluded by the design. However, the consequences are considered to be enveloped by the TOP and LOF events analyzed in detail in Sections 2.1.1 and 2.1.2 of CRBRP-GEFR-00523.

Since the Fermi blockage incident, major changes have been introduced in the design of fuel assemblies and the inlet structures to preclude fuel assembly inlet blockage. The design of the inlet nozzle of the fuel assembly and the inlet module and module liner are presented in Sections 4.2.1 and 4.2.2 of the PSAR respectively.

QUESTIONS 15(b)

If so, please provide the analysis.

ANSWER 15(b)

Not applicable.

NOTE: Question 16 pertains to a hypothetical TOP at startup without scram.

QUESTION 16

Has this accident been analyzed (calculated) for the course it could take (<u>see</u> p. F3-19)? If so, please identify where in the PSAR this accident is discussed. If not, why not?

ANSWER 16

An analysis has not been performed for the continuous control assembly withdrawal at startup with shutdown system failure for the current design. Section 6.3 of CRBRP-GEFR-00103 contains analysis of this event for the homogenous design.

NOTE: Question 17 pertains to a hypothetical OBE without scram.

QUESTIONS 17

(a) Has this been calculated to determine the course it could take?

(b) If so, please provide the analysis.

ANSWERS 17(a) and 17(b)

This event has not been analyzed for the current core design. Parametric analysis in Section 8 of CRBRP-GEFR-00103 shows the effects of step reactivity combined with an LOF-HCDA for the homogeneous core.

QUESTIONS 18

QUESTION 18(a)

What is the basis for the assumption that only one rod will stick in light of the Monticello (BWR) experience where the control rod drive mechanism exhibited internal cracking that would have prevented twenty three rods from scramming? Why not assume two, three, or more stuck rods?

ANSWER 18(a)

This question has been discussed with the Monticello Plant Superintendent and his staff. They state that no such occurrence, or anything relatable to it, occurred at Monticello.

There is no 'assumption' that only one rod will stick. However, there is a basic criterion that requires both the primary and secondary systems to be able to perform their intended design and plant equipment protection function even if one of the shutdown systems fails entirely and one rod sticks in the system that operates. This requirement results in the design having considerable capability to accommodate more than one stuck rod while satisfying safety considerations. Additionally, it is pointed out that diversity between the CRBRP primary and secondary shutdown system mechanical design minimizes the potential for common failure of both systems.

QUESTION 18(b)

Have any CDA or DBA analyses ever been performed assuming more than one stuck rod? If so, please provide the analyses.

ANSWER 18(b)

Yes. The HCDA analyses reported in CRBRP-3 typically assumes no additional control rods are inserted in either of the two shutdown systems following the initiation of the transient. The DBA analyses reported in Chapter 15

of the PSAR typically assume one of the two shutdown systems is inoperable and one rod is stuck in the operable system.

QUESTION 18(c)

How many stuck rods can be tolerated for each of the various CDAs or DBAs considered without exceeding the CRBR design basis — assuming the worse set of stuck rods? (See p. F3-22).

ANSWER 18(c)

See Answer 18(b) above for the failure assumptions used in the analyses. The minimum number of rods required is dependent on the challenging event and when the event occurs. (The number of rods available or required to scram is dependent on variables such as power level, time in fuel cycle, individual rod worths, and the type of transient, e.g., loss of flow or overpower.) No assessment of the minimum number of control rods required to terminate each DBA transient is available. The question does not apply in the case of the HCDA where the assumption of no rods operating is a prerequisite.

QUESTION 18(d)

Identify, describe, and document all abnormal incidences in commercial, experimental, military, and production reactor — domestic and foreign -- related to, or associated with control rod or safety rod failure, including failure to scram, accidental withdrawal, stuck rod(s), and errosion design, installation, repair, and operation. (This question is also related to Contention 2.)

ANSWER 18(d)

The Applicants do not maintain a file which documents abnormal occurrences associated with the control rods as delineated in the question. However,

data sources containing the information are continually and routinely reviewed by design and safety personnel in the course of their work. The data sources are available to the public and include the following:

a) NRC

Quarterly Reports to Congress
Computer Listings of Licensee Events Reports
Operating Unit Status Reports
Special Topical Reports

b) Nuclear Safety Information Center (NSIC/ORNL

Unusual Occurrence Reports
Special Topical Reports

c) Institute for Nuclear Power Operations (INPO)

Additionally, a comprehensive test program has been initiated to verify the design, performance, and safety aspects of the CRBRP control rod system design. This program is described in the PSAR, Appendix C. The program will provide data which is prototypic to CRBRP. Information obtained from the review of the data sources listed above supplements the data obtained from this test program.

QUESTION 19

Does the parallel design involve any core design changes, such as the Doppler coefficient, relative to the reference design?

ANSWER 19

Not applicable. The "parallel design" has been deleted from the license application.

QUESTION 20

Will the sealed HAA be tested to withstand its design basis CDA explosion using simulant explosions? (See p. F4-1). If not, why not?

ANSWER 20

Not applicable. The license application does not contain provisions for sealing the head access area.

QUESTION 21

Identify or supply the analysis which demonstrates that criteria (c) of Sec. F5.3.2 is satisfied (p. F5-4).

ANSWER 21

Criterion (c) of F5.3.2 is no longer applied to in-vessel equipment. The current reactor component design requirements are specified in Section 5.2 of CRBRP-3, Volume 1. The requirements for in-vessel equipment are in Section 5.2.3 of CRBRP-3, Volume 1.

QUESTION 22

The Vessel Closure Head and other components and systems of Section F5.3.3 and F5.3.4 are designed to withstand the structural loads of the design basis CDAs. Has any analysis ever been performed of the consequences of failure of any, several, or all of these components under such loading? If so, please provide the analyses.

ANSWER 22

As discussed in CRBRP-3, Volume 1, the components are required to withstand the hypothetical core disruptive accident dynamic loads without the

formation of missiles or excessive leakage. Designing to the CRBRP-3, Volume 1 (Appendix B) requirements provides assurance that the components will maintain their integrity under HCDA dynamic loading conditions. The CRBRP Safety Study (CRBRP-1) includes analysis of consequences of hypothetical accidents, some of which imply failure of the vessel closure head or other components.

QUESTION 23

Has the Beginning-of-Life (new core; zero burnup throughout) condition ever been analyzed with regard to CDAs using the same qualitative degree of pessimism used in Appendix F to explore the worst possible course the TOP and LOF initiator accidents might take? The concern here is the lack of fission gases which the PSAR expects would likely cause self-shutdown for the cases of BOEC and EOEC.

ANSWER 23

The analysis of a Beginning-of-Life core for the current design is in CRBRP-GEFR-00523. This analysis accounts for lack of fission gases.

QUESTION 24

With respect to the report by H. K. Fauske, identified as Reference 5 on p. F6.2-119, and the conclusion of Section F6.2.1 that no vapor explosion would occur in any core melting event, do the authors of References 5 and 6 on p. F6.2-119 agree with this interpretation?

ANSWER 24

The authors of References 20 and 63 in CRBRP-GEFR-00103 agree with the position that the occurrence of coherent energetic vapor explosion in the CRBRP environment with oxide fuel can be considered highly unlikely. For further details see Section 8.2.6 of CRBRP-GEFR-00523.

QUESTION 25 (PREAMBLE)

The PSAR states that point kinetics is used in SAS3A (p. F6.2-6) and VENUS (p. F6.2-105-106). Point kinetics is an approximation to the method of calculating the reactivity from eigenvalue differences using static diffusion theory, which is an approximation of the FX-2 type factorization method (quasi-static method), which is an approximation of the time-dependent diffusion, which in turn is an approximation of time-dependent neutron transport theory.

QUESTION 25(a)

What specific allowances for the error of each of these approximations have been made, if any, and what is the mathematical basis for the magnitude of these allowances?

ANSWER 25(a)

No specific allowances for the errors which might exist in the approximations mentioned were made, since such errors were judged to be small for CRBRP. The mathematical bases for the magnitudes of such errors are as follows:

(1) With respect to the differences between diffusion theory and transport theory, the estimates of reactivity changes will not be significantly different unless significant void spaces exist, such that the diffusion coefficient becomes infinite over large areas of the core. Such differences do not exist in any of the cases considered in CRBRP-GEFR-00103 or CRBRP-GEFR-00523 so that diffusion theory is adequate. The mathematical basis can be found in discussion on the relationships between neutron diffusion theory and neutron transport theory such as in Reactor Handbook, H. Soodak (Editor) Interscience, John Wiley and Sons, second edition, p. 140.

(2) With respect to the differences among approximate methods of solving the time-dependent diffusion equation, namely, point kinetics, eigenvalue differences using static diffusion theory ("adiabatic" methods), and factorization methods, the mathematical basis can be found in Reference 1 below.

It is shown therein that all three methods are factorization methods, differing only in their methods of testing the time dependence of the flux shape.

QUESTION 25(b)

In Reference 75 (PSAR, p. F6.2-124), the error between the static eigenvalue method and the FX method was found to be 33% non-conservative in the energy yield of the power excursion; and yet a stronger Doppler coefficient was used than the values used in the CRBR PSAR. What is the error between these two methods for the CRBR cases considered, i.e., using CRBR Doppler coefficients and the CRBR CDAs considered.

ANSWER 25(b)

No comparison was made between (i) FX2-VENUS space-time kinetics (ii) FX2-VENUS with point kinetics ("static-eigenvalue method"), and (iii) VENUS-II (point kinetics) for CRBRP.

QUESTIONS 25(c) and 25(d)

(c) What is the basis for not performing the CDA analyses (SAS and VENUS) with at least the quasistatic method?

(d) The PSAR (p. F6.2-106) asserts that the inaccuracies incurred by not using the quasi-static method are not serious, and cites Sha, et al. (Reference 75) for the basis. Yet, Reference 75 concludes that "The cases

described here do not constitute an adequate test of these approximations in any sense" (o. 147). Specifically, how is it judged from Reference 75 that the error of SAS/VENUS relative to the quasi-static neutron kinetics approximation would not be serious?

ANSWERS 25(c) and 25(d)

The Applicants response to Interrogatory II-1(a-c) of the Sixth Set of Interrogatories to Applicant (see p. AA-138) presents an analysis which shows that the quasistatic method is not significantly more accurate than point kinetics with worth tables generated using the linearized leakage treatment for the initiating phase (SAS) for CRBRP analyses. Regarding the disassembly phase, the comparisons provided in Reference 68 in CRBRP-GEFR-00103 for the high temperature initial conditions provide the basis for concluding that the results predicted when not using the quasistatic method do not differ significantly.

QUESTION 25(e)

What is the basis for the PSAR's conclusion that the plant design can safely contain the consequences of a CDA, in view of the fact that Sha, et al., (Reference 75) qualified the quasistatic method (known theoretically to be at least less approximate than the simple neutronic model used for the PSAR) in regard to accuracy, stating, "Obviously, no firm statement can be made concerning the errors introduced by these models (used in the quasistatic method)"? In this regard, it has been noted by Meneley, et al., that the quasistatic method is an approximation of neutron diffusion theory, which is in itself an approximation of neutron transport theory. (See Reference 74, p. 486).

ANSWER 25(e)

The quote, from page 151 of Reference 68 in CRBRP-GEFR-00103 was not intended to be a qualification of the quasistatic method but rather referred to methods used by the "analysts" who "apply a point kinetics

model with coefficients calculated for conditions which are representative of the conditions expected during the transient." In other words, the amount of error introduced by using point kinetics is really case-dependent. The quasistatic method itself is a numerical method of solving the time-dependent diffusion equations, which has been shown (2,3) to converge to the solution as calculated using finite difference methods on a very fine mesh, as the number of shape function calculations is increased. The point kinetics formula is one in which only one shape function is used to describe the space dependence whether this shape is selected to be for the state at the beginning of the transient, or for a state expected at some time during the transient. If the transient under consideration does not include gross material motions, such a use of point kinetics may be adequate. For the CRBRP cases analyzed and presented in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 this was indeed true, as is demonstrated in the answer to Interrogatory II-1 in the Sixth Set of Interrogatories (see p. AA-138).

QUESTIONS 25(f) and 25(g)

(f) Furthermore, it is noted that even though the SAS/VENUS calculations are ultimately based on diffusion theory, Boudreau, et al., (A Proposal for Computer Investigation of LMFBR Core Meltdown Accidents, for Alamor, LA-UR-74-243 (undated), pp. 14, 28), have concluded that time dependent transport theory is necessary, at least in order to investigate the possibility and severity of secondary criticality (secondary power excursions). In fact, in one calculation, the reactivity was underpredicted by diffusion theory by about 3% K reactivity units (p. 16), which would be disastrous if a substantial amount of this reactivity could be "inserted" in an explosive recompaction event.

Is Boudreau, et al.'s analysis correct? (Here, we have reference to a private communication between Webb and D. Ferguson, ANL). If not, why not? If it is correct, in answering (e) above, also be responsible to Boudreau et al.'s analysis and conclusions cited here.

(g) If Boudreau, et al.'s analysis is incorrect, discuss in detail why the Applicants believe diffusion theory is still adequate for all CRBR CDA analyses of interest.

ANSWERS 25(f) and 25(g)

Boudreau's analysis is not correct. An inconsistency was present between the two treatments of the diffusion terms. When this inconsistency is removed, much closer agreement results. The important point to be derived from Figure 3, on p. 16 of LA-UR-74-243, is that, even though the magnitudes differ, the slopes of the curves of k_{eff} vs separation distance are nearly the same. The slope is what determines the reactivity insertion rate; the energy release from a disassembly calculation is dependent on the ramp rate chosen. When the two curves are brought together by moving the TWOTRAN-II curve downward, it will be seen that the slope at prompt critical (Δk units/cm) will be nearly the same for each curve. With a knowledge of the slug velocity at this time, one can then derive a ramp rate (Δk/sec), which can then be used to drive the hydrodynamic disassembly.

QUESTION 25(h)

What (even if considered outside the area of interest) would have to be postulated before diffusion theory would be considered inadequate? Explain in detail.

ANSWER 25(h)

Only a recriticality event in which very large voids were present at the time of prompt criticality would require consideration of the use of transport theory. Such a condition has not been identified in any of the CRBRP analyses performed to date.

QUESTION 25(i)

Please supply Reference 25 in the Boudreau proposal, namely, LA-4432 - Theory and Use of TWOTRAN.

ANSWER 25(i)

This document has been or will be made available for inspection and copying.

QUESTION 25(j)

As noted by Boudreau, et al., transport theory predicted a secondary power excursion, whereas diffusion theory did not, for the case examined (p. 14). In light of this result is the CRBR project planning to reanalyze the CDAs using time-dependent transport theory; or is it not practical to do so? If not, why not? If so, please discuss the details of the proposed program.

ANSWER 25(j)

There is no plan to reanalyze HCDAs using time-dependent transport theory. The reasons are found in the responses to parts (a)-(h) above.

QUESTION 25(k)

On what grounds can the Applicant justify the safety of the CRBR relative to the CDAs analyzed in light of the apparent need for time-dependent transport theory? The answer to this question can be included in (c) above (See (f) above.)

ANSWER 25(k)

As indicated in parts (a)-(j), diffusion theory is adequate for HCDA analyses for the CRBRP.

QUESTION 25(l)

In the Boudreau TWOTRAN transport calculation, which used the SN method, what was the order of quadrature (N = ?)? Presumably, Boudreau used a (rest of question missing)

ANSWER 25(l)

The Boudreau TWOTRAN "transport" calculation was a P_o calculation. The S_n method was not used.

QUESTION 25(m)

Has this order used been established to be sufficiently accurate within the limits of the S_n method, by comparison with higher order calculations?

ANSWER 25(m)

A P_o calculation is no more accurate than diffusion theory calculations.

QUESTION 25(n)

In several of the CDAs a cavity is predicted or postulated to develop. What order quadrature would be necessary to accurately calculate the reactivity? (Present the mathematical analysis in support of the answer).

ANSWER 25(n)

Such an analysis has not been carried out. Diffusion theory is sufficiently accurate to analyze all cases of interest. Therefore, S_2 is also sufficiently accurate.

QUESTION 25(o)

It should be noted that TWOTRAN is a two-dimensional S_n method. Are there any plans to calculate any of the more severe CDA using three-dimensional S_n method, say with N = 4?

ANSWER 25(o)

There are no plans to calculate any HCDAs using three-dimensional S_n methods.

QUESTION 25(p)

Is there any analysis which shows that the additional accuracy afforded by a 3-D, S_4 method is not needed? If so, please provide.

ANSWER 25(p)

As indicated in parts (a) - (j), the methods presently utilized for CRBRP analyses of HCDAs are adequate. No analysis exists which would show the benefits, if any, of utilizing a 3-D, S_4 method of analysis.

QUESTION 25(q)

Has time-dependent transport theory ever been compared with an adiabatic-type transport calculation, similar to the classic space-time calculations of Yasinsky and Henry (N. S. and E., 22: 171-181)? For the CRBR core design?

ANSWER 25(q)

No. An indirect comparison exists in a paper by E. L. Fuller in Reference 4. In this work static diffusion theory, S_4, and time-dependent diffusion

theory (FX2) are compared in an analysis in which void spaces were present. Agreement was very close among these three methods. Since the major differences between "adiabatic" methods and "exact" space-time methods lie in the treatment of the variation of the precursor concentrations, and since the transients resulting from HCDAs are very rapid, it is concluded that time-dependent transport theory would give essentially the same results as eigenvalue differencing using transport theory for the cases of interest.

QUESTION 25(r)

To use time-dependent transport theory, there is the question of the "stability" of the S_n finite difference calculational method. In 1964 Clark and Hansen noted that no proof of numerical computations stability exists (Numerical Methods of Reactor Analysis, p. 223).

(i) Has a proof now been made? Where?

(ii) What is the significance of this?

(iii) Should a calculational result be unstable, can it be shown that it will always be obvious?

(iv) Could it be that a result will be grossly in error without any indication of a calculation being on the verge of being unstable or slightly unstable?

ANSWER 25(r)

(i) The applicant is not aware of such a proof.

(ii) The fact that a mathematical proof for the unconditional stability of the S_n finite difference calculational method does not currently exist is of little significance in regard to the practical application of the method, and is of no significance to CRBRP HCDA evaluations.

(iii) It cannot be mathematically shown that, should a calculational result be unstable, it will always be obvious. However, should a numerical

instability occur, it will usually be discovered by a careful inspection of the calculational results.

(iv) To the best of the Applicant's knowledge it has not been shown that a result could be grossly in error without any indication of a calculation being on the verge of being unstable or slightly stable.

QUESTION 25(s)

What is the status of the computer investigation proposed by Boudreau, et al., at Los Alamos? Have any results been issued in report (including draft) form? If so, please provide these.

ANSWER 25(s)

The SIMMER code development by Boudreau et al. is still in progress.

QUESTION 25(t)

Please supply the one-group and two-group diffusion constants* for the CRBR, for various fuel zones at various depletion states for each of the cases, BOL, BOEC, and EOEC.

*Namely, D, Σ_{tr}, Σ_{a}, ν, Σ_{f}, Σ_{s}, $\Sigma_{1\rightarrow 2}$, etc.

ANSWER 25(t)

One-group and two-group constants do not exist for the CRBRP. The sets that were used for the CRBRP-GEFR-00103 calculations were the 27-group set described on page 5-1 and the nine-group set, mentioned on pages 10-16 of CRBRP-GEFR-00103. A similar set was used in CRBRP-GEFR-00523.

QUESTION 25(u)

In the SAS calculation the use of neutron diffusion theory may be considered by the Applicant as having been shown to be an adequate approximation of transport theory by Ferguson, et al. (Reference 3, PSAR, p. F6.2-119), citing Ferguson, et al.'s, fair agreement using an S_4 calculation. However, Ferguson, et al., emphasized that they assumed that the S_4 calculation is sufficiently accurate (Reference 3, p. IX-68). What plans, if any, does the Applicant have to verify the validity of this assumption?

ANSWER 25(u)

The Applicants have no plans to verify the validity of the assumption mentioned.

QUESTION 25(v)

Do any of the CDAs analyzed in the PSAR involve significantly more fuel slumping and/or voiding of core materials which may show a greater transport effect than the fuel slumping situation calculated by Ferguson, et al.?

ANSWER 25(v)

The fuel slumping situation mentioned involves about the same amount of fuel as is involved in the immediate-reentry analyses in CRBRP-GEFR-00103. In addition, the relative degrees of sodium voiding are similar.

QUESTION 25 (References)

1. K. O. Ott and D. A. Meneley, "Accuracy of the Quasistatic Treatment of Spatial Reactor Kinetics," Nucl. Sci. Eng. 36, 402-411 (1969).

2. E. L. Fuller, "One-Dimensional Space-Time Kinetics Benchmark Calculations," Argonne National Laboratory, Applied Physics Division Annual

Report, July 1, 1970 to June 30, 1971, ANL-7910, pp. 497-502 (Jan. 1972).

3. H. L. Dodds, Jr., "Accuracy of the Quasistatic Method for Two-Dimensional Thermal Reactor Transients with Feedback," Nucl. Sci. Eng. 59, 271-281 (197?)

4. E. L. Fuller, "Reactivity Effects of Core Slumping in Fast Reactors: A Case Study," Argonne National Laboratory, Applied Physics Division Annual Report, July 1, 1971 to June 30, 1972, ANL-8010, pp. 583-587 (1976).

QUESTION 26

Is there any consideration whatsoever being given to core destruct experiments, including partial core destruct experiments? If so, please provide a detailed description of what is planned in this area, including when results are expected. Please provide all documents related to any considerations given to core disruptive experiments.

ANSWER 26

The Applicants have given no consideration to core destruct experiments and has produced no documents relative to such experiments.

QUESTION 27

The SLUMPY fuel motion model contains a pseudo-viscous pressure for the purpose of providing numerical stability.

(a) What is the physical basis for this pseudo pressure?

(b) Is this the von Neumann term which is founded on the Hugoniot shock relation?

ANSWER 27

The SLUMPY pseudo-viscous pressure provides an automatic treatment of shocks that can arise due to the low sonic velocities that can occur in the two-phase system that SLUMPY models. The algorithm is based on the two-phase formulation for two-phase calculations without a significant increase in the widths of the shock fronts (H. U. Wider et al., "An Improved Viscous Pressure Formulation for Two-Phase Compressible Hydrodynamics Calculations," Trans. Am. Nucl. Soc., 17, p. 246, 1973). In a single-phase situation, this formulation indeed reduces to the von Neumann term founded on the Hugoniot shock relation (J. von Neumann and R. D. Richtmyer, "A Method for the Numerical Calculation of Hydrodynamic Shocks," J. of Applied Physics, 21, pp. 232-237, 1950), with the improvement that the pseudo-viscous pressure is zero when the material is undergoing an expansion (as of a free surface) (R. D. Richtmyer, Difference Methods for Initial-Value Problems, Interscience Publishers, Inc., New York, pp. 210-211, 1957).

QUESTION 28(a)

Can the Doppler coefficient be readily varied (reduced) in the core design change without significantly affecting thermal, mechanical, and hydraulic design?

ANSWER 28(a)

The Doppler coefficient in CRBRP cannot be readily varied by core design changes without affecting thermal, mechanical, and hydraulic design conditions. Any fuel or core design changes which would be considered to increase the breeding capability are constrained by pump head (coolant pressure drop), fuel lifetime, and other thermal/hydraulic or mechanical limits. The Doppler effect in CRBRP is primarily a function of the U-238 mass (fertile-to-fissile ratio) and the fraction of the neutron flux in the resonance range (neutron moderation).

QUESTION 28(b)

Please supply a curve of breeding "doubling time" versus the Doppler coefficient (sodium in/out).

ANSWER 28(b)

The CRBRP breeding ratio is a performance parameter and, as such, it is only calculated for nominal (sodium in) reactor conditions. However, a unique curve of breeding ratio versus Doppler coefficient is not meaningful due to the dependency of, for example, Doppler coefficient on U-238 content, fuel composition, neutron moderation (spectrum) from steel and sodium content, etc., and the variety of core design changes which could be proposed to result in a particular breeding ratio change. The change in the Doppler coefficient is assessed explicitly for any particular fuel or core design change considered for CRBRP.

QUESTION 29

In each of the SAS3A predictions of self-shutdown by fuel ejection that were considered in the PSAR, what percentage of the fuel in the core is ejected from the core?

ANSWER 29

CRBRP-GEFR-00523 describes the analysis for the current core design.

QUESTION 30

In the various CDAs analyzed, what are core average and (local) maximum fractions of fission gases that were originally in the core, that bubble out of the molten fuel prior to any recriticality event, including power excursion event?

ANSWER 30

The analysis of recriticality events in CRBRP-GEFR-00103 did not consider the presence of fission gas as a dispersant. Therefore, 100 percent of any fission gas present was assumed to bubble out of the molten fuel prior to any recriticality event, including power excursion event.

QUESTIONS 31 (PREAMBLE)

In view of the four-fold difference in the neutron mean free path between UO_2 and sodium, one might expect a streaming effect of neutrons along the coolant channels (sodium in). When fuel slumping occurs in which there is a substantial loss of core geometry, this streaming effect might conceivably add a significant reactivity effect of core compaction or expansion.

QUESTION 31(a)

Has it been shown theoretically or experimentally that such a streaming effect is negligible?

ANSWER 31(a)

Neutron streaming has a negligible reactivity effect in CRBRP. Even though fuel (UO_2) and coolant (sodium or partially sodium void condition) have substantially different neutron mean free paths, neutron streaming is negligible when these materials are relatively homogeneously mixed because the medium is then essentially isotropic. Consideration of the CRBRP fuel assembly design shows the fuel and coolant are relatively homogeneously mixed, i.e., 0.23 inch diameter fuel rods with a pitch-to-diameter ratio of 1.256.

Neutron streaming can only have a non-negligible effect if the fuel and coolant are arranged in a very heterogeneous manner such that (1) the

radial fuel dimension represents a substantial fraction of the neutron mean free path and/or (2) a two-dimensional streaming path exists. Neither of these conditions exists in CRBRP.

Therefore, geometric considerations alone are sufficient to demonstrate that neutron streaming in CRBRP is negligible.

QUESTION 31(b)

In regard to neutron streaming in the case of sodium out, such as the loss of flow without SCRAM, which is predicted by SAS to have much or most of the core voided of coolant, have any theoretical estimates been made as to the reactivity effect of the streaming alone?

ANSWER 31(b)

Theoretical estimates have been made of the reactivity effects of neutron streaming in CRBRP with sodium out. These results were presented in Reference 1. From the results of three-dimensional Monte-Carlo Calculations (Reference 1) heterogeneous neutron streaming effects are not expected to be important for the sodium void worth and cladding worth in CRBRP.

QUESTIONS 31(c)

The NRCs Reactor Safety Research Program (NUREG 75/058, pp. 26-27) notes that Monte Carlo calculations of streaming are being attempted.

(i) To what extent is the Applicant, its consultants, or other researchers performing these particular calculations? Identify where and by whom this work is being performed.

(ii) Are there any results as to reactivity estimates?

(iii) What is the purpose of these calculations?

(iv) Does (and if so, how does) the Applicant, or its consultants expect to use the Monte Carlo calculations in assessing the safety of the LMFBR?

(v) Please supply all documents relating to research work identified above, including the research proposal, ERDA approval and commenting memoranda.

ANSWER 31(c)

The Applicants have not depended on the document referenced in the interrogatory. As stated in response to parts a) and b) above, the Applicants has considered the potential impacts of neutron streaming and the impact is not expected to be significant.

QUESTION 31(d)

Kohler and Ligow predicted a reactivity effect due to neutron streaming in the Gas-Cooled Fast Reactor of about 1% ΔK (Nuc. Sci. & Eng., 54:357-60). Discuss any and all considerations given by the Applicant or its consultants or experts known to the Applicant to the possibility that during a "hydrodynamic disassembly," e.g., severe power excursion in a core voided substantially of coolant, as the fuel rods swell during the early phase of the excursion that the reduction of neutron streaming may cause an autocatalytic reactivity feedback effect?

ANSWER 31(d)

The only analysis applicable to CRBRP known to the Applicants is that presented in Reference 1. From this analysis it is judged that such considerations are not important for CRBRP.

QUESTION 31(e)

What theoretical effort, if any, has or is being undertaken to include this process (changes in neutron streaming) in the SAS/VENUS CDA analyses? Discuss fully any results of this effort.

ANSWER 31(e)

The Applicants have reviewed reference 2 for its applicability to CRBRP HCDA analysis.

QUESTION 31 (References)

1. F. E. Dunn and R. Lell, "Heterogeneous Neutron Streaming Effects in the Clinch River Breeder Reactor," Trans. Am. Nucl. Soc., 22, 373 (1975).
2. Gerald Lee Goldsmith and Richard B. Nicholson, "Reactivity Due to Neutron Streaming in the Voids of A Bubbly Pool Core, Design Basis Accident Studies, Final Report, Richard B. Nicholson, editor, Ohio State University, COO-2286-3, pp. 104-171 (1974).

QUESTION 32

Professor R. B. Nicholson was an AEC consultant in LMFBR safety research after March 1972. Please supply all of the documents, writings, papers, articles, letters to the AEC, by Dr. Nicholson in which the results of his research are presented and discussed.

ANSWER 32

The Applicants are not familiar with Dr. Nicholson's precise role and activities as an AEC consultant on LMFBR safety research. The Applicants believe that he may have been a consultant to AEC Regulatory. He has not been a DOE consultant or involved in CRBRP safety analyses. The Applicants do not have the documents, writings, papers, etc., by Dr. Nicholson in which the results of his research are presented and discussed.

QUESTION 33

In the various loss of flow accidents, what consideration has been given to the possibility of one or more coolant pumps restarting, which could conceivably lead to rapid compaction of a pliable core — made pliable by overheating — due to the onrush of returning coolant flow? Please supply all relevant analyses and supporting documents.

ANSWER 33

If one considers the extremely low probability case of loss of off-site power and the resulting main pump loss coupled with the failure of the primary and secondary control assemblies to insert, one could attempt to restart the primary heat transport system pumps if off-site electrical power were restored. However, flow would not return automatically on reestablishing off-site power.

However, restarting the primary pumps even if electric power were available is not a simple one-step operation. To satisfy installed interlocks, the operator would have to reestablish lube oil flow to the pump power supplies (motor-generator sets) before reenergizing the pump power supplies. Procedures would then require establishing intermediate flow before attempting to establish primary coolant flow. Even if the procedures are violated, the primary heat transport system pump breakers are interlocked with the shutdown systems to prevent establishing primary flow whenever any scram breaker of the primary electrical subsystem or solenoid valve of the secondary electrical subsystem is open. This interlock with primary pump breakers would prevent reestablishing flow in this postulated extreme low probability case. With this plant design, it is not reasonable to postulate that the LOF HCDA could occur, offsite electrical power is restored, and a primary pump is restarted.

QUESTION 34

What consideration, if any, has been given to whole core destruct tests with zero burnup and fueled only by U-235 (no plutonium)? Such tests would avoid a serious radioactivity hazard.

ANSWER 34

No consideration has been given by the Applicants to whole core destruct test of any type and therefore not to the test proposed

QUESTION 35

Please list all of the physical differences between each of the TREAT experiments and CRBRP design, including but not limited to differences in fuel rod and lattice dimensions, coolant flow rates and temperatures, fuel burnup levels, fuel composition, fuel rod height, axial blanket lengths, power output, fuel temperatures (initial), reactor transient period. (Simply citing references to TREAT and CRBR data would not be an adequate response to this question.)

ANSWER 35

The TREAT reactor has been used for many experiments, a considerable number of which are not applicable to the CRBRP HCDA analysis and which, therefore, have not been used to support CRBRP HCDA evaluations. Within the context of the subject matter questioned, those physical differences believed to be significant and associated only with those TREAT tests that have been utilized to support the first principles modeling and engineering judgments employed in the CRBRP HCDA analyses can be identified.

The TREAT reactor is a transient test facility having a thermal neutron flux with no significant cooling system. It operates on the principle of utilizing the heat capacity in the reactor fuel to absorb the energy generated during the transient. To conduct fuel-in-sodium tests the TREAT

facility requires that the test be conducted in test capsules which contain the sodium and fuel test specimens.

A variety of these sodium filled capsules have been designed for use in the TREAT facility. Three major categories of capsules can be identified: static capsules, circulating sodium capsules (Mark II loop capsules), and transient sodium flow capsules (R-loop capsules). Each of these types of capsules introduces physical differences into the experiments. In addition to the test vehicle introducing its own differences, the experimental capsules may contain a single fuel pin, multiple fuel pins, or actual fuel pins and simulated fuel pins.

The experimenter's choice of capsule and fuel pin test assembly design is a function of the physical phenomena to be investigated. Attempts are made to produce the best approximation to the physical phenomena to be investigated within the constraints of capsule type and test fuel pin design. Another extremely important variable is the instrumentation capability associated with the capsule. In the interest of obtaining greater amounts of measured data, the experimenter may accept additional differences in the test and reactor condition. It should, therefore, be understood that several physical differences may exist between a TREAT experiment and a particular reactor design.

Many of these differences do not have an important effect on the principal physical phenomena that the TREAT test may be designed to investigate.

Table I summarizes what the Applicants consider to be the significant physical differences that exist between the TREAT experiments that have been most directly utilized to support the validity of the CRBRP HCDA analysis and the CRBRP design.

TABLE I. Comparison of TREAT Test Parameters and CRBR Design Parameters

Design Parameter	Units	CRBR Typical Value or Core Wide Range at Power (975 MWt)	TREAT EXPERIMENT VALUE[1] R4,R5, R6,R7, R8	L2	L3	L4	L5	C4A	C4B	C5A,C5B	S11,S12	HOP-3C
Active Fuel Height	cm	91	(4)	34	34	34	86	36	36	61	15	34
Upper Blanket Height[2]	cm	36	17	17	1	30	9	12	12	36	1	
Fuel Pin Diameter	cm	0.53	(4)	(4)	(4)	(4)	(4)	0.64	0.64	0.64	(4)	(4)
Fuel Smear Density As Percent of Theoretical	%	85	(4)	87	87	(4)	(4)	90	90	90	(4)	88
Fuel Specific Power[3]	watt/gm fuel	50-180	(4)	(4)	(4)	(4)	190	988(6)	1475(6)	1057(6)	32,000(6)	214
Fuel Burnup	GWD/T	0.6-110	Zero	Zero	(4)	(4)	(4)	Zero	Zero	(4)	Zero	(4)
Fuel Assembly Coolant Inlet Temperature	°C	333	(4)	480	480	460	400	310	425	~650	~150	~425
Coolant Pressure Drop Over Active Fuel Length	Kpascal	70-110	(4)	(5)	(5)	(5)	(5)	(7)	(7)	(7)	(7)	(7)
Coolant Average Mass Flow Rate Per Pin	gm/sec	80-112	130	(4)	(4)	(4)	49	(7)	(7)	(7)	(7)	(7)
Coolant and Spacer Wire Area to Fuel Pin Area	—	0.75	(4)	(4)	(4)	(4)	(5)	.69	.69	(5)	2.5,2.0	1.12
Number of Fuel Pins Per Assembly	—	217	7	7	7	7	3	1	1	1	1	1

NOTES:
1. Values reported in or calculated from HEDL and ANL Reports.
2. Reflector and/or blanket material comprising thermal heat sink and inertial restraint.
3. Flattop value during TREAT controlled transients.
4. Value is within range of, or approximates, CRBRP design value.
5. Detailed value not reported or readily available.
6. TREAT natural transient, half maximum pulse height of transient.
7. Static capsule design, parameter not applicable.

TABLE I. (Continued) Comparison of TREAT Test Parameters and CRBR Design Parameters

Design Parameter	Units	CRBR Typical Value or Core Wide Range at Power (975 MWt)	TREAT EXPERIMENT VALUE (1)													
			HUT5-3A	H3	H4	H5	E6	E7	L6, L7	L8	H6	E8	J1	F1	F2	R9, R12
Active Fuel Height	cm	91	34	34	34	34	34	34	86	86	34	34	34	34	34	(4)
Upper Blanket Height (2)	cm	36	30	1	35	1	17	30	9	9	40	40	27	30	14	17
Fuel Pin Diameter	cm	0.53	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)
Fuel Smear Density As Percent of Theoretical	%	85	(4)	88	(4)	88	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)
Fuel Specific Power (3)	watt/gm fuel	50-180	246	243	383	(4)	272	231	188	408	158	168	315	198	206	230
Fuel Burnup	GWD/T	0.6-110	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	(4)	0.34	Zero
Fuel Assembly Coolant Inlet Temperature	°C	333	~250	377	368	354	393	407	400	400	470	407	476	25	25	(4)
Coolant Pressure Drop Over Active Fuel Length	Kpascal	70-110	(7)	22	22	22	22	22	(5)	(5)	44	(5)	47	(7)	(7)	(4)
Coolant Average Mass Flow Rate Per Pin	gm/sec	80-112	(7)	(4)	(4)	(4)	(4)	(4)	50	50	(4)	(4)	(4)	(7)	(7)	136
Coolant and Spacer Wire Area to Fuel Pin Area	—	0.75	1.12	.96	.96	.96	.97	(4)	.85	.85	(4)	.95	.89	.82	.82	(4)
Number of Fuel Pins Per Assembly	—	217	1	7	7	7	7	7	3	3	7	7	7	1	1	7

NOTES:
1. Values reported in or calculated from HEDL and ANL Reports.
2. Reflector and/or blanket material comprising thermal heat sink and inertial restraint.
3. Flattop value during TREAT controlled transients.
4. Value is within range of, or approximates, CRBRP design value.
5. Detailed value not reported or readily available.
6. TREAT natural transient, half maximum pulse height of transient.
7. Static capsule design, parameter not applicable.

QUESTION 36

The PSAR assumed reactivity insertion rates starting the TOP as high as 10$/sec (p. F6.2-86). What possible TOP initiator would yield a 10$/sec rate?

ANSWER 36

There is no TOP initiator that would result in a reactivity insertion rate anywhere near 10$/sec. The 10$/sec ramp included in CRBRP-GEFR-00103 was part of parametric calculations to determine if any new phenomena became important at higher than realizable ramp rates.

NOTE: Question 37 pertains to the consequence of assuming midplane failures.

QUESTION 37

The PSAR considers the possibility of midplane fuel failure.

(a) What fraction of the fuel rods were assumed to fail at the midplane, and what was the magnitude of the reactivity feedback from the midplane failure for the various CDAs considered in Appendix F?

(b) What percentage of the core was voided of coolant when the positive reactivity feedback occurred?

(c) The PSAR (p. F6.2-86) analyses 0.5$/sec to 10$/sec ramp rates. Have 0.01 to 0.1$/sec ramp rates been considered when assuming midplane failures?

(d) Would the sodium be more likely to be boiled out of the core for ramp rates lower than 0.5$/sec?

(e) Has the BOEC for TOP been analyzed assuming midplane fuel rod failures for 0.01 to 10$/sec (F6.2.4.3.2 is not clear on this point)? If not, why not? If so, where is this discussed in the PSAR?

ANSWER 37(a-e)

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. Section 6.1 of CRBRP-GEFR-00523 describes the TOP BOC-1 analysis for the current core design.

QUESTION 38

(a) In the transition phase from LOF events, what fractions of the initial fission gas inventory would remain in the molten core for the disassembly phase? (See p. F6.2-93).

(b) Supply the basis for these estimates.

ANSWER 38

The question implies that the transition phase is followed by a disassembly phase. It should be reemphasized that a rapid recriticality leading to hydrodynamic disassembly from the transition phase is highly unlikely because of the inherent dispersal characteristics of a molten core with entrained steel.

The literal answer to part (a) is that no quantitative estimate of the amount of fission gas remaining can currently be given. Qualitatively, the fission gas concentration should be low, as is suggested by analysis of the resolidified molten fuel in TREAT tests and other out-of-pile heating experiments on irradiated fuel. Because precise quantitative estimates of fission gas availability were not available, the calculations of speculative disassembly transients in CRBRP-GEFR-00103 were done in a conservative fashion, assuming no fission gas was present. The fact that the disassembly calculations were conservative is an adequate basis for the

estimate. It is well known that the presence of only a few percent of the fission products could significantly reduce disassembly energetics, although delays in pressurization due to these fission products of only a few milliseconds will substantially reduce the effect (J. F. Jackson and A. M. Eaton, "Pressurization Rate Effects in Irradiated Core Disassembly Calculations." Trans. AM. Nucl. Soc. 22, p. 370, (1975)).

NOTE: Question 39 pertains to extended fuel motion.

QUESTION 39

QUESTION 39(a)

When molten fuel (and molten steel) moves into the blanket region, could it encounter sodium there? (See p. F6.2.5.2.2).

ANSWER 39(a)

The possibility that the molten fuel-steel mixture entering the blanket region will encounter liquid sodium cannot be ruled out.

QUESTION 39(b)

If so, how was this treated in the analyses of CRBR CDAs?

ANSWER 39(b)

This subject is evaluated in Sections 8.2 and 8.3.6 of CRBRP-GEFR-00523.

QUESTION 40

Please supply all program proposals and test reports related to the "Upper Plenum Injection" experiments mentioned on p. F6.2-101.

ANSWER 40

ANL/RAS 76-4, Upper Plenum Injection Tests No. 1 and No. 2, Robert E. Henry, et al., February 1976 is the only documentation available on completed tests. The Applicants have not proposed any programs for additional tests of this type at this time.

QUESTION 41

To explore reactivity effects in a disruptive core, the PSAR assumed an EOEC core:

(a) Shouldn't the BOL and BOEC cases be examined too, since they would contain more excess reactivity? If not, why not? If so, why hasn't this been examined in the PSAR?

ANSWER 41(a)

Reactivity effects for both the BOC-1 and the EOC-4 disrupted cores have been evaluated and are reported in Appendix F of CRBRP-GEFR-00523.

QUESTION 42

(a) Have any core destruct tests ever been considered which could provide the needed experimental confirmation that CDAs will not likely lead into a superprompt critical power excursion and hydrodynamic disassembly (explosion)?

(b) If so, what are the cost estimates of such tests and how many experiments (and of what kind) were or are being considered?

(c) Please supply all documents which describe the core destruct tests (including internal memoranda, etc.) that may have been considered.

ANSWERS 42(a)-(c)

(a) The Applicants have not identified a need for such tests and is not considering such tests for support of CRBRP.

(b) No such information has been developed by the Applicants.

(c) No such documents have been developed by the Applicants.

QUESTION 43

Elaborate on the pessimistic assumptions which must be made regarding fission gas effects, to generate disassembly. (See PSAR, p. F6.2-105).

ANSWER 43

This question appears specific to the homogeneous core which is not the current design.

Accordingly, it requires no answer.

Section 7.2.3 in CRBRP-GEFR-00523 describes the consequences of pessimistic assumptioins regarding fission gas effects in LOF-HCDA of the current design.

QUESTION 44 (PREAMBLE)

The PSAR states that sodium is still largely in the core in TOP cases prior to disassembly (p. F6.2-105).

QUESTION 44(a)

Are there any possible situations, such as lower initial ramp rates, where this may not be true?

ANSWER 44(a)

The Applicants have not been able to identify a situation in the analysis of a TOP disassembly event in which sodium would not be expected to still be largely in the core. It must be emphasized that the failure of each channel must be forced to occur at the core midplane in order to satisfy the conditions necessary to begin a hydrodynamic disassembly calculation.

QUESTION 44(b)

Explain in detail the basis for the answer in Part (a).

ANSWER 44(b)

In the TOP event sodium voiding results from fuel-coolant interaction in the failed channels. The failed channels still contain a large fraction of liquid sodium since the coolant pumps continue to operate during the transient. This is in contrast to the loss-of-flow event, in which an operating pump head is not available to maintain liquid sodium in the channels. In addition, the channels which have not failed in the TOP event are completely filled with sodium, whereas in an LOF event sodium voiding may occur prior to pin failure. Therefore, in TOP events which satisfy the conditions for a hydrodynamic disassembly calculation sodium is still largely in the core.

QUESTION 45

Boudreau, et al., are concerned about the possibility of rapid, regional core compaction during disassembly which may lead to secondary criticality

with high ramp rates. Identify and supply the studies, if any, that have been made to explore this possibility. (See Boudreau and Erdmann's On Autocatalysis, Nuc. Sci. & Eng., 51:206-22; and the Proposal for Computer Investigation . . ., mentioned in question 25 above.)

ANSWER 45

It is the Applicants' understanding that Boudreau, et al., were not concerned with rapid regional core compaction during disassembly, but rather with fuel recompaction following an initial disassembly to cause a recriticality. The reentry cases presented in Section 11.3 of CRBRP-GEFR-00103 address such a situation as it is hypothesized to occur in CRBRP.

Ref. 65 in CRBRP-GEFR-00103 by J. F. Jackson et al., addresses the situation as it is hypothesized to occur in FFTF. Further analyses are provided in the paper by J. E. Boudreau and J. F. Jackson, entitled "Recriticiality Considerations in LMFBR Accidents" presented at the Fast Reactor Safety meeting at Beverly Hills, California in 1974. The Applicants are not aware of any other relevant analyses addressing the concern of Boudreau, excepting for those referred to by the author in the question.

QUESTION 46

What effects do the fission gases have on (possibly) mitigating the power excursion of a $100 per second LOF accident (BOEC and EOEC)?

ANSWER 46

Since the CRBRP-GEFR-00523 and CRBRP-GEFR-00103 analysis supports the conclusion that a power excursion of 100$/sec in the CRBRP (BOEC or EOEC homogeneous design) or in the current design core does not appear to be physically realizable, the situation hypothesized is not very significant to CRBRP. The importance of fission gas in the theoretical problem postulated is not answerable in the general case. The initial part (about 60% of the pulse width) of the power pulse from a 100$/sec excursion is

controlled solely by the Doppler effect and is unaffected by the gas or void distribution in the core. The energy in the pulse tail is affected by gas, sodium and void distribution. However, to decide to what extent fission gas is a mitigating effect requires knowledge of the specific distributions of sodium, void (gas) and fuel.

QUESTION 47

The PSAR asserts that the probable course of CDAs is self-shutdown without explosion due to fuel ejection. Please supply for each CDA considered in the PSAR numerical figures that would show how much margin there is between the most probable course of a CDA and the most pessimistic course considered. That is, identify those parameters and variables which control the course of fuel and coolant motion and associated reactivity effects, and indicate the extent that each parameter and variable would need to be varied for the CDA to take the most pessimistic course considered.

ANSWER 47

Table 2-2 on pages 2-4 through 2-8 of CRBRP-GEFR-00103 and Table 2-1 on pages 2-8 through 2-10 of CRBRP-GEFR-00523 supplies the numerical figures that show the margin that exists between the best-estimate course of TOP and LOF HCDAs and the pessimistic cases considered. The extent to which the particular variable or variables were varied for the HCDAs to take the pessimistic courses considered is discussed in the sections of CRBRP-GEFR-00103 and CRBRP-GEFR-00523 which present the results of the cases summarized in the tables.

QUESTIONS 48(a) and 48(b)

(a) Is sodium hammer (analogous to water hammer) possible in an LMFBR accident situation?

(b) If so, has it been evaluated as a cause of fuel failure and fuel motion in the CRBR?

ANSWERS 48(a) and 48(b)

(a) A sodium hammer pressure transient may result in the CRBR from the closure of the check valve in a primary sodium coolant loop following the hypothetical seizure of a sodium coolant pump.

(b) The effect of a sodium hammer pressure transient was analyzed using the DEMO code. The results are presented in Section 15.3.2.1.2 of the PSAR. The analyses indicated that the most severe situation postulated, that of the instantaneous closure of the check valve, would result in a pressure change of less than 1 psi in each of the other primary sodium coolant loops. The effect of such a small hydraulic perturbation on the operation of the remaining pumps and on the coolant flow in the reactor is considered inconsequential. Therefore, a sodium hammer pressure transient resulting from a check valve closure is predicted to cause no failure of fuel rods in the CRBR reactor.

QUESTION 49

Is the possibility of sodium vapor explosion-driven core recompaction for autocatalysis ruled out? (See PSAR, p. F6.2-101).

(b) Wouldn't core destruct tests be necessary before one could make any firm conclusion that sodium vapor explosions strong enough to drive a core back to criticality are not possible in CDAs?

(c) If not, explain fully on what basis are they unnecessary?

(d) What are the individual opinions of the various ANL experts (and other experts whose opinions are known by the Applicant) on fuel-coolant interactions about the need for core destruct tests?

ANSWERS 49(b)-(d)

This subject is evaluated in Sections 8.2 and 8.3.6 of CRBRP-GEFR-00523. Answers to Parts (b), (c), and (d) are as follows:

(b) No.

(c) Definitive conclusions regarding the possibility for vapor explosion driven fuel compaction can be obtained by carrying out out-of-pile laboratory experiments and analyses to determine in a general way the requirements for vapor explosive events.

(d) To the Applicants' best knowledge, none of the ANL experts (and other experts) believe that core destruct tests would be necessary.

QUESTION 50

Should the value of the gradient at the top of p. F6.2-112 (.1525¢/cm) be .1525$/cm?

ANSWER 50

See CRBRP-GEFR-00103 for the corrected value.

QUESTION 51

The PSAR notes that there is a positive reactivity feedback due to fuel imploding into a cavity upon fuel reentry in an EOEC LOF (see Section F6.2.6.4.1 of p. F6.1-112).

(a) Please elaborate by providing a complete analysis of these disassembly calculations.

(b) Identify how much feedback was estimated.

(c) What was the rate of reactivity feedback?

(d) Provide drawing of the changing configuration of the core (elevation and plan view) during this process.

(e) What fraction of the core or zones was involved in this implosion, e.g, fraction of core planar area which the imploding fuel crossed (horizontal plane through the core)?

(f) Since transport theory was not used, is there any plan to recalculate this CDA using transport theory?

ANSWER 51

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. GEFR00523 describes the analysis for the current core design.

QUESTIONS 52

QUESTION 52(a)

Does the design basis of 102 Mj of sodium slug energy for the head bolts include a safety factor?

ANSWER 52(a)

No, the ~102 MJ is the value calculated explicitly from the Structural Margin Beyond the Design Base energetics. It has not been arbitrarily increased to provide an additional safety factor.

QUESTION 52(b)

What is the value of this safety factor? That is, at what slug energy would the head bolts be expected to fail?

ANSWER 52(b)

The design of the head restraint meets the requirements of Section 5.2.2 of CRBRP-3, Volume 1. It is not known at what slug energy level the head restraint would be expected to fail.

QUESTIONS 53 (Sodium Slug Rebound)

QUESTION 53(a)

After a violent core disassembly event in which the sodium slug slams under the closure head, what consideration has been given to the possibility of the sodium slug rebounding and blasting fuel back into the core region to cause a secondary power excursion by reassembly of enough fuel?

ANSWER 53(a)

The scenario proposed in this question has not been specifically identified for analysis. However, Section 11.3 in CRBRP-GEFR-00103 discusses the potential for termination of the initiating phase. These recriticality analyses provide results characteristic of those that would be generated by the proposed scenario. The results of the VENUS-II analyses for various parameter possibilities is presented in Section 11.3 of CRBRP-GEFR-00103. Evaluations in CRBRP-GEFR-00523 do not identify any mechanisms for this type of accident.

QUESTION 53(b)

Could fuel moving away radically from the core center rebound and return to meet the down-coming fuel mass?

ANSWER 53(b)

There is substantially more fuel dispersal predicted to occur axially than radially in the CRBRP core disruptive analyses and therefore the propensity for recriticality to occur from axial recompaction is greater than for radial recompaction. Since the analyses in Section 11.3 of CRBRP-GEFR-00103 are done with ramp rate as a parameter it is possible to equate these ramp rates to recompaction from combined directions. The CRBRP-GEFR-00103 evaluation was compared to axial recompaction only because it is judged that if significant recompaction is to occur it would be far more likely to come from the axial direction.

QUESTION 53(c)

Identify and supply any analyses that have been done to explore the rapid fuel motion after soidum slug impact.

ANSWER 53(c)

The only analysis that the Applicants have done that is appropriate to explore rapid fuel motion after sodium slug impact is presented in Section 11.3 of CRBRP-GEFR-00103 and in Section 9 of CRBRP-GEFR-00523.

QUESTION 54

What are the planned "Safety Test Facilities" for fast reactor transient testing mentioned in the NRC's Reactor Safety Research Program (NUREG-75/058, p. 41), and how do they fit into the decision-making process for the CRBR?

ANSWER 54

The only project fitting the description of "Safety Test Facilities" for "Fast Reactor, transient in-reactor tests" known by the Applicant to be planned is the TREAT Upgrade, now under way at ANL. The Applicant does not believe that such facilities are necessary to reach a decision on CRBRP.

The Applicants understand that there were studies conducted by NRC relating to "Safety Test Facilities." However, the Applicants are not aware that there are any firm plans by NRC to construct such facilities. The Applicants do not believe that such facilities are necessary to reach a decision on CRBRP.

NINTH INTERROGATORY SET

QUESTIONS (GENERAL)

Each of the following questions is to be answered in 6 parts, as follows [Where appropriate, the parts of the question have been restated to reflect the protocol for discovery agreed to by Applicants, Staff, and Intervenors NRDC et al.]:

(A) Provide the direct answer to the question.

(B) Identify all documents and studies, and the particular parts thereof, relied upon by Applicants, now or in the past, which serve as the basis for the answer. In lieu thereof, at Applicants' option, a copy of such document and study may be attached to the answer.

(C) Identify principal documents and studies, and the particular parts thereof, specifically examined but not cited in B). In lieu thereof, at Applicants' option, a copy of each such document and study may be attached to the answer.

(D) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) who provided the answer to the question.

(E) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicants' answer. This answer need be provided only in cases where Applicants intend to rely upon on going research not included in Section 1.5 of the PSAR at the LWA or construction permit hearing on the CRBR. Failure to provide such an answer means that Applicants do not intend to rely upon the existence of any such research at the LWA or construction permit hearing on the CRBR.

(F) Identify the expert(s), if any, whom Applicants intend to have testify on the subject matter questioned. State the qualifications of each such

expert. This answer need not be provided until Applicants have identified the expert(s) in question or determined that no expert(s) will testify, as long as such answer provides reasonable notice to Intervenors.

ANSWERS (GENERAL)

The following answers are identical for all interrogatories except where supplementary information is provided in the answers which follow.

(A) See direct answers below under heading "ANSWER".

(B) The documents which serve as a basis for the Applicants' answer are identified in the responses below.

(C) Unless otherwise indicated below in regard to the answers under heading "ANSWER (REFERENCES)"; none.

(D) See the attached affidavits.

(E) Except where otherwise noted below, the Applicants' program of further research work is described in Section 1.5 of the PSAR.

(F) At the present time the Applicants have not determined the experts, if any, whom they intend to have testify on the subject matter questioned.

PART A: INTERROGATORIES RELATED TO (ORIGINAL) CONTENTION NO. 8 [NOW CONTENTION NO. 11]

QUESTION I

On page 12.1-2 of the PSAR, the Applicant states that personnel exposure in routinely occupied restricted areas will be limited to approximately 1/10 of the limits of 10 CFR 20. On page 12.1-3, the Applicant indicates that

there will be other zones in the plant where dose rates may range from 2 mrem/hr to 100 mrem/hr. How has it been determined for each of these zones that the radiation level is ALARA?

ANSWER I

The zoning criteria referred to in this question is part of the overall CRBRP radiation protection and shielding design which will fully meet the intent of Regulatory Guide 8.8, Revision 1. It is recognized in this Regulatory Guide that LWR occupational doses have been below the applicable limits of 10 CFR 20. Therefore, the intent of Regulatory Guide 8.8 is to "promote a more formal approach to keeping doses ALARA, to identify and promote continuance of good practices, and to promote further improvements where practicable."

The CRBRP radiation protection design limits the exposure of the individual to 10 CFR 20 occupational limits while keeping the total man-rem dose to the total staff ALARA. As specifically noted in Regulatory Guide 8.8, "It would be inappropriate to hold the individual doses to a fraction of the applicable limit if this resulted in the irradiation of more people and increased the total man-rem dose."

The radiation zone for a given cell is determined by its access requirements. As noted in Section 12.1.5 of the PSAR, the anticipated fractional time spent in Zones II and III is 25% and 5%, respectively. The expected man-hours of occupancy by area and radiation zone is shown on pages Q331.17-1 and 17-2 of the PSAR (Amendment 6). The occupancy requirements of Zones II and III represent approximately 24% and 2.5%, respectively, of the total access requirements, and are therefore consistent with the original design basis.

The man-rem dose for required activities within radiation Zones I, II, and III are discussed on PSAR pages Q331.19-7 (Amendment 8) and Q331.21-1 (Amendment 14). These man-rem doses for operations, maintenance within accessible cells, radwaste operations and refueling/fuel handling operations result in an estimated exposure of 58 man-rem per year, or an average

of about 0.4 rem/year to an individual on the CRBRP plant staff. Based on data provided in Reference 1, the dose from these activities is consistent with that in LWR experience. This data shows the overall radiation exposure due to these operations to be ALARA. The radiation exposure in the various radiation zones are not disproportionate nor has any single activity been identified as having an undue fraction of the allowed dose.

The CRBRP ALARA review program provides a basis for continual review of the activities during design and operation of the facility. This program is discussed in PSAR Q331.1 (Amendment 1) and Q331.3 (Amendment 20) of the PSAR and Q331.2 and Q331.4 (per NRC response).

ANSWER I (REFERENCES)

1. L. A. Johnson, "Occupational Radiation Exposure at Light Water Cooled Power Reactors," NUREG-0323, March 1978.

Documents used as reference material in developing this reply:

1. L. A. Johnson, "Occupational Radiation Exposure at Light Water Cooled Power Reactors," NUREG-0323, March 1978.

Documents examined during preparation of this reply:

1. Pelletier, Charles A., et al., "Compilation and Analysis of Data on Occupational Radiation Exposure Examined at Operating Nuclear Power Plants," Atomic Industrial Forum, Inc.

QUESTION II

Page 8.8-2 and 8.8-3 of Regulatory Guide 8.8 tabulates specific information that should be provided (items a through r) at the construction permit stage to ensure that provisions have been included to achieve ALARA. With respect to each item (a) through (r) separately, precisely how is compliance with this guide being implemented?

ANSWER II

The method of implementing the provisions of items (a) through (r) of Regulatory Guide 8.8 (Revision 1) are given below:

Item (a) General service and access design criteria are included in overall plant design requirements. Features specific to individual systems are included in the system design requirements. Examples of overall plant service and access design criteria are as follows:

1. All components shall be made readily accessible and maintainable with a logical removal path defined and documented. Provisions shall be included where practicable, for isolating components to permit continued operation of the plant. Pad-eyes shall be strategically located in radioactive cells for installation of portable shielding or for mounting pipe restraints or tooling.

2. The plant design shall be such that maintenance can be performed with adequate maintenance access for personnel and for required tools, and with minimization of scaffolding, rigging, and portable shielding required to facilitate the work for both scheduled and unscheduled events.

3. Maintenance access for servicing and/or removal or replacement shall be provided for each component that is to be maintained. All system components shall be designed for removal and replacement.

4. Clearance shall be provided between adjacent components and structures for personnel access, installation, and operation of tooling, and installation of temporary shielding. Overhead room shall be provided for equipment removal and replacement. The following represents specific maintenance envelope requirements:

a. A nominal 3'-0" maintenance clearance space shall be provided for all major components and piping 24" and larger.

b. For in-service maintenance requiring cutting and rewelding of pipe, access space must be provided for manual and/or automatic cutting and welding equipment. The most restrictive clearance is expected to be for cutting the pipe. Specific access requirements as a function of pipe radial and axial dimensions have been developed for project use.

5. A minimum of 7'-0" clearance from the floor to any overhead obstruction shall be provided on all stairs, walkways, and other personnel access ways.

Item (b) The general service and access design criteria for the overall plant requires that all electrical junction boxes and instrument and instrument junctions shall be external to all normally inaccessible areas or cells. This criteria excludes the location of these components in high radiation areas which are inaccessible during operation. For example, the flux monitoring instrumentation and calibration equipment will be located in the HAA. Provisions have been made to remove nuclear detectors as required from the reactor cavity through the HAA. Thus, entry to the reactor cavity would not be required for this activity.

Item (c) Wherever possible nonradioactive plant components are located in accessible areas as a part of the overall plant design criteria. The maintenance system design criteria provides for the removal and cleaning of several major components such as the PHTS primary pump, check valves, et al. The response to PSAR Q331.4 (Amendment 20) discusses in detail the accessibility and removability of the following systems:

1. Liquid, Gaseous and Solid Radwaste
2. Closure Head Operations
3. Refueling and Fuel Handling Systems
4. Control Rod Drive Removal Operations
5. Maintenance Work on Large Equipment

Item (d) The overall plant design criteria recognize the importance of "best" grade components to minimize radiation exposure. Specifically, design-dictated maintenance will be reduced through application of fail-safe features, designating components which require little or no preventive

maintenance and assigning tolerances which allow for use and wear throughout the equipment's useful life.

The specific problem related to valves is well recognized by all systems and discussed in the response to PSAR Question 331.4.

Item (e) The design requirements for penetrations are discussed in PSAR p. 12.1-5 through 12.1-6. Specific design criteria are part of the overall plant design criteria as follows:

Location of Penetration	Description	Allowable Dose Rate Range at Penetration
Restricted Area, and Radiation Area, Cell Accessible, Penetration Normally Inacessible	Penetration radiation peak 9 feet or more above normal working surface. Access requires special platform or ladder.	$\leq$200 mrem/hr. The dose rate at accessible locations from normally inacessible penetrations shall also be limited to not exceed the rerequirement stated below.
Restricted Area, and Radiation Area, Cell Accessible, Penetration Normally Accessible	Penetration radiation peak less than 9 feet above normal working surface.	Factor of 3 greater than cell or area design dose rate (general area dose rate increase at work location limited to 1.2 times the level without penetrations).
Unrestricted Area, Penetration Accessible or Inaccessible	-----------	$\leq$2 mrem/hr (10 CFR 20 limit).

Item (f) Radioactive sources in accessible areas are controlled as required to meet the PSAR radiation zoning criteria. The radiation zoning criteria is a part of the overall plant design criteria. Where transport of a substantial source through an accessible zone is required, radiation exposure will be controlled by shielding, access restrictions or both. The dose rate due to a transient source is limited by the overall plant design criteria to 200 mrem/hr. Any exceptions to this will require specific design and operational features to provide positive protection to personnel. Further discussion of implementing this ALARA feature can be found

in response to PSAR Questions Q331.6 (Amendment 1) and Q331.4 (Amendment 20).

Item (g) The overall plant design criteria sets the following general requirements:

1. Facilities shall be provided for convenient inspection, removal, and repair or replacement of reactor internal components. Provision for interim storage of components to be replaced shall be considered.

2. Adequate local lay-down space shall be provided for all equipment such as shield plugs required for maintenance operations. On-site storage space shall be provided for maintenance equipment and tooling, spare parts, temporary shielding, etc.

3. Cell liners, drip and splash pans, and/or other devices shall be provided as required to limit damage caused by sodium leaks and to facilitate cleanup. Cell finishing should provide smooth nonporous surfaces and eliminate hard-to-reach corners and pockets so as to ease decontamination.

4. The ventilation system shall be designed to facilitate the flow of potentially contaminated air from the less contaminated to the more contaminated area, thus minimizing the spread of contamination.

Additional design features being implemented by individual systems design criteria are discussed in response to PSAR Q331.4 (Amendment 20).

Item (h) The following overall plant design requirement has been included to limit the number of undrainable locations:

Liquid containing systems and/or components shall be designed to facilitate complete drainage. For components that cannot be completely drained by normal means, provisions shall be included in the design of the component to permit use of other liquid removal methods utilizing maintenance equipment.

The design of sodium-containing equipment and/or components shall minimize crevices and pockets which make complete sodium removal difficult.

This requirement to limit sodium-containing pockets also serves to limit potential crevices for solids.

Item (i) The design requirement discussed under item (h) also serves to permit flushing. A complete discussion of the large component cleaning and decontamination system can be found in response to PSAR Q331.4 (Amendment 20). These facilities are provided as part of the CRBRP Maintenance System.

Item (j) The overall plant design requires that area configurations, gas purges, and differential pressures should be established to assure leakage occurs from less contaminated to more contaminated areas. The Nuclear Island HVAC system has the additional requirement to limit the spread of airborne radioactive materials, where they may be present within the NI building.

Item (k) A complete radiation and airborne contamination monitoring system with both fixed and remote readouts/alarms has been implemented in the design. This system is discussed in detail in PSAR Section 12.2.4.

Item (l) CRBRP cells which contain significant heat and radiation sources are inerted with Argon or Nitrogen and are cooled by the recirculating gas cooling system (RGCS). The components requiring maintenance for the cooling system are located outside the radioactive cells in radiation Zone III as described in PSAR 12.1. The radiation level in these cells is designed to give ALARA radiation exposures under the following conditions:

1. The RGCS shall be designed for contact maintenance during plant full power operation and normal fuel handling operations.

2. Provisions shall be made for fan, blower, and cooler replacement during plant full-power operation.

3. Ventilation is not provided except as required to acquire cell access during maintenance periods after the cell radiation has sufficiently decayed.

Item (m) The scope and extent of the CRBRP shield is discussed in PSAR Section 12.1. The doses have been shown to be ALARA (See response to I-1).

Item (n) The overall plant design criteria require the following provisions for temporary shielding:

1. Radiation from sources within the cell shall be limited either by removal (e.g., primary sodium), or by permanent or temporary local shielding as required.

2. All components shall be made readily accessible and maintainable with a logical removal path defined and documented. Provisions shall be included, where practicable, for isolating components to permit continued operation of the plant. Pad-eyes shall be strategically located in radioactive cells for installation of portable shielding or for mounting pipe restraints or tooling.

3. Clearance shall be provided between adjacent components and structures for personnel access, installation and operation of tooling, and installation of temporary shielding. Overhead room shall be provided for equipment removal and replacement.

Item (o) Detailed information on the radioactive waste shielding requirements are discussed in PSAR Chapter 11 and 12. Detailed information on the ALARA aspects of the radioactive waste disposal system are given in response to PSAR Q331.4 (Amendment 20).

Item (p) The overall plant design requires that each system consider the following:

The logistics of all maintenance operations shall be considered, including the paths all equipment must follow; the availability, capacity, lift, and

area coverage of handling devices; port and hatch size and locations, rotating and other special handling requirements; operator stationing with respect to safety and visibility; and requirements for pits or other temporary storage or transfer areas. Special equipment shall be identified as required by each system.

Examples of remote handling equipment are given in response to PSAR Q331.4 (Amendment 20).

Item (q) The plant radiation protection and shielding design source terms are based on maximum expected failures of fuel elements and "conservative" analysis. A complete discussion of these design requirements are provided in Sections 11.1 and 12.1 of the PSAR.

Item (r) The manned access points for sampling sites are in radiation Zone II. The radiation zoning has been found to be ALARA in general (see response to I-1) and included required sampling locations. Additional information on sample handling equipment is discussed in response to PSAR Q331.5. Source terms at sampling stations are discussed in PSAR Section 12.1.

Pages AB-89--AB-93

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PART B: INTERROGATORIES RELATED TO (ORIGINAL) CONTENTION NO. 14 [NEW CONTENTION NO. 8]

QUESTION I

The Draft EIS on the CRBR (NUREG-0024, p. 10-4) identifies 9 civilian nuclear power facilities that were or are in the process of being decommissioned. With respect to the CRBR and each of these 9 facilities, please identify and compare at the time of decommissioning:

a. composition of the reactor vessel, vessel internals and the concrete shielding. In responding to this question, we request that you designate the materials for the components of the reactor, e.g., plate, course and flange forgings, nozzle forgings, boltings, nuts, support, pipe, weld rod, cladding, etc. By designate we mean list current ASTM designations, or ASME Section III specifications, including chemical composition in weight percent of C, Ni, Fe, and Mn.

b. the principal activation products (with particular attention to Fe-55, Ni-59, Co-60, and Ni-63) within the reactor vessel and the vessel internals;

c. the principal activation products in the immediate concrete shielding;

d. the reactor shielding exposure in Mw-days or some other convenient unit; and

e. the flux density, the neutron fluence, nvt, to which the reactor vessel and concrete shielding has been exposed, with particular attention to the vessel inner surface at the beltline.

ANSWER I

This response deals only with those components of the Clinch River Breeder Reactor Plant (CRBRP) which are not designed to be removable. Any activated removable reactor internals would be transported to facilities for re-processing or storage of radioactive materials at the time of decommissioning.

The Applicants are not in possession of the information sought which pertains to the other nine nuclear facilities. Information relating to CRBRP is available to the Applicants and such information will form the basis for the following responses.

(a) Table 1 lists the components, and the RDT standards (1,2,3) and material types required for the current design of CRBRP. Table 2 gives chemical composition of the steels as specified in the ASME references (4,5,6), along with any additional constraints imposed by RDT standards or specific design requirements. The assumed composition of the primary concrete shield is given in Table 3.

(b) Table 4 lists the principal activation reactions which occur in the CRBRP permanent steel components (8,9). The activity of the most highly activated component, the Fixed Radial Shield, and of the Reactor Vessel wall is given for the time of decommissioning. The buildup of radioisotopes was calculated based on the conservative assumption that the reactor will operate for 30 effective full power years (EFPY).

The activity in $\mu Ci/CM^3$ was calculated based on an average neutron flux spectrum at the inner surfaces of the fixed radial shield and reactor vessel over a two foot high surface area centered at the radial midplane of the reactor core (i.e., the location of the maximum neutron flux level).

(c) Table 5 lists the principal activation reactions which occur in the CRBRP Primary Concrete Shield (9). The activity in $\mu Ci/CM^3$ was calculated based on a neutron flux spectrum incident on the concrete surface at the core midplane. This is the location of maximum neutron flux and thus

maximum neutron activation in the concrete shield. The conservative assumption was made that the reactor will operate for 30 effective full power years (EFPY).

(d) The design power rating of the CRBRP is $975MW_t$. The design reactor lifetime is 30 years, with a 75 percent plant capacity. This corresponds to a total reactor power generation of about 8.0×10^6 Mw days over the in-service life of the CRBRP.

(e) The total neutron flux, neutron energy spectrum and neutron fluence for the inner surface of the CRBRP Reactor Vessel at the core midplane elevation are given in Table 6. The total neutron flux, neutron energy spectrum, and neutron fluence at the surface of the Primary Concrete Shield at core midplane elevation are given in Table 7. Fluence levels are based on a 30 effective full power years of operation.

QUESTION II

Harwood, et al., in "Activation Products in a Nuclear Reactor," state on page 6:

The length of time the reactor vessel must remain isolated from the environment depends on the criteria for safe radiation levels. The criteria which has been adopted in several decommissionings is contained in 10 CFR Part 20.105(b) (1):

'Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour...'

Do you agree or disagree with this assertion? If you disagree, please explain the basis for the disagreement.

ANSWER II

The Applicants understand that the quoted criteria have previously been applied as a guideline for decommissioning. It should be noted that pursuant to 10 C.F.R. 50.82, decommissioning can be undertaken upon Commission approval of the Applicants' decommissioning plan, and that the criteria for decommissioning would be established pursuant to review and approval of a given decommissioning plan.

TABLE 1

MATERIAL REQUIREMENTS FOR CRBR PERMANENT STEEL COMPONENTS

Component	Product Form	RDT Standard*	Grade or Type
Reactor Vessel	Plate	M 5-1	304
	Plate	M 5-1	316
	Forging	M 2-2	F-304
Suppressor Plate	Plate	(ASME SA-240)	316
Guard Vessel	Plate	(ASME SA-240)	304
Core Former Structure	Plate	M 5-1	304
	Plate	M 5-1	316
	Forging	M 2-2	F-316
Core Support Structure	Plate	M 5-1	304
	Forging	M 2-2	F-304
	Forging	M 2-4	F-8
Fixed Radial Shield	Plate	M 5-1	316
Horizontal Baffle	Plate	M 5-1	316
Bypass Flow Module	Forging	M 2-2	F-304
	Forging	M 2-4	F-8

*RDT M 5-1 is ASME SA-240 with additional requirements(3)

RDT M 2-2 is ASME SA-182 with additional requirements(1)

RDT M 2-4 is ASME SA-336 with additional requirements(2)

TABLE 2

CHEMICAL COMPOSITION OF MATERIALS REQUIRED FOR CRBR PERMANENT STEEL COMPONENTS

	Composition (weight %)				
	SA-240 (6)	SA-240 (6)	SA-182 (4)	SA-182 (4)	SA-336 (5)
Element	Type 304	Type 316	Type F-304	Type F-316	Type F-8
Carbon (max)	0.08	0.08	0.08	0.08	0.08
Manganese (max)	2.00	2.00	2.00	2.00	2.00
Phosphorus (max)	0.045	0.045	0.040	0.040	0.040
Sulfur (max)	0.030	0.030	0.030	0.030	0.030
Silicon (max)	1.00	1.00	1.00	1.00	1.00
Nickel	8.00-10.50	10.00-1400	8.00-11.00	10.00-1400	8.00-11.00
Chromium	18.00-20.00	16.00-18.00	18.00-20.00	16.00-18.00	18.00-20.00
Molybdenum	---	2.00-3.00	---	2.00-3.00	---
Columbium 1),2) (max) Tantalum (Max)	0.02	0.02	0.05	0.05	0.05
Titanium 2) (max)	0.05	0.05	0.05	0.05	0.05
Cobalt 3) (max)	0.10	0.10	0.10	0.10	0.10

1) Columbium and Tantalum are controlled together

2) Applicable to welded austenitic stainless steel items which may be subjected to service at temperatures over 800 F.

3) Applicable to core former structure, core support structure, and bypass flow module steels in contact with the sodium pool.

TABLE 3

CHEMICAL COMPOSITION OF THE CRBRP PRIMARY SHIELD

Composition (weight %)

Element	Ordinary Concrete (Portland)
Iron	1.22
Hydrogen	0.56
Oxygen	49.83
Magnesium	0.24
Calcium	8.26
Sodium	1.71
Silicon	31.58
Aluminum	4.56
Sulfur	0.12
Potassium	1.92

TABLE 4

PRINCIPAL ACTIVATION PRODUCTS IN THE CRBRP PERMANENT STEEL COMPONENTS

Activation Reaction	Activity* In Fixed Radial Shield ($\mu Ci/cm^3$)	Activity* In Reactor Vessel ($\mu Ci/cm^3$)
$Cr^{50}(n,\gamma)\ Cr^{51}$	9.8×10^4	8.7×10^3
$Fe^{54}(n,\alpha)\ Cr^{51}$	5.1×10^0	1.5×10^{-3}
$Fe^{54}(n,p)\ Mn^{54}$	1.2×10^2	3.2×10^{-2}
$Mn^{55}(n,2n)\ Mn^{54}$	1.3×10^{-1}	2.2×10^{-5}
$Fe^{58}(n,\gamma)\ Fe^{59}$	3.7×10^3	2.5×10^3
$Co^{59}(n,p)\ Fe^{59}$	1.5×10^{-2}	2.6×10^{-6}
$Ni^{62}(n,\alpha)\ Fe^{59}$	1.1×10^{-3}	1.3×10^{-7}
$Ni^{58}(n,p)\ Co^{58}$	5.1×10^2	9.2×10^{-2}
$Co^{59}(n,2n)\ Co^{58}$	5.2×10^{-3}	8.4×10^{-7}
$Co^{59}(n,\gamma)\ Co^{60}$	2.1×10^5	8.9×10^3
$Ni^{60}(n,p)\ Co^{60}$	4.4×10^0	7.7×10^{-4}
$Ta^{181}(n,\gamma)\ Ta^{182}$	7.4×10^4	5.0×10^3
$Fe^{54}(n,\gamma)\ Fe^{55}$	5.1×10^5	1.2×10^4
$Ni^{58}(n,\gamma)\ Ni^{59}$	5.7×10^1	3.1×10^0
$Ni^{62}(n,\gamma)\ Ni^{63}$	5.9×10^3	4.2×10^2
$Nb^{93}(n,\gamma)\ Nb^{94}$	4.5×10^0	7.3×10^{-2}

*After 30 effective full power years (EFPY) of reactor operation and averaged over a two foot high section of the inner surface of each component at radial midplane.

TABLE 5

PRINCIPAL ACTIVATION PRODUCTS IN THE CRBRP PRIMARY CONCRETE SHIELD

Activation Reaction	Maximum Activity* In The Concrete Shield (Ci/cm^3)
$O^{17}(n,\alpha)\ C^{14}$	2.0×10^{-3}
$Na^{23}(n,\gamma)\ Na^{24}$	7.3×10^{1}
$Mg^{26}(n,\gamma)\ Mg^{27}$	9.3×10^{-2}
$Al^{27}(n,\gamma)\ Al^{28}$	6.9×10^{1}
$Si^{30}(n,\gamma)\ Si^{31}$	6.5×10^{0}
$S^{33}(n,p)\ P^{33}$	8.8×10^{-5}
$S^{34}(n,\gamma)\ S^{35}$	9.5×10^{-2}
$S^{36}(n,\gamma)\ S^{37}$	4.9×10^{-4}
$K^{39}(n,\gamma)\ K^{40}$	3.0×10^{-6}
$K^{41}(n,\gamma)\ K^{42}$	6.4×10^{0}
$Ca^{40}(n,\gamma)\ Ca^{41}$	2.9×10^{-2}
$Ca^{40}(n,\alpha)\ Ar^{37}$	8.1×10^{-1}
$Ca^{44}(n,\gamma)\ Ca^{45}$	6.6×10^{0}
$Ca^{46}(n,\gamma)\ Ca^{47}$	5.0×10^{-3}
$Ca^{48}(n,\gamma)\ Ca^{49}$	7.6×10^{-1}
$Fe^{54}(n,\gamma)\ Fe^{55}$	1.1×10^{1}
$Fe^{58}(n,\gamma)\ Fe^{59}$	2.1×10^{-1}

*After 30 effective full power years (EFPY) of reactor operation at core radial midplane.

TABLE 6

NEUTRON FLUX AT THE CRBRP REACTOR VESSEL INNER SURFACE AT CORE MIDPLANE ELEVATION

	Flux (n/cm^2 sec)	Fluence* (n/cm^2)
Total	4.3×10^{11}	3.1×10^{20}
Thermal	1.9×10^{9}	1.3×10^{18}
E>0.1 MeV	5.9×10^{9}	4.2×10^{18}
E>1.0 MeV	2.2×10^{7}	1.6×10^{16}

*Fluence is based on 30 effective full power years (EFPY) of reactor operation.

TABLE 7

NEUTRON FLUX AT THE CRBRP PRIMARY CONCRETE SHIELD
AT CORE MIDPLANE ELEVATION

	Flux (n/cm^2 sec)	Fluence* (n/cm^2)
Total	1.1×10^{10}	7.8×10^{18}
Thermal	2.5×10^{9}	1.8×10^{18}
E>0.1 MeV	1.3×10^{8}	9.4×10^{11}
E>1.0 MeV	8.6×10^{4}	6.1×10^{13}

*Fluence is based on 30 effective full power years (EFPY) of reactor operation.

ANSWERS (PART B) (REFERENCES)

Documents used as reference material in developing this reply:

1. RDT M2-2T, "Stainless and Low Alloy Steel Forgings," December, 1974.

2. RDT M2-4T, "Alloy Steel Forgings," November, 1974.

3. RDT M5-1T, "Stainless Steel Plate, Sheet, and Strip," November, 1974.

4. ASME SA-182, "Specification for Forged or Rolled Alloy-Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service."

5. ASME SA-336, "Specification for Alloy Steel Forgings for Seamless Drum, Heads, and Other Pressure Vessels."

6. ASME SA-249 "Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion - Welded Unfired Pressure Vessels."

7. Jaeger, R. G., et al., "Engineering Compendium on Radiation Shielding, Vol. 1," Springer-Verlag, New York, 1968, p. 177.

8. HEDL-TME-72-135, "Multigroup Reactor Cross Sections for FTR Application," R. B. Kidman.

9. Nuclear Data Tables, Vol. All, No. 8-9, "Neutron Activation Cross Sections, Measured and Semiempirical," July, 1973.

TENTH INTERROGATORY SET

GENERAL QUESTION

Each question is instructed to be answered in 6 parts, as follows [Where appropriate the parts of the question have been restated to reflect the protocol for discovery agreed to by Applicants, Staff, and Intervenors NRDC et al.]:

(A) Provide the direct answer to the question.

(B) Identify all documents and studies, and the particular parts thereof, relied upon by Applicants, now or in the past, which serve as the basis for the answer. In lieu thereof, at Applicants' option, a copy of each such document and study may be attached to the answer.

(C) Identify principal documents and studies, and the particular parts thereof, examined but not relied upon by Applicants, which pertain to the subject matter questioned. In lieu thereof, at Applicants' option, a copy of each such document and study may be attached to the answer.

(D) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) who provided the answer to the question.

(E) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicants' answer. This answer need be provided only in cases where Applicants intend to rely upon on going research not included in Section 1.5 of the PSAR at the LWA or construction permit hearing on the CRBR. Failure to provide such an answer means that Applicants do not intend to rely upon the existence of any such research at the LWA or construction permit hearing on the CRBR.

(F) Identify the expert(s), if any, whom Applicants intend to have testify on the subject matter questioned. State the qualifications of each such

expert. This answer need not be provided until Applicants have identified the expert(s) in question or determined that no expert(s) will testify, as long as such answer provides reasonable notice to Intervenors.

GENERAL ANSWERS

The following responses are identical for all interrogatories with the exception of those instances where additional or supplementary information is provided in the responses to the interrogatories themselves:

(A) See numbered responses below.

(B) The documents which serve as the basis for the Applicants' answer are identified in the appropriate numbered response and have been or will be made available for inspection and copying.

(C) The Applicants have examined and evaluated numerous documents pertaining to the subject matter questioned, however, unless otherwise indicated in the responses below, documents and other studies pertaining to the subject matter have been examined but not relied upon by the Applicants. This does not imply that the Applicants have examined all documents in existence which could pertain to the subject matter questioned. Of the documents examined by the Applicants which might pertain to the subject matter questioned, only that material relied upon by the Applicants has been retained in retrievable form by the Applicants. This material is identified in the response to Subpart B.

(D) See the attached affidavits.

(E) Except where otherwise noted below, the Applicants' further research work is described in Section 1.5 of the PSAR.

(F) At the present time, the Applicants have not determined the experts, if any, whom they intend to have testify on the subject matter questioned.

QUESTION I

Attached to the letter Van Nort to Boyd (April 30, 1976) was the PMC Summary of the reliability program meeting between the CRBR Project and NRC. On page 2 of the summary the following appears which is attributed to R. Denise:

> (b) A goal of 1×10^{-6} events/year for events leading to consequences exceeding 10 CFR 100 guidelines is considered acceptable by NRC,

In the CRBRP Reliability Program report, dated January 1976, the following appears on page 7:

> 1.2.1 Rationale
>
> Regulations for licensing of power reactors do not establish probability guidelines which can be used directly in this reliability program, although regulatory documents do provide guidance. This guidance has been considered in establishing the following numerical reliability criterion:
>
> > The probability of exceeding 10CFR100 guidelines shall be less than one chance in one million per reactor year.

With respect to the goal of 1×10^{-6} events/year or the numerical reliability criterion of one chance in one million per reactor year, please answer the following questions:

1. What factual information was utilized to support the implied contention that this specific goal or criterion is in fact achievable?

(a) In answering this question, please refrain from theoretical discussions. We are only interested here in factual information — that

is, documentation that a numerically comparable reliability goal was proposed and achieved in a project comparable in complexity to the CRBRP.

(b) In answering this question, please cite only factual documented information that is relevant to a project comparable in complexity, in being essentially first-of-a-kind and where the criterion was introduced at the design stage and was required to be implemented during the construction phase and to be effective in the subsequent operational phase.

(c) In answering the above question, please specify those portions of the documentation that specifically treat the matters of human error and common mode failure during the design, construction and operational phases.

2. If the answer to the above question is that there is no such factual information, is factual information available relative to a higher probability (10^{-5}, 10^{-4}, 10^{-3}) goal or criterion?

(a) In answering this question please be responsive to (a), (b), and (c) of Interrogatory I.1.

3. Since one can often learn from past mistakes, are there documented cases wherein a goal or criterion of 10^{-3}, 10^{-4}, 10^{-5}, or 10^{-6} was established but was not realized?

(a) In answering this question be responsive to (a), (b), and (c) of Interrogatory I.1.

(b) In answering this question please indicate precisely what was learned from the failure and how it can be utilized in the CRBR project.

4. In the event that the above questions are conceived as being too restrictive, what other factual information is available to demonstrate that a goal or criterion of 10^{-6}, 10^{-5}, 10^{-4} or 10^{-3} is achievable?

ANSWER I (1-4)

The Applicant agrees that in the NRC letter to the Applicant dated May 6, 1976, the probability of one chance in a million per year resulting in consequences exceeding 10 CFR 100 Guide Lines is the "safety objective aiming point" rather than a fixed number which must be demonstrated for a given plant.

The factual information which supports the Applicants' position that the CRBRP reliability goal is achievable is the experience that has been gained with nuclear reactor systems and the NRC conclusions regarding achievement of system reliabilities presented in WASH-1270. Since this goal is only one aspect of the over-all design safety approach for the CRBRP, it is not intended nor necessary that the achievement of this goal will be confirmed in a rigorous statistical fashion. Therefore, documentation that a numerically comparable goal had been proposed and achieved in a project comparable in complexity to CRBRP would only be of academic interest unless an overall design safety approach utilizing comparable regulations, regulatory guides, and other design constraints, performance criteria, and successful precedents, had also been imposed on the hypothetical project in question.

The operating experience that has been gained on current LWR plants which are comparable to the CRBRP in complexity do provide an indication of the achievability of the criterion for the purpose for which it was intended, i.e., as one aspect of a balanced design safety approach. The CRBRP utilizes concepts and equipment very similar to those employed in LWRs. Where differences do exist by necessity (e.g., sodium vs. water coolant) specific attention has been given to those differences to assure they are well understood and accommodated in the CRBRP design and factored into the CRBRP reliability program. Experience to date for shutdown systems for LWRs supports the attainment of an unreliability of less than 10^{-4} failures per year for single systems. When redundant and diverse systems are employed to accomplish a particular function, it is reasonable to expect to be able to achieve net unreliabilities in the range of 10^{-6} to 10^{-7} failures per year.

Further, since there are no known instances where a commercial reactor incident has led to significant impacts upon the health and safety of the public, confirmation as to whether or not reactor systems can achieve such a goal is more appropriately derived from actual operating experience rather than by hypothetical data which might support or contradict attainment. With regard to whether or not a numerically comparable reliability program has been proposed and achieved in a project comparable in complexity to the CRBRP, the Applicants agree with NRDC's suggestion that the questions posed by NRDC are somewhat restrictive. The Applicants believe that the more appropriate comparisons and analogies should be made with the commercial LWR experience. Commercial LWRs involve projects of similar complexity to the CRBRP and also have demonstrated a high degree of reliability in protecting the public. The commercial reactor experience includes all generations of reactors including those which were at the time "first of a kind." It also includes all types and aspects of human error and common cause failures. The fact that a numerically comparable reliability goal was not set prior to the achievement of their demonstrated high reliability in protecting the public (an achievement of an unprecedented public safety record) should not really be the pertinent consideration.

Although not directly relied upon, the United Kingdom has reached similar conclusions and actions relative to the use of reliability techniques in reactor design. The United Kingdom began work in the area of reliability for reactor plant design and development during the early 1960's. Dedicated reliability programs have been utilized in the United Kingdom and methods for implementation of such programs have been described in reasonable detail by A. E. Green (the Reliability Assessment of Emergency Electrical Supplies: Proceedings 1975 Annual Reliability and Maintainability. Symposium - Washington, D. C. Jan. 1975), M. C. Pugh (The Use of Probability Techniques in a Reactor Design Office - SRS/GR/5-UKAEA), and others. Although not specifically stated in these references, such procedures and techniques have been employed by the UKAEA in nuclear projects.

The Prototype Fast Reactor (PFR), presently operating in the United Kingdom, utilized a reliability program developed for the various systems

essential to safe operation of that plant, and reliability program elements (similar to those applied on the CRBRP) were applied throughout the design stage. This program involved implementation of reliability requirements in design, assessments of system reliability, and data gathering and application. This program was designed to carry on through long-term operation of the PFR. However, detailed descriptions of this program are not available at the present time and no documentation is expected in the near future. The brief description provided here was obtained by a personal communication with Mr. John Bourne of the UKAEA, Directorate of Safety and Reliability. Both the British and the CRBRP project have recognized the effectiveness of a reliability program and have integrated similar programs into design.

Reliability methodology and confirmation techniques have evolved over a long period of time on many and varied programs. Intensive application of reliability techniques in a formal manner began over two decades ago and has matured through application and development in large numbers of military, space, transportation and industrial programs during this time. This has resulted in a vast "storehouse" of tried and proven approaches and methods which can be applied to the project with confidence. Thus, this project is drawing on the vast experience and varied techniques developed on these programs to provide additional means of assuring a safe and reliable power generator. There is more than enough history of application and successful results to provide guidance for a well-balanced program of goal-setting and confirmatory analyses and tests. By taking maximum advantage of this extensive history and the experience gained from LWRs, the reliability program on this project will be an additional factor in assuring that the CRBRP will indeed be safe and reliable. The project has and is making a concerted effort to judiciously apply the ordered discipline of Reliability Assurance to CRBRP to assure that these objectives are met.

The reliability technology referred to above has been applied with varying degrees of discipline to a variety of systems and programs which range from simple to complex. Information from these programs has a direct bearing in advancing the state-of-the-art of the reliability discipline and enhancing

the reliability data base. However, since the CRBRP Project is unaware of any project which has combined the use of a reliability goal, comparable regulations, guides, codes and standards, and successful precedents, direct numerical results from and conclusions reached by reliability programs for any other projects would have to be qualified prior to comparison with the CRBRP Program to such an extent that they might be of doubtful significance in any evaluation of the CRBRP reliability program.

QUESTION II (PREAMBLE)

On page 3 of the Reliability Meeting Summary the following appears and is attributed to Dr. Ian Wall:

> As human errors, test and maintenance activities and common mode failures are major contributors to system unavailability, and these are difficult if not impossible to accurately quantify, attainment of the goals cannot be demonstrated by analysis. Neither can they be demonstrated by test because of the nature of the equipment and such rare events would require an unrealistic amount of time to test.

With respect to this statement, please answer the following questions:

QUESTIONS II-1, II-2

1. Does the Applicant agree with this statement?

(a) If not, why not?

(b) In answering this question, please reconcile your answer with your answers to Interrogatories I-1 through I-4 above.

2. If the answer to II.1 above is yes, then explain how it will be possible to demonstrate that the CRBR design is acceptable.

ANSWERS II-1, II-2

The Applicants agree that human errors, test and maintenance activities, and common mode failures are major contributors to system unavailability in a system which has been engineered for high availability, such as the CRBRP shutdown and shutdown heat removal systems. The Applicants agree that mathematically rigorous quantification of the unavailability contributed by these causes is difficult. The Applicants agree that system testing to demonstrate achievement of very high availability goals requires an unrealistic amount of time and equipment.

However, the Applicants do not agree that the attainment of such goals cannot be demonstrated. Attainment of such goals for high availability with respect to common mode type failures can be assessed by comparison with operating system data and evaluations to determine the degree of design immunity to common mode or human failures. Comparison to operating systems shows tha a single system can achieve a given availability in practice when subjected to test, maintenance, or other human errors. These data for single systems can be applied to diverse systems by appropriate combinations of these single systems and by taking into account the extent to which the unavailability of the diverse systems depends upon estimated single system unavailabilities. Careful evaluation of common mode failure potential (including human, maintenance, and test) provides the basis for determining the degree of independence, for eliminating or minimizing the influence of common causes through design or procedural changes, and for determining the relative remoteness of postulated failures. Therefore, through the use of operating data, evaluation of the design regarding common mode failure, careful consideration of human errors, and test and maintenance systems, and assessments based on component failure rates, a high availability can be estimated. Given the use of reliability techniques and goals as one aspect of a balanced design safety approach, this attainment by qualitative and quantitative arguments is consistent with the answer to I above.

QUESTION II-3

Is it possible to quantify the probability of deliberate human acts such as sabotage?

(a) Is it possible that the probability of a deliberate act of sabotage is as large as 10^{-4}, 10^{-3}, or 10^{-2} per year?

(b) Is it possible that a deliberate act could produce a CDA and/or a situation wherein 10 CFR 100 criteria could be exceeded?

(c) If deliberate acts such as sabotage were considered as accident initiating events, would it not be possible to include design and operational features that could significantly reduce the residual risk of and from such acts?

i. Are such acts being considered with respect to the CRBR and if not, why not?

ii. If the answer to (i) above is yes, explain in detail those specific design and operational features that have been included for this purpose.

ANSWER II-3

(a) The Applicants have not found any technically credible way to quantify the probability that an act of sabotage would be attempted. For the reasons stated in their response to Subparts (b) and (c) below, the Applicants believe that it is highly improbable that an act of sabotage would be successful. Moreover, specific plant design features and the CRBRP physical security program will be implemented to further reduce the probability of such acts.

(b) As described in detail in the Applicants' response to a prior interrogatory propounded by NRDC in NRDC's Eighth Set of Interrogatories to the

Applicants, it is possible, but highly improbable, that a deliberate act could produce a CDA.

(c) As described in the Applicants' response to the interrogatory referenced above, multiple layers of controls and safeguards to preclude such acts have already been incorporated in the plant design and physical security systems.

QUESTION III

Considering your answers to the above interrogatories (I and II) and considering that we are concerned with determining the precision (quantitatively) with which the residual risks from the operation of the CRBR can be determined, how can it be safely concluded that CDAs can be excluded as DBAs?

1. In answering this question please be responsive to the remark attributed to R. Denise on page 2 of the Summary of the Reliability Program meeting:

> The documentation which NRC has received is basically not acceptable for their audit and review due to the extensive use of engineering judgment.

2. In answering this question please be responsive to the following which appears on page 6 of the CRBRP Reliability Program report:

> The overall design of the CRBRP is based on the natural three levels of design which Regulatory uses to evaluate the adequacy of proposed nuclear power plants . . . The third level provides assurance that the public is protected even in the event of extremely unlikely circumstances of failures or malfunctions.

With respect to this quotation, please also answer the following questions:

(a) What specific design features are included in this third level?

(b) To what extremely unlikely circumstances are they directed?

i. In answering this, please consider your answers to all the above interrogatories and indicate how unlikely are these circumstances.

ANSWER III

It may be safely concluded that the hypothetical core disruptive accidents (HCDAs) can be excluded as a Design Basis Accident. The preventive design features included in the CRBRP and augmented by reliability program activities renders HCDAs to hypotethical events. As described in the PSAR, the preventive features that meet the CRBRP Design Criteria and appropriate Federal Regulations and Criteria are:

- Provided in accordance with the three levels of safety approach

- Comparable to LWR preventive features which have been shown effective in practice

- Under continuing scrutiny through the interaction of reliability and engineering to further reduce the likelihood of failure.

The engineering design (and subsequent steps through to final operation) of these deterministic criteria are augmented in the CRBRP through the reliability program to provide further assurance of the sufficiently low probability of the initiation of an HCDA.

It is important to recognize that exclusion of HCDAs from the list of DBAs is not synonomous with exclusion from consideration altogether. On the contrary, as is explained several times in the information contained in the Public Record (see e.g., References 1-2), events beyond the design base have been examined by the Applicants to determine the impacts associated

with them and, more particularly, the impacts associated with HCDAs have been evaluated in the CRBRP Application.

The treatment applied to HCDAs is described in References 1 and 2, which form a part of the CRBRP Application. As shown in Reference 2, the postulated off-site doses for HCDAs are not excessive considering the highly improbable occurrence of HCDAs. This provides the necessary information to ascertain the residual risks associated with events beyond the design basis.

The Interrogatory requests that the Applicants' answer be responsive to a remark attributed to R. Denise on page 2 of the Summary of the Reliability Program meeting. The remark as quoted in the interrogatory has been taken out of the context in which it was presented. The summary of Mr. Denise's remarks is given in Reference 3. The Applicants' understanding of Mr. Denise's position, as derived from the totality of his remarks, is that the Reliability Program should be re-oriented so as to place less emphasis on numerical allocation of reliability and more emphasis on conformance to other elements of licensing practice, such as design criteria. Quantitative reliability studies on the CRBRP are directed toward effecting design improvements. In order to correctly identify any areas in the design where improvements may be necessary, realistic reliability models and calculations utilizing sensitivity studies are necessary. Very conservative models, calculations, and data can distort the realistic nature of these studies and result in erroneous conclusions. Developing realistic models, data, and assumptions requires the utilization of sound engineering judgment. Therefore, the use of qualified engineering judgment is consistent with and appropriate for the intents and purposes of the CRBRP Reliability Program.

The Interrogatory also quotes statements from page 6 of the CRBRP Reliability Program, and poses certain other questions relating thereto. The statements originate from Section 1.1.2.1 of the PSAR, where essentially identical statements are made, and are amplified into a full description of the design safety approach for CRBRP.

Table 1.1-2 of the PSAR illustrates the classification of events into the three various levels of design. The detailed classification of a much broader spectrum of events is presented in Table 15.1.3-2 of the PSAR. Since the PSAR was docketed, the evolution of the CRBRP design has resulted in the augmentation of the design by the addition of Margin Beyond the Design Base features. These are reported in References 1 and 2.

The Margin Beyond the Design Base features are directed at mitigating the consequences of a postulated accident resulting in the melt-through of the reactor vessel. The scenario used for purposes of this evaluation is described in Section 3 of Reference 2.

It is the judgment of the Applicants, concurred in by the Nuclear Regulatory Commission (Reference 4), that events leading to a scenario of this kind are sufficiently improbable that they need not be included in the list of design basis events for the CRBRP.

References for III

1. CRBRP-3, Volume 1: Energetics and Structural Margin Beyond the Design Base

2. CRBRP-3, Volume 2: Assessment of Thermal Margin Beyond the Design Base

3. Attachment to letter, P. S. Van Nort to R. S. Boyd, "Summary of Meeting Held Between CRBRP Project and NRC to Discuss CRBRP Reliability Program and Related Documentation," April 30, 1976, Docket No. 50-537.

4. Letter, R. P. Denise to L. W. Caffey, May 6, 1976, Docket No. 50-537.

QUESTION IV

If one of the design features specified in III-2(a) above is not a core catcher, precisely how was it excluded as a design feature?

1. In answering this, please consider all of the above interrogatories (I, II and III) and indicate how unlikely are the circumstances that would require a core catcher.

ANSWER IV

A core catcher is not included as a design feature for CRBRP. It is not necessary to include such a device since the anticipated consequences of the postulated event which it might otherwise have been argued would require a core catcher, have been shown to be acceptable without the inclusion of this devicce. See CRBRP-3, Volume 2, Assessment of Thermal Margin Beyond the Design Base.

QUESTION V (PREAMBLE)

Questions V relate to July 14, 1976 letter to Roger Boyd from Lochlin Caffey.

QUESTION V(1)

Are Applicants aware of the NRC practice of establishing the source term for site suitability which could only occur if an accident substantially more severe than the DBA occurred?

ANSWER V(1)

Yes.

QUESTION V(2)

If the answer is yes, what is Applicants understanding of the basis for that approach?

ANSWER V(2)

The basis for that approach is the requirement for compliance with 10 CFR 100.11(a), using the guidance given in the footnote thereto.

QUESTION V(3)

Why do you believe such an approach is not appropriate for the CRBR? In your answer, take into account the relative lack of information and experience with LMFBRs as compared to LWRs.

ANSWER V(3)

The assumption that CRBRP is not in compliance with 10 CFR 100.11(a) is invalid. The CRBRP has been committed to meet the Federal Regulations, including the requirements of 10 CFR 100.11(a).

QUESTION V(4)

Provide a copy of the Project's evaluation of the experimental data base for the alternation/1 processes for LWR, HTGR, and LMFBR facilities. If no such evaluation exists, describe in detail how the evaluation was done, what data was analyzed, who conducted the analysis, how long it took, and when it was completed.

/1 In the responses that follow, the Applicants assume that NRDC meant "attenuation" instead of the term "alternation"

It is presumed that this question relates to the following sentence, which is contained in Mr. Caffey's letter of July 14, 1976:

"Regarding the halogen source term, the Project has evaluated the experimental data base and further compared the attenuation processes for LWR, HTGR and LMFBR facilities."

This is to be interpreted as meaning that the Project has evaluated the experimental data base relative to the halogen source term for LMFBRs, and has, in addition, compared the attenuation processes for LWR, HTGR, and LMFBR facilities. Based on the precedent established in the HTGR, the provision of a technical case with supporting experimental data should suffice to permit credit for physical attenuation processes. The Applicants have not and do not intend to evaluate the data base for either HTGR or LWR since the detailed experimental support for attenuation in these reactors is not relevant to an LMFBR. The relevant factor is the precedent for credit where experimentally supported arguments are presented.

The Project evaluation of the experimental data base relative to the halogen source term for LMFBRs is contained in Mr. Caffey's letter to Mr. Boyd dated March 12, 1976. This is now in the Public Record and attention is directed to Sections 2.1.1, 2.1.2, and 2.1.3 of the attachment to that letter.

Further information relating to experimental determination of halogen attenuation in sodium under conditions in which large bubbles of fission gas are released is contained in References V-1 and V-2.

QUESTION V(5)

Provide a copy of the Project's further comparison of the alternation/1 processes for LWR, HTGR and LMFBR facilities. If no such comparison exists, describe in detail how the comparison was done, what data was analyzed, who conducted the comparison, how long it took, and when it was completed.

ANSWER V(5)

Comparison of LWR, HTGR and LMFBR attenuation processes was conducted as follows:

For LWRs, the guidance given in Reference 3 is that 50% of the halogens are to be assumed to be released into the reactor building, and, of this fraction, 50% is to be assumed to be absorbed onto internal surfaces of the reactor building or adhere to internal components. These assumptions are predicated on a postulation of LOCA with degraded performance of engineered safety features which will result in a substantial reduction in water level within the reactor vessel during the period in which fission products are being released. This is in contrast to the situation for LMFBRs, in which there would be no change in sodium level in the reactor vessel.

For HTGR, the attenuation mechanism consists of hold-up of fission products within the fuel particle coatings during adiabatic heat-up (Reference 4). Thus, it is established that credit may be given to such attenuation mechanisms as may exist due to the unique characteristics of a given reactor design.

QUESTION V(6)

What is the Project's firm reason to believe that the source term is overly conservative and that insufficient credit for halogen alternation/1 is given?

ANSWER V(6)

The Project's firm reasons for belief that the source term is overly conservative are stated in Mr. Caffey's letter to Mr. Boyd of March 12, 1976.

QUESTION V(7)

Describe in detail the alternation/1 mechanisms which the Project claims exist and for which warrant further credit should be given.

ANSWER V(7)

The answer to this question is contained in Sections 2.1.1, 2.1.2, 2.1.3 of Mr. Caffey's letter to Mr. Boyd dated March 12, 1976.

QUESTION V(8)

Provide copies of all the research data upon which the claim for greater alternation/1 mechanisms is based.

ANSWER V(8)

The data is that contained in References V-1 and V-2 quoted above, together with that from References 4, 6, 7, 9, 10, and 12 of Mr. Caffey's March 12 letter.

QUESTION V(11)

Explain the Project's reliance upon the general approach and result of the Reactor Safety Studies for Release Categories 1-7 for LWRs as a basis for a licensing decision on containment venting in light of the following Commission Interim General Statement of Policy (39 Fed. Reg. 30964) (August 27, 1974):

> Accordingly, it is the interim position of the Commission that, pending completion and detailed evaluation of the final (Reactor Safety Study) study, including public comment thereon, (1) no changes in the Commission's safety or environmental regulations pertaining to nuclear power plants are now warranted, (2) the Commission's existing requirements should not be relaxed, and (3) the contents of the draft study are not an appropriate basis for licensing decisions.

ANSWER V(11)

As stated in the letter from Mr. Caffey to Mr. R. S. Boyd dated July 14, 1976, "the Project is using the general approach (realistic assessments) and results of the Reactor Safety Studies for release categories 1-7 for LWRs to gauge the consequences associated with core disruptive and core melt accidents." The Project does not rely on this as the sole basis for a licensing decision.

The cited statement in the interrogatory must be considered in the light of the remarks of then NRC Chairman W. A. Anders, at the time of the release of the final version of the Reactor Safety Study.

"The Commission believes that the Reactor Safety Study report provides an objective and meaningful estimate of the public risks associated with the operation of present-day light water power reactors in the United States. The final report is a soundly based and impressive work. Its overall conclusion is that the risk attached to the operation of nuclear power plants is very low compared with other natural and man-made risks. The report reinforces the Commission's belief that a nuclear power plant designed, constructed and operated in accordance with NRC's comprehensive regulatory requirements provides adequate protection to public health and safety and the environment. Of course, such regulatory requirements must be continually reviewed in the light of new knowledge, including that derived from a vigorous regulatory research program."

Thus, NRC is not using the results of the Reactor Safety Study directly in the licensing of LWRs since one of the major conclusions of WASH-1400 is that the current NRC licensing procedures provide adequate protection for the general public. However, the Reactor Safety Study results are being utilized indirectly in LWR licensing in areas not previously addressed by licensing procedures and regulations (See Reference V-6). Similarly, it is appropriate that the general approach and results contained in the Reactor Safety Study be utilized as one measure of the comparability of the CRBRP to LWRs.

QUESTION V (References)

1. TC-537 LMFBR Source Term Attenuation by D. R. Dickinson and F. H. Nanumaker, December 1975.

2. AI-ERDA-13172 Quarterly Technical Progress Report, January March 1976, pp. 7-17.

3. TID 14844 Calculation of Distance Factors for Power and Test Reactor Sites by J. J. DiNunno <u>et al.</u>, March 23, 1962.

4. NUREG 75/004 Safety Evaluation of the Summit Power Station, Docket No. 50-450, January 1975, pp. 15-19 through 15-22.

5. WASH-1400, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," Final Report, October 1975.

6. Letter from B. C. Rusche, Director, Office of Nuclear Reactor Regulation, to J. E. Ward, Atomic Industrial Forum, Inc., dated August 13, 1976.

QUESTION VI (GENERAL)

Considering your answers to the previous interrogatories in this set, how is it possible to justify the statement on page 11 of the CRBRP Reliability Program report:

Based on the preceding discussion, it is concluded that consistent with Nuclear Regulatory requirements and LWR precedents, a spectrum of events (which include floods and earthquakes in excess of the stipulations of the Regulations and Regulatory Guides and aircraft impacts) is not appropriate for inclusion in the design bases of the CRBRP.

ANSWER VI (GENERAL)

As was stated in the response to the questions in Section III of these interrogatories, exclusion of events from the CRBRP Design Base is not the same as exclusion from consideration altogether. Events of the type noted in the question (floods, earthquakes and aircraft impacts) have been considered in the design. Based on the principle of comparability with light water reactors, the approach adopted has been to establish design bases in precisely the same manner as for light water reactors, and then to test the capability of the design to accommodate some larger event.

For example, in the case of earthquakes beyond the design base, a considerable amount of relevant data was presented at the June 23, 1976, meeting of the Advisory Committee on Reactor Safeguards, CRBRP Subcommittee. See e.g. pp. 167 - 186 of the transcript of that meeting.

Thus, material in the public record has already demonstrated the large margin of reserve capability which exists to accommodate earthquakes more severe than those within the design base, and more severe than required for siting at the overwhelming majority of locations within the continental United States.

With regard to flooding, the protection provided is by purely conventional means, e.g., assurance of flood barriers, placement of building access above maximum flood levels, etc. See Section 2.4, PSAR. Consideration of aircraft proximity is given in Section 2.2.2 of the Preliminary Safety Analysis Report, in which it is shown that there are no commercial flight paths within 10 miles of the Clinch River site, and that the nearest airport (Meadowlake, 10 miles away) only handles small sport-type civil aircraft.

Consideration of other events beyond the design base is included in the response to questions in Section III.

QUESTION VI(1)

In answering this interrogatory, be responsive to the consideration that this is essentially a first-of-a-kind facility and answer separately for each of the objectives of the CRBR relative to the LMFBR Program as stated on page 8-2 of NUREG-0024:

The identified role of the CRBRP was stated in the PFES as follows: It is a key element in both the engineering and manufacuring phase and the

utility commitment phase. In its role as an LMFBR demonstration plant, the principal objectives of the CRBRP are to:

(a) Demonstrate safe, clean and reliable operation with high availability in a utility environment.

(b) Focus the development of systems and components.

(c) Develop industrial and utility capabilities to design, control, operate and maintain an LMFBR.

(d) Demonstrate the licensability of LMFBRs.

ANSWER VI(1)

(a) As has been demonstrated above, there exists considerable capability to protect the public from the consequences of a range of such events beyond the design base.

(b) One element of focusing the development of systems and components is the establishment of appropriate licensing-related requirements on these systems and components. It is clearly desirable, if possible, to do this by reference to existing licensing requirements. For that reason, the process adopted has been that of establishing design bases in these areas on the same premise as is done for light water reactors, and then to evaluate the capability to accommodate events beyond the design base. The demonstration that use of design base requirements in this manner provides a significant level of additional capability is a major step in establishing that certain traditional light water reactor licensing requirements can be directly translated into LMFBR licensing requirements.

(c) Apart from the design element, which is addressed in (b), such considerations are not really directly related to end-of-spectrum events.

(d) The response to this question is contained in (b), above.

QUESTION VI(2)

In answering this interrogatory, explain how, without land use or other regulations, an airport or an FAA airway can be excluded from the CRBR vicinity or that of future LMFBRs.

(a) Also explain how some other potentially undesirable neighbor can be excluded.

ANSWER VI(2)

As indicated above, the nearest commercial flight path is 10 miles away from the CRBRP and the nearest airport is 10 miles away. In addition, the question of land use planning in the CRBRP has been addressed in question response 310.34 (See Section 2.2.3 of the PSAR).

QUESTION VI(3)

In answering this interrogatory, consider the implications relative to the above-quoted objectives of the events recently revealed at the North Anna seismic fault site, the recently revealed magnitude of the fault and seismic events near Diablo Canyon, and of the size of the SSE and OBE at sites such as those along the Pacific Coast.

(a) We are concerned here not only with the licensability of the CRBRP but also with the demonstration of meeting the overall LMFBR Program objective of the CRBR.

ANSWER VI(3)

Consistent with the practice utilized in the licensing of LWRs, the CRBRP plant design will be shown to be compatible, with appropriate margins, for the site location stipulated in the license application. As discussed in response (1) of this interrogatory, there is significant reserve seismic

capability within the current CRBRP design. The existing margin is considered more than adequate for siting at the overwhelming majority of locations within the continental United States.

QUESTION VII (GENERAL)

The following appears on page 6 of the CRBRP Reliability Program report:

> Due to the lack of precedents for LMFBR plants, the CRBRP design approach utilizes reliability techniques extensively to provide a systematic determination of events to be included in the plant design basis.

On page 7 of the same report, the following appears:

> The probability objective of one chance in a million was selected after a review of appropriate nuclear safety oriented literature such as WASH-1270[2] Chauncey Starr's work[3], WASH-1400[4], WASH-1250[5], WASH-1285[6], WASH-1318[7] and others.

Since the first quotation above describes the purpose of the reliability program and the second cites only some of the reliability or safety oriented literature reviewed, please supply us with a more complete listing of the pertinent literature reviewed by the Applicants.

ANSWER VII (GENERAL)

The literature determined to be more pertinent to the stated Project objectives are those referenced in the introduction of the interrogatory. It should be noted that it is not reasonable to cite all references which provide a source of information used to establish a design approach, program objectives, or assessment techniques. The reference having the greatest influence on the specific value chosen was WASH-1270. The remainder of the quoted references were supportive in nature, in that they provided additional evidence that the objective chosen was (1) reasonable

in relation to assuring adequate protection of the public and (2) achievable using proven design concepts.

In addition to those references already noted, the following references are typical of the types of literature which were reviewed and factored into the CRBRP decision-making process as appropriate:

- Farmer, F. R., "Reactor Safety and Siting: A Proposed Risk Criterion," Nuclear Safety, Vol. 8, No. 6, Nov.-Dec. 1967.

- Otway, H. S. and Erdmann, R. C., "Reactor Siting and Design from a Risk Criterion," Nuclear Engineering Design, 1970.

- Godbout, P., "Appendix III, Probabilistic Safety Analysis of a Hypothetical 1000 MWe Liquid Metal Fast Breeder Reactor," Public Health Risks of Thermal Power Plants, Report No. UCLA-ENG-7242, School of Engineering and Applied Science, UCLA, May 1972.

- U.S. Atomic Energy Commission, "Nuclear Power Plants: Seismic and Geologic Siting Criteria," No. 10 CFR 100, Federal Register, Vol. 36, No. 228, 1971.

- Salvatori, R., "Systematic Approach to Safety Design and Evaluation," IEEE Transactions on Nuclear Science, Vol. NS-18, February 1971.

- IEEE, "Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection System," IEEE Std. 352-1975, 1975.

- Morcos, A., "The Role of Probability in Nuclear Plant Design," Consulting Engineer, December 1973.

- Rasmussen, N., "Reactor Safety: Real Probabilities," Combustion, June 1974.

- Bozovsky, I., "Reliability T ory and Practice," Prentice Hall, Englewood Cliffs, New Jersey, 1961.

- Green, A. E. and Bourne, A. J., "Reliability Technology," Wiley Interscience, New York, New York, 1972.

- Epler, E. P., "Common Mode Failure Considerations in the Design of Systems for Protection and Control," Nuclear Safety, Vo. 10, No. 1, Jan.-Feb., 1969.

During the five year period since 1977, considerable attention has been given to formalizing Probabilistic Risk Assessment (PRA) methods and goals. The Applicant has monitored these efforts and reviewed the documentation produced. The objective of the monitoring and review was to ascertain that the Applicants' Reliability Progı m objectives and goals continued to be current with the evolving industrial practices.

The following list of documents identifies the principal documentation reviewed during the last five years:

1. U.S. Nuclear Regulatory Commission, Safety Goals for Nuclear Power Plants: A Discussion Paper (NUREG-0880) (Feb 1982);

2. Nuclear Regulatory Commission Statement of Risk Assessment in Light of the Risk Assessment Review Group Report (Jan. 18, 1979);

3. NRC proposed rule requiring improvements in reactor design to reduce the risks from anticipated trnasients without scram ("ATWS") events (46 Fed. Reg. 57521) (Nov. 24, 1981);

4. Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission (NUREG/CR-0400);

5. Swain, A.D., A.G. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Draft Report (NUREG/CR-1278) (Oct. 1980);

6. Proceeding of the International ANS/ENS Topical Meeting on Probabilistic Risk Analysis (Sept. 20-24, 1981) Port Chester, NY;

7. U.S. Nuclear Regulatory Commission, NRC Action Plan Developed as a Result of the TMI-2 Accident (NUREG-0660) (Aug 1980);

8. "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License. (Mar 1981) (NUREG-0718) (Rev. July 10, 1981);

9. PRA Procedures Guide, A guide to the performance of Probabilistic Risk Assessments for Nuclear Power Plants (Review Draft) (1982) (NUREG-2300, Rev. 1).

QUESTION VII(1)

Has the Applicant reviewed any literature that is skeptical or critical of reliability assessments or estimates such as those presented in WASH-1400?

ANSWER VII(1)

The Applicants are aware of viewpoints that differ from the consensus of opinion regarding the application of reliability technology in general and the assessments presented in WASH-1400 in particular.

With regard to WASH-1400, the Applicants are aware of specific skepticisms or criticisms which have been published. Typical of these are comments by the Union of Concerned Scientists, National Intervenors, Californians for Safe Nuclear Energy, Joel Yellin and William Bryan. These comments were considered in light of LWR experience, reliability experience with other programs, state-of-the-art of reliability technology, a detailed review of WASH-1400 and its implications, and a continuing interaction with personnel involved in the preparation of that report. In those cases where the concerns reflected constructive criticisms, they have been appropriately

considered in their application to the CRBRP Reliability Program. Many of the criticisms, however, were found to be without merit.

The more recent literature which the Applicants have reviewed is overwhelmingly in support of the application of reliability methods.

QUESTION VIII

Mr. William M. Bryan, former manager of reliability studies in the NERVA and Apollo programs, has made the following criticism of WASH-1400 (a copy of his entire report is attached; the section quoted below is found on page 4):

> As an example, the Apollo 4th stage rocket engine had an assigned reliability of .999 (or 1 failure allowed per 10,000 missions). This assigned value was derived in the early 1960's quite similarly to the quantitative values calculated in the WASH 1400 report. The highest reliability estimate achieved by this engine after thousands of actual tests was approximately 0.96 (or 4 failures per 100 missions). Thus, in this case an error or uncertainty factor of 400 existed between the predicted and actual reliability. Use of this type of reliability assignment (prediction) techniques consistently led to such an overstatement of reliability, which was one of the main reasons these techniques were abandoned by NASA. By comparison to a nuclear power plant, this engine was a very simple system which represented an off-the-shelf technology. Many of the failures that occurred were due to human errors during manufacture and assembly, errors which got through unnoticed despite the most sophisticated quality control procedures ever utilized to discover and prevent them (quality control procedures and funding considerably superior to those currently used in nuclear power plant construction). Many others were due to design mistakes or lack of knowledge. None were planned. Each time the system was tested, the engineers and analysts expected it to fully succeed, much the same as the technical optimism that is evident in WASH 1400.

With respect to this statement by Mr. Bryan, please answer the following questions:

1. Does the Applicant agree with the statement? If not, why not?

2. Considering that one of the major objectives of the Apollo program was the safety of the astronauts, does the Applicant consider that the incineration of only 3 astronauts on the pad at Cape Kennedy is representative of an adequate program? If not, why not?

ANSWER VIII

The comments referred to in this interrogatory are part of an overall set of comments on WASH-1400 by Mr. Bryan. The principal point of these comments is that the analytical techniques used in the study lead to optimistic conclusions. It was further asserted that the National Aeronautics and Space Administration (NASA) and the Aerospace Industry abandoned the use of these techniques for this reason.

In arriving at a decision as to the validity of these comments, it is instructive to review NASA's own comments on the study's methodology. Their comments are contained in a letter of June 16, 1975, from the Administrator of NASA to the Chairman of the U.S. Nuclear Regulatory Commission (a copy of which is provided as Attachment 1). Briefly, the NASA letter states that the techniques used in the study are effective and are capable of producing numerical assessments of value if the data base from which failure probabilities are determined has sufficient accuracy and content in light of the quantification being performed.

Additional insight into this question is provided by A. E. Green, Manager of the Systems Reliability Service (SRS) operated by the United Kingdom Atomic Energy Authority. A copy of a letter from Mr. Green to the Project Staff Director of the WASH-1400 study is also included as Attachment 2. Mr. Green states that where they have applied quantitative reliability techniques for prediction, there has been reasonable agreement with field experience when it became known, generally within a factor of two. In support of this realistic prediction capability a graph from Reliability Technology is provided as Attachment 3, which shows the close agreement the SRS group has experienced between predicted and observed system failure rates for some 50 system elements.

An objective review of these and other comments leads to the conviction that the analytical techniques used in WASH-1400 were current, applied appropriately, and can serve to provide realistic system failure predictions.

The foregoing is not intended as a justification of WASH-1400, rather it is an indication of the awareness of the Applicants of the state-of-the-art of reliability technology and the conviction that when appropriately applied it forms an effective part of the CRBRP development process.

THIRTEENTH INTERROGATORY SET

PREAMBLE TO QUESTIONS

There are several LMFBR safety systems known to us that are currently being discussed in the literature at reactor safety meetings, etc., but that are not presently incorporated in the CRBR design. These include (a) the use of a third shut-down system that is self-actuated and that does not rely on any instrumentation, e.g., using curie point magnet as a release mechanism, (b) the "parfait" core arrangement, i.e., interspacing the core and blanket materials; (c) use of heavy flywheels on the primary sodium pumps; (d) designing the reactor to operate at a lower sodium temperature, thereby sacrificing efficiency for safety, and (e) the use of a core catcher and other design alternatives presently being considered but not incorporated in the present CRBR design.

The third self-actuated shutdown system was discussed by Demetrious L. Basdekas of the NRC Staff at the December 3, 1976, ACRS meeting (Transcripts, pp. 2[illegible]3-241 and attachments). The "parfait" core was discussed at the International Meeting on Fast Reactor Safety and Related Physics in Chicago, October 3-8, 1976. The use of flywheels on the sodium pumps is utilized in the Phenix to delay onset of a CDA in the LOF scenario. The use of a lower sodium operating temperature has been proposed to ERDA personnel by Hans Bethe. The core catcher is well known to the Applicants/Staff and is incorporated in the German LMFBR demo design. We are interested in identifying all such safety systems that are not presently incorporated in the CRBR design, obtaining complete documentation on these systems, determining the effects and safety advantages that these systems would have if incorporated into the CRBR design, including the implication on criteria related to site suitability, CDA probability and energetics and, finally, the effects on cost and schedule if these systems were added to the CRBR.

Each of the following questions is to be answered in six parts, as follows [Where appropriate, the parts of the question have been restated to reflect the protocol for discovery agreed to by Applicants, Staff, and Intervenors NRDC et al.]:

(a) Provide the direct answer to the question.

(b) Identify all documents and studies, and the particular parts thereof, relied upon by the Applicants, now or in the past, which serve as the basis for the answer. In lieu thereof, at Applicants' option, a copy of such document and study may be attached to the answer.

(c) Identify principal documents and studies, and the particular parts thereof, specifically examined but not cited in (b). In lieu thereof, at Applicants' option, a copy of each such document and study may be attached to the answer.

(d) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) who provided the answer to the question.

(e) Explain whether the Applicants are presently engaged in or intend to engage in any further research or work which may affect the Applicants' answer. This answer need be provided only in cases where Applicants intend to rely upon on going research not included in Section 1.5 of the PSAR at the LWA or construction permit hearing on the CRBR. Failure to provide such an answer means that Applicants do not intend to rely upon the existence of any such research at the LWA or construction permit hearing on the CRBR.

(f) Identify the expert(s), if any, whom the Applicants/Staff intend to have testify on the subject matter questioned. State the qualifications of each such expert. This answer need not be provided until Applicants have identified the expert(s) in question or determined that no expert(s) will testify, as long as such answer provides reasonable notice to Intervenors.

QUESTION 1

Please identify all safety systems, materials, concepts and significant design alternatives, including those discussed above that are not presently incorporated in the CRBR design but that could conceivably impact on the probability or energetics of a CRBR CDA.

ANSWER 1

The safety systems, materials, and design concepts incorporated into the CRBRP design that have an impact on the probability of CDAs are discussed in the following PSAR Sections: 1.6 (Reference 10), 3.1, 3.2, 3.8, 4.1, 4.2, 4.3, 4.4, 5.2, 5.3, 5.4, 5.5, 5.6, 7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 8.2, 10.4, 15.1, 15.2, 15.4.

(a) Third Shutdown System

The Applicants have incorporated two shutdown systems in the reactor shutdown system to ensure that all postulated off-normal occurrences are terminated without initiation of an HCDA. Any new shutdown systems would be less certain and less tested. There is no reliability data for them, as there are for the existing primary and secondary shutdown systems. All identified failure modes are addressed by the primary and secondary shutdown systems; a new system does not address any other failure modes. Based on the guidance of WASH-1270 (ATWS), the May 6, 1976, letter from R. Denise to L. W. Caffey, and Applicants' assessments, the Applicants do not believe that the addition of a third shutdown system would have a significant impact on further reducing the probability of an HCDA, and therefore, the third system was not included in the CRBRP design.

(b) Parfait Core Concept

The Applicants have not conducted a detailed analysis of the particular "parfait" core concept as studied at MIT and discussed at the ANS International Meeting on Fast Reactor Safety and Related Physics. Many "Parfait"

and "Radial Parfait" or heterogeneous concepts have been postulated to enhance breeding characteristics of LMFBRs. Heterogeneous core concepts were evaluated by the Applicants. One suited to the CRBRP design objective was adopted and is described in the PSAR as the current reference design.

The preliminary indications are that the energetics potential for the initiating phase of the HCDA may be even less with the heterogeneous design than with the earlier homogeneous fuel management scheme. The heterogeneous core is described in PSAR Section 4. The reference fuel management scheme has been extensively analyzed by the Applicants and has been shown to have negligible potential for energetics resulting from a postulated HCDA (See CRBRP-GEFR-00523).

Since the design of the systems which prevent initiation of HCDAs did not materially change for the heterogeneous core design, there is a negligible difference in the probability of initiation of an HCDA occurring with either fuel management scheme. The previous high degree of redundancy and diversity in the CRBRP design is retained with the heterogeneous fuel management concept.

(c) Heavy Flywheels

The CRBRP primary flow coastdown characteristics have been selected by balancing two requirements, which are:

1. To provide adequate coolant flow to the core and radial blanket for all design basis events including postulated loss of power to all the three primary pumps.

2. To minimize the thermal transients associated with reactor and plant trips.

The required flow coastdown characteristics will be provided for the CRBRP by building sufficient inertia directly into the pump drive motor rotor such that the momentum of the pump-drive motor assembly will be available for these purposes. This inertia satisfies both of the above requirements

and obviates the need for the addition of a separate flywheel for these purposes.

Lower probability events which are beyond the design base have also been considered. The probability or the resultant scenario of the postulated transient overpower (TOP) events, which assume failure of both reactor shutdown systems, would not be changed by modified primary pump inertia since no pump trip is involved. The effect of increased primary pump inertia for the postulated loss of flow (LOF) events, which also assume failure of both reactor shutdown systems, would be to slightly change the time scale for the events but not the overall conclusions. The time for initiation of boiling might increase, but once boiling is initiated, the sequence of events would be controlled by the phenomenon related to boiling, such as void reactivity insertion, cladding dryout, fuel failure and post-failure fuel and cladding motion. These phenomena lead to a prediction of event termination in a non-energetic manner. Increased pump inertia would not change the probability of sodium boiling and the resultant consequences. Thus, increased pump inertia would be ineffective in significantly impacting the probability or consequences of a LOF HCDA, and, therefore, was not included in the CRBRP design.

(d) Operation at Lower Temperature

The operating temperatures of the plant have been selected based upon detailed plant performance analyses which evaluated a variety of factors, including operation of the plant at high thermal efficiency consistent with fuel breeding goals and material capabilities. Section 15 of the PSAR, "Accident Analysis," clearly demonstrates that the selected plant operating conditions are acceptable.

The effect of choosing a lower plant operating temperatures would not significantly change the TOP HCDA consequences because the current TOP scenario results in molten fuel release from the pin before coolant boiling occurs. Thus, the overall conclusions regarding the TOP HCDA would not be influenced by a choice of lower operating temperature.

The effect of lower operating temperatures on the probability and consequences of a LOF HCDA is similar to that described in (c) above. The time to initiate boiling would be slightly increased, but the probability or consequences of sodium boiling would not change. Thus, a lower operating temperature would not significantly impact the probability or consequences of a LOF HCDA.

(e) Core Catcher

The core catcher would not have any impact on the probability or energetics of an HCDA; rather the presence or absence of a core catcher would impact upon the level of consequences to occur in the unlikely event of an HCDA with core melt. The Applicants' current assessment entitled Third Level Thermal Margin (TLTM) indicates that the current plant design, without the core catcher, provides acceptable protection to the public health and safety.

QUESTION 2 (PREAMBLE)

Questions 2(a) through 2(f) pertain to each of the systems, etc. Identified in the response to Question 1.

QUESTION 2(a)

Please identify and supply complete and current documentation of these systems, materials, concepts and design alternatives.

ANSWER 2(a)

Refer to the PSAR sections and documentation enumerated in the response to Question 1 above.

QUESTION 2(b)

Describe to the fullest extent possible and quantify where possible the impact each would have on the probability and/or energetics of CDAs if each was incorporated in the CRBR design.

ANSWER 2(b)

See the response to Question 1.

QUESTION 2(c)

What effect in terms of CRBR cost and scheduling would be felt if it was incorporated in the CRBR design?

ANSWER 2(c)

There is no impact on CRBRP cost and scheduling for those safety systems, materials, and design concepts already incorporated into the design of the CRBRP as discussed in the PSAR. The Applicants have not performed detailed cost and schedule analyses to assess the impact of those additional design features identified in the background provided with this interrogatory.

QUESTION 2(d)

What was the basis for the Applicants determination that it should be excluded from the CRBR design?

ANSWER 2(d)

The additional design features proposed by this interrogatory would not provide significant additional protection for the health and safety of the public nor are they necessary to meet NRC criteria including those set

forth in the NRC May 6, 1976, letter. Those features that are necessary are discussed in the PSAR sections identified above.

QUESTION 2(e)

To what extent do the Applicants' conclusions regarding (i) CDA energetics, (ii) probability of CDA, (iii) adequacy of the current design to cope with CDAs with respect to their prevention or mitigation, or (iv) the adequacy of the NRC criteria set forth in the May 6, 1976, letter by Denise impact on whether it should be included or excluded from the CRBR design?

ANSWER 2(e)

The Applicants' conclusions regarding CDA energetics, the probability of a CDA, and the adequacy of the current design to cope with CDAs with respect to their prevention or mitigation and the impact which these features had on the Applicants' decision to exclude these additional design features is presented in 2(a) through 2(d) above. The CRBRP design features specified in the PSAR are in compliance with the NRC criteria set forth in the May 6, 1976 letter by Denise.

QUESTION 2(f)

If the Applicants' conclusions regarding items (i) through (iv) in Question 2(e) above were substantially found to be in error and non-conservative, how large an error or what change in criteria would be required before the Applicants would likely consider incorporating it in the CRBR design?

ANSWER 2(f)

The Applicants have provided adequate margins in the CRBRP design to provide protection for the health and safety of the public, as discussed in the PSAR and the Third Level Thermal Margin Report. These margins also

provide sufficient protection against uncertainties which might exist in the data base. The incorporation of the additional features mentioned above into the CRBRP design would not provide significant additional protection to the public health and safety for the reasons stated in response to Question 1.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
DEPARTMENT OF ENERGY	)	
PROJECT MANAGEMENT CORPORATION	)	DOCKET NO. 50-537
TENNESSEE VALLEY AUTHORITY	)	

AFFIDAVIT OF PAUL W. DICKSON, JR.

being duly sworn, deposes and says as follows:

1. That he is employed by Westinghouse Electric Corporation as Technical Director, Clinch River Breeder Reactor Project, Westinghouse Advanced Reactors Division, Post Office Box W, Oak Ridge, Tennessee 37830

2. That he is duly authorized to answer the Interrogatories numbered 1,2,18-21,33,48,52, and 54 in NRDC's Seventh set of Interrogatories, and all Interrogatories except V (9,10) in NRDC's Tenth set of Interrogatories.

3. That the above-mentioned and attached answers are true and correct to the best of his knowledge and belief.

Paul W. Dickson, Jr.
SIGNATURE

SUBSCRIBED and SWORN to before me this 28th day of April, 1982.

Inez W. Minor
Notary Public

My Commission Expires April 28, 1984

04M/28D 20:11 GMT CRBRP OAK RIDGE FTS 626-6100

My commission expires

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)		
UNITED STATES DEPARTMENT OF ENERGY)		Docket No. 50-537
PROJECT MANAGEMENT CORPORATION)		
TENNESSEE VALLEY AUTHORITY)		

AFFIDAVIT OF RICHARD KEPPLE DISNEY

Richard K. Disney, being duly sworn, deposes and says as follows:

1. That he is employed by,

Westinghouse Electric Corporation

Advanced Reactors Division
P.O. Box 158
Madison, Pennsylvania 15663

as Manager, Shielding Analysis.

2. That he is duly authorized to answer the Interrogatories numbered I and II for (Old) Contention 8 in NRDC's 9th set of Interrogatories, and I and II of (Old) Contention 14 in NRDC's 9th set of Interrogatories.

3. That the above mentioned and attached answers are true and correct to the best of his knowledge and belief.

Richard K. Disney
(Signature)

Subscribed and sworn to before me this 28th day of April, 1982.

Mary G. Hayden
Notary Public

My Commission expires MARY G. HAYDEN, NOTARY PUBLIC SEWICKLEY TWP., WESTMORELAND COUNTY MY COMMISSION EXPIRES FEB 13, 1984 Member, Pennsylvania Association of Notaries

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
UNITED STATES DEPARTMENT OF ENERGY	)	DOCKET NO. 50-537
PROJECT MANAGEMENT CORPORATION	)	
TENNESSEE VALLEY AUTHORITY	)	

AFFIDAVIT OF DR. CARL ALBERT ANDERSON, JR.

Dr. Carl A. Anderson, being duly sworn, deposes and says as follows:

1. That he is employed by Westinghouse Electric Corporation
Advanced Reactors Division
Post Office Box 158
Madison, Pennsylvania 15663

as Manager, Reactor Projects.

2. The he is duly authorized to answer the Interrogatories numbered 1 and 2 (a) thru (f) in NRDC's 13 set of Interrogatories.

3. That the above-mentioned and attached answers are true and correct to the best of his knowledge and belief.

Carl A. Anderson
(Signature)

Subscribed and sworn to before me this 21st day of April, 1982.

Mary G. Hayden
Notary Public

My Commission expires MARY G. HAYDEN, NOTARY PUBLIC
SEWICKLEY TWP. WESTMORELAND COUNTY
MY COMMISSION EXPIRES FEB. 15, 1984
Member, Pennsylvania Association of Notaries

(1)

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)

UNITED STATES DEPARTMENT OF ENERGY) DOCKET NO. 50-537

PROJECT MANAGEMENT CORPORATION)

TENNESSEE VALLEY AUTHORITY)

AFFIDAVIT OF LEE E. STRAWBRIDGE

Lee E. Strawbridge, being duly sworn, deposes and says as follows:

1. That he is employed by the Westinghouse Advanced Reactors Division as Manager, Nuclear Safety and Licensing, P. O. Box 158, Madison, Pennsylvania 15663.

2. That he is duly authorized to answer the Interrogatories numbered 7-16, 22, 26, 28, 34, 36, and 42 of NRDC's Seventh Set of Interrogatories.

3. That the above-mentioned and attached answers are true and correct to the best of his knowledge and belief.

Lee E. Strawbridge
(Signature)

Subscribed and sworn to before me this 29th day of April, 1982.

Notary Public

My Commission expires [illegible].

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	
UNITED STATES DEPARTMENT OF ENERGY	)	DOCKET NO. 50-537
PROJECT MANAGEMENT CORPORATION	)	
TENNESSEE VALLEY AUTHORITY	)	

AFFIDAVIT OF DENNIS M. SWITICK

Dennis M. Switick, being duly sworn, deposes and says as follows:

1. That he is employed by the General Electric Company as Manager, Safety Analysis, Advanced Reactor Systems Department, 310 De Guigne Drive, Sunnyvale, California 94086.

2. That he is duly authorized to answer the interrogatories numbered 3-6, 17, 23, 24, 25, 29-32, 37-41, 43-47, 49-51, and 53 of NRDC's Seventh Set of Interrogatories.

3. That the above-mentioned and attached answers are true and correct to the best of his knowledge and belief.

Dennis M. Switick
(Signature)

Subscribed and sworn to before me this 28th day of April, 1982.

Notary Public

My Commission expires 9/23/84.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	
UNITED STATES DEPARTMENT OF ENERGY	)	DOCKET NO. 50-537
PROJECT MANAGEMENT CORPORATION	)	
TENNESSEE VALLEY AUTHORITY	)	

AFFIDAVIT OF L. WALTER DEITRICH

L. Walter Deitrich, being duly sworn, deposes and says as follows:

1. That he is employed by the Reactor Analysis and Safety Division of Argonne National Laboratory, 9700 So. Cass Avenue, Argonne, Illinois 60439, as Associated Division Director.

2. That he is duly authorized to answer the Interrogatories numbered 27 and 36 of NRDC's Seventh Set of Interrogatories.

3. That the above-mentioned and attached answers are true and correct to the best of his knowledge and belief.

Walter Deitrich
(Signature)

Subscribed and sworn to before me this 27th day of April, 1982.

Notary Public

My Commission expires 9/23/84.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor Plant)	Docket No. 50-537

CERTIFICATE OF SERVICE

Service has been effected on this date by personal delivery or first-class mail to the following:

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DATED: April 30, 1982

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