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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 #1 TO NATURAL RESOURCES DEFENSE COUNCIL, INC. AND SIERRA CLUB INTERROGATORIES (SECOND, THIRD, FOURTH, FIFTH AND SIXTH 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 Second, Third, Fourth, Fifth and Sixth Sets of Interrogatories to the Applicants, dated December 23, 1975, December 31, 1975, January 14, 1976, February 12, 1976 and April 7, 1976, 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 #1 supercede all prior answers to the interrogatories as to which they are applicable.

In many 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 Appendix F was withdrawn from the application in 1976.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 desiign and are not appropriately applicable to the current design Applicant has so noted.

SECOND INTERROGATORY SET

PREAMBLE TO QUESTIONS

With respect to the following requests for information we are concerned with four distinct validations relative to the models and computer codes:

i) Validation that the code's output is the correct numerical calculation that should result from a given set of input data and the model assumptions;

ii) Validation of the models against actual experimental data;

iii) Validation that the models can be extended to the CRBR; and

iv) Validation that the input assumptions for the CRBR case are adequate with respect to the CDA analysis, i.e., are supported by experimental evidence. By "adequate", here and below, we mean that the calculations will not underestimate the CDA work potential (i.e., forces and resulting energetics of a CDA) or overestimate the containment capability of the reactor with respect to CDA.

QUESTION I

With respect to each of the following codes and each sub-routine of each of the following codes:

(A) SAS3A (including SASBLOK),

(B) VENUS,

(C) PLUTO,

please provide the following information [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.]:

1) Complete, current documentation (i.e., a writeup) of the codes and the subroutines;

2) Identify, by name and affiliation, the author, or authors of each model, subroutine, or portion of each subroutine, which each contributed or worked on;

3) Identify by name affiliation (including organization, division, branch, title, etc.) each applicant employee, or consultant, that has intimate working knowledge of the code and each subroutine, or parts thereof, including its validity. Where more than one person is involved, delineate which portion of the code or subroutine with which each has an intimate working knowledge;

4) Describe fully the procedures by which Applicant has assured itself and continues to assure itself, that the various computer programs (codes) accurately reproduces the models as described in the PSAR and its references (see Validation (i) above);

5) Indicate which models (including subroutines, or portions of subroutines) have not been validated as described in Validation (i);

6) Indicate the models (including subroutines, or portions of subroutines) or assumptions that have not been validated as described in Validation (ii);

7) For each model, portion of the model, or assumption that has been validated (against experimental (or other) data, see Validation (ii) above) describe fully the procedure by which it was validated, and the results, including all uncertainties and limitation of the validation. Indicate the source of the experimental, or other data, that was used in the validation;

8) Explain fully all instabilities in the numerical performance in the models, what causes them, and how they are avoided, and the extent to which this introduces uncertainties in the calculations and limits the validity of the model (cf., p.F6.2-10, para. 2).

9) To the extent that any answers to the above questions are based on referenced material, please supply the references;

10) 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.

11) 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.

ANSWER I(A)

These answers provide the information requested relative to the SAS3A (including SASBLOK) computer code and have been revised to address the current application of SAS3D.

(1) References 6, 7, 8, 9, and 10 on page 11-1 of CRBRP-GEFR-00523, "An Assessment of HCDA Energetics in the CRBRP Heterogeneous Reactor Core", S.K. Rhow, et al., describe the SAS3A code, the fuel-coolant interaction model, the clad motion model and the fuel motion model in the SAS3A code. The SAS3A sodium film motion model is documented in: G. Hoppner, "Sodium Flow Motion Model of SAS3A," ANL/RAS 74-22, 1974. The SAS3A primary loop model is documented in: Ref. 30 in CRBRP-GEFR-00103. The SAS3D code, now being used, evolved from SAS3A which evolved from the SAS2A code which evolved from the SAS1A code. Documentation for the SAS3A code is applicable to the SAS3D code. The SAS2A code is documented in Reference 7 in CRBRP-GEFR-00523, and SAS1A is documented in ANL-7607, "SAS1A, A Computer Code for the Analysis of Fast Reactor Power and Flow Transients," by D.R. MacFarlane et al. The SASBLOK algorithms used in the SAS3A and SAS3D codes are documented in CRBRP-GEFR-00103, "An Analysis of Hypothetical Core

Disruptive Events in the Clinch River Breeder Reactor Plant", J.L. McElroy, et al.

(2) The SAS3A and SAS3D codes are complex code systems which have been developed over a period of years by the Reactor Analysis and Safety Division of Argonne National Laboratory. The SASBLOK algorithm was developed by the General Electric Company. The principal contributors to the SAS3A and SAS3D code development are identified as authors of the references in Response 1.

(3) The following staff members of Argonne National Laboratory and General Electric have a working knowledge of the codes, including their range of applicability and the efforts that have been made to validate them: L. Walter Deitrich, Associate Director, Reactor Analysis and Safety Division, Argonne National Laboratory; David P. Weber, Manager, Accident Analysis Section, Reactor Analysis and Safety Division, Argonne National Laboratory; Dennis M. Switick, Manager, Safety Analysis, General Electric Advanced Reactors Systems Department.

(4) The entire SAS3A and SAS3D codes, including all subroutines, have been checked and rechecked to assure that the numerical algorithms which are implemented in them to solve the equation sets which constitute these codes, behave in a stable fashion (both individually and collectively) and produce accurate solutions to the original equation sets. This was carried out by comparing SAS3A and SAS3D results with the output from other codes, with the results of hand calculations, and with what sound engineering judgement deemed to be physically reasonable.

(5) All models have been validated as discussed in (4) above.

(6) - (7) The experimental basis for the SAS3A code as of April 1974 has been documented in the paper, "Current Status and Experimental Basis of the SAS LMFBR Accident Analysis Code System," Proc. Am. Nucl. Soc. Fast Reactor Safety Conf., Beverly Hills, California, CONF-740401, pp. 1303-1318.

Additional comparisons of the SAS3A code with experiments have been made since that time and are documented in the following references:

(1) Ref. 32 in CRBRP-GEFR-00103.

(2) Ref. 59 in CRBRP-GEFR-00103.

(3) Ref. 8 in CRBRP-GEFR-00523, pp. 54-62.

(4) Ref. 28 in CRBRP-GEFR-00103, pp. 64-100.

(5) L. W. Deitrich, "Analysis of Transient Fuel Failure Mechanisms, Selected ANL Programs, "Presented at the International Working Group on Fast Reactors Specialists' Meeting on Fuel Failure Mechanisms, Seattle, Washington, May 11-16, 1975.

(6) E. Barts, et al., "Summary and Evaluation, Fuel Dynamic Loss-of-Flow Experiments (Tests L2, L3, and L4)," ANL 75-57, September 1975.

The experimental basis for the SAS3A is applicable to SAS3D and additional experimental basis is documented in the following:

(1) Ref. 35 in CRBRP-GEFR-00523.

(2) "Final Report on the SLSF In-pile Experiment P3A," T.E. Kraft and L.R. Kelman, ANL/RAS 81-20, June, 1981.

(3) W.A. Ragland, "LMFBR Loss-of-Flow Simulations in the Sodium Loop Safety Facility," ASME Paper 80-C2/NE-22, presented at the Century 2 Nuclear Engineering Conference, San Francisco, Aug. 19-21, 1980.

It should be noted that many of the models used in SAS3A and SAS3D are parametric in nature and justification for the particular parameters used in the analysis is given in CRBRP-GEFR-00103 and CRBRP-GEFR-00523. Because of this parametric nature of the SAS3A and SAS3D codes, they can be used to draw valid conclusions relative to the course of hypothetical accidents in an LMFBR even though each subroutine may not have been completely validated by experiments, since parameters can be varied to determine the sensitivity of the results to variations in parameters.

(8) Mathematically, practically all of the models in SAS3A and SAS3D consist of sets of coupled ordinary differential or integro-differential

equations in time or of coupled partial differential equations in space and time. Numerically, these equation sets are solved by applying appropriate linearization and finite-differencing techniques. Some of these temporal finite-differencing techniques are fully implicit and are unconditionally stable. Other models, such as that which treats the time-dependent radial heat transport from the fuel pin into the coolant, have their equation sets solved by semi-implicit temporal finite-differencing techniques. It is well known that solutions obtained by semi-implicit differencing can exhibit bounded oscillations if time steps which are too large are taken. Thirdly, some equation sets, such as the SLUMPY compressible hydrodynamics equations, are solved with fully explicit methods. Here, taking time steps that are too large can produce solutions which become unstable.

Throughout SAS3A and SAS3D provisions have been made to insure that the time step sizes being used for advancing the various solutions in time are kept sufficiently small so that the solutions behave stably and are accurate. These time step sizes are chosen by monitoring both the solutions and their rates of change and applying step size selection criteria based on both known analytical constraints, where they are available, and on experience gained in applying the code to a variety of situations. These step size selection criteria are explained in detail in the references provided in part 1 above. It is still possible, however, to occasionally force a model in the SAS3A or SAS3D code to utilize a time step size which is so large that stability problems result. It is also possible for the user to try to utilize SAS3A or SAS3D to analyze cases which are not intended to be modeled by SAS3A or SAS3D. In these cases, the results predicted by SAS3A or SAS3D may tend to become unrealistic and physically meaningless. Both of these problems can and are generally dealt with by carefully scrutinizing the computer output and comparing it against engineering judgment.

(9) The reference documents have been or will be made available for inspection and copying.

(10) The Applicants are currently analyzing this area and have documented the planned program of research in Appendix A to CRBRP-3, Vol. 1. Applicants have not yet determined whether they will rely on the results of future analysis.

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

ANSWER I(B)

These answers provide the information requested relative to the VENUS computer code.

(1) The VENUS-II code is documented in Ref. 5 in CRBRP-GEFR-00523.

(2) The principal contributors to the VENUS-II code are identified as the authors of the reference in response 1.

(3) The following staff members of Argonne National Laboratory have a working knowledge of the code, including its range of applicability and the efforts that have been made to validate it: L. Walter Deitrich, Associate Director, Reactor Analysis and Safety Division, Argonne National Laboratory, and David P. Weber, Manager, Accident Analysis Section, Reactor Analysis and Safety Division, Argonne National Laboratory.

(4) The entire VENUS-II code has been thoroughly checked to assure that the equation sets and algorithms given in Ref. 5 in CRBRP-GEFR-00523 are accurately programmed into VENUS-II. Because these equation sets are relatively simple, this was done by comparing output from the various subroutines against hand calculations.

(5) All models have been validated as described in (4) above.

(6) - (7) The VENUS-II code has been validated against the KIWI-TNT experiment and SNAPTRAN-2 AND SNAPTRAN-3 experiments. See: "Improvement and Verification of Fast Reactor Safety Analysis Techniques," Progress Report,

Jan. 1, 1977 to Mar. 31, 1977, COO-2571-2, by Dee H. Barker, Terry F. Bott, Paul A. Wheeler, Larry Lamonica, and James F. Jackson, Department of Chemical Engineering, Brigham Young University, Provo, Utah; also T.F. Bott and J.F. Jackson, "Experimental Comparison Studies with the VENUS-II Disassembly Code," Proc. Intl. Mtg. on Fast Reactor Safety and Related Physics, Chicago, October 1976, pg. 1139.

(8) The numerical algorithm utilized in VENUS-II to solve the compressible hydrodynamics equations in two-dimensional cylindrical geometry involves an explicit finite differencing of the temporal derivatives. A satisfactory method has been implemented in VENUS-II to control time step size so that its calculations remain stable and accurate. This method is summarized in Ref. 5 in CRBRP-GEFR-00523 and is described in detail in the paper, J. F. Jackson, R. B. Nicholson, and W. T. Sha, "Numerical Stability Problems in the VENUS Disassembly Code," Proc. of Conf. on New Developments in Reactor Mathematics and Applications, CONF-710302, Vol. 1, pp. 152-165, 1971. This reference will be made available for inspection and copying.

(9) The reference documents have been or will be made available for inspection and copying.

(10) The Applicants are not doing development work on VENUS-II. No such development work is currently planned.

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

ANSWER I(C)

These answers provide the information requested relative to the PLUTO 1 and PLUTO 2 Computer Codes.

(1) References 21 in CRBRP-GEFR-00103 and 25 in CRBRP-GEFR-00523 describe the PLUTO 1 and PLUTO 2 Codes respectively.

(2) The author of the PLUTO 1 and PLUTO 2 codes is Hartmut U. Wider, Associate Section Manager, Accident Analysis Section of the Reactor Analysis and Safety Division, Argonne National Laboratory.

(3) The following staff members of the Argonne National Laboratory have a working knowledge of the code, including its range of applicability and the efforts that have been made to validate it: L. Walter Deitrich, Associate Director, Reactor Analysis and Safety Division, Argonne National Laboratory and David P. Weber, Manager, Accident Analysis Section, Reactor Analysis and Safety Division, Argonne National Laboratory.

(4) The PLUTO 1 and PLUTO 2 codes have been checked and rechecked to assure that the numerical algorithms which are implemented in them to solve the equation sets have been programmed correctly. Furthermore, test calculations were performed to assure that these numerical algorithms behave in a stable fashion and produce accurate solutions to the original equation sets. This was carried out by comparing PLUTO 1 results with the output from another code (see Ref. 21 in CRBRP-GEFR-00523) with the results of hand calculations, and with what sound engineering judgement deemed to be physically reasonable. PLUTO 2 results have been compared with PLUTO 1 results (see H. U. Wider, PLUTO 2: A Computer Code for the Analysis of Overpower Accidents in LMFBRs, TANSAO 27, p. 533, 1977) and with EPIC results (see H. U. Wider, et al., The PLUTO 2 Overpower Excursion Code and Comparison with EPIC, Proc. of the International Meeting on Fast Reaactor Safety Technologies, Seattle, 1979, p. 120).

(5) All models have been validated as described in (4) above.

(6) and (7) The sodium voiding rates calculated by PLUTO 1 and PLUTO 2 strongly depend on a few input parameters concerning the fuel-coolant interaction and the fuel pin pressures at the pin failure time. These input parameters can be chosen such that voiding rates similar to those in TREAT in-pile experiments are calculated.

The rapid sodium voiding which occurred in the H4 TREAT test was analyzed with PLUTO 1. This led to important information concerning fuel-coolant

interaction parameters. (See, H. U. Wider and A. E. Wright, "Analysis of a Sodium Reentry Event in the H4 TREAT Test," Trans. Am. Nucl. Soc., TANSAO 22, p. 428, 1975.) PLUTO 1 has also been used to analyze the sodium voiding in the E8 and H6 tests. PLUTO 2 has also been used for analyzing part of the H6 TREAT test (see Ref. E-3 in CRBRP-GEFR-00523). PLUTO 2 has also been used for analyzing the voiding in the L8 TREAT test (see Ref. E-4 in CRBRP-GEFR-00523).

(8) Mathematically, PLUTO 1 and PLUTO 2 consist of sets of coupled hyperbolic partial differential equations which are of first order in time and of first and second order in space. All the equations but one are solved with fully explicit methods. Stability of the solution is ensured by using a time step which satisfies the Courant criterion in all nodes. The compressible calculation in the purely liquid sodium slugs in PLUTO 1 always requires the smallest time step and it is also very much the same for all times. Therefore, no automatic time step control is necessary in PLUTO 1 and a constant time step which initially satisfies the Courant criterion in the liquid sodium slugs is being used in PLUTO 1. In PLUTO 2, however, with its incompressible liquid sodium slugs, an automatic time step control is employed.

During the development of the PLUTO 1 and PLUTO 2 codes it has been recognized that the explicit solution of the momentum equation for the light sodium/fission-gas mixture can lead to instabilities of the mixture velocity. This is caused by the action of two large but opposite forces (pressure gradient and drag) on the light mixture. A semi-implicit solution of the sodium/fission-gas momentum equation resolved the above-mentioned, stability problem (see, Ref 9, CRBRP-GEFR-00523).

(9) The reference documents have been or will be made available for inspection and copying.

(10) The Applicants development work has been identified in Appendix A to CRBRP-3, Vol. 1.

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

QUESTIONS II (General)

Request for the following information is based on our concerns with respect to Validations (iii) and (iv) above. In the Applicant's answers to the generic questions (b) and (c) below, the Applicant is requested to be responsive to these concerns.

With respect to each statement, assertion or assumption (from Section F6.2 of the PSAR) identified below, please provide the following information (unless noted otherwise). (NOTE: the following numbered Interrogatories are identified by the page and/or paragraph number from the PSAR in parentheses). [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) Identify, by name, title and affiliation the primary Applicant employee(s) or consultant(s) that has the expert knowledge required to support the statement, assertion, or assumption.

b) Describe in detail the supporting evidence for the statement, assertion, or assumption and where appropriate the rationale for the approach taken.

c) Provide any additional information requested following each statement, assertion, or assumption.

d) To the extent that any answers to the above questions are based on referenced material, please supply the references.

e) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicants' answer. Identify such research or work. 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 II (General)

The following responses are identical for all interrogatories except where supplementary information is provided in responses II-1 through II-69 below.

(a) See the attached affidavits.

(b) and (c) See responses numbered 1-69 below.

(d) The reference documents, except as otherwise noted hereinabove, have been or will be made available for inspection and copying.

(e) The Applicants' program of R&D is identified in Section 1.5 of the PSAR. Additional R&D work has been identified in CRBRP-3, Volume 1, Appendix A for the SMBDB area and in CRBRP-3, Volume 2, Appendix A for the TMBDB area.

(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.

NOTE: Questions II-1 through II-3 pertain to SAS3A Introduction.

QUESTION II-1

(P6.2-6, par. 2) The assumption that the core can be adequately represented by ten channels each containing one pin.

ANSWER II-1 (b) and (c)

The SAS3D code accommodates up to 34 channels while SAS3A is limited to ten channels. The core subassemblies are assigned to a group of subassemblies with similar neutronic, thermal, and hydraulic characteristics. Technical judgment is necessary in making the selections such that each member of a group may be expected to respond to a transient event in a similar way. Each such group of subassemblies is assigned a "channel number" and given properties representative of all subassemblies in the group. This procedure is standard engineering practice and is commonly used in analysis of multiple parallel channel flow systems. For the effect of varying the number of channels on a similar system see: Ref. 4 of CRBRP-GEFR-00523, and "Multi-channel Grouping Techniques for Conducting Reactor Safety Studies," A. E. Waltar, N. P. Wilburn; ANS Transactions, Vol. 22, November 1975, page 375.

QUESTION II-2

(P6.2-6, par. 2) The assumption that the solution can be adequately represented by point kinetics equations with reactivity feedback obtained by summing over stationary fuel worth curves.

ANSWER II-2 (b) and (c)

The use of a point kinetics model with fuel displacement feedback obtained by summing over fuel worth tables is judged adequate, so long as small, local displacements are considered. Gross relocation of fuel in large segments of the core can be addressed by recomputation of the fuel worth tables with a multi-group diffusion code when such recomputation is judged

necessary. See answer to question II(1)(a)-(c) in the Sixth Set of Interrogatories to the Applicant (p. AA-138).

QUESTION II-3

(F6.2-6, par. 3) The assumption that the iteration algorithm linking the SAS3A subroutines adequately represent the time sequence of events during a CDA. In this regard it is noted in para. 5 that, "At the present time, the cladding and fuel motion modules cannot operate simultaneously in the same channel with the FCI model, etc.

ANSWER II-3 (b) and (c)

The two distinct modes of failure identified by the terms "Slumping" and "Fuel-Coolant Interaction" are the mutually exclusive extremes of a continuous spectrum of failure modes. There is an area between these extremes where it is recognized that neither of these models precisely predicts the physical phenomena. To handle this area with present analysis tools requires technical judgment to ensure that limiting conditions for the available subroutines are applied to bound the resulting energy release. SAS3A and SAS3D are sufficiently flexible to control selection of the appropriate model or introduction of results from external parallel calculations, as necessary to assure that the resulting energy release is bounded.

NOTE: Questions II-4 through II-8 pertain to the Fuel Pin TOP Failure Model.

QUESTION II-4

(F6.2-7, par. 2) The assumption that the deterministic "burst" model suggested by Stuart and formulated for SAS3A by Smith is an adequate representation of fuel pin failure phenomena.

ANSWER II-4 (b) and (c)

The deterministic burst model suggested by Stuart and formulated for SAS3A and SAS3D by Smith is a mathematical model of fuel pin failure developed from analysis of TREAT transient overpower experiments. The behavior of many of the CRBR mixed oxide fuel pins during the hypothetical transient overpower accidents is generically similar to fuel pin behavior during the transient overpower experiments from which the mechanistic burst model was developed. Uncertainties which may exist in the fuel pin failure model are accounted for by performing parametric calculations in which pin failures are conservatively forced at the core midplane. The applicants are currently evaluating preliminary results from the Sodium Loop Safety Facility W-2 test as they may be applicable to the hypothetical TOP accident analysis.

QUESTION II-5

(F6.2-7, Par. 2) The Statement "The slope of the cladding strength as a function of temperature significantly influences the degree of bias in pin failure toward the upper part of the pin."

(c) What is meant by "significantly"? How is pin failure influenced by inhomogenieties in the fuel and cladding, e.g., pin hole leaks, fabrication errors, corrosion, swelling, fuel-cladding gap conductance, migration, cracking, and similar phenomena?

ANSWER II-5 (b) and (c)

(b) The clad strength vs clad temperature curves shown in the figure on page 3-43 of CRBRP-GEFR-00103 exhibit a change in slope at about 600°C. Beyond 600°C the steeper slope of the curve indicates that clad failure is more sensitive to increases in temperature than to increases in stress. Temperatures exceeding 600°C are attained in the upper part of the pin during transient overpower accidents as analyzed in CRBRP-GEFR-00523.

Therefore, failure is expected to occur in the upper part of the pin due to the clad axial temperature distribution during the transient, the clad loading mechanism, and the nature of the clad failure property.

(c) The adjective "significantly" in this context means that beyond 600°C, clad failure is much more sensitive to increases in temperature than to increases in stress as quantified by the clad failure curves on page 3-43 of CRBRP-GEFR-00103.

Inhomogenieties in the fuel and cladding are accounted for by performing parametric calculations in which pin failures are conservatively forced to occur at the core axial midplane.

QUESTION II-6

(F6.2-7, par. 3) For cladding strength the use measured data from 20% CW 316SS specimens irradiated in EBR II.

(c) What is the basis for extrapolating the EBR II data to the CRBR environment?

ANSWER II-6 (b) and (c)

The data from Ref. [illegible] in CRBRP-GEFR-00103 has been used as the basis of data for fast flux irradiated cladding prototypic of that to be used in CRBR. CRBR fuel pin cladding operating parameters outside the range of the EBR-II data are accounted for by performing parametric calculations in which pins are conservatively forced to fail at the core axial midplane.

QUESTION II-7

(F6.2-8, par. 1) The use of the Gruber's simplified correlation to FRAS.

ANSWER II-7 (b) and (c)

It is shown in Reference 13 in CRBRP-GEFR-00523 that Gruber's simplified correlation is an adequate representation of FRAS results.

QUESTION II-8

(P6.2-8, par. 3) The fuel-cladding relative heating rates are dominated by the fuel-clad gap conductance, fuel thermal conductivity, and reactor power.

ANSWER II-8 (b) and (c)

The fuel-cladding relative heating rates during reactor transients are calculated in the SAS3A and SAS3D code by using coupled heat transfer models of the fuel, gap, clad, coolant and reactor structure and values of physical properties and heat transfer coefficients based on experiments using prototypic CRBR materials.

The statement that the fuel-cladding relative heating rates are dominated by the fuel-clad gap conductance, fuel thermal conductivity, and reactor power is a qualitative statement which means that heat transfer calculations show greater sensitivity to these parameters than to the other heat transfer parameters in the models.

NOTE: Questions II-9 through II-20 pertain to the SAS/FCI Summary.

QUESTION II-9

(P6.2-8, par. 3) The use of Ross-Stoute heat transport model.

ANSWER II-9 (b) and (c)

The Ross-Stoute gap conductance type model for application to fast reactor fuel rods is given by Dutt (D. S. Dutt, R. B. Baker, and R. J. Jackson, "Interim Fuel Thermal Performance Models for LIFE-2," W/FFTF 73518, January 15, 1973). The contact conductance algorithm of this correlation was adopted for use in the CRBR HCDA analysis as the best available formulation of this component of fuel-cladding heat transfer.

QUESTION II-10

(F6.2-8, par. 3) The assumption that cladding hardness is inversely proportional to the cladding yield strength.

ANSWER II-10 (b) and (c)

The word "inversely" was a misprint on page F6.2-8. Appendix F has been deleted from the application. It is correctly stated in CRBRP-GEFR-00103 that the cladding hardness is assumed proportional to cladding strength.

QUESTION II-11

(F6.2-8, par. 4) The assumption that a one-dimensional model using Lagrangian cells can adequately represent interaction of fuel and coolant.

ANSWER II-11 (b) and (c)

In SAS/FCI, it is assumed that the interaction between fuel and coolant is occurring uniformly throughout the interaction zone. This is an obvious simplifying assumption which makes it possible to treat the interaction zone without resorting to one-dimensional compressible hydrodynamics treatment. It is because of this simplifying assumption and others made within SAS/FCI that auxiliary calculations were made with the PLUTO 1 and PLUTO 2 (Ref. 25 in CRBRP-GEFR-00523) codes in order to more accurately

determine the voiding rates and fuel relocation dynamics associated with these pin failures into non-voided subassemblies. In PLUTO 1 it is indeed assumed that a one-dimensional model using Lagrangian cells can adequately represent interaction of fuel and coolant. PLUTO 2 also uses a one-dimensional treatment but it utilizes the Eulerian approach. As pointed out in the above reference, the one-dimensional treatment is adequate, based on the long length in the axial direction of the coolant channel in comparison to the relatively small distances between adjacent pins.

QUESTION II-12

(F6.2-8, par. 4) (Generic answers (a) and (b) are not required.)

(c) Define more fully what is meant by rip length and how it is determined.

ANSWER II-12 (c)

Within the context of the SAS/FCA model, the rip length is that length of the pin over which the cladding is initially assumed to fail. The rip length determines the length of the initial interaction zone and defines the length over which the fuel-fission gas mixture continues to be ejected from the failed pin into the coolant channel. The location of the rip is determined by centering the rip length on the center of the SAS3A or SAS3D axial node at which the failure criterion is exceeded by the greatest amount at the time step when the failure criterion is first exceeded at one or more nodes in the channel. A discussion of the rip length used in the studies is documented on pages 4-16, 7-27 and 7-28 of CRBRP-GEFR-00103.

QUESTION II-13

(F6.2-8, par. 5) (Generic answers (a) and (b) are not required.)

(c) Please clarify this statement: "Three failure groups based upon time from initial failure reflects for the incoherence of fuel-pin failure within fuel subassemblies."

ANSWER II-13 (c)

There was a typographical error in the last sentence of par. 5 on p. F6.2-9. Appendix F has been deleted from the application. CRBRP-GEFR-00103 correctly states that "Moreover, three failure groups based upon time from initial failure reflects the incoherence of fuel pin failure within fuel assemblies." On p. 7 of Ref. 9 in CRBRP-GEFR-00523, the manner in which these failure groups are treated is discussed in more detail.

QUESTION II-14

(F6.2-9, par. 1) The assumption that the Cho-Wright model is an adequate representation of fuel to sodium heat transfers.

(c) Identify all alternative models (including MFCI models) that have been considered and rejected. What is the basis for rejecting these other models? (For each model rejected explain in detail.)

ANSWER II-14 (b) and (c)

A summary of models to calculate fuel to sodium heat transfer is given in the following reference: Hans K. Fauske, "CSNI Meeting on Fuel-Coolant Interactions," Nuclear Safety, 16, pp. 436-442, 1975. The Cho-Wright model is representative of the state-of-the-art; and since it is a parametric model, a wide variety of situations can be simulated by variation of the particle diameter, the mixing time constant, and the other parametric additions used in SAS/FCI. This model is adequate for simulating the mild

interactions which have been observed in experiments, as well as more hypothetical situations. A further discussion of the comparison between analytical models and the experimental data base on fuel pin postfailure transient behavior is given in the following references:

(1) H.U. Wider and A.E. Wright, "Analysis of a Sodium Reentry Event in the H4 TREAT Test," TANSAO 22, p. 428, 1975.
(2) Ref. E-3 in CRBRP-GEFR-00523.
(3) Ref. E-4 in CRBRP-GEFR-00523.

QUESTION II-15

(F6.2-9, par. 1) The assumed values for the thermodynamic properties of fuel, cladding and sodium.

ANSWER II-15 (b) and (c)

The sodium thermodynamic properties for both single-phase and two-phase sodium used in SAS/FCI are given in Chapter 6 of Ref. 9 in CRBRP-GEFR-00523. The thermodynamic properties of cladding and fuel used in SAS/FCI are identical to those in the remainder of the SAS3A and SAS3D code. These are summarized in Section 4.0 of CRBRP-GEFR-00103.

QUESTION II-16

(F6.2-9, par. 2) The assumption that the sodium void reactivity can be determined adequately from the average, smeared sodium density of the interaction zone.

ANSWER II-16 (b) and (c)

One purpose of doing auxiliary PLUTO 1 and PLUTO 2 calculations was to check the SAS/FCI computed material relocation reactivities with a more detailed model. The PLUTO 1 and PLUTO 2 calculations indicated that the

magnitude of the sodium void reactivity following pin failure in the hypothetical accidents considered is not a dominating factor in determining the hypothetical accident progression in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 since the fuel motion reactivity quickly dominates the progression of the hypothetical accident after initiation of a fuel-coolant interaction.

QUESTION II-17

(F6.2-10, par. 1) (Generic answers (a) and (b) are not required.)

(c) Please explain what is meant by "Reactivity feedbacks are determined from the projection of the cell lengths and masses onto the normal SAS3A axial node lengths."

ANSWER II-17 (c)

In Section 7.2.2 of Ref. 9 in CRBRP-GEFR-00523 the complete fuel motion reactivity feedback model, including the statement to be clarified by this response, is described in detail.

QUESTION II-18

(F6.2-10, par. 2) (Generic answers (a) and (b) are not required.)

(c) What is meant by "input default options"?

ANSWER II-18 (c)

The original SAS/FCI model, as documented in Ref. 9 of CRBRP-GEFR-00523, required that a number of parameters it used be supplied by the user as input to the SAS3A and SAS3D code. After gaining experience with the model, it was determined that it would be more appropriate to actually calculate the values assigned to some of these parameters based on conditions within the SAS3A or SAS3D channel at the time that each parameter was

actually required. The parameters discussed in par. 4 on p. 3-5 of CRBRP-GEFR-00103 are those parameters for which provision was made within the SAS/FCI subroutines of SAS3A and SAS3D to calculate them for purposes of the cases discussed in CRBRP-GEFR-00103 and CRBRP-GEFR-00523.

QUESTION II-19

(F6.2-10, par. 2) The assumption that the rip area is the cross-sectional area of the internal pin cavity.

ANSWER II-19 (b) and (c)

Once the fuel and fission gas which are located immediately behind the rip in the cladding (the SAS/FCI reservoir) are deposited into the interaction zone at the time of failure, any additional fuel and fission gas which are ejected through the rip must come from the central cavity in the pin. This cavity is quite long compared to its cross-sectional area. Thus the rate at which the fuel fission gas mixture can be transported to the rip and into the interaction zone is controlled by the cross-sectional area of the cavity. Thus, it is appropriate to utilize this area as the rip area in the Bernoulli equation which is used to compute the ejection rate.

QUESTION II-20

(F6.2-10, par. 2) The assumption that the flow is adequately represented by a one-dimensional Bernoulli equation for unsteady flow.

ANSWER II-20 (b) and (c)

The one-dimensional Bernoulli equation as used in SAS/FCI is a parametric algorithm to calculate fuel ejection from the central pin cavity into the coolant channel. The model is described in Ref. 9 in CRBRP-GEFR-00523. When the fuel pin cavity can greatly change in size following fuel pin failure, e.g., for a core midplane failure in a prompt-critical situation,

the model can significantly overpredict the mass-flow-rate of fuel within the pin to the failure location. The model utilized is judged to enhance the conservative nature of the analysis. A less conservative treatment would result in a decrease in the calculated accident-energetics in many of these situations.

NOTE: Questions II-21 through II-27 pertain to the PLUTO Summary.

QUESTION II-21

(F6.2-11, par. 3) The fuel-fission gas flow in the molten pin region is treated as homogeneous, compressible, one-dimensional flow with a non-uniform flow cross section.

ANSWER II-21 (b) and (c)

The assumption of a one-dimensional flow is adequate because the molten pin cavities are several tens of cm's long and only a few tenths of a cm in diameter. The assumption of homogeneous compressible two-phase flow (i.e., no slip between fuel and gas) is supported by H. J. Willenberg and A. Padilla, Jr., "Analysis of Transient Compressible Two-Phase Flow with Heat and Mass Sources Using the Method of Characteristics," Computational Methods in Nuclear Engineering, CONF-750413, Vol. 1, p. II-107, 1975.

QUESTION II-22

(F6.2-11, par. 3) The representation of all failed pins in a given sub-assembly by a single pin model.

ANSWER II-22 (b) and (c)

The assumption of treating all fuel pins in a given subassembly with a single pin model is made necessary by computational limitations. This

approach is consistant with the one dimensional treatments of fuel and coolant dynamics used in SAS3A, SAS3D, PLUTO 1, AND PLUTO 2. Use of one pin per channel "which may represent several subassemblies" results in a coherent treatment of fuel coolant interactions, sodium voiding, intra-pin fuel motion, and fuel sweepout. For short times after failure, the important effects governing reactivity are intra-pin motion and sodium voiding. The single pin treatment will tend to accentuate these effects and, if positive feedback is predicted such as would be the case for mid-plane failure, lead to a conservatively high positive feedback. It is the short-time positive feedback which would tend to produce an energetic transient. In a longer time, fuel sweepout due to hydraulic forces becomes important. The coherent single pin treatment may over-estimate the hydraulic forces available for sweepout. Overall, the single pin treatment will emphasize positive reactivity feedback effects in the short-time after failure, and yield a conservative result.

QUESTION II-23

(F6.2-11, par. 4) The axial motion of material in the coolant channel is treated as two-component slip flow.

ANSWER II-23 (b) and (c)

The density, velocity, and internal energy changes of both components are calculated by solving a set of two mass, two momentum, two energy equations and an equation of state. The two momentum equations contain the interactive (or drag) forces which couple the flows of the two components (see Ref. 13 in CRBRP-GEFR-00103).

QUESTION II-24

(F6.2-11, par. 4) The liquid sodium, or the mixture of liquid coolant, vaporized coolant, and fission gas (Na/FG) is regarded as one component and its flow is modeled with compressible Lagrangian hydrodynamics.

ANSWER II-24 (b) and (c)

Slip between liquid sodium and sodium vapor or fission gas can be disregarded because the flow under consideration is mostly in a slug flow regime. The flow is modeled using Lagrangian hydrodynamics in PLUTO 1 and Eulerian hydrodynamics in PLUTO 2.

QUESTION II-25

(F6.2-11, par. 4) The other component is the fuel, which is assumed to be in the form of particles. The motion of the fuel particles is calculated by solving the momentum equation for representative (or "master") particles.

ANSWER II-25 (b) and (c)

The assumption that the fuel is in the form of particles can be justified, as long as liquid sodium is close to the fuel. If the coolant channel is voided, the assumption of annular or bubbly fuel flow, which is made in PLUTO 2, is more appropriate.

QUESTION II-26

(F6.2-11, par. 5) Although sodium vapor condensation on cold cladding is accounted for in PLUTO, the condensate is currently assumed not to adhere to the cold wall but rather to be torn off instantaneously and mixed with the sodium in the coolant channel at the same axial location.

ANSWER II-26 (b) and (c)

Hot fuel particles moving through narrow coolant channels should quickly vaporize a liquid sodium film. Moreover, the gas and vapor streaming in the coolant channels will lead to flooding of the liquid sodium film since

the physical vapor and gas velocities calculated by PLUTO 1 and PLUTO 2 far exceed the necessary flooding velocity of about 15 ft/sec. Flooding of the liquid film leads to an unstable sodium film interface and a significant increase in film to vapor frictional coupling. Hence, any sodium film will travel in the direction of the moving vapor with good interfacial frictional coupling, and the PLUTO 1 treatment is reasonable. PLUTO 2 incorporates the treatment of a liquid sodium film which can be evaporated or entrained by high gas velocities or it can be torn off by fuel flows.

QUESTION II-27

(P6.2-11, par. 6) For the fission-gas temperatures, mass-weight averages between liquid sodium and fuel are used.

ANSWER II-27 (b) and (c)

If there is much sodium at a certain location the fission gas temperature will be close to the sodium temperature and if there is much fuel, the fission-gas temperature will be close to the fuel temperature according to PLUTO 1 and PLUTO 2.

NOTE: Questions II-28 through II-47 pertain to the SASBLOK Summary.

QUESTION II-28

(P6.2-12 par. 1) (Generic answers (a) and (b) are not required.)

(c) Explain fully why neither SAS/FCI nor PLUTO is capable of treating fuel blockages, or of continuing the calculation beyond FCI initiation for more than a few hundred milliseconds.

ANSWER II-28 (c)

Neither SAS/FCI nor PLUTO 1 consider "freezing" of molten fuel to cold surfaces. Both models assume a constant, average particle size which may be a liquid drop or a solid particle. PLUTO 2 incorporates a freezing model (Ref. 25, CRBRP-GEFR-00523).

SAS/FCI will not continue calculations beyond a few hundred milliseconds because of coding limitations within the model. As the transient progresses, vapor bubbles may form in the channel below the FCI zone and may attempt to merge with the zone. The present SAS/FCI does not have the coded logic and numerical models to treat this occurrence. The calculations are continued with the SASBLOK option in the SAS3A and SAS3D code. PLUTO 1 and PLUTO 2 are not limited to running time of a few hundred milliseconds.

The PLUTO 1 and PLUTO 2 codes can only be run separately from SAS3A and SAS3D using one of them to generate input for PLUTO 1 and PLUTO 2.

QUESTION II-29

(F6.2-12, par. 1) The assumption that the hydraulic effect on assembly flow can be adequately represented from packed particle bed correlations and represented as a local hydraulic loss.

ANSWER II-29 (b) and (c)

The flow resistance characteristics of the packed particle bed used in the blockage model were taken from experimental data (Ref. A-1 in CRBRP-GEFR-00103) in which the material used in the test was a packed bed of UO_2 having a particle-size distribution similar to those found to result from molten-fuel-sodium contact. Results of these tests showed good agreement with the correlation used in the SASBLOK analysis. Representing the packed bed as a local hydraulic loss is a standard technique used in flow testing where complicated pressure losses are represented by an equivalent local

hydraulic loss coefficient. The use of these coefficients in the SASBLOK analysis produced an equivalent effect upon the flow within the core.

QUESTION II-30

(F6.2-12, par. 1) Power generation in the damaged region is proportionately reduced by fuel loss.

ANSWER II-30 (b) and (c)

Pin failure results in fuel loss from the pin. The power produced in the damaged channel must be reduced in proportion to the amount of fuel ejected from the pin.

For numerical simplicity, the power level is reduced uniformly in the core region of the modeled channel although fuel would actually be removed preferentially from the higher worth core midplane region. This simplified model underestimates the power reduction and is therefore adequate for the purpose of analysis.

QUESTION II-31

(F6.2-12, par. 1) (Generic answers (a) and (b) are not required.)

(c) Explain fully what is meant by quasi-steady formulation.

ANSWER II-31 (c)

The term "quasi-steady" was used in reference to the temperature distribution within the blockage and the coolant temperature increase due to the heat generation within the blockage. To compute these values, a constant heat generation rate, taken at the time of initial formulation of the blockage, and steady state formulas for the temperature distribution within the blockage were used. The short duration of the flow transient in the

channel relative to the change in the heat generation rate justified the use of the quasi-steady formulation.

QUESTION II-32

(F6.2-12, par. 1) (Generic answers (a) and (b) are not required.)

(c) Is it correct to assume that SASBLOK is not utilized in analysis of LOF CDA events?

ANSWER II-32 (c)

Yes, SASBLOK was not used for analysis of any LOF-HCDA event reported in CRBRP-GEFR-00103 or in CRBRP-GEFR-00523.

QUESTION II-33

(F6.2-12, par. 2) (Generic answers (a) and (b) are not required.)

(c) Explain fully the purpose and operation of the "tabular fission gas release module".

ANSWER II-33 (c)

The tabular fission gas release model is an option built into the SAS codes which permits construction of user specified bubble growth and collapse within a channel. Time dependent bubble interface locations must be specified. Its purpose is to allow an explicit investigation of the effects of voiding at specified locations and times during the course of an accident. The input bubble interface locations and pressures are supplied by auxiliary calculations or estimates. A complete description of the option is given beginning on page 79 of Ref. 7 in CRBRP-GEFR-00523.

QUESTION II-34

(F6.2-12, par. 2) (Generic answers (a) and (b) are not required.)

(c) Explain fully the "programmable reactivity option."

ANSWER II-34 (c)

The programmable reactivity option in SAS is a table look-up operation to provide user specified reactivity vs time. Input tables of reactivity and time are provided by the code user. The reactor net reactivity is the programmed or driving reactivity plus the reactivity components calculated by the code. Interpolation is performed in the code to find the programmed reactivity at any particular time.

QUESTION II-35

(F6.2-12, Par. 2) The exit loss coefficient is increased to the blocked value.

(c) What does this mean?

ANSWER II-35 (b) and (c)

During a SASBLOK calculation, the value of the channel exit loss coefficient is increased linearly from the initial input value to the calculated blocked value over a period of time equal to twice the fuel ejection time, beginning at the time of initial fuel ejection. The purpose of the ramp change in exit loss coefficient is to represent the buildup of the blockage over a period of time. Twice the ejection time is judged to be a reasonable estimate of the build-up time.

QUESTION II-36

(P6.2-12, par. 2) Explain fully how the locus of coolable blockage configurations for each degree of hydraulic disturbance is generated and justify each assumption.

ANSWER II-36 (b) and (c)

The approach and calculations for determining the locus of coolable blockage configurations were presented in detail for the EOEC Reactivity Insertion Base Case (Section 6.1.1.2 of CRBRP-GEFR-00103). The assumptions have been justified in Appendix A and Section 6.1.1.2 of CRBRP-GEFR-00103.

QUESTION II-37

(P6.2-12, par. 2) The assumption of complete core outlet blockage for an unstable two-phase solution.

(c) What is meant by unstable two phase solution?

ANSWER II-37 (b) and (c)

An assumption of complete outlet blockage was made in order to result in an analysis which is (a) conservative and (b) numerically stable. This assumption implies a complete subassembly meltdown and reactivity estimates of fuel-slumping are performed as described in the response to Question 38.

QUESTION II-38

(P6.2-12, par. 2) (Generic answers (a) and (b) are not required.)

(c) How are reactivity estimates on the effects of fuel slumping performed?

ANSWER II-38 (c)

The reactivity estimates on the effects of fuel slumping were performed using diffusion theory models. Values of K_{eff} were computed for various positions of the fuel in the coolant channels, starting from the point at which it formed the blockage, to the point of maximum reactivity change after it moved downward. The core model and cross sections used were the same as for the other neutronic calculations that were done for CRBRP-GEFR-00103. These are described in Section 5 of CRBRP-GEFR-00103.

QUESTION II-39

(F6.2-12, par. 2) (Generic answers (a) and (b) are not required.)

(c) How are the "various analysis path" selected?

ANSWER II-39 (b) and (c)

The analysis path determination is based on whether reactivity estimates of relocating fuel in the uncoolable assemblies would lead to a critical or supercritical reactor state. These two analysis paths are indicated in the lower left hand portion of the SASBLOK flow chart in Figure 3-7 in CRBRP-GEFR-00103.

QUESTION II-40

(F6.2-13, par. 2) The assumption that porous blockage is adequately represented by a mean particle size of 420μ.

ANSWER II-40 (b) and (c)

The mean particle size is a parameter that characterizes the particle distribution used in the packed bed analysis. The 420 value used in the SASBLOK analysis was taken from Ref. A-1 in Appendix A of CRBRP-GEFR-00103.

This value characterized the particle distribution used in flow tests of packed beds. The 420μ value was calculated by methods recommended by the correlation which showed good agreement with the test data. The particle distribution used in the tests was similar to those found to result from molten-fuel-sodium contact. Therefore, the porous blockage is adequately represented by the mean particle size of 420μ.

QUESTION II-41

(F6.2-13, par. 2) The assumption of a friction factor that correlates with tests involving water through uranium dioxide.

ANSWER II-41 (b) and (c)

The friction factor is a dimensionless parameter which depends upon relative roughness and the Reynolds number. The correlations of test data with friction factor can be seen in Appendix C of Ref. A-1 in Appendix A of CRBRP-GEFR-00103 where pressure drop versus velocity has been plotted for both the test data and the Leva correlation using a mean particle diameter of 420μ.

QUESTION II-42

(F6.2-13, par. 2) (Generic answers (a) and (b) are not required.)

(c) Explain fully what is meant by the second sentence beginning with ". . .since the particle bed correlation. . ."

ANSWER II-42 (b) and (c)

The particle distribution used in the packed bed test (Ref. A-1 in Appendix A of CRBRP-GEFR-00103) was similar to those found to result from molten-fuel-sodium contact. This similarity forms the basis for using the correlation in the SASBLOK analysis.

QUESTION II-43

(F6.2-13, par. 2) The use of the Leva correlation.

ANSWER II-43 (b) and (c)

The Leva correlation was found to be in good agreement with test data (Ref. A-1 in Appendix of CRBRP-GEFR-00103) and this agreement provides a basis for its application in the SASBLOK analysis.

QUESTION II-44

(F6.2-13, par. 3) The assumption that flow blockage can be adequately modeled by the average loss coefficient representation.

ANSWER II-44 (b) and (c)

The use of an average loss coefficient to simulate a flow blockage is a standard engineering practice in flow modeling where overall flow resistance modeling is the objective. Complicated blockages can be modeled with simple loss coefficients which produce equivalent flow in the system. The SASBLOK calculations included a parametric variation of the loss coefficient so that the range of expected porous blockages is adequately covered.

QUESTION II-45

(F6.2-14, par. 4) The assumption that the blockages form as a contiguous mass of material located in the fission gas plenum region where either the original geometry has been destroyed or in the coolant channel where the geometry has been maintained.

ANSWER II-45 (b) and (c)

The blockage was assumed to form a contiguous mass because this would represent the worst case analysis, i.e., it would be the most difficult condition to maintain in a coolable and stable condition. On the other hand, distributed blockages could be more easily cooled due to the presence of greater coolant access to blockages. The location of the blockage was selected as one of several locations where agglomeration of the material could occur, and the cooler regions of the core provide such a location.

QUESTION II-46

(F6.2-16, par. 3) (Generic answers (a) and (b) are not required).

(c) Explain fully the elementary methods that were applied to the gravity effects and the generalized discussion of the coolant effects; with respect to the elementary methods, is this simply the "falling film analysis" and "falling of liquid drops" analysis described in the following paragraphs?

ANSWER II-46 (c)

The elementary methods refer to analyses of "the falling film" and "falling of liquid drops" which are described in Appendix A of CRBRP-GEFR-00103. Both are described in some detail in pages A-8 and A-9 of Appendix A and further details can be found in Reference A-6, page A-15 of CRBRP-GEFR-00103.

QUESTION II-47

(F6.2-18, par. 1) (Generic answers (a) and (b) are not required).

(c) Identify the "considerable uncertainties" in molten fuel penetration methods.

ANSWER II-47 (c)

The response to this interrogatory is identical to the response number II-27 of the Third Set of Interrogatories (p. AA-75).

NOTE: Questions II-48 through II-59 pertain to the Sodium Voiding Model Summary.

QUESTION II-48

(P6.2-18, par. 3) (Generic answers (a) and (b) are not required.)

(c) What is meant by "axial interface areas"?

ANSWER II-48 (c)

The axial interface area is the cross-sectional area of the coolant channel. Section IIB of Ref. 7 in CRBRP-GEFR-00523 provides a detailed description of most of the sodium boiling model in SAS3A, and Ref. 6 in CRBRP-GEFR-00523 lists the SAS3A additions to the SAS2A boiling model. The physics represented in the SAS3D boiling model is identical to that in SAS3A.

QUESTION II-49

(P6.2-18, par. 3) The sodium vapor and liquid film are assumed to be at saturation conditions determined by channel pressure as opposed to non-equilibrium super heat conditions.

ANSWER II-49 (b) and (c)

Ref. 71 of CRBRP-GEFR-00523, pp. 20-22, describes the supporting evidence for believing the liquid superheat in a reactor will be small, and have little effect on the voiding process.

The thin film of sodium left on the clad in a voided region is in contact with the vapor bubble and the film will vaporize before it superheats.

The vapor will only superheat if it passes over hot, dried-out clad. Before the liquid film on the clad dries out, heat removal due to vaporizing the film will prevent the clad surface temperature from rising much above the sodium saturation temperature. After extensive voiding, extensive film dry-out, and significant heating of the dried-out clad past the sodium saturation temperature, some superheating will be limited by the relatively poor heat transfer coefficient between hot, dried-out clad and cooler sodium vapor. By the time that any appreciable superheating of the vapor might occur in the LOF cases discussed in CRBRP-GEFR-00103 and CRBRP-GEFR-00523, the active core region of the subassembly is voided and dried out; and it remains voided, with little or no heat removal from the fuel pins to the voided coolant channel. At this point, many of the details of the voiding in the subassembly, including any superheating of the vapor, will be largely unimportant, since any coolant voiding reactivity insertion has already occurred, and the sodium will no longer remove much heat from the voided, dried-out core.

QUESTION II-50

(P6.2-18, par. 4) All the assumptions concerning reentry of lower liquid slug discussed in the paragraph.

(c) Describe what happens when the channel becomes blocked.

ANSWER II-50 (b) and (c)

Reentry of the lower liquid slug is described in Section IIB of Ref 7 in CRBRP-GEFR-00523. When a partial blockage is formed in the coolant channel, the hydraulic diameter and coolant flow area are reduced. This leads to an increase in the friction pressure drop due to any vapor streaming past the blockage, and causes the vapor pressure to buildup below the blockage, if

the vapor is coming from below, or above the blockage, if the vapor is streaming from above the blockage. The increase in vapor pressure usually stops liquid slugs before they get to the blockage. If a liquid slug does pass a partial blockage, then the friction pressure drop in the liquid slug is increased due to the reduction in hydraulic diameter.

QUESTION II-51

(F6.2-18, par. 5) (Generic answers (a) and (b) are not required.)

(c) Explain fully what is meant by "reasonably good qualitative agreement." How did the results compare quantitatively?

ANSWER II-51 (c)

For a quantitative comparision of the results, see Reference 26 in CRBRP-GEFR-00103.

QUESTION II-52

(F6.2-18, par. 5) (Generic answers (a) and (b) are not required.)

(c) Explain fully how ". . .the experimentally observed dryout mechanism is considered."

ANSWER II-52 (c)

A static film dryout model was used for the calculations reported in Ref. 26 in CRBRP-GEFR-00103 for the Karlsruhe experiments of Peppler. When the calculations were repeated using the film motion model described in Ref. 71 of CRBRP-GEFR-00523, good quantitative agreement was achieved between the calculated and experimentally observed dryout times (see G. Hoeppner, F. E. Dunn, and T. J. Heames, "The SAS3A Sodium Boiling Model and Its Experimental Basis," Trans. Am. Nucl. Soc., 20, 519, April 1975). The film

motion model was used for the CRBR calculations described in CRBRP-GEFR-00103, CRBRP-GEFR-00523 and Ref. 4 in CRBRP-GEFR-00523.

QUESTION II-53

(F6.2-19, par. 1) (Generic answers (a) and (b) are not required.)

(c) Explain fully the comparison of the film dryout models results with experimental observed dryout time.

ANSWER II-53 (c)

See Answer II-52(c).

QUESTION II-54

(F6.2-19, par. 2) ". . .full assembly voiding would be expected somewhat earlier than the average pin model predicts. . ."

ANSWER II-54 (b) and (c)

The radial growth of voiding within a subassembly, and the applicability of the average pin model are discussed in detail in Section 4.2 of Ref. 71 in CRBRP-GEFR-00523.

QUESTION II-55

(F6.2-19, par. 2) "However, the difference would not be expected to be great."

ANSWER II-55 (b) and (c)

See Answer II-54(b) and (c).

QUESTION II-56

(P6.2-19, par. 3) "The present SAS3A voiding model does not include a meaningful consistant treatment of gas release into a voided channel."

ANSWER II-56 (b) and (c)

The predictions of the boiling and plenum gas release models in SAS3A are used only for three main purposes in the CRBR analysis. These purposes are: (1) the calculation of the rate at which voiding reactivity is inserted, (2) the calculation of heat removal from pins in voided regions, determining the time of melting and rate of melting of clad and fuel, and (3) determining the impact that the presence of sodium vapor or sodium liquid could have on the relocation of molten clad or fuel in voided regions. Because of the strength vs. temperature characteristics of the clad, the cladding will not fail and release plenum gas during a hypothetical LOF accident until the sodium has boiled extensively, the film on the clad has dried out, and the clad temperature has risen considerably above the sodium saturation temperature. At this point, failure of the pins and release of plenum gas would have some influence on the voiding profile; and it could affect L-series test voiding measurements; but it would have very little impact on the voiding and voiding reactivity insertion because: (1) the core region of the subassembly would already be voided before gas release, so the coolant voiding reactivity insertion would have already occurred, (2) the liquid film would already have dried out from the clad in most of the hotter parts of the pins, and no heat removal from these areas would be predicted, whether the plenum gas is released or not.

QUESTION II-57

(F6.2-19, par. 3) However, at the present time, release of plenum gas into a voided channel is expected to have only a slight effect on the overall voiding behavior.

ANSWER II-57 (b) and (c)

See Answer II-56(b) and (c).

QUESTION II-58

(F6.2-19, par. 3) This should not greatly affect the voiding process once full-assembly voiding has occurred and cladding film has dried out or has been stripped off.

ANSWER II-58 (b) and (c)

See Answer II-56(b) and (c).

QUESTION II-59

(F6.2-19, par. 4) . . .this has been suggested to be due to the presence of noncondensible fission gases.

ANSWER II-59 (b) and (c)

See Answer II-56(b) and (c).

NOTE: Questions II-60 through II-62 pertain to the Cladding Relocation Model (CLAZAS) Summary.

QUESTION II-60

(F6.2-19, par. 5) How are molten cladding-coolant interactions similar to molten fuel coolant interactions?

ANSWER II-60 (b) and (c)

CLAZAS was written to analyze the reactivity and thermal effects of cladding relocation before oxide fuel pin disruption. In this phase of a CRBR hypothetical LOF accident, the molten cladding-coolant interaction question is not pertinent, since a long length of hot unmelted cladding and a zone producing large quantities of sodium vapor separate the molten cladding from the nearest liquid sodium.

QUESTION II-61

(F6.2-20, par. 1) The assumption that effects of clad melting and relocation can be adequately modeled with three radial nodes comprising 25%, 70% and 5% of the clad respectively.

ANSWER II-61 (b) and (c)

The clad radial mesh was chosen due to numerical considerations involved in cladding-liquid sodium heat transfer. During cladding melting and relocation, the cladding thermal conductivity and thickness suggest that three nodes are more than adequate, and perhaps even one node would be sufficient. For example, the thermal diffusivity of molten cladding is 0.05 cm^2 sec^{-1}. This leads to a thermal response time for 0.038 cm thick cladding of ~0.03 sec. The time for absorbing the cladding heat of fusion is usually a few tenths of a second, or approximately an order of magnitude greater. Hence, the cladding will melt essentially as a unit and the details of the radial melting profile are unimportant to the analysis.

QUESTION II-62

(P6.2-20, par. 3) Thus, it would appear that CLAZAS overestimates the degree of upward cladding relocation, and that, if the time interval between clad melting and fuel melting is as short as expected for CRBR, very little, if any, net clad motion would result.

ANSWER II-62 (b) and (c)

The basis for this judgment is the R-5 experiment discussed on page 3-28 of CRBRP-GEFR-00103 and Ref. 32 in CRBRP-GEFR-00103.

NOTE: Questions II-63 through II-69 pertain to the SLUMPY Summary.

QUESTION II-63

(P6.2-21, par. 4) (Generic answers (a) and (b) are not required.)

(c) Explain fully what is meant by "detailed initial conditions" supplied to VENUS-II. Describe the limitations of one-dimensional motion.

ANSWER II-63 (c)

If the computer models suggest that a hydrodynamic disassembly calculation should be performed, the SAS3A and SAS3D codes with SLUMPY contain all the information, with respect to material location and material internal energy, that is required to set-up the VENUS-II geometric core representation.

One-dimensional motion refers to the restriction of only allowing axial fuel relocation within the confines of any SAS3A or SAS3D channel. Two different hypothetical transient initiating accident situations must be examined, that of low power and of high power. In the context of this question, low power is defined as a few times nominal reactor operating

levels or lower. Here, one-dimensional motion does not allow for intra-subassembly incoherence effects and hence tends to yield an exaggerated description of material relocation, e.g., any fuel slumping is predicted to be too coherent, and calculations cannot be done on material moving upward at one radial location and downward at another. As the hypothetical accident power level increases, the radial and axial power profiles guarantee that fuel will move coherently away from the locations of peak power. At high power (defined as disassembly power levels or approximately several hundred to more than a thousand times nominal power), the main limitation of axial one-dimensional motion concerns the lack of a description on how the hypothetical accident energetics will be mitigated by fuel expansion in more than one direction. Here axial one-dimensional motion is too conservative and a two-dimensional capability such as VENUS-II is desirable.

QUESTION II-64

(F6.2-21, par. 4) In most SLUMPY calculations, fuel motion is assumed to begin when melting begins in unrestructured fuel.

(c) What is meant by "equiaxed region"?

ANSWER II-64 (b) and (c)

(b) Ref. 8 in CRBRP-GEFR-00103 describes the experimental basis for assuming fuel motion because of fission gas release when melting begins in unrestructured fuel.

(c) Fast reactor fuels operate with close to radially flat power profiles and at a high linear power rating. Hence, a large radial temperature gradient exists inside of the oxide fuel pin. As the fuel temperature rises above approximately 1350 C, the grains of the fuel pellet exhibit grain growth, without preferred orientation. The fuel takes on an equiaxed grain structure. At still higher temperatures, as low as 1550 C at high burnup, migration of the fabricated porosity to the center of the fuel pin occurs resulting in a "columnar grain" region. An examination of some

microstructural effects in fast reactor fuels is given in Ref. 53 in CRBRP-GEFR-00103.

QUESTION II-65

(F6.2-21, par. 4) ". . .and the more recent R- and L-series experiments also seem to verify this latter assumption".

ANSWER II-65 (b) and (c)

The assumption under discussion is modeling of fuel as an intact column during the time following cladding melting but preceding fuel melting. Restructuring during irradiation is known to cause sintering of fuel pellets. Hence, this assumption appears to be reasonable for irradiated fuel. The L-series loss-of-flow tests run with irradiated fuel have been consistent with this assumption.

Figure 11 and Fig. 12 of the following reference: E. W. Barts et al., "Summary and Evaluation Fuel Dynamics Loss-of-Flow Experiments Tests L2, L3, and L4," ANL 75-57, 1975, show several seconds between the time of stainless steel melting indications and the time of axial fuel motion. Later L-series tests, L5, L6, L7 also show results consistent with modeling the fuel column as intact prior to failure associated with fuel melting. See Ref. 31, 32, 33, pg. 11-3, CRBRP-GEFR-00523. Fresh fuel tests such as L2, and the R-series tests (Ref. 23, pg. 11-2, CRBRP-GEFR-00523) do not indicate fuel motion upon cladding melting, but do indicate initiation of fuel collapse once melting occurs in the flow tube which provides the radial restraint for the test section. This loss of radial restraint will not occur in the reactor situation.

QUESTION II-66

(F6.2-21, par. 6) The assumed equation of state composed of six parts.

ANSWER II-66 (b) and (c)

Details of the equation-of-state used in the SLUMPY compressible region are described on pp. 15-19 of Ref. 8 of CRBRP-GEFR-00523.

QUESTION II-67

(F6.2-22, par. 2) (Generic answers (a) and (b) are not required.)

(c) Explain more fully the basis for the conclusions, "A relatively simple single-pin model like SLUMPY cannot adequately analyze this situation. Although the model can be forced to fit the experiment, the mechanisms are not presently identified clearly enough to allow extrapolation to another system." Indicate in detail the uncertainties that this introduces into the subsequent calculations modeling the transition phase and the disassembly (VENUS) phase.

ANSWER II-67 (c)

The SLUMPY model assumes that fuel motion occurs only in the axial direction. In addition, it is assumed in SAS3A and SAS3D that the power density in each fuel pin is azimuthally symmetric. Hence, it is not capable of modeling the pellet stack crumbling process, which would be three-dimensional in nature, nor can it analyze the effect of the power density gradient across the pin, as was present in the L2 TREAT test.

Because several of the physical processes modeled in SLUMPY are treated parametrically, it is possible to simulate observed experimental results by a suitable choice of parameters, so long as the dominant mode of fuel motion is axial motion.

Uncertainties in SLUMPY calculated fuel motion have been addressed by parametrically varying an equivalent gravity force and by matching these effects to appropriate test data (see CRBRP-GEFR-00523 Sections 7.1.1,

7.2.1, and Appendix C). Core conditions which result from the above variations are carried directly into either the meltout or disassembly phase evaluations.

QUESTION II-68

(F6.2-23, par. 1 (Generic answers (a) and (b) are not required.)

(c) Explain more fully (qualitatively and quantitatively) the statement that "this mechanism is somewhat sensitive, and slight variations in the assumptions can cause the fuel to either rise, fall, or move in both directions.

ANSWER II-68 (c)

The statement should be interpreted in the context of the L3 and L4 loss-of-flow TREAT tests. At the steady-state power levels under which these tests were conducted there are several competing effects tending to cause fuel motion in various directions once fuel melting starts, e.g., gravity, production of steel vapor pressure, and any remaining fission gas. Hence, calculated results do become sensitive to the rate of heat transfer from fuel to stainless steel, the rate of release of fission gas from fuel near its melting point, and the assumed radial heat losses. The quantitative SLUMPY analysis of these tests is presented on pp. 54-60 of Ref. 8 in CRBRP-GEFR-00523. It is felt that uncertainties in SLUMPY analysis are covered by the range of parametric cases which are presented in CRBRP-GEFR-00103. One initial purpose of the F-series tests (Question II-69) was to provide an increased experimental data base to aid in the modeling of fuel motion in a reactor HCDA analysis.

QUESTION II-69

(F6.2-23, par. 3) (Generic answers (a) and (b) are not required).

(c) What uncertainties are likely to be resolved by the F-series experiment and what uncertainties are likely to remain?

ANSWER II-69 (c)

Experiment F-1 provides insight into the behavior of highly irradiated fuel when heated to melting at a near nominal power level with small radial temperature gradients. Analysis of this test shows that the observed fuel motion would produce little or no reactivity increase although the motion was not strongly dispersive (see Ref. 34, pg. 11-3, CRBRP-GEFR-00523). Experiment F-2 showed that fuel having very small irradiation exposure is dispersive when exposed to a power burst leading to dispersal at elevated (~6 times nominal) power. Uncertainties remain in effects of details of fuel characterization (burnup, fission product content) and thermal history. However, the body of experimental data presents a consistent picture. See: L. W. Deitrich, "An Assessment of Early Fuel Dispersal in the Hypothetical Loss-of-Flow Accident," Proc. Fast Reactor Safety Meeting, Seattle, Aug. 19-23, 1979, pg. 615. The applicant is aware that experiments F-3 and F-4 have been completed, but it has not been determined to what extent these will be relied on.

THIRD INTERROGATORY SET

PREAMBLE TO QUESTIONS

With respect to the following requests for information we are concerned with four distinct validations relative to the models and computer codes:

i) Validation that the code's output is the correct numerical calculation that should result from a given set of input data and the model assumptions;

ii) Valdiation of the models against actual experimental data;

iii) Validation that the models can be extended to the CRBR; and

iv) Validation that the input assumptions for the CRBR case are adequate with respect to the CDA analysis, i.e., are supported by experimental evidence. By "adequate," here and below, we mean that the calculations will not underestimate the CDA work potential (i.e., forces and resulting energetics of a CDA) or overestimate the containment capability of the reactor with respect to a CDA.

QUESTION I

With respect to each of the following codes and each subroutine of each of the following codes:

(A) FXVARI
(B) REXCO-HEP

please provide the following information [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.]:

1) Complete, current documentation (i.e., a writeup) of the codes and the subroutines;

2) Identify, by name and affiliation, the author, or authors, of each model, subroutine, or portion of each subroutine, which each contributed or worked on;

3) Identify by name affiliation (including organization, division, branch, title, etc.) each applicant employee, or consultant, that has intimate working knowledge of the code and each subroutine, or parts thereof, including its validity. Where more than one person is involved, delineate which portion of the code or subroutine with which each has an intimate working knowledge;

4) Describe fully the procedures by which Applicant has assured itself and continues to assure itself, that the various computer programs (codes) accurately reproduce the models as described in the PSAR and its references (see Validation (i) above);

5) Indicate which models (including subroutines, or portions of subroutines) have not been validated as described in Validation (i);

6) Indicate the models (including subroutines, or portions of subroutines) or assumptions that have not been validated as described in Validation (ii);

7) For each model, portion of the model, or assumption that has been validated (against experimental (or other) data, see Validation (ii) above) describe fully the procedure by which it was validated, and the results, including all uncertainties and limitation of the validation. Indicate the source of the experimental, or other data, that was used in the validation.

8) Explain fully all instabilities in the numerical performance in the models, what causes them, and how they are avoided, and the extent to which this introduces uncertainties in the calculations and limits the validity of the model (cf., p.P6.2-10, par. 2).

9) To the extent that any answers to the above questions are based on referenced material, please supply the references.

10) 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.

11) 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.

ANSWER I(A)

The processing of neutron cross-section data for the heterogeneous core analyzed in CRBRP-GEFR-00523 did not use the FXVARI or its subroutines. Accordingly, responses to I(A) 1 through I(A) 11 are not required. (The CRBRP-GEFR-00103 homogenous core analysis used FXVARI).

The methods used for processing neutronic input data for SAS3D analysis in CRBRP-GEFR-00523 are the same as those used in Section 4.3 of the PSAR.

ANSWER I(B)

These answers provide the information requested for the REXCO-HEP Computer Code.

(1) Reference 1 on p. 5-45 of CRBRP-3, Vol. 1, "Energetics and Structural Margins Beyond the Design Base" is the latest and most complete document describing the REXCO-HEP code.

Documentation of the subroutines which constitute the REXCO-HEP code are contained in this reference. Other improvements are described in J. Gvildys and Y.W. Chang, "REXCO-HEP (Release 4) Users Manual," ANL/RAS 78-42 (September, 1978).

(2) The REXCO-HEP code is a complex system developed over a span of several years. Its development started in the Reactor Engineering Division and continued in the Reactor Analysis and Safety Division of Argonne National Laboratory. The major contributors are authors of Reference 1 on p. 5-45 of CRBRP-3 Vol. 1.

(3) The following staff members of Argonne National Laboratory have a working knowledge of the code, including its range of applicability and the extent of its validation: Stanley H. Fistedis, Manager, Engineering Mechanics Program, and Yao W. Chang, Manager, Structural Mechanics Section, Reactor Analysis and Safety Division, Argonne National Laboratory.

(4) The hydrodynamic and solid mechanic principles on which the code is based are established scientific facts. The individual subroutines and the entire code were checked and rechecked both individually and in its entirety for correctness of results. Extensive checking of the code against other established analytical solutions was performed and the comparisons were cited in Ref. 1 on p. 5-45 of CRBRP-3, Vol. 1 and also in J. Gvildys and Y.W. Chang, "REXCO-HEP (Release 4) Users Manual," ANL/RAS 78-42 (September, 1978).

(5) All of the models have been validated as in (4) above.

(6) There has been extensive experimental validation of the code as described in Item (7) below. In the few areas where experimental validation does not exist at this time, issues were resolved by making conservative assumptions.

(7) Substantial validation of the REXCO code system was performed. Predictions of the REXCO code system were compared against a variety of experiments. The comparisons are documented in the following publications:

(1) "Comparison of a Two-Dimensional Hydrodynamics Code (REXCO) to Excursion Experiments for Fast Reactor Containment," ANL-7911, January 1972.

(2) "Comparison of a 2-D Hydrodynamics Code (REXCO) to Excursion Experiments for Fast Reactor Containment," ANL-7911 Supplement 1, July 1972.

(3) "Comparison of FFTF Simple-Model Tests with REXCO Predictions," ANL-8071, January 1974.

(4) "REXCO Predictions of Elastic and Elastoplastic Deformation of Fluid Filled Pipes and Comparisons with Experiments of 1/10 Scale FFTF Pipe Models," ANL 75-61, September 1975.

(5) Y.W. Chang and J. Gvildys, "Comparison of REXCO Code Predictions with Flexible Vessel Experiments," ANL/RAS 78-9 (February, 1978).

(6) Y.W. Chang and J. Gvildys, "Comparison of REXCO Code Predictions with Rigid Vessel Experiments," ANL/RAS 78-30 (June, 1978).

(7) Y.W. Chang and J. Gvildys, "Comparison of REXCO Code Prediction with SRI SM-2 Experimental Results," ANL-78-18 (August, 1978).

(8) In the explicit integration of the equations of motion, if the time step is too large, the computed response may result in numerical instability. In REXCO-HEP, the time step used is based upon the White stability criterion. This criterion is explained in detail and referenced in Ref. 1 on p. 5-45 of CRBRP-3, Vol. 1.

(9) The referenced documents have been or will be made available for inspection and copying.

(10) The Applicants are not doing development work on REXCO-HEP. No such development is currently planned.

QUESTION II (GENERAL)

Request for the following information is based on our concerns with respect to Validation (iii) and (iv) above. In the Applicant's answers to the generic questions (b) and (c) below, the Applicant is requested to be responsive to these concerns.

With respect to each statement, assertion or assumption from Section F6.2 of the PSAR) identified below, please provide the following information (unless noted otherwise). (NOTE: the following numbered Interrogatories are identified by the page and/or paragraph number from the PSAR in parenthesis.) [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) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) that has the expert knowledge required to support the statement, assertion, or assumption;

b) Describe in detail the supporting evidence for the statement, assertion, or assumption and where appropriate the rationale for the approach taken.

c) Provide any additional information requested following each statement, assertion, or assumption.

d) To the extent that any answers to the above questions are based on referenced material, please supply the references.

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.

ANSWER II (GENERAL)

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

(a) Dennis M. Switick, Manager, Safety Analysis, General Electric Company, Fast Breeder Reactor Department, 310 De Guigne Dr., Sunnyvale, California 94806, has the expert knowledge required to support the responses identified in the attached affidavit.

(b) and (c) See responses 1-29 below.

(d) The referenced documents have been or will be made available for inspection and copying.

(e) The Applicants are currently analyzing this area and will provide pertinent information as it becomes available.

(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.

NOTE: Question II-1 pertains to the potential for transition phase occuring from TOP events in the EOEC configuration.

QUESTION II-1

(F6.2-92, par. 1-4) The last two sentences in the first paragraph beginning with, "If the blockage . . . . the first and last sentence in the second paragraph; and all of paragraphs three and four.

(c) With respect to the statements in these first four paragraphs concerning blockage, relocation and reactivity effects and the conclusion that "potential for a transition phase occurring in the BOEC TOP event is negligible," depend on assumptions made in the SAS3A (including SASBLOK) evaluations, to what extent are each of the statements and the conclusion sensitive to assumptions concerning each of the following:

(i) the mechanical effects involved in fuel lodging;
(ii) the location of fuel blockages above the core;
(iii) the location and degree of fuel failure;
(iv) the position of the bulk sodium level above the ejected fuel slug;
(v) the dynamic pressure across the slug;
(vi) the rate at which sodium vapor is produced;
(vii) the variations in driving pressure;
(viii) the rate of vapor production from various heat transfer processes in conjunction with the rate of condensation of these vapors which play a role in determining the dynamic pressures acting on the ejected material on the above core structure;
(ix) the use of equilibrium thermodynamics;
(x) kinematic processes included;
(xi) the molten fuel coolant interaction model assumed;
(xii) the choice of uncertain reactor parameters including fuel, steel and sodium reactivity worths and reactor loading patterns as a function of burnup.

Please provide detailed responses for each case (i) through (xii).

ANSWER II-1 (b) and (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

NOTE: Question II.2 pertains to BOEC configuration.

QUESTIONS II-2

(P6.2-92, par. 5) Generic answers (a) and (b) are not required.

(c) With respect to the statements in these two paragraphs and the conclusion". . . it is believed that energy released from a partial core meltdown would be bounded by those events which could follow termination of the initiating phase of the BOEC LOF accidents which are considered next," to what extent are each of the statements and the conclusion sensitive to assumptions concerning each of the following:

(i) the mechanical effects involved in fuel lodging;
(ii) the location of fuel blockages above the core;
(iii) the location and degree of fuel failure;
(iv) the position of the bulk sodium level above the ejected fuel slug;
(v) the dynamic pressure across the slug;
(vi) the rate at which sodium vapor is produced;
(vii) the variations in driving pressure;
(viii) the rate of vapor production from various heat transfer processes in conjunction with the rate of condensation of these vapors which play a role in determining the dynamic pressures acting on the ejected material on the above core structure;
(ix) the use of equilibrium thermodynamics;
(x) kinematic processes included;
(xi) the molten fuel coolant interaction model assumed;
(xii) the choice of uncertain reactor parameters including fuel, steel and sodium reactivity worths and reactor loading patterns as a function of burnup.

Please provide detailed answers for each case (i) through (xii).

ANSWER II-2 (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

NOTE: Questions II-3 and II-4 pertain to the potential for transition phase from LOF events.

QUESTION II-3

(P6.2-93, par. 2) Generic answers (a) and (b) are not required.

(c) What criteria are used to define termination of the initiating phases of LOF accidents? Of TOP accidents?

What criteria are used to define initiation of the transition phase of LOF accidents? Of TOP accidents? Describe in detail each case considered where conditions in the core are such that a true hydrodynamic disassembly calculation is justified as a means of continuing the analysis to permanent shutdown.

ANSWER II-3 (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

QUESTION II-4

(P6.2-93, par. 2 and 3) Generic answers (a) and (b) are not required.

(c) The statements and conclusions in paragraph 2 beginning at "In these less energetic . . ." and the two statements in paragraph 3, "Thus, the

core. . . is proceeding gradually into a completely molten state," and "This is judged to be one of early removal of large amounts of fuel. . . and below the core," depend on assumptions made in the SAS3A LOF CDA evaluations. To what extent are each of the statements and the conclusions sensitive to assumptions concerning each of the following:

(i) the rate at which sodium vapor is produced;
(ii) the use of equilibrium thermodynamics;
(iii) the equations of state assumed;
(iv) the molten fuel-coolant interaction model assumed;
(v) the choice of uncertain reactor parameters including fuel, steel, and sodium reactivity worths and reactor loading patterns as a function of burnup.

Please provide detailed answers for each case (i) through (v).

ANSWER II-4 (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

NOTE: Questions II-5 through II-8 pertain to the potential for the existence of steel blockages.

QUESTIONS II-5

(F6.2-93, par. 4) Generic answers (a) and (b) are not required.

(c)(i) Identify all TREAT tests that were designed to simulate unprotected LOF CDA phenomena; (ii) with respect to these, identify all TREAT tests that focused on coolant and cladding behavior; (iii) identify and supply all ANL documentation of results of those TREAT tests identified in (ii)

above; (iv) identify all TREAT tests identified in (ii) above where upper cladding blockage was observed; where lower cladding blockage was observed.

ANSWER II-5 (c)

Section 10.2 in CRBRP-GEFR-00103 and Section 8.2 in CRBRP-GEFR-00523 discuss the same material that this Interrogatory makes reference to.

(c)(i) The TREAT tests that were designed to simulate unprotected LOF CDA phenomenon were the L, R, and F series experiments.

(c)(ii) The test series that focused on coolant and cladding behavior were the L and R series experiments.

(c)(iii) Documentation of the results of the TREAT tests identified in (ii) above can be found in the following documents:

(1) Reference 59 in CRBRP-GEFR-00103.
(2) E. Barts, et al., "Summary and Evaluation, Fuel Dynamics Loss-of-Flow Experiments (Tests L2, L3, L4)," ANL 75-57, Sept. 1975.
(3) Reference 36 in CRBRP-GEFR-00103
(4) Reference 55 in CRBRP-GEFR-00103
(5) Reference 21 in CRBRP-GEFR-00523
(6) Reference 23 in CRBRP-GEFR-00523
(7) Reference 28 in CRBRP-GEFR-00523
(8) Reference 29 in CRBRP-GEFR-00523
(9) Reference 31 in CRBRP-GEFR-00523
(10) Reference 32 in CRBRP-GEFR-00523
(11) Reference 33 in CRBRP-GEFR-00523
(12) Reference 34 in CRBRP-GEFR-00523
(13) Reference 46 in CRBRP-GEFR-00523
(14) R. Simms, et al., "Fuel Motion in Experiments Simulating LMFBR Loss-of-Flow Accidents," ANL/RAS 80-25, Nov., 1980.
(15) R. Simms, et al., "TREAT Test L7 Simulating an LMFBR Loss-of-Flow with FTR-Type Fuel," ANL-80-112, Nov., 1980.

(c)(iv) This information is found in the documents listed in (iii) above.

QUESTION II-6

(F6.2-93, par. 4) Generic answers (a) and (b) are not required.

(c) Clarify how Reference 12 is supportive of the conclusion, "It is presently believed that such blockages do not form or are incomplete in almost all subassemblies."

ANSWER II-6 (c)

The reference referred to in the question is Ref. 57 in CRBRP-GEFR-00103 and the discussion of this subject has been revised by more recent data from Ref. 49 in CRBRP-GEFR-00523. See Section 8.2.1 of CRBRP-GEFR-00523.

QUESTION II-7

(F6.2-95, par. 1) Since incomplete blockages will be immediately melted out when molten fuel passes through them, these blockages can be neglected in the analysis.

ANSWER II-7 (b) and (c)

The discussion of this subject has been revised by more recent data from Ref. 49 in CRBRP-GEFR-00523. See Section 8.2.1 of CRBRP-GEFR-00523.

QUESTION II-8

(F6.2-93), par. 4) What is the basis for assuming a molten steel-coolant interaction would not occur as the molten cladding contacts liquid sodium at the top and bottom of the core? If it does occur how is this interaction modeled?

ANSWER II-8 (b) and (c)

Movement of molten cladding prior to fuel movement would occur into voided regions of the fuel assembly where the steel would freeze. No evidence of steel-sodium interactions were observed in the TREAT R-series experiments in which early cladding relocations occurred (Ref. 23 in CRBRP-GEFR-00523). The potential for a dispersed flow regime ejection of molten fuel-steel-gas mixtures was discussed on page 10-7 of CRBRP-GEFR-00103 and more recent evaluations are provided in Section 8.2 of CRBRP-GEFR-00523. Neglecting the molten steel-sodium interactions is primarily founded on the dispersed flow regime negating the potential for large scale, intimate liquid-liquid contact occurring. Out-of-pile experiments with both simulant materials and reactor materials support this conclusion (Ref. 63 in CRBRP-GEFR-00103).

NOTE: Questions II-9 through II-16 pertain to extended fuel motion.

QUESTION II-9

(F6.2-95, par. 2) The development of fuel and steel vapor pressures strongly suggest that the fuel motion will be monotonically dispersive.

ANSWER II-9 (b) and (c)

References 57 and 60 in CRBRP-GEFR-00103 and more recent analysis in Section 8.3 of CRBRP-GEFR-00523 supported by Reference 76 in CRBRP-GEFR-00523 provide the basis for the statement set forth in the interrogatory.

QUESTION II-10

(F6.2-95, par. 2) Entrainment of clad could arise. . . criteria provided in Ref. 69.

(c) Under what conditions could entrainment of clad be expected not to occur? If cladding steel sloshing could occur, could fuel sloshing also occur? To what extent, if any, did entrainment take place in the in-pile LOF meltdown experiments? Identify and discuss all uncertainties in these in-pile LOF meltdown experiments with respect to the applicability of these results to CRBR LOF CDA conditions. In the post-test analysis of the LOF meltdown experiments to what extent was the relocated fuel and steel heterogeneous? Identify and discuss all uncertainties in the applicability of the stability criteria provided in Reference 69 to CRBR LOF CDA conditions. Discuss.

ANSWER II-10 (b) and (c)

Reference 57 in CRBRP-GEFR-00103 provides the bases for this statement. Clad-fuel entrainment is likely to occur for all postulated core-disruptive hypothetical accidents. Fuel sloshing in the same sense as clad sloshing is considered very unlikely since following fuel melting a large fraction of the neutron heating will appear as latent heat of vaporization. The resultant driving force is much larger than that represented by the sodium vapor streaming (which leads to clad sloshing) and therefore results in essentially monotonic dispersal of the fuel. Even with no entrainment, considering the fact that the steel-fuel system is largely predispersed by design, no significant change in the accident sequence as depicted in the CRBRP-GEFR-00103 or CRBRP-GEFR-00523 is anticipated.

QUESTION II-11

(F6.2-95, par. 2) Heat Transfer from the fuel to the clad will result in rapid clad vaporization and dispersal of fuel.

ANSWER II-11 (b) and (c)

Reference 57 in CRBRP-GEFR-00103 (same as Ref. 50 in CRBRP-GEFR-00523) and updated analysis in Section 8.2 of CRBRP-GEFR-00523 provide the basis for the statements set forth in the interrogatory.

QUESTION II-12

(P6.2-95, par. 2) Only a small fraction of the available clad material is necessary, since the liquid-to-vapor density ratio is in the order of 10^4.

ANSWER II-12 (b) and (c)

Relatively small vapor velocities ($\sim$2m/sec) are required to fluidize the fuel for void fractions of interest (40 to 50%). Hence, only a small volume of liquid steel is necessary to produce these velocities. Even small puddles of steel left behind on the fuel pins would be sufficient. (See Ref. 57 in CRBRP-GEFR-00103)

QUESTION II-13

(P6.2-95, par. 2) Furthermore, because of the above entrainment processes and since molten steel is known to wet oxide fuel, the local heat transfer between fuel and clad can be approximated by equilibrium conditions.

ANSWER II-13 (b) and (c)

The final outcome is not sensitive to the assumptions of equilibrium conditions. Local nonequilibrium conditions between fuel and steel will lead to similar conclusions regarding fuel dispersal and boil-up.

QUESTION II-14

(P6.2-95, par. 2) The vaporization rates are therefore more than sufficient to fluidize and to maintain a dispersed fuel-steel system.

ANSWER II-14 (b) and (c)

See Section 8.2 of CRBRP-GEFR-00523.

QUESTION II-15

(F6.2-95, par. 2) Generic answers (a) and (b) are not required.

(c) Provide more detail models of possible phenomena and events taking place between clad melting and fuel dispersal above the gas plenum region, giving estimates of the time sequence of events, material description, movements and relocations.

ANSWER II-15 (c)

Presently available models of the phenomena and events taking place between clad melting and fuel dispersal during the initial stages of core disruption are those used by the Applicant in CRBRP-GEFR-00103 and CRBRP-GEFR-00523, i.e., SAS3D, PLUTO 1, and PLUTO 2. The Applicant's updated analysis of fuel penetration into assembly rod structure is given in Section 8.2.2 of CRBRP-GEFR-00523.

QUESTION II-16

(F6.2-97, par. 4) This process can continue since experiments (Reference 6) (with both simulant materials and reactor system and a cold liquid is unlikely to result in sustained interaction pressures larger than the vapor pressure or system pressure of the hot fluid.

(c) What is the basis for rejecting the analysis and conclusions presented in "The Role of Spontaneous Nucleation in Thermal Explosions, Freon/Water Experiments," S. J. Board, R. W. Hall, G. E. Brown, Proc. Fast Reactor Safety Meeting, April 24, 1974, Beverly Hills, Calif., USAEC Report CONF-740401-P2 (1974), pp. 933-936.

ANSWER II-16 (b) and (c)

(b) Reference 63 in CRBRP-GEFR-00103 supports the conclusion that contact temperature larger than the spontaneous nucleation temperature is required for sustained pressure generation. This condition is not satisfied for the dispersed fuel-steel-sodium system.

(c) The experiments and analysis reported in "The Role of Spontaneous Nucleation in Thermal Explosions, Freon/Water Experiments," by S. J. Board et el., support the criteria stated in 16(b) above. For further classification, see the following reference: R. E. Henry, H. K. Fauske, and L. M. McUmber, "Vapor Explosions with Subcooled Freon," Trans. Am. Nucl. Soc., Vol. 22, 1975. See Section 8.0 of CRBRP-GEFR-00523 for recent Applicant evaluations.

NOTE: Questions II-17 through II-23 pertain to fuel behavior following postulated core plugging.

QUESTION II-17

(F6-2-98, par. 4) The disruption of the fuel in different subassemblies is relatively coherent across the core due to the high power levels in the initiating phase of the accident. All of the subassemblies experience fuel disruption within a few seconds of each other.

(c) Define in more detail the extent of incoherence that might be involved, including the extent of the incoherence in the position of fuel temperatures, reactivity, insertion mechanisms and their spatial distribution, and time sequence of events.

ANSWER II-17 (b) and (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

QUESTION II-18

(F6.2-98, par. 4) Generic answers (a) and (b) are not required.

(c) What happens to the control rods during the transition phases of the LOF and TOP CDAs? How are these effects modeled?

ANSWER II-18 (c)

The control rods are neglected in the transition phase analyses. Control rod material is assumed to be absent from the homogenized pools that eventually form.

QUESTION II-19

(F6.2-98, par. 4) Generic answers (a) and (b) are not required.

(c) Explain in detail all alternative modes of accident progression, if it is assumed that the melting of wrapper can steel is not faster than the removal of the postulated frozen steel-fuel blockage above the core.

ANSWER II-19 (c)

The identified modes of hypothetical accident progression for the requested assumption that the melting of wrapper can steel is not faster than the removal of the postulated frozen steel-fuel blockage above the core were

described in detail in Section 10.1 of CRBRP-GEFR-00103 and are updated for more recent data in Section 8.2 of CRBRP-GEFR-00523.

QUESTION II-20

(F6.2-98, par. 4) Generic answers (a) and (b) are not required.

(c) Explain in detail all alternative modes of accident progression if the core cannot be adequately represented by a single coherent boiling region.

ANSWER II-20 (c)

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

Section 8 of CRBRP-GEFR-00523 describes the analysis for the current core design.

QUESTION II-21

(F6.2-98, par. 5) Generic answers (a) and (b) are not required.

(c) The basis for assuming a vanishing small viscosity, thereby eliminating laminar and turbulent flow regimes.

ANSWER II-21 (c)

The basis for this statement is found in Reference 58 of CRBRP-GEFR-00103.

QUESTION II-22

(F6.2-101, par. 2) Generic answers (a) and (b) are not required.

(c) Provide detailed description of the series of experiments being performed to investigate fuel-coolant and steel-coolant interaction postulated if hot fuel and steel are driven out the upper and lower core structure encounter sodium, and all writeups (including internal memoranda) of results (final and preliminary) of these experiments.

ANSWER II-22 (c)

Section 8.2.6 of CRBRP-GEFR-00523 provides an updated summary of the information requested.

QUESTION II-23

(F6.2-101, par. 3) Since melt-through of the structure is anticipated well before the power level has dropped to 1%, a significant fraction of the fuel-steel mixture is likely to be rapidly ejected.

ANSWER II-23 (b) and (c)

Evaluations on the current design in Section 8.3.5 of CRBRP-GEFR-00523 indicate that small fractions of fuel-steel mixture are more likely to be ejected.

NOTE: Questions II-24 through II-29 pertain to reactivity effects in a disrupted core.

QUESTION II-24

(F6.2-101, par. 5) The assumption that reactivity calculations on a disrupted core can be adequately modeled using two-dimension r-z geometry?

ANSWER II-24 (b) and (c)

The types of reactivity calculations made in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 are adequate with two-dimensional r-z geometry because the reactivity changes considered are relatively large, and primarily due to axial fuel motions and sodium voiding. More sophisticated models (e.g., three-dimensional hex-z geometry) are useful primarily in obtaining accurate estimates of physics parameters related to normal operation. Using such models to estimate reactivity changes due to sodium voiding and fuel motion would not yield significantly different results than would be obtained from an r-z model. This is a standard engineering approach, the adequacy of which is borne out from physics calculations performed for conditions of normal operation. In such calculations it is standard practice to use simplified models to compute reactivity changes, after having verified the approach by periodic checking with three-dimensional calculations.

QUESTION II-25

(P6.2-101, par. 5) The assumption that reactivity calculations on a disrupted core can be adequately modeled using the nine group cross-section set.

ANSWER II-25 (b) and (c)

The types of reactivity calculations made in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 are adequately modeled with a nine-group cross-section set because boundaries were chosen to ensure that the important physical phenomena were treated properly. Taking such care results in very good agreement with calculations made with finer-group cross-section sets.

QUESTION II-26

(F6.2-102, par. 1) Generic answers (a) and (b) are not required.

(c)(i) What is meant by "significant" in". . . results has a significant implication for the transition phase . . ."? (ii) What are the uncertainties involved? (iii) Identify the higher power subassemblies where steel in the core is nearing the boiling point. (iv) What are the uncertainties involved in this estimate? (v) Quantify "nearing the boiling point." (vi) What are the uncertainties? (vii) Quantify what is meant by "imminent" in" . . . rapid production of steel vapor is imminent . . ." (viii) What are the uncertainties involved? (ix) What are the uncertainties that lead to the choice of the word "should" in ". . . should uncertainties relative to the time of onset of steel vapor production. (xi) Quantify what is meant by ". . . delayed for any significant period of time . . ." (xii) What are the uncertainties involved? (xiii) Quantify what is meant by "mild" in ". . . another mild burst would follow. . .?" (xiv) What are the uncertainties? (xv)-(xxviii) With respect to each of the above questions, (i) through (xiv), relative to (F6.2-102, par. 1), discuss in detail the implications of incoherences in the phenomena involved and consequent lack of symmetry in time sequences and geometry.

ANSWER II-26 (c)

(c) (i) through (xxviii). Sections 8. and 9.4 of CRBRP-GEFR-00523 provide the current assessment of the meltout and large-scale pool phase of the HCDA.

QUESTION II-27

(F6.2-102, par. 2) Generic answers (a) and (b) are not required.

(c) Whose "current best estimate" is being referred to in the second sentence? Are there differing views known to the applicant as to what

constitutes the "best estimate" as to whether fuel ejected into the blanket will travel through then without plugging? If so, present in detail the basis for these alternative views.

ANSWER II-27 (c)

The extent of penetration of flowing molten materials through various reactor structures and its basis are discussed in Section 8.2 of CRBRP-GEFR-00523.

QUESTION II-28

(P6.2-102, par. 3) Generic answers (a) and (b) are not required.

(c) What are the implications of nonuniform removal of fuel from the core?

ANSWER II-28 (c)

In terms of calculation of the reactivity due to fuel relocation, the mode of fuel removal is insignificant. The actual state of the reactor for which the reactivity is being computed has far greater effects upon the resulting energetics.

QUESTION II-29

(P6.2-103, par. 2) Generic answers (a) and (b) are not required.

(c) Whose "best-estimate path" is being referred to here? Are there differing views known to the applicant as to what constitutes the "best-estimate path"? If so, present in detail the basis for these alternative views.

ANSWER II-29 (c)

See Section 8.2.2 of CRBRP-GEFR-00523.

FOURTH INTERROGATORY SET

PREAMBLE TO QUESTIONS

In previous interrogatories information was requested concerning four distinct validations relative to the models and computer codes used in the analysis of CRBR CDAs, namely,

i) Validation that the code's output is the correct numerical calculation that should result from a given set of input data and the model assumptions;

ii) Validation of the models against actual experimental data;

iii) Validation that the models can be extended to the CRBR; and

iv) Validation that the input assumptions for the CRBR case are adequate with respect to the CDA analysis, i.e., are supported by experimental evidence. By "adequate", here and below, we mean that the calculations will not underestimate the CDA work potential (i.e., forces and resulting energetics of a CDA) or over estimate the containment capability of the reactor with respect to a CDA.

With respect to the following requests for information we are concerned primarily with the fourth validation--validation that the input assumptions for the CRBR case are adequate with respect to the CDA analysis. Here we are not so much concerned with the validity of the model expressions as with the uncertainties in the VENUS work energy calculations due to propagation of uncertainties in a) the parameters used and b) the model input data and due to any synergisms among these uncertainties and the model assumptions.

QUESTION I

With respect to the calculations identified below, under (A) through (D), please provide the following information [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.]:

1) List and identify all model input data (exclusive of coding flags and inputs that specify coding options, criteria, printout formats, etc.) and all model parameters that come into play in each of the models utilized in the coupled-code accident analysis calculations, including but not limited to input data and parameters in SAS3A and VENUS-II. Exclude parameters not called into use because a subroutine, or part thereof, was not utilized.

2) Describe in detail the basis for the choice of each input datum and model parameter listed above.

i) In each case quantify the uncertainty in the value selected;

ii) In each case indicate whether the value is based on first principles, experimental measurements, unvalidated hypothesis, output of a coupled model (e.g., VENUS-II input obtained from SAS3A output), etc.,

iii) In each case indicate whether the choice of the input datum or model parameter was selected to represent the "best estimate", or a bounding or "conservative value", where "conservative value" here means a value chosen so as not to underestimate the accident consequences, e.g., work potential.

3) For each input datum and model parameter with uncertainty listed in 1) above, indicate in quantitative terms the magnitude of the uncertainty introduced into the final calculation of the work energy by the uncertainty in the input datum or model parameter. In addition, discuss in detail any synergistic effects resulting from combinations of uncertainties in the input values, model parameters, and model assumptions. In each case discuss the basis for the estimate of how the uncertainties propagate, e.g., include and discuss all parameters analysed used to test the effect of uncertainties.

4) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) that has intimate working knowledge of the basis for the selection of the parameter or input datum.

5) To the extent that any answers to the above questions are based on referenced material, please supply the references.

6) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicant's 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.

7) 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.

(A) The three SAS3A/VENUS-II calculations for the LOP-BOEC configuration considered in Section F6.2.6.2.1 (i.e., 40$/sec, 50$/sec and 100$/sec ramp rates) and summarized in Table F6.2-2d. In the latter two cases (50$/sec and 100$/sec) it is not necessary to duplicate information previously provided with respect to the 40$/sec case, e.g., much of the SAS3A input data. Here, indicate only those parameters and input data that differ from those previously listed.

(B) The one SAS3A/VENUS-II calculation for the LOF-EOEC configuration considered in F6.26.2.2 is summarized in Table F6.2-22.

(C) The four SAS3A/VENUS-II calculations for TOP-EOEC configurations considered and summarized in Table F6.2-23. In the last three cases, (i.e., the second 50$/sec, 75$/sec, and 100$/sec) it is not necessary to

duplicate information previously provided with respect to the first 50$/sec case.

(D) The four VENUS-II disassembly calculations for the EOEC LOF immediate re-entry case and three VENUS-II disassembly calculations for homogenized core re-entry considered in P6.2.6.4 and summarized in Table P6.2-24. It is not necessary to duplicate information in each case; indicate only those parameters and input data that differ from those previously listed.

ANSWER I

1) This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. CRBRP-GEFR-00523 describes the analyses for the current core design. The SAS3D and VENUS-II input data used in the calculations have been set forth for selected cases in CRBRP-GEFR-00523 on page 4-19 and in Appendices D and G. Physical descriptions of the numerical input data listings are found in References 5 and 6 of CRBRP-GEFR-00523.

2) This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. CRBRP-GEFR-00523 describes the analyses for the current core design. The basis for the choice of the SAS3D base case input parameter values is presented in Section 4.3 of CRBRP-GEFR-00523.

Referring to the VENUS input description in ANL-7951, the following data arises from the geometric model chosen, and from options chosen which were deemed appropriate for the analyses made:

(a) All data on cards 2-10, 14, 15, 21, 22.
(b) Data for all regions on cards 23-27, and 33.

The power densities on cards 11, the material worths on cards 28, and the Doppler broadening feedback parameters on card 31 were obtained from calculations similar to those described in Section 4.3 of the PSAR. The

data on cards 16 and 17 were obtained from SAS3D output, as was PZERO on card 20, and data on cards 32. The rest of the data on card 20, are deemed appropriate for the analyses model.

The data on card 30 are based on experimental measurements. The data on card 50 are based on SAS3D output, which was used to obtain an average core temperature, which was in turn used by VENUS to obtain an appropriate r-z temperature distribution. Input parameter changes in the base case input decks are described in Section 9 of CRBRP-GEFR-00523.

3) The impact of uncertainties with respect to input data is discussed in CRBRP-GEFR-00523. Various parameters were varied to determine sensitivities to data and modeling uncertainties.

4) Dennis M. Switick, Manager, Safety Analysis, General Electric Company, Fast Breeder Reactor Department, 310 DeGuigne Drive, Sunnyvale, California 94086.

5) The referenced documents have been or will be made available for inspection and copying.

6) The Applicants are currently analyzing this area and will provide pertinent information as it becomes available.

7) 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 II (General)

Request for the following information is based on our concerns with respect to validation (iii) and (iv) noted previously. In the Applicant's answers to the generic questions (b) and (c) below, the Applicant is requested to be responsive to these concerns.

With respect to each statement, assertion or assumption (based on Section F6.2 of the PSAR) identified below, please provide the following information (unless noted otherwise). NOTE: The following numbered interrogatories are identified in parentheses by the page and/or paragraph number from the PSAR or by a code number identifying an NRC question addressed to the Applicant. [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) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) that has the expert knowledge required to support the statement, assertion, or assumption.

b) Describe in detail the supporting evidence for the statement, assertion, or assumption and where appropriate the rationale for the approach taken.

c) Provide any additional information requested following each statement, assertion or assumption.

d) To the extent that any answers to the above questions are based on referenced material, please supply the reference.

e) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicant's 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.

ANSWER II (General)

The answers to questions II(a), (d), (e) and (f) are the same as those for questions I(4), (5), (6) and (7) respectively.

NOTE: Questions II-1 through II-4 pertain to analysis of hydrodynamic disassemblies.

QUESTION II-1

NRC Question 001.497. In this question to the Applicant the Staff states "...disassembly calculations results will depend on the results of ...the equation of state for disassembly phase ..." For the disassembly phase perhaps the single most important uncertainty is the equation of state especially for temperatures and pressures close to the critical point of fuel vapor.

Generic answers (a) and (b) are not required.

(c) Does the Applicant agree with this conclusion? If not, why not? In either case (if so or if not), discuss in more detail the basis for this conclusion. Document the uncertainties in the equation of state and indicate the effect these uncertainties could have on the results of the hydrodynamic disassembly analysis.

ANSWER II-1 (c)

Jackson et al., has studied the effects of using various formulations of the fuel vapor pressure behavior proposed for use in disassembly calculations, and showed that only minor differences emerged from using the different models. (J. F. Jackson, A. M. Eaton, R. M. Hall, T. F. Bott

(Brigham Young University), "The Influence of Equation-of-State Uncertainties on Fast Reactor Disassembly Calculations, Trans. Am. Nucl. Soc. 22, p. 368 (1975). They further showed that the Menzies formulation currently in VENUS-II predicted the greatest energy release. Uncertainties that might exist close to the critical point are not of great concern, because none of the disassembly cases presented in CRBRP-GEFR-00103 or CRBRP-GEFR-00523 lead to temperatures anywhere near the critical temperature.

QUESTION II-2

(F6.2-105, par. 2) Each of these assumptions contradicts present understanding of the phenomena and the combination of all three is highly improbable.

ANSWER II-2 (b) and (c)

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-3

(F6.2-106, par. 2) In comments on the proposed final environmental statement - liquid metal fast breeder reactor (with reference to p. 4.2-148 of the PFES-LMFBR), the NRC Staff stated:

"...the bases for concluding that the total energy generated in a series of small power bursts will be no greater than that generated in a single, large, permanently-dispersive burst requires further study. Further, the safety significance of such a conclusion, even if justified, is not clear at this time. For example, further work is required to evaluate the effective mechanical damage from repeated pulses if they do occur."

Generic answers (a) and (b) are not required.

(c) Does the Applicant agree with this conclusion? If not, why not? In either case (if so or if not), how are these considerations treated in the CRBR CDA disassembly analyses? Is it not possible that a series of small power bursts could occur in such a fashion that they would lead to a large reactivity insertion at or near prompt critical and lead to a sustained superprompt critical burst? Is it not possible that the small power bursts could be due to phenomena having space as well as time, assymetries. For example, in such a fashion that axial symmetry, as assumed in VENUS, would be inappropriate for modeling the phenomena? Discuss in detail the basis for the answers to the above.

ANSWER II-3 (c)

CRBR HCDA disassembly analyses have not resulted in predictions of sequences of small power bursts.

A series of small power bursts occurring in such a fashion that they would lead to a large reactivity insertion at or near prompt critical and cause a sustained superprompt critical burst is judged to be highly unlikely.

To obtain such a situation requires making a continuing series of arbitrary assumptions regarding coherency of fuel motion.

It is true that the VENUS-II model does not allow for external treatment of fuel motion. However, the reactivity effects of fuel motion can be estimated and included in the reactivity insertion rate input to VENUS. The resultant energetics are judged to be reasonable estimates, provided the void volume is handled in a conservative manner. (See R. B. Nicholson and J. F. Jackson, "A Sensitivity Study for Fast Reactor Disassembly Calculations," ANL-7952 (1974).)

QUESTION II-4

(P6.2-106, par. 2) In all cases, care was taken to begin the disassembly calculation early enough to ensure that conservative estimates of the energy generated were made in VENUS-II.

Generic answers (a) and (b) are not required.

(c) Is it not true that SAS3A may predict a disassembly ramp rate which is either unreasonable in sign or in magnitude at a specified core fuel temperature (See Blewbis et. al., Proc. of the Fast Reactor Safety Meeting, Beverly Hills , California CONF 740401, p. 1324). How is this consideration taken into account in modeling the transition from termination of SAS3A and/or the "transition phase to VENUS-II? Describe how the VENUS ramp rate is formulated in light of the above.

ANSWER II-4 (c)

SAS3A or SAS3D, in and of themselves, do not predict a "disassembly ramp rate." It is up to the user to decide when the disassembly calculation should begin. With respect to how this is done, a clear description appears on p. II-3 and II-4 of CRBRP-GEFR-00103. The Applicant cannot ascertain how the authors of the referred to paper obtained the driving reactivities; however, the results seem to be at variance with our experience in CRBR calculations.

NOTE: Questions II-5 through II-9 pertain to transition recriticality considerations.

QUESTION II-5

(P6.2-109, par. 3) First, the pressure from vapor generation in the boiling material in the core could tend to levitate the blockage.

(c) Provide a time dependent profile of the pressures above and below the assumed blockage.

ANSWER II-5 (b) and (c)

Detailed calculations of the time dependent pressure above and below the blockage were not performed since a detailed calculational model is not currently available. Qualitatively, the pressures below the blockage will be higher than those above the blockage. Therefore, the blockage will be levitated.

QUESTION II-6

(F6.2-109, par. 3) Any material re-entering the core would most probably not come in as coherent slugs but rather come in gradually as the upper blockages are melted and "washed" out by the boiling turbulence below.

(c) Quantitatively compare the probability of reentry due to gradual melting and washout with the probability of re-entry due to pressure relief downward.

ANSWER II-6 (b) and (c)

Calculations to quantitatively compare the probability of re-entry due to gradual melting and washout with the probability of re-entry due to pressure relief downward have not been performed since an appropriate calculational model is not currently available.

QUESTION II-7

(F6.2-109, par. ` Generic answers (a) and (b) are not required. Material injected into the blankets would tend to have a temperature profile that is steadily decreasing away from the core...

(c) Provide a quantitative assessment of the time dependent temperature profile across the blockage.

ANSWER II-7 (c)

Calculations of the time dependent temperature profile across the blockage were not performed since a detailed calculational model is not available. Qualitatively, the sentence set forth in the interrogatory is supported by the statements which immediately follow it on page 11-11 of CRBRP-GEFR-00103.

QUESTION II-8

(F6.2-110, par. 1) A second reason that a recriticality is unlikely is that the reactivity of the system, after the early part of the transition phase, would very probably be too low for fuel re-entry to return the system to critical.

ANSWER II-8 (b) and (c)

Evaluation of the current design in Sections 8.2 and 8.3 of CRBRP-GEFR-00523 indicate that recriticality may occur,, but large coherent reactivity events are not foreseen.

QUESTION II-9

(F6.2-110, par. 1) The assumption that the remaining inner and outer core fuel is homogenized.

(c) What are the implications in terms of possible ramp rates if it is assumed that the fuel is not homogenized?

ANSWER II-9 (b) and (c)

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

NOTE: Questions II-10 through II-16 pertain to the EOEC LOF immediate re-entry case.

QUESTION II-10

(F6.2-110, par. 4) The assumption that the re-entry can be adequately modeled by limiting the coherent re-entry consideration to the 36 sub-assemblies in the innermost ring of the outer enrichment zone.

(c) How are the control rods modeled in the EOEC LOF Immediate Re-Entry Case?

ANSWER II-10 (b) and (c)

This question appears specific to the homogeous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-11

(F6.2-110, par. 5) Assume that, in 75 percent of the subassemblies in the innermost ring of the outer core zone, blockages form in the lower portion of the upper blanket.

(c) What is the basis for the choice of the location of the blockage in the upper blanket?

ANSWER II-11 (b) and (c)

This question appears specific to the homogeous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-12

(P6.2-328, Figure P6.2-157) Generic answers (a) and (b) are not required.

(c) Why is the slope of the curve discontinuous?

ANSWER II-12 (c)

The curve referred to in this interrogatory is now Figure 11-2 in CRBRP-GEFR-00103 and should in fact be drawn as continuous and smooth.

QUESTION II-13

(P6.2-111, par. 3) ...the inner and outer core fuels are homogenized.

ANSWER II-13 (b) and (c)

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-14

(P6.2-111, par. 3) Some 17 percent of the core fuel is located in the blanket.

ANSWER II-14 (b) and (c)

This question appears specific to the homogeneous core which is not the current design. Accordingly, it requires no answer. Section 8 of CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-15

(P6.2-111, par. 4). The fuel and steel which has been ejected is then postulated to fall out of the blanket in rings two and three.

(c) What is the basis for selecting the sub-assemblies in rings two and three?

ANSWER II-15 (b) and (c)

This question appears specific to the homogeous core which is not the current design. Accordingly, it requires no answer. CRBRP-GEFR-00523 describes the analyses for the current core design.

QUESTION II-16

(P6.2-111, par. 4) Generic answers (a) and (b) not required.

(c) What is meant by "interpenetrate" in the phrase

"The material is allowed to interpenetrate the material already in the core"? Describe in detail the geometry of the materials involved and the manner of penetration.

ANSWER II-16 (c)

"Interpenetrate" means that, when the neutronic calculations were performed, the fuel was assumed to be filling space that would otherwise be filled by a liquid fuel-steel mixture. That is, in the regions where the

fuel was postulated to the re-entering, the fuel density was increased; thus, the neutronics calculations reflect the change in flux shape due to this effect.

FIFTH INTERROGATORY SET

PREAMBLE TO QUESTIONS

In previous interrogatories information was requested concerning four distinct validations relative to the models and computer codes used in the analysis of CRBR CDAs, namely:

i) Validation that the code's output is the correct numerical calculation that should result from a given set of input data and the model assumptions;

ii) Validation of the models against actual experimental data;

iii) Validation that the models can be extended to the CRBR; and

iv) Validation that the input assumptions for the CRBR case are adequate with respect to the CDA analysis, i.e., are supported by experimental evidence. By "adequate", here and below, we mean that the calculations will not underestimate the CDA work potential (i.e., forces and resulting energetics of a CDA) or overestimate the containment capability of the reactor with respect to a CDA.

With respect to the following requests for information we are concerned with these same four validations relative to the radiological source term and site suitability analysis.

QUESTIONS I

With respect to each of the following codes and each subroutine of each of the following codes:

(A) COMRADEX - II

(B) HAA - 3

please provide the following information [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.]:

1) Complete, current documentation (i.e., a writeup) of the codes and the subroutines;

2) Identify, by name and affiliation, the author, or authors, of each model, subroutine, or portion of each subroutine, which each contributed or worked on;

3) Identify by name affiliation (including organization, division, branch, title, etc.) each applicant employee, or consultant, that has intimate working knowledge of the code and each subroutine, or parts thereof, including its validity. Where more than one person is involved, delineate which portion of the code or subroutine with which each has an intimate working knowledge;

4) Describe fully the procedures by which Applicant has assured itself and continues to assure itself, that the various computer programs (codes) accurately reproduce the models as described in the PSAR and its references (see Validation (i) above);

5) Indicate which models (including subroutines, or portions of subroutines) have not been validated as described in Validation (i);

6) Indicate the models (including subroutines, or portions of subroutines) or assumptions that have not been validated as described in Validation (ii);

7) For each model, portion of the model, or assumption that has been validated (against experimental) or other) data, see Validation (ii) above) describe fully the procedure by which it was validated, and the results, including all uncertainties and limitation of the validation. Indicate the source of the experimental, or other data, that was used in the validation.

8) Explain fully all instabilities in the numerical performance in the models, what causes them, and how they are avoided, and the extent to which this introduces uncertainties in the calculations and limits the validity of the model (cf., p. F6. 2-10 par. 2).

9) To the extent that any answers to the above questions are based on referenced material not previously provided, please supply the references.

10) 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.

11) 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.

ANSWER I(A)

The information provided in these answers pertains to the COMRADEX-III Computer Code, which is used for site suitability source term analysis.

(1) Reference 1 on page A-19 of the PSAR is the current COMRADEX-III documentation.

(2) The information is available in the documentation listed in Response 1 above.

(3) The reference in Response 1 also identifies the contributors to, and the supervisors responsible for, code and subroutine development.

(4) Independent hand calculations were made to check intermediate calculations of the Code. Simpson's Rule integration routines were checked by comparison with calculations which can be reproduced analytically.

(5) All COMRADEX-III programming has been checked as indicated in (4) above to verify that the code performs the correct numerical calculations.

(6) None.

(7) All models used in COMRADEX-III are based on first principles. Reference (2), page A-19 of the PSAR provides additional discussion on this matter.

(8) Program instabilities could result if the time-step size were too large. The code selects the time-step according to predetermined criteria which assures stability.

(9) References mentioned in items (1) and (7) have been or will be made available for inspection and copying.

(10) No further research work of Applicant in this area has been identified.

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

ANSWER I (B)

The information provided in these answers pertains to the HAA-3B computer code, which is used for site-suitability source term analysis.

(1) Reference 1 on page A-140 of the PSAR is the current HAA-3B documentation.

(2) The information is available in the documentation listed in Response 1.

(3) The reference in Response 1 identifies the contributors to, and the supervisors responsible for, code and subroutine development.

(4) The HAA-3B code, including its subroutines, has been thoroughly checked to assure that the numerical algorithms in the HAA-3B code have been programmed correctly. In addition, test cases were performed to assure that the code could reproduce previously calculated results.

(5) All HAA-3B programming has been validated as in (4) above.

(6) None.

(7) References 1 through 6 on page A-140 of the PSAR contain detailed descriptions of the validation of the HAA-3B code.

(8) The integro-differential equation of the model in HAA-3B is solved by applying the moment method and using a log-normal particle distribution which results in three, simultaneous, first order differential equations. In some analyses, more time steps are required to cover the time interval desired than are allowed by array dimensions. In these cases it is necessary to restart the code to continue the calculation. Under certain conditions, following a restart, a slight input parameter manipulation is required to achieve continuity. This does not impact the accuracy of the results.

(9) References mentioned in Items (1) and (7) have been or will be made available for inspection and copying.

(10) No further research work of Applicant in this area has been identified.

(11) 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 II

With respect to the following request for information we are concerned primarily with the fourth validation — validation that the input assumptions for the CRBR case are adequate with respect to the source term and site suitability analysis. Here we are not so much concerned with the validity of the model expressions as with the uncertainties in the site boundary doses due to propagation of uncertainties in a) the parameters used and b) the model input data and due to any synergisms among these uncertainties and the model assumptions.

With respect to the calculations identified below, under (A) through (C), please provide the following information [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.]:

1) List and identify all model input data (exclusive of coding flags and inputs that specify coding options, criteria, printout formats, etc.) and all model parameters that come into play in each of the models utilized in the site suitability radiological analysis calculations, including but not limited to input data and parameters in COMRADEX-II and HAA-3. Exclude parameters not called into use because a subroutine, or part thereof, was not utilized.

2) Describe in detail the basis for the choice of each input datum and model parameter listed above.

i) In each case quantify the uncertainty in the value selected;

ii) In each case indicate whether the value is based on first principles, experimental measurements, unvalidated hypothesis, output of other models, arbitrary assumptions, etc.;

iii) In each case indicate whether the choice of the input datum or model parameter was selected to represent the "best estimate", or a bounding or "conservative value", where "conservative value" here means a value chosen so as not to underestimate the accident consequences, e.g., site boundary and low population zone radiological doses.

3) For each input datum and model parameter with uncertainty listed in 1) above, indicate in quantitative terms the magnitude of the uncertainty introduced into the final calculations of the site boundary 2-hour and the low population zone accident duration doses, respectively, due to the uncertainty in the input datum or model parameter. In addition, discuss in detail any synergistic effects resulting from combinations of uncertainties in the input values, model parameters, and model assumptions. In each case discuss the basis for the estimate of how the uncertainties propagate, e.g., include and discuss all parametric analyses used to test the effect of uncertainties.

4) Identify by name affiliation (including organization, division, branch, title, etc.) each applicant employee or consultant that has intimate working knowledge of the basis for the selection of the parameter or input datum.

5) To the extent that any answers to the above questions are based on referenced material not previously provided, please supply the references.

6) Explain whether Applicants are presently engaged in or intend to engage in any further research or work which may affect Applicant's 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.

7) 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.

(A) The Reference Design site suitability source term dose analysis summarized in Table 15.A.3-5 (white pages)

(B) The Parallel Design site suitability source term dose analysis summarized in Table 15.A.3-4 of Appendix F, Part II (yellow pages)

(C) Any subsequent site suitability source term dose analyses based on source terms (and other parameters) recommended by the NRC Staff.

ANSWER II (GENERAL)

The Applicants have a single design as described in the PSAR. The site suitability source term used by the Applicants in Section 15.A of the PSAR is consistent with that recommended by the NRC Staff. Consequently, it is not necessary to provide separate responses to Parts (A), (B) and (C).

Responses to the list of questions are provided first for COMRADEX-III and then for HAA-3B in Answers II(A) and II(B), below.

ANSWER II(A)

The information provided in these answers pertains to the COMRADEX-III Computer Code.

1) The following input data were used in COMRADEX III calculations for the Site Suitability Source Term dose analysis:

Input release fractions of the core inventory of radioactive isotopes released to the RCB as discussed in Section 15.A.1 of the PSAR are:

Fuel Material	1.0%
Solid Fission Products	1.0%
Halogens	50%
Noble Gases	100%

The source term is hypothesized as a bounding core related release. Time dependent clean-up factors used (calculated by the HAA-3B computer code) are shown in Table 15.A-6 of the PSAR. The leak rate from the RCB used is the design basis leak rate of 0.1% vol/day for the duration of the evaluation.

2) The core inventory release fractions listed in (1) above were utilized in compliance with specific direction by the Nuclear Regulatory Commission (see Reference 2 on page 15.A-9 of PSAR), and are considered by the Applicant to be highly conservative. The RCB leak rate utilized is based on an RCB pressure of 10 psig, which is also considered to be a conservative value since no design basis accidents result in pressures approaching 10 psig.

3) The data and models used to calculate the site boundary 2-hr and the low population zone accident duration doses were chosen to yield upper bound values or conservative doses. Items 1) and 2) provide additional information concerning the bounding analyses and conservative set of assumptions utilized in the site suitability assessment.

4) Development of the input parameters was done under the supervision of L. E. Strawbridge, Manager, Nuclear Safety and Licensing, Westinghouse Advanced Reactors Division.

5) None

6) No further research work of Applicant in this area has been identified.

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

ANSWERS II(B)

The information provided in these answers pertains to the HAA-3B computer code.

1) The following input data were used in the Site Suitability Source Term dose analysis to compute depletion factors as described in 15.A.2.2 of the PSAR:

Source Term Attenuation Within Containment

Initial particle number Concentration of Aerosols (particles/cm^3)	XIN(1) = 1.337×10^8
Aerosol Volume Variance	SIGAIR = 8.000
Aerosol Mass Mean Volume (m^3)	VAIR = 1.000×10^{-3}
Density of Aerosol Material (g/cm^3)	RHO = 10.55
Viscosity of Air (dyne sec/cm^2)	VISC = 2.264×10^{-4}
Temperature (^oK)	TEMP = 3.940×10^2

Diffusional Boundary Layer Thickness	DELTA $= 4.000 \times 10^{-5}$
α	ALPHA $= 1.000 \times 10^{-1}$
ε	EFF $= 1.000$

2) The initial particle number concentration and the density of aerosol material were based on the initial airborne mass concentration and the effective density, respectively of the mass associated with the RCB non-gaseous source term species. The selection of aerosol volume variance and mass mean size was based on the experimental measurement summarized in Reference D-7 on page D-6 of CRBRP-3 Volume 2. The air viscosity used is based on the RCB design temperature, which is conservatively assumed to be the atmospheric temperature for this analysis. The value of $\alpha\,\varepsilon$ was calculated by a conservative linear extrapolation measure which was based on the experimental data reported in Reference D-1 on page D-6 of CRBRP-3 Volume 2.

3) The data and models used to calculate the clean-up factors which were used to calculate the site boundary 2 hr. and the low population zone accident duration doses were chosen to yield conservative doses.

4) All questions regarding the solution of parameters or input datum should be referred to L. E. Strawbridge, Manager, Nuclear Safety and Licensing, Westinghouse Advanced Reactors Division.

5) References from Item 2 above have been or will be made available for inspection and copying.

6) No further research work of Applicant in this area has been identified.

7) 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 III (GENERAL)

Request for the following information is based on our concerns with respect to Validation (iii) and (iv) above. In the Applicant's answers to the generic questions (b) and (c) below, the Applicant is requested to be responsive to these concerns.

With respect to each statement, assertion or assumption (from 15.A in Part II of Appendix F of the PSAR) identified below, please provide the following information (unless noted otherwise). (NOTE: The following numbered Interrogatories are identified by the page and/or paragraph number from the PSAR in parentheses.) [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) Identify by name, title and affiliation the primary Applicant employee(s) or consultant(s) that has the expert knowledge required to support the statement, assertion, or assumption.

b) Describe in detail the supporting evidence for the statement, assertion, or assumption and where appropriate the rationale for the approach taken.

c) Provide any additional information requested following each statement, assertion, or assumption.

d) To the extent that any answers to the above questions are based on referenced material, please supply the references.

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.

ANSWER III (GENERAL)

The questions in this part (III) of this fifth set of interrogatories are based on statements, assertions and assumptions in Section 15.A of Part II of Appendix F of the PSAR, as it existed in 1976. Appendix F in its entirety has been deleted from the PSAR and the Parallel Design is not part of the license application. Hypothetical Core Disruptive Accidents are discussed in detail in CRBRP-3, Volumes 1 and 2. Because of the changes since 1976, Questions 1 through 5 and 6c are no longer applicable. The responses below are, therefore, limited to questions 6 and 7.

The following responses a, d, e, and f are identical for interrogatories 6 and 7:

(a) The work in this area was performed under the supervision of L. E. Strawbridge, Manager, Nuclear Safety and Licensing, Westinghouse Advanced Reactors Division.

(d) The referenced documents have been or will be made available for inspection and copying.

(e) No further research work of Applicant in this area has been identified.

(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 III-1

(15.A-2, par. 2) Define precisely the first containment barrier boundary which includes the reactor cavity and the SHAA.

ANSWER III-1 (b) and (c)

This question is no longer applicable; refer to the general response (above).

QUESTION III-2

(15.A-2, par. 2) Define all potential leakage paths from the first containment boundary.

ANSWER III-2 (b) and (c)

This question is no longer applicable; refer to the general response (above).

QUESTION III-3

What are the design leakage rates for each of the leakage paths identified in (3), and what is the design leakage path for the first containment barrier defined in (2)?

ANSWER III-3 (b) and (c)

This question is no longer applicable; refer to the general response (above).

QUESTION III-4

Describe in detail the test program that will demonstrate that the leakage in (4) does not exceed the design leakage as required by 10 CFR 50, Appendix J.

ANSWER III-4 (b) and (c)

This question is no longer applicable; refer to the general response (above).

QUESTION III-5

(15.A-7, par. 2) Some of the volatile fission products may escape to the RC during the transition of the fuel from the vessel to the core catcher but this will not affect the results significantly since these will contribute to the equilibrium concentration which will be established between the isotopes in the sodium and in the atmosphere.

ANSWER III-5 (b) and (c)

This question is no longer applicable; refer to the general response (above).

QUESTION III-6

(15.A-9, par. 3) The use of the containment design leak rate (0.1% Vol/Day) for the duration of the site suitability source term evaluation is conservative.

(c) What is the basis for assuming the core catcher will work?

ANSWER III-6 (b) and (c)

The conservatism in the use of the containment design leak rate for the duration of the site suitability source term is addressed in the response to Question II-2 of this Fifth Set of Interrogatories as it relates to the COMRADEX Code.

Part (c) of this question is no longer applicable; refer to the general response (above).

QUESTION III-7

(15.A-10) Generic answers (a) and (b) not required here.

(c)(i) Were the F-factors (rem/curie) utilized in the site suitability source term radiological dose evaluations the same as those presented in Table 7.1-3 (p. 7.1-47) of the ER?

(ii) Were these same F-factors utilized as a basis for excluding the effects of certain transuranium isotopes from the site source term calculations? Which isotopes were excluded?

(iii) Are we correct in assuming that References 5 through 8 cited in Table 7.1-3 of the ER are the sole references used to determine the F-factors? If not, please identify and supply all references used to determine the F-factors.

(iv) In reviewing the literature relative to the F-factors, were not some references encountered that suggested higher F-factors? If so, explain in detail the basis for rejecting these higher values for the F-factors?

(v) There are many fission and activation products in the EOEC radionuclide inventory of the CRBR. Upon what basis have each and every of these radionuclides been included or excluded from the dose calculations? Explain this basis in detail by including all relevant information such as

inventory, activation cross-section and data related to F-factors. In answer to this question, please be responsive to (iv) above.

ANSWER III-7

(c)(i) The F-factors utilized in the site suitability source term radiological dose analysis are based on References 2, 3 and 4 on page 15.A-9 of the PSAR. The F-factors in Table 7.1-3 of the ER are based on Reference 3 above, which is NUREG-0172.

(c)(ii) The basis for excluding the effects of certain transuranium isotopes, the calculational procedure used, and the particular transuranium isotopes excluded were identified in detail during the January 22, 1976 meeting between the CRBRP Project and the NRC Staff in Bethesda, Maryland. The information presented by the Project showed that excluding the transuranium isotopes resulted in underestimating the potential bone and lung doses by only 4% and 3%, respectively. A detailed summary of the information presented by the Project is provided in "Summary of Meeting with CRBRP Representatives" prepared by the NRC Staff, dated February 2, 1976.

(c)(iii) The references identifed in Item (i) above are the sole references used to determine the F-factors.

(c)(iv) No specific literature search was conducted with the objective of finding the highest F-factors reported in the literature. The major F-factor references used are part of documentation provided by the NRC for guidance in doing dose analyses.

(c)(v) Two bases are generally used to exclude particular isotopes from dose calculations. The first is a consideration of the isotope half-life. Isotopes with very short half-lives (<1 minute) necessarily undergo rapid radioactive decay and it is physically unrealizable for such isotopes to be released from containment barriers and be transported off-site prior to their decay to insignificant activity levels. The second basis consists of a comparison of a particular isotope's inventory and potential biological hazard (measured by its Maximum Permissible Concentration per 10CFR20 or

alternately by its F-factor for a particular body organ) to the inventory and biological hazard associated with some controlling radioisotope, i.e., the radioiosotope whose inventory and biological hazard results in it being a major dose contributor to a particular body organ. For example, the biological characteristics of Iodine coupled with its inventory following irradiation result in it being the most significant, by a substantial margin, contributor to thyroid exposure. Thus by comparing the biological characteristics and inventory of a particular isotope to Iodine, it may be shown that this particular isotope can be neglected when computing exposures, without significantly underestimating these potential exposures. This procedure can be implemented for any isotope and organ as, for example, was the case for the transuranium isotopes discussed in Item (ii) above.

Note that the COMRADEX computer code, used for the evaluation of the site suitability source term, includes the dose contributions of over 100 individual isotopes. A substantial number of these individual isotopes could be eliminated from the dose analysis without significantly affecting the results. However, this large number of isotopes is retained for completeness and only those isotopes, based on sound engineering judgment in conjunction with the procedures discussed above, that are clearly negligible are eliminated from the dose analysis.

QUESTION

The sixth interrogatory set requests additional information beyond that supplied by the Applicants in the responses to the second interrogatory set.

QUESTION I(A)(1)

In response to Interrogatory I(A)(1) [of the Second Set of Interrogatories to Applicant], the Applicant indicated SAS3A evolved from the SAS2A code which evolved from the SAS1A code, and these are documented in References, 1, 22, 29, 30 and 31 on page F6.2-119, F6.1-120, and F6.2-121 of the PSAR.

Separately, for the main routine, the drive routines and for each subroutine identified in Figure 2 (p.26) of ANL/RAS 75-17 (PSAR, Ref. 1, p.F6.2-119) please provide the following information:

(a) Indicate whether the routine appears in
    (i) SAS1A
    (ii) SAS2A

(b) Where the routine appears in more than one code, including SAS3A (e.g., in SAS2A and SAS3A) indicate whether the routine is exactly the same in each code, or whether coding changes have been made but the name of the routine remained unchanged;

(c) Where coding changes have been made describe fully the changes that were made, why they were made and precisely where these changes are documented in the references identified above.

[NOTE: Interrogatory I(A)(1) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-3 through AA-6.]

ANSWER I(A)(1)

Parts (a) - (c) of this interrogatory concerning development of the SAS3A and SAS3D code are being responded to as a whole in the form of an appropriately-footnoted table. Before explaining the form and substance of the table, a general comment must be made concerning the evolution of the SAS3A and SAS3D code. As stated, this interrogatory appears to be based on the concept that the SAS2A code evolved from SAS1A in some incremental fashion and that SAS3A evolved from SAS2A in the same fashion. On the one hand it is true that perhaps a few thousand FORTRAN language statements from the original SAS1A coding still survive intact in SAS3A. However, it is neither useful nor instructive to think of SAS3A as having evolved from SAS2A and SAS2A from SAS1A in the sense of SAS3A being SAS2A plus a few model improvements plus minor model additions. In fact, only the most fundamental of modeling concepts (e.g., modeling a collection of like subassemblies by an average pin, using point kinetics, treating only radial heat transfer in the pins, etc.) have survived intact from SAS1A into SAS3A. Both when it was decided to develop SAS2A and then to develop SAS3A, the basic approach was to make a fresh start at determining what phenomena relevant to LMFBR accident scenarios would be modeled and then what would be required to model these phenomena to a sufficient level of detail. Only after giving these a great deal of consideration did the developers return to the previous code to determine what parts could be lifted reasonably intact and used in the new code. On the other hand, the SAS3D code was developed directly from SAS3A using the same physical models; the major differences between SAS3A and SAS3D are in the treatment of the data management to allow greater geometric modeling flexibility (i.e., more channels) and in reprogramming to obtain better efficiency.

In the table which follows, the relationship of the subroutines in SAS3D to SAS3A, SAS2A and SAS1A is explained in a five-column format. Column 1 of the table simply lists the SAS3D subroutines, one line per subroutine (including the MAIN routine). In Column 2, an "x" appears on the same line if a subroutine of the same name (but consisting of FORTRAN statements which may or may not be similar to the statements in the like-named SAS3D subroutine) also was found in SAS3A. A hyphen appears in Column 2 on that

line if a subroutine with that name did not appear in SAS3A. Column 3 uses the same notation to relate SAS3D subroutines to SAS2A. Column 4 uses the same notation to relate SAS3D subroutines to SAS1A. Three exceptions to this notation occur in Column 2 and 3 and need further clarification. Subroutines INPOT1 and INPOT2, the input and editing subroutines in SAS3A, contain some coding similar to that found in subroutine INPOUT in SAS2A and an identically-named subroutine in SAS1A. Coding found in subroutines COOLFB and FUELFB was lifted to form a part of subroutine FEEDBK in SAS2A, and a like-named routine appears in SAS3A. Likewise, a part of the coding from subroutine TSCOOL in SAS1A went into subroutine TSC1 in SAS2A. When going from SAS3A to SAS3D, a number of subroutines were split into separate routines, mainly to improve code portability. These splits are indicated in the table. In Column 5 of the table, the numbers found on the line opposite a particular SAS3D subroutine indicate that the particular footnotes identified by those numbers and found at the end of the table apply to that subroutine.

There are only ten subroutines in SAS3A which are identical to subroutines in SAS2A and/or SAS1A. These subroutines are CHIN, FISGAS, FITZ, PIPFLO, PREA, SHAPE, SSPK, THERCO, TSCB AND TSPK. Since SAS3A can only be considered conceptually as being an entirely new code which "borrowed" collections of FORTRAN statements from SAS2A, it is not meaningful to discuss "changes" that were made to SAS2A to obtain SAS3A. Thus, these "changes" have never been specifically documented as such. Rather, the SAS3A documentation (which currently consists of parts or all of several reports, as indicated in the response to Part (1) of Interrogatory I(A) of the Second Set of Interrogatories) provides a description of the overall conceptual framework of the code and its numerous models, presents the formulae which form the mathematical basis of each model, and discussed briefly the algorithms used to solve these equation sets. A great deal of the algorithmic detail of the models is left for the interested reader to discover by physically examining the FORTRAN coding comprising the models.

The programming changes made in generating SAS3D from SAS3A were carried out in order to achieve the following goals: a) to modify the way problem data were stored to allow greater flexibility in channel specification, b)

to update the FORTRAN code to bring it into consistency with standard practices, and c) to improve the performance (speed) of the overall code. In the process of generating SAS3D from SAS3A, programming changes were validated by demonstrating that the intermediate versions of SAS3A and SAS3D could produce computationally identical results. In addition, some subroutines were split into multiple subroutines and FORTRAN source changes were made to promote consistency with accepted standard programming practices and to allow code exportability. These changes may be seen in the FORTRAN code. A number of subroutines were added to provide the new data management stragegy.

Relationship of SAS3D Subroutines to SAS3A

SAS2A and SAS1A Subroutines

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
MAIN	X	X	X	1
ADITIT	—	—	—	13
AXMESH	--	—	--	13
BLENDR	—	—	—	13
BLENDZ	--	--	--	13
BLOWUP	X	—	—	6
CAVINT	X	--	--	7
CHIN	X	X	—	1
CLADIT	—	--	--	13
CLAZAS	X	—	—	5
CMCOPY	--	--	--	14
CONFIS	X	—	—	6
COOLIT	--	--	--	13
CROEF	—	—	—	17
CROFUL	X	--	--	10
DATMOV	—	—	—	14
DATSET	--	--	--	14
DEFORM, DEFINT, BOUND	DEFORM	DEFORM	DEFORM	1
DTFND, DTFCHN	DTFND	DTFND	DTFND	1,6,7
DNYNALL	--	—	--	14

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
EDITIT	--	--	--	13
EKT	--	--	--	17
EQCLAZ	--	--	--	13
EQCOLC	--	--	--	13
EQCOMC	--	--	--	13
EQFCIC	--	--	--	13
EQFUEL	--	--	--	13
EQHEAT	--	--	--	13
EQSLUM	--	--	--	13
EQTBAR	--	--	--	13
EQZFI	--	--	--	13
ERRORS	--	--	--	17
EXCHAN	--	--	--	13
FALLDN	FALLDN	--	--	6,16
FALL2				
FALL3				
FALL4				
FALL5				
REVEL				
FBKCLZ	FEEDBK	X	COOLFB, FUELFB	1,5,6, 7,16
FBKCOL				
FBKFCI				
FBKHET				
FBKSLM				
FISGAS	X	X	--	2
FITZ	X	X	X	1

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
FK				
CRO				
CPF				
PROGRE				
RHOS				
RHOL	K	K	K	1,16
SCRAM				
YELDPT				
YOUNG				
FUAREA	X	—	—	6
FUELCI	—	—	—	13
FUELIT	—	—	—	3
GEOEDT	—	—	—	13
GETRDY	X	—	—	6
INCHEK	—	—	—	13
INEDIT	—	—	—	13
INPOTO				
INPOT1	INPOT1			
INPOT2	INPOT2	INPOUT	INPOUT	5,6,7,16
INPT21				
FCIO				
TBSCAN				
FOUT				
IOUT				
AOUT				
DATOUT				
SETC				
INTERP	X	X	X	1
INTIRP	X	X	—	1
INTRP	X	—	—	10
KFUEL	X	—	—	10
KMCOPY	—	—	—	14

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
LINES	—	--	--	15
LOCHEX	—	—	—	13
MAPDRV	--	--	--	13
MOVPK2	—	—	—	14
NAPRES	X	--	--	6
OVERLAY	—	—	—	16
OVLY61	—	--	--	13
OVLY62	—	—	—	13
OVLY63	--	--	--	13
OVLY64	—	—	—	13
OVLY65	--	--	--	13
PIPFLO	X	X	—	2
PNORMZ	--	--	--	13
POINST	—	—	—	14
POWADJ SLUMP2	SLUMPF	--	--	6
POWREA	--	--	--	13
PREA PROGRE	PREA	PREA	PREA	1
PRESS	X	--	--	6
PRFCI	X	—	—	7
PRIMAR SSPRIM PRIMDT	PRIMAR	PRIMAR	--	2,9
READEC	--	--	--	14
READI	—	—	—	14
READIN	X	X	X	1
REED	—	—	—	14
REGMAP	--	--	--	13
RESTAR	X	X	—	3
REZONE	X	--	--	6

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
RHOL RHOLFL	RHOLFL	--	--	10
RHOS RHOSFL	RHOSFL	--	--	10
RNGPOS	--	--	--	13
RPOWRE	--	--	--	13
SASFCI INTGRL FCIZ	SASFCI	--	--	7,16
SETFLT	--	--	--	13
SETINP	--	--	--	13
SETINS	--	--	--	13
SETINT	--	--	--	13
SHAPE	X	X	X	1
SLUMP2 TSOV46	SLUMPF	--	--	6,16
SSCOOL	X	X	X	1
SSDRIV	X	X	X	1,11
SSFUEL	X	--	--	11
SSHTR	X	X	X	1,10,11
SSPK	X	X	X	1,12
SSPRNT	X	X	X	1,11
STATUS	--	--	--	14
TEMPER	--	--	--	13
TEMPUL	X	--	--	6
THERCO	X	X	--	2
TLEFT TELAPS	TLEFT	TLEFT	TLEFT	
TRIGBO	--	--	--	13

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
TSCA	TSCA	TSCA	--	2,9
PRIMUP				
TSCB	X	X	--	2
TSCC	TSCC	--	--	8
TSCC2				
TSCC3				
TSCC1	M7	--	--	8
TSC2	X	X	—	2
TSC3	X	X	--	2
TSC31	HCON	HCON	—	2
TSC4A	TSC4	TSC4	--	3,5,6,7, 8,16
TSC4B				
TSC41A				
TSC41B				
TSC42A				
TSC42B				
TSC43A				
TSC43B				
TABFIS				
TSC5	X	X	—	2,8
TSC6	X	X	—	2
TSC7	X	X	--	2
TSC8	TSC8	TSC8	--	2,7,16
TSC82				
TSC83				
TSC84				
TSC85				
TSC86				
TSCBUB				
TSC9	X	X	—	2

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
TSDRIV				
FAILUR				
SCRAM	TSDRIV	TSDRIV	TSDRIV	1,2,6,7,
TSOV30				9,16
TSOV50				
INVERP				
TSHTR				
FUELTP	TSHTR	TSHTR	TSHTR	1,5,6,7,16
TSOV40				
TSOV41				
TSOV42				
TSOV43				
TSOV44	TSC1	TSC1	TSCOOL	1,2,7,16
TSOV45				
TSOV47				
TSCINT				
TSCSET				
TSPK	X	X	X	1,12
TSPLOT				
TSPRNT				
PRINT2				
PRINT3	TSPRNT	TSPRNT	TSPRNT	1,5,6,7,8,
PRINT5				9,16
SSPLOT				
FORGAS				
SHORT				
TWOGEO	--	--	--	13
VELSCN	X	--	--	6
VFCHAN	--	--	--	13
VFRITE	--	--	--	13
VOIDIT	--	--	--	13

SAS3D Subroutine Name	Found in SAS3A	Found in SAS2A	Found in SAS1A	Applicable Footnotes
WGEODS	--	--	--	13
WRAPIT	--	--	--	13
WRITEC	--	--	--	14
WRITEI	--	--	--	14
XSRITE	--	--	--	13
ZAPPA	--	--	--	14
ZAPPF	--	--	--	14
ZBLEND	--	--	--	13
ZCORE	--	--	--	13
--	--	VENINP	--	4
--	--	SINP	--	4

Footnotes to Table:

1. The Multichannel concept, as described in Sec. I of ANL-8138, "The SAS2A LMFBR Accident Analysis Computer Code," by F. E. Dunn, et al., required that a channel subscript be added to a number of the FORTRAN arrays which were lifted from SAS1A. Subroutines from SAS2A noted with this footnote contained pieces of coding lifted from SAS1A which referenced some of these arrays.

2. These subroutines were added or were extensively modified in implementing the new voiding model in SAS2A, as described in Sect. I.B. of ANL-8138.

3. These subroutines provide the restart capability for their respective codes, as described in Sect. III.F.9 of ANL-8138 for SAS2A and in Sect. V.F.1 of ANL/RAS 75-17.

4. These subroutines provide direct coupling between SAS2A and the VENUS-II code. This capability is not operational in SAS3A.

5. These subroutines were coded new or were modified from their SAS2A form in implementing the CLAZAS clad motion model in SAS3A.

6. These subroutines were coded new or were modified from their SAS2A form in implementing the SLUMPY fuel motion model in SAS3A.

7. These subroutines were coded new or were modified from their SAS2A form in implementing the SAS/FCI fuel-coolant interaction model in SAS3A.

8. These subroutines were coded new or were modified from their SAS2A form in implementing the sodium film motion model in SAS3A.

9. These subroutines were coded new or were modified from their SAS2A form in implementing the PRIMAR-2 primary loop model in SAS3A.

10. These subroutines were added or modified from their SAS2A form to improve computational speed, as described in Appendix A of ANL/RAS 75-17.

11. These subroutines were added or modified from their SAS2A form in implementing the steady-state fuels categorization model, as described in Sect. III.B of ANL/RAS 75-17.

12. These subroutines were added or modified from their SAS2A form in implementing the decay heat treatment, as described in Sect. III.A of ANL/RAS 75-17.

13. These subroutines were added to SAS3D for the steady-state neutronics coupling.

14. New routines added to SAS3D for data management.

15. Routines added or modified in SAS3D to provide improved computational speed or additional printout.

16. Routines added or modified in SAS3D for improved code portability.

17. Additional option in SAS3D.

QUESTION I(A)(2)

In response to Interrogatory I(A)(4) [of the Second Set of Interrogatories to Applicant], the Applicant indicated

> The entire SAS3A code, including all subroutines, has been checked and rechecked to assure that the numerical algorithms which are implemented in SAS3A to solve the equation sets which constitute the SAS3A code as well as with the specific model being added to assure that these numerical algorithms, both individually and collectively, behave in a stable fashion and produce accurate solutions to the original equation sets. This was carried out by comparing SAS3A results with the output from other codes, with the results of hand calculations, and with what sound engineering judgment deemed to be physically reasonable.

Separately for each routine identified in Figure 2 (p. 26) of ANL/RAS 75-17 (PSAR, Ref. 1, p.F6.2-119), please provide the following information:

(a) Was the routine verified by comparison with other codes, or by comparison with the results of hand calculations, or by comparison with what sound engineering judgment deemed to be physically reasonable?

(b) If the routine was verified by comparison with other codes, how was the other code or codes verified? Identify the other code or codes.

(c) If the answer to (a) or (b) above is that the routine was verified by hand calculations, please supply the hand calculations or the appropriate documentation, i.e.,

(i) the name(s) of the individual(s) who performed the calculations and made the comparison; and

(ii) the laboratory notebook, memorandum or other written record that documents the comparison.

(d) If the answer to (a) or (b) above is that the routine was verified by comparison with what sound engineering judgment deemed to be physically reasonable, please describe in detail the nature of and basis for the engineering judgment. In addition, supply:

(i) the name(s) of the individual(s) who rendered the judgment and made the comparison; and

(ii) the laboratory notebook, memorandum or other written record that documents the comparison.

(e) Did the author(s) of the models actually perform the coding? If not, identify the programmer(s).

[NOTE: Interrogatory I(A)(4) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-3 through AA-6.]

ANSWERS I(A)(2)

Parts (a), (b), (c), and (d) of this interrogatory are being responded to as a whole. Any checkout of new or extensively-modified coding does not generally proceed on a subroutine-by-subroutine basis. Rather, it is carried out at the very least on a model-by-model basis, where each model (fuel motion, clad motion, coolant dynamics, etc., in the case of SAS3A and SAS3D) could consist of a number of whole subroutines plus parts of others (where it is coupled to the rest of the code). The collection of subroutines comprising one of these models is generally referred to in the SAS vernacular as a module. Thus, in the case of SAS3A and SAS3D as in the case of many other large-scale codes, the checkout proceeded on a module-by-module basis.

Comparisons of the output of SAS3A and SAS3D modules and the entire code with the output of other codes, with simple hand calculations, and with what engineering judgment deemed to be reasonable have been and continue to be carried by the model and code developers. However, except as explained in the next paragraph, such efforts are not formally or informally documented.

The documentation that exists is in the form of the references provided in the response to interrogatory I(A)(1) of the Second Set of Interrogatories. These ANL reports serve to document the mathematical bases and provide a broad overview of the computational algorithms associated with each of the

models and the code as a whole. It is implicit in the publication of these reports that the authors have satisfied themselves that the FORTRAN programming in the code is correct.

(e) It is standard practice within the Accident Analysis Section of the Reactor Analysis and Safety Division of Argonne National Laboratory that the authors of the SAS3A and SAS3D models, as identified by the authors listed in the documents referenced in the above paragraph, do their own coding and subsequently actually perform or directly supervise any subsequent modifications to that coding.

QUESTION I(A)(3)

How does the Applicant continue to assure itself that the overall code and its subroutines accurately reproduce the models as described in the PSAR and its references?

ANSWER I(A)(3)

The Applicant continues to assure itself that the overall code and its subroutines accurately reproduce the models as described in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 and their references by careful inspection of the output results for every case analyzed and by comparison of the output results for each case analyzed with the results of previous cases which are similar in part or in whole to the particular case analyzed. In addition, the computer system messages are checked to assure that the job was properly executed, without error, by the computer system.

QUESTION I(A)(4)

Please identify and provide all Intra-Laboratory Memoranda generated by personnel in the Accident Analysis Section, the Coolant Dynamics Section and other Sections of the ANL Reactor Analysis and Safety Division that critique or otherwise evaluate the models developed by other personnel in

these respective sections, limited to the development of any and all models and subroutines that are used in SAS3A (i.e., routines identified in Figure 2, p.26 of ANL/RAS 75-17). Also provide all subsequent memoranda that are responses to criticisms or evaluations identified above or that represent a continuation of the dialogue related to the model evaluation.

ANSWER I(A)(4)

See schedule of documents "Applicants' Response to NRDC Interrogatories" dated August 30, 1976. The files and documents have been and will be available for inspection at the Argonne National Laboratory and provisions have been and will be made for copying. The schedule of documents is being updated and the update will be furnished upon completion. Documents referred to in the update will be available for inspection and copying at Argonne National Laboratory.

QUESTION I(A)(5)

Please identify pertinent sections of all ANL policy and procedures manuals that discuss policies and procedures related to validation of models and codes, including those codes that are part of the ANL library.

ANSWER I(A)(5)

The pertinent sections of ANL Policy and Procedures Manuals are the following:

(1) Sections II-3.0 and II-4.0 of the Reactor Analysis and Safety Policy and Procedures Manual, For Trail Use Only, dated March, 1972 (applicable to SAS3A, SAS3D, VENUS-II, and PLUTO 1) and Sections II-3.0 and II-4.0 of the Reactor Analysis and Safety Policy and Procedures Manual dated May, 1979 (applicable to PLUTO 2 only).
(2) The Quality Assurance Policy from the ANL Policy and Practice Guide.
(3) Sections I and III of the ANL Quality Assurance Policy and Procedures Manual.

(4) Argonne Code Center Installation Representative Guide.

QUESTION I(B)(1)

In response to Interrogatory I(B)(4) [of the Second Set of Interrogatories to Applicant], the Applicant stated:

> The entire VENUS-II code has been thoroughly checked to assure that the equation sets and algorithms given in Ref. 2 on p.F6.2-119 of the PSAR are accurately programmed into VENUS-II. Because these equation sets are relatively simple, this was done by comparing output from the various subroutines against hand calculations.

Please identify each and every routine in the entire VENUS-II code.

(a) Separately, for each routine identified above, please supply the hand calculations or the appropriate documentation, i.e.,

(i) the name(s) of the individual(s) who performed the calculations and made the comparison; and

(ii) the laboratory notebook, memorandum or other writeen record that documents the comparison.

(b) Did the author(s) of the models actually perform the coding? If not, identify the programmer(s).

[NOTE: Interrogatory I(B)(4) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-4 and AA-6.]

ANSWER I(B)(1)

The table on the following page lists each of the subroutines in the VENUS-II code and contains a brief description of each of them.

(a) As explained in the response to Part I(A)(2) of this interrogatory, no formal or informal documentation of the verification activity associated with checking out VENUS-II exists, nor is it reasonable to expect it to exist. The authors of VENUS-II carried out these comparisons prior to their releasing the code for general use (and subsequently to the Argonne Code Center). This is implicit in the release of the code and the publication of the topical report describing VENUS-II, ANL-7951 (Ref. 2, in CRBRP-GEFR-00103).

(b) The authors of the VENUS-II code, as identified in ANL-7951, performed or directly supervised all of the coding of VENUS-II.

VENUS-II Subroutine Names and Brief Description

SUBROUTINE NAME	DESCRIPTION
MAIN	Master routine which calls input routine, hydrodynamics, neutronics feedback and prints edits
INPUT	Reads input from cards and sets constants
HYDRO	Calls for point kinetics calculation, determines new densities and energies and calls equation-of-state routine
HYDRIN	Entry point in HYDRO which sets hydrodynamic and thermodynamic initial conditions
EQUSTA	Determines pressure and temperature from density and internal energy
INTEGR	Determines material motion feedback contribution to reactivity
DOP	Determines Doppler contribution to reactivity
PKFETS	Solves point kinetics equations
INTERP	Calculates constants needed for material motion feedback (worth gradients) and normalizes power as well as worth gradients
ITERAT	Used to obtain coefficients for quadratic time series used to find material and Doppler feedback
FUNCTO	Used to determine reactivity when reactivity input in tabular form
FDEN	Used to allow for a void fraction for non-voided initial conditions
FITZ	Quadratic interpolation to determine reactivity from time series
DISPLY	Writes limited accuracy edit
CUTWAY	Prints out pictorial view of core under investigation
HOLLER	Used in pictorial
PICTURE PLCAL PLOT3D DSCALE TRNMAT PHI DAXIS DRAW WRITE	Plotting routines for 3-D plots and time history plots

QUESTION I(B)(2)

How does the Applicant continue to assure itself that the overall code and its subroutines accurately reproduce the models as described in the PSAR and its references?

ANSWER I(B)(2)

The Applicant continues to assure itself that the overall code and its subroutines accurately reproduce the models as described in the PSAR and its references by careful inspection of the output results for every case analyzed and by comparison of the output results for each case analyzed with the results of previous cases which are similar in part or in whole to the particular case analyzed. In addition, the computer system messages are checked to assure that the job was properly executed, without error, by the computer system.

QUESTION I(B)(3)

Please identify and provide all Intra-Laboratory Memoranda generated by personnel in the Accident Analysis Section, the Coolant Dynamics Section and other Sections of the ANL Reactor Analysis and Safety Division that critique or otherwise evaluate the models developed by other personnel in these respective sections, limited to the development of any and all models and subroutines that are used in VENUS-II (i.e., routines identified in response to (1) above). Also provide all subsequent memoranda that are responses to criticisms or evaluations identified above or that represent a continuation of the dialogue related to the model evaluation.

ANSWER I(B)(3)

See schedule of documents "Applicants' Response to NRDC Interrogatories" dated August 30, 1976. The files and documents have been and will be available for inspection at the Argonne National Laboratory and provisions have been and will be made for copying. The schedule of documents is being

updated and the update will be furnished upon completion. Documents referred to in the update will be available for inspection and copying at Argonne National Laboratory.

QUESTION I(C)(1)

In response to Interrogatory I(C)(4) [of the Second Set of Interrogatories to Applicant], the Applicant stated:

> The PLUTO code has been checked and rechecked to assure that the numerical algorithms which are implemented in PLUTO to solve the equation sets have been programmed correctly. Furthermore, test calculations were performed to assure that these numerical algorithms behave in a stable fashion and produce accurate solutions to the original equation sets. This was carried out by comparing PLUTO results with the output from another code (see, H. U. Wider, J. F. Jackson, L. L. Smith, and D. T. Eggen, An Improved Analysis of Fuel Motion During an Overpower Excursion, Proc. of the Fast Reactor Safety Meeting, CONF-740401-P3, p.1541, 1974) with the results of hand calculations, and with what sound engineering judgment deemed to be physically reasonable. This reference will be made available for inspection and copying.

Please identify each and every routine in the PLUTO code.

[NOTE: Interrogatory I(C)(4) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-4 and AA-11.]

ANSWER I(C)(1)

The PLUTO 1 code consists of four subroutines:

(1) The main driver calls the other three routines and also solves the set of compressible hydrodynamic equations describing the fuel, sodium, and fission-gas motion in the coolant channels. Furthermore, this routine calculates the ejection of fuel and fission gas from the pins.

(2) The subroutine PLREZO has as its main function the rezoning of the two Lagrangian numerical grids in the channel. This subroutine also performs the mapping of one of these grids on the other (e.g., for determining the fuel density or the fuel temperature or the fuel velocity on the sodium grid).

(3) The subroutine SMITH solves the set of compressible hydrodynamic equations which describe the fuel and fission-gas motion inside the pin.

(4) The subroutine PLIO reads the input and produces the output. Moreover, it calculates the reactivity changes caused by the fuel and sodium motion.

The PLUTO 2 subroutines are as follows:

PLUDRV - Main PLUTO2 driver; calls all PLUTO 2 routines except PLSAIN, PLINPT, and PLSET. Includes automatic time step calculation, fuel and voiding reactivity calculation, and writes the output. Called by TSTHRM.

PLSAIN - Picks up data from SAS which are necessary to set up the interaction zone. Called by FAILUR which is called by DFORM3.

PLINPT - Initiates mostly channel variables, edits PLUTO 2 input. Called by FAILUR which is called by DFORM3. Is shared with LEVITATE.

PLSET - Initiates mostly pin cavity variables, calculates auxiliary terms used in the code. Also edits PLUTO 2 input. Called by FAILUR which is called by DFORM3. Is shared with LEVITATE.

PLSET2 - Called by PLUDRV. Reinitializes temporary variables whenever control is transferred to PLUDRV from TSTHRM.

PLIF - Calculates slug interface locations and interface locations of the fuel, fission-gas, and fuel vapor regions in the channel. Also calcualtes the cladding rupture propagation. Called by PLUDRV.

PLREZO - Adds or deletes channel cells whenever the liquid sodium plug interfaces cross mesh-cell boundaries. Called by PLUDRV.

PLMACO - Solves the channel mass conservation equations for the 2-phase sodium mixture, fuel, fission, gas, and fuel vapor. Called by PLUDRV.

PLVOFR - Calculates fuel, liquid sodium, and gas void fractions. Calculates the thickness of the liquid sodium film. Determines the fuel flow regions for each node. Called by PLUDRV.

PLMISC - Calculates various channel heat transfer and friction coefficients. Determines the frozen fuel geometry. Solves the channel energy equations for mobile and plated out fuel. Called by PLUDRV.

PLTECS - Calculates cladding and structure temperatures. Called by PLUDRV.

PLNAEN - Solves two-phase and single-gas-phase energy equations for the mixture of sodium and fission gas. Called by PLUDRV.

PL1PIN - Solves mass and energy equations in the pin cavity. Also calculates the fuel and gas ejection rates into the channel as well as the fuel melt-in rates. Called by PLUDRV.

PL2PIN - Solves the fuel/fission gas momentum equations inside the pin and produces pin related output. Called by PLUDRV.

PLMOCO - Solves the fuel and sodium/fission gas momentum equations in the channel. Also calcualtes the sodium slug velocities.

PLFREZ - Determines the amount of frozen fuel plateout and release if the underlying clad is melted.

QUESTION I(C)(2)

Separately, for each routine identified in (1) above, please supply the following information:

(a) Was the routine verified by comparison with other codes, or by comparison with the results of hand calculations, or by comparison with what sound engineering judgment deemed to be physically reasonable?

(b) If the routine was verified by comparison with other codes, how was the other code or codes verified? Identify the other code or codes.

(c) If the answer to (a) or (b) above is that the routine was verified by hand calculations, please supply the hand calculations or the appropriate documentation, i.e.,

(i) the name(s) of the individual(s) who performed the calculations and made the comparison; and

(ii) the laboratory notebook, memorandum or other written record that documents the comparison.

(d) If the answer to (a) or (b) above is that the subroutine was verified by comparison with what sound engineering judgment deemed to be physically reasonable, please describe in detail the nature of and basis for the engineering judgment. In addition, supply:

(i) the name(s) of the individual(s) who rendered the judgment and made the comparison; and

(ii) the laboratory notebook, memorandum or other written record that documents the comparison.

(e) Did the author(s) of the models actually perform the coding? If not, identify the programmer(s).

ANSWER I(C)(2)

As with the response to Question I(A)(2) of this interrogatory set, Parts (a), (b), (c), and (d) are being responded to collectively here. Similarly to SAS3A and VENUS-II, the PLUTO 1 and PLUTO 2 codes were extensively checked during their development and at their completion. For the same reasons given for SAS3A and VENUS-II, these efforts were not documented in other than the formal PLUTO 1 and PLUTO 2 reports and also in several meeting abstracts and proceedings as noted in the paragraph that follows.

During its development, the PLUTO 1 code consisted of only two subroutines which were strongly connected and therefore, tested together. A rather stringent check on the two-phase PLUTO 1 hydrodynamics was made by calculating a shock propagation through a two-phase medium (H. U. Wider, J. F. Jackson, and D. T. Eggen, "An Improved Viscous Pressure Formulation for Two-Phase Compressible Hydrodynamics Calculations," Trans. Am. Nucl. Soc., 17, p. 246, 1974). In addition, comparison calculations with the SAS/FCI model have been performed, Ref. 49 in CRBRP-GEFR-00103, with certain aspects such as the FCI, Fuel Injection into the channel and the fuel motion in the channel.

During the PLUTO 2 development, comparison calculations with the PLUTO 1 code were made (see H. U. Wider, "PLUTO 2: A Computer Code for the Analysis of Overpower Accidents in LMFBRs," TANSAO 27, p. 533, 1977 and also see H. U. Wider, et al., ANL-RDP-63, p. 6.8). PLUTO 2 was also compared with the EPIC code (see H. U. Wider, et al., "The PLUTO 2 Overpower Excursion Code and a Comparison with EPIC", Proceedings of the International Meeting on Fast Reactor Safety Technology, Vol. 1, p. 120, Seattle, 1979). The multiphase hydrodynamics model was checked by analyzing standard flow expansion and contraction problems as well as by attempting to achieve steady-state conditions with this time-dependent compressible code (see A. M. Tentner and H. U. Wider, "Pressure Drop in Variable Area, Multiphase, Transient Flow," 2nd Multi-Phase Flow and Heat Transfer Symposium - Workshop, p. 1137, Miami Beach, 1979). PLUTO 2 was also compared to in-pile experiments (see CRBRP-GEFR-00523, References E-3 and E-4).

(e) The author of the PLUTO 1 and PLUTO 2 models performed or directly supervised all of the coding contained in the PLUTO 1 and PLUTO 2 codes.

QUESTION I(C)(3)

How does the Applicant continue to assure itself that the overall code and its subroutines accurately reproduce the models as described in the PSAR and its references?

ANSWER I(C)(3)

I(C)(3) The Applicant continues to assure itself that the overall code and its subroutines accurately reproduce the models as described in the PSAR and its references by careful inspection of the output results for every case analyzed and by comparison of the output results for each case analyzed with the results of previous cases which are similar in part or in whole to the particular case analyzed. In addition, the computer system

messages are checked to assure that the job was properly executed, without error, by the computer system.

QUESTION I(C)(4)

Please identify and provide all Intra-Laboratory Memoranda generated by personnel in the Accident Analysis Section, the Coolant Dynamics Section and other Sections of the ANL Reactor Analysis and Safety Division that critique or otherwise evaluate the models developed by other personnel in these respective sections, limited to the development of any and all models and subroutines that are used in PLUTO (i.e., subroutines identified in (1) above). Also provide all subsequent memoranda that are responses to criticisms or evaluations identified above or that represent a continuation of the dialogue related to the model evaluation.

ANSWER I(C)(4)

See schedule of documents "Applicants' Response to NRDC Interrogatories" dated August 30, 1976. The files and documents have been and will be available for inspection at the Argonne National Laboratory and provisions have been and will be made for copying. The schedule of documents is being updated and the update will be furnished upon completion. Documents referred to in the update will be available for inspection and copying at Argonne National Laboratory.

QUESTION II (GENERAL)

Please answer part (e) of questions 1-69 of Part II of the Second Set of Interrogatories to the Applicant (pp. AA-13 through AA-14).

ANSWER II (GENERAL)

The requested information is provided in the (revised) responses to the second interrogatory set (see p. AA-14).

QUESTIONS II-1

In response to Interrogatory II 2(b) [of the Second Set of Interrogatories to Applicant], the Applicant stated:

> The use of point kinetics model with fuel displacement feedback obtained by summing over fuel worth tables is judged adequate, so long as small, local displacements are considered. Gross relocation of fuel in large segments of the core can be addressed by use of FXVARI or similar diffusion type codes to recompute the fuel worth tables when such recomputation is judged necessary. Section F6.2.1 of the PSAR discusses the approach used.

(a) What is the basis for the first sentence of this response? Explain in detail.

(b) Quantify what is meant by "small" displacements and "gross" relocation of fuel.

(c) What criteria are used to decide when recomputation of the fuel worth tables is necessary? Describe in detail.

(d) Which CDA calculations identified in F6.2 of the PSAR met these criteria? In each case were fuel worth tables recomputed using FXVARI? If not, why not?

[NOTE: Interrogatory II-2(b) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-15 and AA-16.]

ANSWER II-1(a) through (c)

See Section 3.2.13 of CRBRP-GEFR-00103.

ANSWER II-1(d)

None of the HCDA analyses reported in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 met the criteria for recomputation of material worth values. Therefore, fuel worth values were not recomputed using FX-2.

QUESTIONS II-2

In response to Interrogatory II 3(b) [of the Second Set of Interrogatories to Applicant], the Applicant stated:

> The two distinct modes of failure identified by the terms 'Slumping' and 'Fuel-Coolant Interaction' are mutually exclusive extremes of a continuous spectrum of failure modes.

(a) What is meant by "extremes"?

(b) Describe in detail the basis for the above statement that the two failure modes are on the extremes.

(c) Is it not possible to have both extremes as part of the same accident scenario?

(d) Describe fully and precisely the nature of, and the application of, "technical judgments used to ensure that limiting conditions for the available subroutines are applied to bound the resulting energy release."

[NOTE: Interrogatory II-3(b) of the Second Set of Interrogatories to Applicant and the updated answer are on p. AA-16.]

ANSWER II-2(a), (b)

The SLUMPY Model was devised to represent fuel motion following disruption of fuel geometry in a voided coolant channel. The SAS/FCI Model was

devised to represent the fuel-coolant interaction process which by definition requires the presence of liquid coolant. Since the SLUMPY module requires the absence of coolant to function as intended and the SAS/FCI module requires the presence of liquid coolant to function as intended, they are on the opposite ends of the spectrum of all possible scenarios for fuel motion with or without coolant interaction.

ANSWER II-2(c)

Both SLUMPY and FCI analyses may be used in the same accident scenario but not in the same channel at the same time.

ANSWER II-2(d)

If neither model can be eliminated by reference to first principles of physical reality, the procedure is to perform both types of analysis and choose the path leading to higher energetics.

QUESTION II-3 (PREAMBLE)

The Applicant's responses to Interrogatories II-5(b) and (c) in the Second Set of Interrogatories are inadequate.

With regard to Applicant's response to II-5(c), we are requesting a quantitative rather than a qualitative response which the Applicant should keep in mind when answering (a) through (d) below. We see very little difference between "significant" and "much more sensitive."

QUESTION II-3(a)

Please provide a detailed and rigorous quantitative response to II-5(b) [of the Second Set of Interrogatories to Applicant] that includes the time histories of (i) clad temperature, (ii) pressure at the fuel pin center

line and at the fuel cladding interface, (iii) clad stress, and (iv) clad strain, each as functions of the height from the reflector bottom.

[NOTE: Interrogatory II-5(b) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-17 and AA-18.]

ANSWER II-3(a)

A quantitative assessment of the influence of the clad failure characteristics on the axial location of pin failure can be obtained by comparison of the SAS calculated BOEC and EOEC TOP $0.50/sec pin failure location predictions with failures predicted by the Damage Parameter empirical pin failure correlation, described on pages 6-32 and 6-33 of CRBRP-GEFR-00103. Comparison of the predicted axial failure locations in corresponding channels given in Table 6-6, page 6-51 of CRBRP-GEFR-00103 shows that both failure models predict failure at comparable axial locations above the midplane. This comparison is applicable to the TOP analysis in CRBRP-GEFR-00523.

This comparison quantitatively confirms that use of the experimental cladding failure strength as a function of temperature predicts pin failure toward the upper part of the pin in hypothetical unprotected TOP events (2.4 to 50¢/sec ramp rates). The uncertainty is greatest at low ramp rates. Recent information on the SLSF W-2 test at low ramp rates is being evaluated.

The effect of clad inhomogeneities on cladding failure during a hypothetical transient overpower or loss of flow transient has not been quantitatively assessed since no statistical data on cladding failure due to clad inhomogeneities is available. Qualitatively, the effects of cladding inhomogeneities may cause an individual pin to fail randomly along its length.

Cladding inhomogeneities are not expected to result in the coherent failure of large groups of pins. The possible failure of a small number of pins during the transient due to clad inhomogeneities is not expected to alter

the failure scenarios and energetics resulting from the failure of large groups of pins as predicted by current pin failure models using the best available experimental data for irradiated and unirradiated cladding failure strength.

The temperature dependent clad failure strength curves used for fresh and irradiated cladding are based on the best available experimental data, which are given in Refs. 14 and 42 in CRBRP-GEFR-00103.

The use of the experimental clad failure data with the transient cladding stress and temperature calculations in the SAS code result in the predicted axial location of cladding failure. The statement "The slope of the cladding strength as a function of temperature significantly influences the degree of bias in pin failure toward the upper part of the pin" is a general statement, applicable to TOP transients in the range of approximately 2.4 to 50¢/sec, which is supported by the results of SAS code calculations using the referenced experimental failure data.

The clad temperature-time history for the BOEC and EOEC TOP 50¢/sec transients is given in Figure 6-66 on page 6-117 and in Figure 6-32 on page 6-83, respectively, in CRBRP-GEFR-00103. Clad temperature histories demonstrate the trend of the cladding temperature increases during low ramp rates (2.4 to 50¢/sec) hypothetical TOP transients. The pressure at the fuel pin center line in channel 10, the first channel to fail in the best-estimate BOEC TOP 10¢/sec transient, is shown in Figure 6-69 on page 6-120 in CRBRP-GEFR-00103. This figure illustrates the trend of the pressure increase at the fuel pin center during the low ramp rate TOP transients. Pin failure is predicted to occur where the clad circumferential stress and the corresponding clad midpoint temperature satisfies the experimental failure strength data. The mechanistic burst pressure failure criterion in SAS is based on clad failure strength rather than cladding strain.

QUESTION II-3(b)

Please provide the quantitative probability of clad failure with the appropriate uncertainties as a function of time and height above the reflector.

ANSWER II-3(b)

A quantitatiave probability of clad failure with the appropriate uncertainties as a function of time and axial position does not exist nor is it necessary since for each transient analyzed clad failure was explicitly calculated.

There is no axial reflector in the CRBRP.

QUESTION II-3(c)

Please provide a detailed write-up of the Stuart model as formulated for SAS3A. Ref. 10 on page F6.2-120 of the PSAR is an abstract and does not provide sufficient detail.

ANSWER II-3(c)

A 'Stuart Model' as such was not formulated for SAS3A and SAS3D. Rather, Smith formulated a mathematical model of clad loading based on Stuart's fission gas pressure loading theory.

The essential elements of Stuart's theory are stated in paragraphs four and five on page 655 in Ref. 6 in CRBRP-GEFR-00103. Smith's clad loading model was based on Stuart's approach that fission gas released from the fuel during a transient overpower event will add to the steady state fission gas contained in the central cavity to load the cladding.

A detailed description of Smith's clad loading model used in the burst pressure failure criterion in SAS is given in Refs. 7 and 13 in CRBRP-GEFR-00103.

QUESTION II-3(d)

Does the model assume the pressure at the fuel-cladding interface is independent of the height above the reflector? If so, please justify.

ANSWER II-3(d)

The TOP fuel pin failure model described in Section 3.2.3 of CRBRP-GEFR-00103 does not assume that the pressure at the fuel cladding interface is independent of the height above the lower blanket.

QUESTION II-3(e)

Please provide a detailed description of the curves in Figure F6.2-14 (p.F6.1-176). What are Y.P.$^{(1,2)}$, UT$^{(4)}$, Burst$^{(3)}$, Burst irradiated clad$^{(5)}$, and Smith-Stevenson? What do these symbols and references mean? Please cross-reference the curves in Figure F6.2-14 with the curves in Reference 52 on p.F6.2-122 of the PSAR.

ANSWER II-3(e)

Pointer 50 on page 4-14 of CRBRP-GEFR-00103 provides a detailed description of the clad strength curves in Figure 3-3 (p. 3-43) of CRBRP-GEFR-00103. References which contain the data upon which the curves are based are also cited. The curve in Figure 3-3 of CRBRP-GEFR-00103 is the same as Figure F6.2-14.

The cladding strength table values for fuel types 1 and 2, denoted as Y.P.$^{(1,2)}$, are yield strength values for unirradiated 20% CW 316 SS,

property code 2102, page revision 1, 6-10-74 in Ref. 42 in CRBRP-GEFR-00103.

The cladding strength table values for fuel type 3, denoted as BURST(3), are burst strength values for unirradiated 20% CW 316 SS, property code 2203, page revision 2, 4-16-75 in Ref. 42.

The cladding strength table values for fuel type 4, denoted as UT(4), are ultimate tensile strength values for unirradiated 20% CW 316 SS, property code 2101, page revision 1, 6-10-74 in Ref. 42.

The cladding strength table values for fuel type 5, denoted as BURST IRRADIATED CLAD(5), are irradiated clad burst strength values for 20% CW 316 SS cladding heated at a transient heating rate of 100°F/sec. This curve was developed from a logarithmic interpolation between the 10°F/sec and 200°F/sec curves given in Ref. 14 in CRBRP-GEFR-00103.

As stated in Ref. 7 in CRBRP-GEFR-00103, the SMITH-STEVENSON curve for ultimate tensile strength is given by a 1/T fit to high strain rate data for unirradiated 20% CW 316 SS.

QUESTION II-3(f)

In the above response, please consider the implications of the model and calculations presented by H. G. Bogensbirger and C. Ranchi (Nuclear Technology, Vol. 29, April 1976, pp. 73-85).

(i) How does this model differ from that used in SAS3A and in the PSAR?

ANSWER II-3(f)

The Applicant can make no quantitative assessment of the implications of the model and calculations presented by H. C. Bogensberger and C. Ronchi (Nuclear Technology, Vol. 29, April 1976, pp. 73-85) relative to the CRBRP. The reference applied the fission gas behavior model in the analysis of a

$5/sec unprotected TOP event in the SNR-300 Mark I core. No physical bases for ramp rates of the order of $5/sec have been identified in the CRBRP.

The implications of the use of the referenced fission gas behavior model in the analysis of lower ramp rate TOP events is not discussed in the reference, and the Applicant can make no quantitative assessment of the implications of the model on the lower ramp rate (2.4 to 50¢/sec) hypothetical TOP events reported in CRBRP-GEFR-00103.

QUESTION II-4

With regard to Applicant's response to II.11.(b) [of the Second Set of Interrogatories to Applicant] we note that Figure F6.2-1 on p.F6.2-116 of the PSAR (cited on p.F6.2-9) indicates the SAS/FCI model uses one dimensional Lagrangian cells.

(a) Is this a correct interpretation of the SAS/FCI model?

(b) Has the Applicant rigorously tested whether one-dimensional Lagrangian cells is an adequate formalism, or is the Applicant simply assuming it is adequate because of the "long length in the axial direction of the coolant channel in comparison to the relative small distances between adjacent pins?"

[NOTE: Interrogatory II-11(b) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-20 through AA-21.]

ANSWERS II-4

The question most probably refers to Fig. F6.2-4 on p. F6.2-166 of the PSAR because Fig. F6.2-1 on p. F6.2-116 did not exist. Figure 3-4 on p. 3-44 of CRBRP-GEFR-00103 is the same as Figure F6.2-4.

(a) These are not Lagrangian cells in a rigorous sense because the velocities of the cell boundaries are not calculated by solving the momentum conservation equation. Rather the velocity is determined by linear interpolation of the interface velocities of the two constraining liquid sodium loops.

(b) The assumption of the one-dimensional pseudo-Lagrangian cells has been made because the coolant channels are very long in the axial direction compared to the small distances between adjacent pins. These pseudo-Lagrangian cells are used only for accounting for fuel motion in the channel for calculating fuel motion reactivity feedback. The reactivity feedback effect of any radial fuel motion in the channel would be totally negligible.

QUESTION II-5

The Applicant apparently failed to answer Interrogatory II.14(c) [of the Second Set of Interrogatories to Applicant]. The last sentence under 14 in the Applicant's response is totally inadequate.

Please answer II-14(c) [of the Second Set of Interrogatories to Applicant] fully, identifying each model by author(s) and reference and then explain fully the basis for rejecting these other models.

[NOTE: Interrogatory II-14(c) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-22 through AA-23.]

ANSWER II-5

Models in which a rapid superheating of the liquid sodium and subsequent explosive vaporization is assumed to occur are not considered to be relevant for an LMFBR environment with its abundance of nucleation sites. All in-pile tests done to date support the above statement (see Ref. 56 in CRBRP-GEFR-00103).

The current models which assume the sodium to be in thermodynamical equilibrium are all similar to the Cho-Wright model. This model can simulate a wide variety of situations by varying the fuel particle diameter, the mixing and fragmentation time constant, and the fuel sodium heat transfer coefficient as a function of the sodium void fraction. This model is adequate for simulating the mild interactions which have been observed in experiments, typical of LMFBR environments, as well as hypothetical more energetic interactions.

A heat transfer model which takes sodium condensation into account and is otherwise similar to the Cho-Wright model was used for the successful analysis of the fresh fuel H-2 experiment (see Ref. 16 in CRBRP-GEFR-00103).

The SAS/FCI heat transfer model is based on the Cho-Wright model and it also takes sodium condensation into account. Moreover, the SAS/FCI model calculates the rate of fuel injection into the sodium.

The PLUTO 1 and PLUTO 2 heat transfer model contains a space-dependent fuel-coolant interaction calculation in which a SAS/FCI type calculation is being performed in many axial nodes in the coolant channel.

Other current heat transfer models are reviewed in the following reference: H. K. Fauske, "CSNI Meeting on Fuel-Coolant Interactions," Nuclear Safety, 16, p. 436-422, 1975. The heat transfer modeling utilized in SAS/FCI and PLUTO 1 and PLUTO 2 are considered to be adequate for inclusion in a whole-core analysis code such as SAS3A and SAS3D.

QUESTION II-6 (PREAMBLE)

In response to II-15(b) [of the Second Set of Interrogatories to Applicant], the Applicant cited Chapter 6 of Ref. 22, p.F6.2-122 of the PSAR for sodium thermodynamic properties. This reference on p.96 states:

> Lack of data and inability to maintain consistency in the superheated vapor region and the region above the sodium critical temperature have led to a neglection of sodium properties in these regions. Since most calculations with the SAS/FCI model do not require properties in these regions, the equation-of-state model is adequate. However, extremely high heat transfer rates can lead to temperatures and pressures above the critical values as well as to superheated vapor. A more comprehensive equation of state will be necessary to handle these cases.

and on p.101:

> Although this method for the subcooled region does satisfy the basic thermodynamic relations at the saturation line, it does have some problems away from the saturation line. The most glaring problem is that for large pressures Eq. (6.15) may yield a negative isothermal coefficient of bulk compressibility. Similar effects may occur for the thermal expansion coefficient of Eq. (6.17). Except in severe cases, the pressures and temperatures predicted by the SAS/FCI model generally do not result in these anomalies.

With respect to each CDA analysis presented in F6.2 of the PSAR (see I(A) through (D) of NRDC's Fourth Set of Interrogatories), please provide the following information (separately for each CDA analysis):

[NOTE: Interrogatory II-15(b) of the Second Set of Interrogatories to Applicant and the updated answer are on p. AA-23.]

QUESTION II-6(a)

Does sodium in the core lead to temperatures and pressures above the critical value? Is superheated vapor produced? Explain fully the basis for the answer.

ANSWER II-6(a)

An examination of the cases in CRBRP-GEFR-00103 indicates that the highest liquid sodium temperature produced in SAS/FCI was approximately 1000°K below the critical temperature. Exceeding the critical temperature was not predicted.

Also, no case appears to have resulted in the production of superheated vapor. This is reasonable since the smallest initial interaction zone length used was 5 cm. A minimum condition to produce superheated vapor in SAS/FCI is to vaporize all the initial liquid sodium present in the interaction zone. This requires a volume expansion by a factor of 100 to 1000 or more, i.e., to an interaction zone 500 to 5000 cm long. The active core is less than 100 cm in height. Heat losses due to condensation will thus arrest the vaporization process before such an expansion can occur.

QUESTION II-6(b)

If the answer to (a) is yes, describe fully why the Applicant believes the SAS/FCI model is an adequate representation of the sodium equation of state.

ANSWER II-6(b)

The answer to II-6(a) is "no".

QUESTION II-6(c)

Do large pressures yield a negative isothermal coefficient of bulk compressibiity or thermal expansion coefficient anywhere in the core? Explain fully the basis for this answer.

ANSWER II-6(c)

Pressures large enough to yield these negative values are not calculated anywhere in the core. Such single phase pressures would be very large. For example, at a temperature of 1400°K, a pressure of approximately 4750 atm is required to produce negative values of either of these coefficients. Pressures of this magnitude are not anticipated, since fission gas is generally present in the interaction zone to act as a cushion, and known heat transfer rates between oxide fuel and sodium cannot produce explosive conditions with expected HCDA phenomenology. In any case, the maximum calculated single phase sodium pressures in CRBRP-GEFR-00103 and CRBRP-GEFR-00523 are generally more than an order of magnitude below the values required for negative coefficients.

QUESTION II-6(d)

If the answer to (c) is yes, describe fully why the Applicant believes the SAS/FCI model is an adequate representation of the sodium equation of state.

ANSWER II-6(d)

The answer to II-6(c) is "no".

QUESTION II-6(e)

If the answer to (a) or (c) is no, which are the minimum changes in the more sensitive parameters (e.g., reactivity ramp rate) that would be necessary before the answer to either (a) or (c) is yes?

ANSWER II-6(e)

No reasonable, i.e., physically meaningful, change to any sensitive parameters can be made to change the response in sections (a) and (c) from no to yes.

QUESTION II-7

The Applicant also stated the thermodynamic properties of cladding and fuel used in SAS/FCI are summarized in Section 6.2.2.3 of the PSAR.

(a) Precisely where in Section 6.2.2.3 are these properties summarized?

(b) Explain fully (rather than summarize) the basis for the choice of these properties.

(c) Explain fully why the Applicant believes the choice of these properties is adequate.

ANSWERS II-7

(a) SAS input for SAS/FCI is summarized on page 3-9 and 4-16 of CRBRP-GEFR-00103.

(b) No special input for clad and fuel thermodynamic properties is needed for SAS/FCI. The SAS/FCI module uses the standard cladding and fuel properties.

(c) It is mainly the uncertainties in accident phenomenology, not the uncertainties in the thermodynamic properties employed, that lead to the spectrum of hypothetical accident scenarios presented in CRBRP-GEFR-00103 and CRBRP-GEFR-00523. A complete study of the influence of all possible property variations has not been done, but the remaining uncertainties are not expected to lead to a broadening of the spectrum of hypothetical accident scenario. As a dramatic example, present oxide vapor pressure uncertainties were shown to have only a small effect in fast reactor disassembly calculations. See the following reference: J. F. Jackson et al., "The Influence of Equation-of-State Uncertainties on Fast Reactor Disassembly Calculations," Trans. Am. Nuc. Soc., 22, p. 368, 1975.

QUESTION II-8

Document the basis for the Applicant's response to Interrogatory II 16(b) [of the Second Set of Interrogatories to Applicant].

[NOTE: Interrogatory II-16(b) of the Second Set of Interrogatories to Applicant and the updated answer are on pp. AA-23 through AA-24.]

ANSWER II-8

The second sentence in the response to Interrogatory II 16(b) of the Second Set of Interrogatories to Applicant may have been misinterpreted. Whether or not the sodium voiding reactivity associated with a fuel-coolant interaction is larger than or smaller than the associated fuel motion reactivity is dependent on the particular FCI event being discussed, although the fuel motion reactivity is generally the dominant reactivity once the event passes the first few milliseconds. The original interrogatory asked for support of the assumption that the sodium voiding reactivity can be adequately determined in SAS/FCI from the average, smeared sodium density of the interaction zone. In support of this asumption, calculations of FCI events with the PLUTO 1 and PLUTO 2 codes have been done a number of times. In these codes, the sodium voiding reactivity is determined from the detailed axial distribution of sodium in the interaction zone provided by the Lagrangian mesh. The PLUTO 1 and PLUTO 2 calculations predict sodium voiding reactivities that do not differ significantly from those calculated by SAS/FCI if the constraining sodium slug velocities predicted by both models are similar (e.g., see page 7-84 in CRBRP-GEFR-00103). The reason for this is that the detailed distribution of the small amount of sodium in the interaction zone as calculated by PLUTO 1 and PLUTO 2 is not important for the sodium voiding reactivity feedback calculation.

QUESTION II-9

Identify in Appendix F of the PSAR (by page and paragraph) the SASBLOK calculations that represent the parametric variation of the loss coefficient (see Applicant's Response to Interrogatory II 44(b) [of the Second Set of Interrogatories to Applicant].

[NOTE: Interrogatory II-44(b) of the Second Set of Interrogatories to Applicant and the updated answer are on p. AA-37.]

ANSWER II-9

The SASBLOK calculations that represent the parametric variation of the loss coefficient referred to in the Applicant's response to Interrogatory II 44(b) are identified in CRBRP-GEFR-00103 (by page and paragraph) as follows: 6-4, para. 2 and 3; p. 6-5, para. 1 and 2; and p. 6-7, para. 2.

QUESTION II-10

With regard to the Applicant's Answer to Interrogatory II 45(b) [of the Second Set of Interrogatories to Applicant], identify the other locations of the blockages that were considered. Discuss the sensitivity of the CDA energetics to the location of the blockage.

[NOTE: Interrogatory II-45(b) of the Second Set of Interrogatories to Applicant and the updated answer is on pp. AA-37 through AA-38.]

ANSWER II-10

The other locations where agglomeration of material could occur are the upper blanket and the core.

The HCDA energetics are not expected to be significantly influenced by the location of the blockage. Fuel blockages are progressively more difficult to maintain in a stable coolable configuration as the blockage location

approaches the core mid-plane. As the reactor power continues to increase due to the continued control rod withdrawal, blockages closer to the core are expected to be partially or totally dispersed, depending on the portion of the blockage material which cannot be cooled below the melting point.

The analysis of Section 10.1.1 of CRBRP-GEFR-00103 was performed to assess the pessimistic assumption that fuel blockages could not be sufficiently cooled and would slump upon melting. The results of those calculations showed that slumping of the melted blockages would not result in recriticality. Therefore, the location of the blockage is not expected to have a significant effect on possible HCDA energetics which might result from the slumping of melted blockages.

QUESTION II-11

Please review the Applicant's Response to Interrogatory II 47(b) [see 47(c) of the Second Set of Interrogatories to Applicant] for correctness. The Applicant's response here is to refer NRDC to its Response to Interrogatory 27. Interrogatory 27 addresses the method of estimating fission-gas temperatures, an unrelated subject. Perhaps the Applicant meant to refer to another interrogatory.

[NOTE: Interrogatory II 47(c) of the Second Set of Interrogatories to Applicant and the updated answer is on pp. AA-38 through AA-39.]

ANSWER II-11

The original cross-reference has been corrected. See page identified above and p. AA-75 for answer.

QUESTION III

In response to Interrogatories I(A)(10), I(B)(10), and I(C)(10) [of the Second Set of Interrogatories to Applicant], the Applicants stated:

> "(10) The Applicants are currently analyzing this area and will provide pertinent information as it becomes available."

We find this response to be inadequate and request the following information:

(a) What is the precise nature of the analysis or analyses currently being performed in this area, and what is the nature of the uncertainty(ies) to be resolved by the analysis or analyses?

(b) Who is performing the analysis or analyses?

(c) When is the analysis or analyses expected to be completed?

[NOTE: Interrogatories I(A)10, I(B)10, and I(C)10 of the Second Set of Interrogatories to Applicant and the updated answers are on pp. AA-5 and AA-9, pp. AA-5 and AA-10, and pp. AA-5 and AA-12 respectively.]

ANSWER III

The original answers to interrogatories I(A)(10), I(B)(10) and I(C)(10) have been updated. See pages noted above.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	
UNITED STATES DEPARTMENT OF ENERGY	)	DOCKET NO. 50-537
PROJECT MANA[illegible] CORPORATION	)	
TENNESSEE [illegible] AUTHORITY	)	

AFFIDAVIT OF STANLEY H. FISTEDIS

Stanley H. Fistedis, 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 Manager of the Engineering Mechanics Program.

2. That he is duly authorized to answer the Interrogatory numbered I(8) of NRDC's Third Set of Interrogatories.

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

(Signature)

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

Notary Public

My Commission expires 9/28/84.

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 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 I(A) (that material related to the SASBLOK module), II(1-8), II(28-47) in NRDC's Second Set of Interrogatories, I(A), II(1-4), II(6-29) in NRDC's Third Set of Interrogatories, I(1-7), II(1-16) in NRDC's Fourth Set of Interrogatories, and II(2,3), II(9-11), and III in NRDC's Sixth Set of Interrogatories and that answers to Interrogatories II(1-8) (Second Set), and I(1-7) (Fourth Set) include results of work performed and developed under the Reactor Analysis and Safety Division, Argonne National Laboratory and that the answers to II(6-29) (Third Set) and I(1-7), II(1-16) (Fourth Set) include results of work performed by Fauske and Associates Incorporated.

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 27th day of April, 1982.

Notary Public

My Commission expires 9/28/84.

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 Associate Division Director.

2. That he is duly authorized to answer the Interrogatories numbered I(A), I(B), I(C), II(9-27), and II(48-69) in NRDC's Second Set of Interrogatories, excepting material related to the SASBLOK module in the response to Interrogatory I(A), Interrogatory II(5) of NRDC's Third Set of Interrogatories, and Interrogatories I(A), I(B), I(C), II(1), and II(4-8) of NRDC's Sixth Set of Interrogatories.

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

(Signature)

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

Notary Public

My Commission expires 9/13/84.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)

UNITED STATES DEPARTMENT OF ENERGY)

PROJECT MANAGEMENT CORPORATION)

TENNESSEE VALLEY AUTHORITY)

DOCKET NO. 50-537

AFFIDAVIT OF LEE E. STRAWBRIDGE

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

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

2. That he is duly authorized to answer the Interrogatories in NRDC's Fifth 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 2nd day of April, 1982.

Notary Public

My Commission expires 5/28/82.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

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 28, 1982

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