10/1/82

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF'S ANSWERS TO NATURAL RESOURCES DEFENSE COUNCIL, INC. AND SIERRA CLUB TWENTY-SEVENTH SET OF INTERROGATORIES TO STAFF

Pursuant to the Licensing Board's Scheduling Order of August 31, 1982, the Nuclear Regulatory Commission Staff ("Staff") hereby responds to "Natural Resources Defense Council, Inc. ("NRDC") and the Sierra Club Twenty-Seventh Set of Interrogatories and Request to Produce to Staff", filed on September 17, 1982, relating to the Draft Supplement to the Final Environmental Statement related to construction and operation of Clinch River Breeder Reactor Plant, NUREG-0139, Supplement No. 1, Draft Reprot (July 1982). Attached hereto are the Staff's answers to those interrogatories, together with the affidavits of the sponsors of the Staff's answers. To the extent not provided herewith, signed and notarized affidavits of NRC Staff employees who participated in preparing the attached answers will be provided as soon as possible.

Pursuant to 10 C.F.R. § 2.744, the Staff will respond to "Natural Resources Defense Council, Inc. and the Sierra Club Third Request to Staff for Production of Documents" filed on September 17, 1982 and the

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documents requested in the "Natural Resources Defense Council, Inc. and the Sierra Club Twenty-Seventh Set of Interrogatories and Request to Produce to Staff" by October 18, 1982, unless the documents are provided in this response.

On March 4, 1982, the parties in this proceeding developed a Protocol for Discovery, pursuant to which NRDC and the Sierra Club have requested that answers to interrogatories be provided in six parts, as follows:

(a) Provide the direct answer to the question.

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

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

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

(e) Explain whether Staff is presently engaged in or intends to engage in any further, ongoing reserach program which may affect Staff's answer. This answer need be provided only in cases where the Staff intends to rely upon ongoing 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 Staff does not intend to rely upon the existence of any such research at the LWA or construction permit hearing on the CRBR.

(f) Identify expert(s), if any, which Staff intends to have testify on the subject matter questioned, and state the qualifications of each such expert. This answer may be provided for each separate question or for a group of related questions. This answer need not be provided until Staff has in fact identified the expert(s) in question or determined that no expert will testify, as long as such answer provides reasonable notice to Intervenors.

For all responses to interrogatories in this set, the following are the Staff's answers to the requests set forth above:

(a) Direct answers are provided for each question.

(b) All documents and studies, and the particular parts thereof, relied upon by the Staff now or in the past which serve as the basis for the answer are identified in the answer to the question, unless otherwise noted.

(c) There are no principal documents and studies specifically examined but not cited in (b), unless otherwise indicated herein.

(d) The name, title and affiliation of the primary Staff employee(s) or consultant(s) who provided the answer to the question are set forth in the attached affidavits, unless otherwise indicated herein.

(e) The Staff is not presently engaged in nor does it intend to engage in any further, on-going research program which may affect the Staff's answer, unless otherwise noted.

(f) At this time, the Staff has not determined who will testify on the subject matter questioned. Reasonable notice will be given to all parties after the Staff has made this determination. At that time, a statement of professional qualifications will be provided for each witness.

Finally, in the Staff's June 18, 1982 answer to NRDC's 2nd set of document requests, relating to contention 1, the Staff indicated that Dr. Kelber had been requested to not distribute his response to the questionnaire referenced in that request. Dr. Kelber has now received permission to publicly distribute his response, and a copy is therefore being attached as Enclosure C.

Respectfully submitted,

Daniel T. Swanson
Counsel for NRC Staff

Bradley W. Jones
Counsel for NRC Staff

Geary S. Mizuno
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 1st day of October, 1982.

Responses to NRDC 27th Set of Interrogatories

I. Appendix D

Interrogatory I.1

On page D-2, the staff has stated that it used Amendment XIV to the applicants' CRBR Environmental Report as a basis for its independent assessment of the environmental effects of the CRBRP fuel cycle. What is the basis for the staff's conclusion that this assessment is independent if the staff has simply used the numbers reported by the applicants in the Environmental Report?

Response to I.1

This interrogatory by NRDC completely miscontrues the staff statement on page D-2. The full statement is, "The staff has used Amendment XIV as a basis* for performing an independent assessment of the environmental effects of the CRBRP fuel cycle."

Thus it is clear that the staff used a number of items for its independent review and that the DOE Amendment XIV to its ER is merely one of a number of bases for the staff's independent review. Further, the NRDC inference that the staff merely used "the numbers reported by the applicants in the ER," is an obvious misstatement of fact. Even a casual review of Appendix D would show that the staff considered the material provided by the applicant in the ER and in many instances used different values than those proposed by the applicant. A few typical examples of this are noted below:

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* Underlining provides emphasis for this answer

a. The quantities noted in Fig. D.1 of NUREG-0139, Supplement No. 1 are independently developed by the staff and are not the same as those provided by the applicant in Fig. 5.7-1 - CRBR Equilibrium Fuel Cycle of Amendment XIV to its ER.

b. For the fuel fabrication step, the staff realistically considered the plutonium composition expected and planned for the CRBR fuel as contained in Section 5.8.3 of the applicants' ER, and compared it to the basis used by DOE in its environmental appraisal for the FMEF project. (See Table D.5 and Table D.6). The staff used the higher of the two values for each isotope in its assessment.

c. For the fuel reprocessing and waste management steps, the staff had independent calculations performed by the ORIGEN 2 code at ORNL to estimate the composition of spent fuel. This is clearly noted on page D-12 and in Tables D.7 and D.8 of NUREG-0139, Supplement No. 1. Further, the staff considered the values developed in this manner with those developed by DOE and for conservatism used the highest of the values as the basis for its assessments.

Interrogatory I.2

On page D-3, the staff has identified the current fuel cycle proposed by the applicants for the CRBRP (page D-2). Identify any and all reasonably foreseeable alternative fuel cycles for the CRBRP.

Response to I.2

Contention 6 by NRDC requested an analysis of, "... the environmental impact of the fuel cycle associated with the CRBR ..."

The staff requested from the applicants a description and assessment of such a fuel cycle. The applicant provided this material in Amendment XIV of its ER. The staff is not aware of any alternative fuel cycles proposed by or planned by the applicants. Accordingly, the staff has no basis for predicting any alternative fuel cycles for the CRBR and thus is unable to answer this question. In some portions of the fuel cycle provided by the applicant, alternative facilities were noted as possibilities; for these operations, the staff has considered the alternatives and has based its assessment on what is believed to be a set of conditions that would conservatively bound the alternatives.

Interrogatory I.3

Identify the CRBRP initial loading (Table D.1) that would be required for the heterogenous core if reactor-grade plutonium from reprocessing commercial reactor fuel were utilized to supply the plutonium for the CRBR.

Response to I.3

The CRBRP construction permit application, including the Environmental Report, is based on the use of FFTF grade fuel (i.e. approximately 12% Pu-240; see section 5.8.3 of the ER and 8/24 Transcript at 1833, lines 4-8 and 18-24). Neither applicants nor staff have analyzed the initial CRBRP core loading that would be required if reactor grade plutonium from reprocessing commercial reactor fuel were used.

Interrogatory 1.4

In Figure D.1 at page D-3, the staff has assumed plutonium losses to waste storage of .5% of the plutonium throughput for both the reprocessing plant and the fuel fabrication plant. For each year of operation, or alternatively over the plant lifetime, identify the actual plutonium losses to waste storage (as a percent of plutonium throughput) that occurred at the following plants:

a. the NFS West Valley reprocessing plant,
b. the Savannah River plant, F Canyon,
c. the Hanford Purex plant,
d. the Kerr-McGee MOX Fuel Fabrication plant,
e. the NUMEC Plutonium Fabrication Operations plant,
f. the NFS Erwin Plutonium Fabrication Operations plant,
g. the fuel processing or fabrication plant where these data are known to the NRC staff.

Response to I.4

The staff judged that the DOE estimate of 0.5 percent losses of plutonium in a reprocessing plant is reasonably conservative. Part of the basis for the staff's judgement on this value is founded on data like that contained in Table 4.19, "Overall Decontamination and Recovery," of Chapter 4 of Volume II, "Fuel Reprocessing," of The Reactor Handbook, Second Edition, which indicates that losses significantly below this level (0.3%) were obtainable two decades ago. The staff has also considered that one of the principle purposes for the reprocessing of the CRBRP fuel is to demonstrate the recovery of plutonium to establish its breeding ratio, thus there would be a strong incentive to limit plutonium losses to low levels.

With regard to commercial reprocessing operations, the reprocessing plant operated by Nuclear Fuel Services, Inc. at West Valley, New York had overall plutonium losses of about 2.6 percent. These losses were a product of inefficient operations and a decision not to rework waste solutions to remove plutonium products since the expenditure of funds for better separation was not economically justified by Pu demands. Thus they should not be considered a valid basis for establishing levels of losses. Further, the exact value of plutonium losses is not essential to the environmental impact assessment for the reprocessing plant since such losses would not be to plant effluents, but would be principally included in wastes which are separately disposed of and evaluated for environmental impacts.

Operational data for the DOE reprocessing facilities should be obtained from the DOE.

There has been virtually no experience in the USA to demonstrate the material balance in a closed plutonium breeder cycle. It is recognized however that plutonium losses in the fuel cycle will have the effect of reducing the overall breeding efficiency of the system. Therefore, there is an incentive to limit losses of such important fuel values in a demonstration as a forecast of what may be achievable in a fast breeder program.

There has been limited USA commercial experience in the fast test reactor fuel fabrication. Based upon classified information that has been compiled by the

Office of Inspection and Enforcement relative to a pilot plant that has produced fuel for fast reactor use over a five year period losses approximated about 0.7% of throughput.

Considering that losses to waste of about 0.7% has been achieved and that there is a high incentive to further limit losses in a demonstration cycle, leads the staff to believe that losses of about 0.5% are achievable and may be reasonably expected.

References:

As stated in response. Also Safety Analysis Report, Vol. I., Nuclear Fuel Services, Inc., Reprocessing Plant, West Valley, New York, Docket No. 50-201, 1973, p. I-2-3.

Interrogatory I.5

In Figure D.1 at page D-3, how much plutonium did the staff assume was initially stored in the facility labelled "Pu Storage Inventory" that would be available for use in the CRBR as initial core and reload materials?

Response to I.5

The staff did not assume that any plutonium was initially stored in the "Pu Storage Inventory." That unit operation was included in the overall flow diagram in NUREG-0139, Supplement No. 1, to account for the excess plutonium bred in operation of the CRBRP, not as a basis for storage of plutonium feed material from other DOE sources.

Interrogatory I.6

At page D-4, paragraph 4, the staff states that it based its evaluation on the equilibrium mode with burnups shown in Table D.3. How many reprocessing cycles (identified in Figure D.1) are necessary before the plutonium isotopic concentration in fresh CRBR fuel reaches equilibrium under the fuel cycle assumed in Figure D.1?

Response to I.6

The following answers relate to I.6 through I.9. The CRBRP ER is based upon the use of FFTF grade plutonium (see answer to I.3). The applicant has not provided the staff with information on composition of plutonium resulting from repeated recycling of fissile material through the CRBRP. However, the staff does not believe that any change in plutonium composition resulting from such considerations would affect the fuel cycle requirements in a significant manner (i.e. would not change more than 20%).

Interrogatory I.7

What is the plutonium isotopic concentration in fresh CRBRP fuel at the time after the CRBRP fuel cycle has reached its equilibrium, with regard to fresh fuel plutonium concentration? In other words, identify the plutonium isotopic concentration at equilibrium in weight percent for (a) Pu-236, (b) Pu-238, (c) Pu-239, (d) Pu-240, (e) Pu-241, (f) Pu-242, and (g) Pu-243.

Response to I.7

See response to Interrogatory I.6.

Interrogatory 1.8

What is the plutonium isotopic concentration of CRBR fresh fuel for the fuel cycle in Figure D.1 after 1, 2 and 4 recycles respectively, for each isotope identified in question 7 above.

Response to I.8

See response to Interrogatory I.6.

Interrogatory I.9

In Table D.3, at page D-6, the staff has assumed that the fissile plutonium content represents 88% of the total plutonium of each charge to the reactor core. What is the basis for the staff's assumption that the fissile content will not be substantially lower at equilibrium due to recycling of the CRBR fuel as shown in Figure D.1 at page D-3?

Response to I.9

See response to Interrogatory I.6.

Interrogatory I.10

On page D-9, the staff states that the applicants and the staff both assumed a clean-up factor of 1.25 E-8 for the atmospheric transuranic releases from the core fuel fabrication operations. For each year of operation, or alternatively, over the lifetime of the facility

a. What clean-up factor was achieved by the plutonium operations at the Kerr-McGee facility that was used to fabricate FFTF fuel?

b. What clean-up factor was achieved by the plutonium operations at the NFS Erwin Facility?

c. What clean-up factor was achieved by the plutonium operations at the NUMEC facility?

d. What clean-up factor was achieved by the plutonium operations at Rocky Flats?

e. In light of the experience at Rocky Flats, what is the basis for the staff's assumption that, averaged over the lifetime of the plant, accidental releases will not exceed routine releases through the banks of HEPA filters?

Response to I.10

Clean-up factors applicable to individual filters, or banks of filters in series, are not routinely measured during operations involving plutonium due to a number of reasons such as widely varied operations, range of source concentrations, etc. Therefore, NRC has no directly obtained quantitative values of annual clean-up factor for the operations mentioned. However, such factors are determined for systems prior to filtration system use. Licenses issued for these types of operations issued under 10 CFR Part 70 require that equipment and facilities be adequate to protect health and minimize danger to life. Regulatory Guide 3.12 presents methods acceptable to the Regulatory staff for complying with § 70.23 of that regulation relative to filtration systems. In order to assure adequate clean-up factors Regulatory Guide 3.12(C)(8)(f) states that "HEPA filter systems should be tested after filter installation using a "cold DOP" test. Acceptance should be based on an efficiency of 99.95% or better...." Similar guidance for DOE operations, including all DOE contractor operations, is provided in DOE 5480.1A Environmental Protection, Safety, and Health Protection Program for DOE Operations, of 8-13-81. A system with three filters in series (as planned for the SAF line) that passes this test will have a calculated clean-up factor of 1.25×10^{-10}. For these reasons the staff feels that derating a filtration system by a factor of 100 (clean-up factor 1.25×10^{-8}) is a sufficiently conservative basis for estimating gaseous effluent quality.

Interrogatory I.11

On page D-14, the staff refers to alternative reprocessing plants to the DRP. For each isotope element identified in Table D.8 at page D-15, identify the containment factor utilized by the staff, or by DOE if larger than the staff's assumption, for the DRP and for each of the following plants:

a. The F-Canyon at the Savannah River plant,
b. The NFS West Valley facility,
c. The Hanford Purex plant,
d. Any foreign reprocessing plant where the data is known to the NRC.

Response to I.11

The staff has not utilized containment factors for any of the facilities listed in the interrogatory. The containment factors, used by the staff for the DRP are taken from Table 5.7-3 of DOE's Amendment XIV as explained in Table D.8 in footnote (a). These containment factors for most isotopes, with the exception of ruthenium, were judged by the staff to be appropriate for the following reasons:

1. The commitment on the part of DOE to use current guides and standards in the design of the DRP.
2. The general conservatism used by the applicant in estimating the performance of plant final filters.
3. The availability of current technology to achieve these objectives.

The staff used a more conservative containment factor for ruthenium taken from Data Sheet No. 256 of DOE/ET-0028.

Interrogatory I.12

On page D-21 the staff states, "It is estimated that for CRBRP these releases would range from about 6 x 10^{-5} Ci/yr from a repository in salt to about .5 Ci/yr from a repository in granite (1/100th of values reported in DOE 1980)."

a. What is the basis for the staff's assumption that 1/100th of the values reported in DOE 1980 should be taken?

b. What independent analysis, if any, has the staff conducted to verify that the release rate assumed in the reported DOE document are correct; i.e., 6 x 10^{-3} Ci/yr from a repository in salt and 50 Ci/yr from a repository in granite?

c. Display all calculations that form the basis for the staff's estimate of this curie release.

d. Identify each and every staff person and consultant who conducted this independent review. Identify the employer and location of each consultant.

e. Identify when this review took place.

f. Identify and produce all documents examined and relied upon by the staff in the conduct of this review.

g. Are the above curie releases to the "accessible environment"? If not, what are these releases to?

h. How does the staff define "accessible environment" in this regard?

Response to I.12.a

Two factors were taken into consideration in arriving at the 1/100 figure: 1) waste characteristics and 2) waste quantities.

The characteristics of the CRBRP wastes were presented in Table D.10 of NUREG-0139, Supplement No. 1. This table shows that the isotopic composition of CRBRP HLW are similar to that of LWR HLW. Furthermore, the radioactivity, thermal power, and ingestion toxicity for CRBRP HLW and LWR HLW wastes would be essentially similar as shown in the independent work performed by ORNL for NRC as shown in Reference NRC 1982a.

The amount of high level waste from a nuclear fuel cycle is generally quantified in terms of the amount of heavy metal fuel that it results from.

With regard to the quantity of CRBR waste, the average annual fuel requirements for the CRBRP, as illustrated by Figure D.1, page D-3, is 11.87 MTH. Thus, over the 30-year operating life of CRBRP, the fuel cycle waste disposal requirement would be based upon about 360 MTHM of fuel. In DOE 1980b (see page 5.41, Table 5.3.7), the waste capacity of the conceptual waste repositories range from wastes resulting from 30,500 MTHM to 69,000 MTHM of fuel. Thus, the CRBRP wastes represent approximately 1/100th of this range of capacity for the conceptual waste repositories.

Response to I.12.b

The NRC staff did not perform an independent analysis of projected release rates from conceptual DOE repositories. Actual repositories will be subject to detailed review by NRC during the required licensing process. Until that licensing analysis is performed, NRC has adopted the values reported in the DOE Final Environmental Impact Statement pursuant to Part 1506 of the CEQ regulations.

Response to I.12.c

Table 5.4.9 in DOE 1980b displays annual releases of naturally occurring radioactivity to air from construction of a geologic repository. The nuclide values in Table 5.4.9 were added together for each of the four geologic media. This yielded total releases ranging from a 5.9×10^{-3} Ci/yr for a salt repository to 53 Ci/yr for a granite repository. Based on the rationale in response to Interrogatory 12.a, the fraction of CRBRP waste is projected to be 1/100 of the total repository waste inventory. As a result, the releases attributed to CRBRP on a prorata basis for a salt repository are approximately 6×10^{-5} Ci/yr and those of a granite repository are 5×10^{-1} Ci/yr.

Response to I.12.d

Robert McCallum, PNL, Richland, WA

Iral Nelson, PNL, Richland, WA

Regis Boyle, NRC, Washington, DC

Homer Lowenberg, NRC, Washington, DC

John P. Colton, NRC, Washington, DC

Response to I.12.e

The review was carried out during the period of April - June, 1982.

Response to I.12.f

The staff relied on the following documents cited in the NUREG-0139, Supplement No. 1:

1) DOE 1980b
2) DOE 1979
3) NRC 1982a

These documents will be made available as part of the staff's response to your request for documents.

Response to I.12.g

DOE 1980b states on page 5.52 that the releases identified in Table 5.4.9 are to the biosphere which is defined on page 8.2 of DOE 1980b as "The part of the earth in which life can exist including the lithosphere, hydrosphere, and atmosphere; living beings together with their environment."

Response to I.12.h

The term "accessible environment" has not been used in NUREG-0139, Supplement No. 1. NRC understands this term, as defined by EPA, to be "(i) the atmosphere, (ii) land surfaces, (iii) surface waters, (iv) oceans, and (v) parts of the lithosphere containing significant amounts of groundwater; the accessible environment also includes (vi) parts of the lithosphere containing insignificant amounts of groundwater that are more than ten kilometers in any direction from the original location of the radioactive wastes in a disposal system."

Interrogatory I.13

At page D-21, the staff states: "The resulting annual dose to the regional populations in the vicinity of the repository would range from about 7×10^{-5} person-rems for a repository in salt to about 1 person-rem for a repository in granite."

a. What is the basis for the staff's assessment of the person-rem calculations presented here? Present all calculations and assumptions.

b. Identify and produce all documents relied upon by the staff as a basis for these calculations.

c. Identify the staff personnel and consultants, by name and affiliation, that performed these calculations and analyses. Were these calculations performed by DOE or DOE consultants? If so, identify who performed the calculations.

d. Define precisely the "regional populations" referred to in this statement, including the size, i.e., number of people; and extent, i.e., distance from the repository.

e. Identify the time period over which the dose estimates (on an annual basis) are summed.

f. Identify each isotope that was considered in the summation of the dose commitment.

g. Identify the contribution to the total dose commitment from each isotope over the prescribed period of summation or integration.

h. Identify the dose conversion factors utilized in the above calculations.

i. Identify all other important assumptions made in the same calculations.

j. What is the staff's estimate of the range of uncertainties on both the estimates of the curie values released for each of the isotopes involved (see question 12 above) and for the overall dose commitment in person-rems? What is the basis for this estimate? Display all calculations and identify and produce all sensitivity studies, and documents relied upon by the staff in response to this question.

k. What is the basis for the staff's reporting the dose commitment in person-rems, presumably meaning whole-body dose, given that most of the dose would be to internal organs, such as the bone surface, for both salt and granite repositories.

l. What is the staff's estimate of the total dose commitment to bone surfaces?

m. What is the basis for this estimate of total dose to the bone surfaces? Display all calculations and identify and produce all documentation relied upon by the staff.

Response to I.13.a

The basis for the staff's assessment of the person-rem calculations is DOE 1980b. As discussed in our response to Interrogatory 12.b, the staff adopted DOE 1980b since the EIS (DOE 1980b) has met the standards of an adequate EIS under CEQ regulations.

The total body doses to the regional population were obtained from DOE 1980b, Table 5.4.10, page 5.53, by dividing the values presented by 100. These values represent 70-year whole-body doses. The annual doses to the regional population from the conceptual repository are presented in DOE/ET-0029, Environmental Aspects of Commercial Radioactive Waste Management, Volume 2, Table 9.1-8, page 9.1.7, and range from 3.1×10^{-4} person-rem/year for a salt repository to 2.6 person-rem/year for a granite repository. Prorating these doses for the CRBRP portion of a repository (1/100) results in a range of 3.1×10^{-6} person-rem/year to 2.6×10^{-2} person-rem/year.

Response to I.13.b

The document relied upon for the 70-year whole-body doses was DOE 1980b. The document relied upon for the annual whole-body doses was DOE/ET-0029, Volume 2. These documents will be included in the staff's response to your document request.

Response to I.13.c

1. Robert McCallum, PNL, Richland, WA
2. Dennis Strenge, PNL, Richland, WA
3. Iral Nelson, PNL, Richland, WA
4. Regis Boyle, NRC, Washington, DC
5. Homer Lowenberg, NRC, Washington, DC
6. John P. Colton, NRC, Washington, DC

Response to I.13.d

DOE 1980b defines regional population as the population within an 80 km radius of a waste facility, i.e., 2 million, (see page 3.16 of Volume 1 and Appendix F of Volume 2 for additional discussion).

Response to I.13.e

The doses on page D-21 were 70-year accumulated whole-body doses to the regional population from exposure to radon and its decay products which are released over a seven year construction period.

Response to I.13.f

The isotopes considered were ^{220}Rn, ^{222}Rn, ^{210}Pb, ^{212}Pb, ^{214}Pb, and ^{210}Bi.

Response to I.13.g, h, i, j

The contribution to the total dose commitment from each isotope over the prescribed period of summation was not presented in DOE 1980b. Such detail should be able to be obtained by contacting DOE.

Assumptions regarding specifics of the models used to estimate dose are presented in Appendix D of DOE 1980b.

Pursuant to CEQ Guidelines Part 1506 the results of the analysis in DOE 1980b were adopted for the waste management analysis of the CRBR fuel cycle. It is the staff's view that it is meaningless to perform additional detailed calculations at this time on a conceptual repository, since the staff will perform a detailed review of any repository for CRBR high-level wastes at the time that such an application is submitted by DOE.

Response to I.13.k

Doses presented are for "enhanced" release of ^{220}Rn and ^{222}Rn (plus daughters) during mining of the repository. The dose to internal organs is mainly due to ^{210}Pb because of the short half-life of the other ^{223}Rn daughter products. The total body doses presented represent an average dose over all body organs and was used as a convenient measure of potential dose. The calculation of the total body dose is performed using methods and data of ICRP Publication 2 with respiratory system response described by the Task Group on Lung Dynamics lung model of ICRP Publication 19. The dose calculation includes all material deposited in all internal organs.

Response to I.13.l

The total population dose to bone surfaces is estimated to be 8×10^{-4} person-rem for a salt repository and 12 person-rem for a granite repository.

Response to I.13.m

The doses reported in response l above were estimated from the "total body" doses as follows. Because the dose to internal organs is primarily from ^{210}Pb, the dose to cortical bone is first estimated using the ratio of inhalation dose conversion factors (for chronic uptake over 1 year nd a 50-year dose commitment period) for bone and total body as presented in Strenge et.al. (1980). This reference gives dose conversion factors for ^{210}Pb (page D.56) as calculated by the computer programs DACRIN (Houston, et al. 1976) using the methods that were used for the dose calculations in the reference report (DOE 1980). This ratio implies the bone dose is 32 times the total body dose. The dose tc bone surfaces can be estimated using data from Dunning et al. (1979) describing dose to internal organs. For ^{210}Pb, the dose to bone surfaces is estimated to be 0.37 that of the dose to the cortical bone (page 71). The dose to bone surfaces is thus estimated by multiplying the total body dose by 12 (32 times 0.37).

References:

ICRP Publication 2.

ICRP Publication 19.

Strenge, D. L. et al., 1980. ALLDOS - A Computer Program for Calculation of Radiation Doses from Airborne and Waterborne Release. PNL-3524, Pacific Northwest Laboratory, Richland, Washington.

Houston, J. R., D. L. Strenge, and E. C. Watson, 1976. DACRIN - A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation. BNWL-B-389. Pacific Northwest Laboratory, Richland, WA.

DOE (1980) DOE/EIS-0046F.

Dunning, D. E., Jr., October 1979. Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occuring in Routine Releases from Nuclear Fuel Cycle Facilities Volume II. NUREG/CR-0150 Volume 2. ORNL/NUREG/TM-190/V2. Oak Ridge, Tennessee.

These documents will be included in the staff's response to your document request.

Interrogatory I.14

On page D-29, the staff discusses the calculation of the dose commitments from blanket fuel fabrication and estimates the total whole-body dose commitment to be less than 0.1 person-rem annually.

a. Given that the primary radiological effluents from blanket fuel fabrication are U-235 and U-238 (see page D-7), what is the basis for the staff's view that only the whole-body dose should be calculated rather than include internal organ doses for the critical organs for U-235 and U-238 exposure?

b. What are these critical organs and what would be the corresponding organ exposures?

c. What dose conversion factor was utilized in calculating the whole-body dose commitment of .1 person-rems annually?

d. Identify and produce the documentation used as a reference source for dose commitment factor.

e. Explain in detail how the RABGAD code has been validated and produce all documentation related to the RABGAD validation.

f. Identify each staff personnel and consultant who conducted the RABGAD (i) validation, and (ii) calculations.

g. Identify and produce all documentation of the RABGAD dose calculations conducted by the staff.

h. With respect to the environmental dose commitments for both the blanket fuel fabrication and the core fuel fabrication, what is the basis for the staff's assumption that the integration period should be limited to 100 years? What would the dose estimate be if the integration period were extended to cover (i) half-life and (ii) 6 half-lives of the pertinent isotopes?

i. In the staff's calculation of the dose commitment of less than 0.1 person-rems for core fuel fabrication, provide a breakdown of the dose commitment by (i) isotope identified in Table D.4, (ii) pathway identified in the second to last paragraph on page D-29, and (iii) organ, including the whole body.

Response to I.14.a

(a) As stated on page D-29 of the Draft Supplement, the staff reviewed the dose estimates in the Environmental Impact Appraisals for existing commercial U.S. uranium fuel fabrication plants. In addition, on p. D-33 (Footnote "a" to Table D.17) of the Draft Supplement, it is stated that "the annual population doses to the bone, lung, kidney and GI tract are also less than 1 person-rem." Using those data bases, the Staff concluded that the population fabrication facility would be a small fraction of the estimated population dose from the entire fuel cycle, and a more sophisticated analysis was not necessary.

(b) See references listed in §D.2.4.1 of the Draft Supplement.

Response to I.14.b

See response to Interrogatory I.14.a.

Response to I.14.c

The dose estimation of the Environmental Impact Appraisals for two of the facilities (i.e., Exxon, and General Electric) were based on dose conversion factors documented in a report by Houston, J. P., et al, DACRIN - A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation BNWL-B-389 Battelle Pacific Northwest Laboratories, Richland, Washington (1975). The dose estimates in the Environmental Impact Appraisal for the Westinghouse facility were based on dose conversion factors documented in a report by Moore, R. E., et al, AIRDOS - A Computer Code for Estimating Population and Individual Doses Resulting from Atmospheric Releases of Radionuclides from Nuclear Facilities, ORNL/TM-4687, Oak Ridge National Laboratory, Oak Ridge, Tenn. (1975).

Response to I.14.d

See response to Interrogatory I.14.c.

Response to I.14.e

(a) The RABGAD computer code was developed in 1975 by K. F. Eckerman, who is now with Oak Ridge National Laboratory. Extensive validation of RABGAD was conducted at that time. The principal documentation of the computer code is described in NUREG-0002. E. Branagan has made some draft comparisons of estimated doses using the RABGAD code with dose estimates using the GASPAR computer code.

(b) The principal documentation is in NUREG-0002 and NUREG-0597.

Response to I.14.f

(a) The main validation of RABGAD was done by K. F. Eckerman (see response to f). The doses for the fuel cycle facilities that are included in Appendix D of the Supplement were estimated by E. F. Branagan, Jr. and L. Fairobent.

(b) The principal documentation is NUREG-0002, NUREG-0597 and the Draft.

Response to I.14.g

The RABGAD computer code was not used to estimate doses from exposure to radioactive effluents from the blanket fuel fabrication facility.

Response to I.14.h

The staff limited its estimates of population and health effects to an environmental dose commitment time of 100 years because predictions over long time periods (even as great as 100 years) are subject to great uncertainties. These uncertainties result from but are not limited to political and social considerations, population size and distribution, and competing health risk characteristics for time periods on the order of thousands of years, additional uncertainties result from geologic and climatologic effects.

Response to I.14.i

(a) Estimates of the population doses from exposure to radioactive effluents from the core fuel fabrication facility are contained in a computer printout identified as "A" (see Enclosure A). Estimates of the population doses to various body organs are given according to radionuclide, and pathways.

(b) Printout "A".

Interrogatory I.15

With regard to the estimate of total population dose commitment of less than .1 person-rem, identify the dose commitment factor utilized for each isotope and internal organ, including the whole body used in this calculation.

a. Identify where these dose conversion factors are documented, and produce any and all such documentation.

Response to I.15

(a) As stated in response to Interrogatory I.14(a), a sophisticated analysis of the doses from the blanket fuel fabrication facility was not necessary (see response to I.14(a) & (c)). The dose conversion factors used in estimating doses from exposure to radioactive airborne effluents from the core fuel fabrication facility are included in computer printout A (i.e., Enclosure A). The bases for these dose conversion factors is described in Chapter IV, Section J, Appendix A of NUREG-0002, Volume III.

(b) The main documents are Printout A, and NUREG-0002.

Interrogatory I.16

The following questions relate to the staff's calculation of the dose commitments from fuel reprocessing, estimated to be 140 person-rems.

a. At page D.30, with respect to the tritium dose calculation, identify the total person-rem dose from tritium exposure within the 50-mile limit,

b. Identify the contribution to the tritium exposure for the U.S. population beyond the 50-mile limit,

c. Identify the tritium dose commitment for the population beyond U.S. boundaries (i.e., remainder of the northern hemisphere).

d. - f. Answer questions a. through c. above for carbon-14.

g. - i. Answer questions a. through c. above for noble gases (e.g. Kr-85).

j. - l. Answer questions a. through c. above for halogens (e.g. I-131, I-129).

m. What is the bone surface dose commitment within the 50-mile limit with respect to each of the dose commitment from fuel reprocessing (Section D.2.4.3)?

Response to I.16.a

Doses to the U.S. population from exposure to airborne radioactive effluents from the fuel reprocessing plant were derived from the last page of the computer printout identified as "B". The estimated doses to the total body of the U.S. population from exposure to H-3, C-14, Kr-85 and I-129 are about 75, 66, 0.4, and 0.02 person-rems, respectively. The staff did not compute the dose to the population within 50 miles of the plant because a specific site for the plant has not been selected. However, the doses to the population within 50 miles of the plant would be less than the preceding values. Computer printout "B" is Enclosure B.

Response to I.16.b, c, d - f, g - i, j - l

See response to Interrogatory I.16.a.

c, f, i, l

The Staff has not made conclusions, nor would it be appropriate to do so, regarding environmental impacts beyond the U.S. boundary. (See Board Order, May 27, 1982).

Response to I.16.m

(a) As stated on p. D-30 of the Draft Supplement, over 90% of the estimated dose to the total body (i.e., 140 person-rems to the U.S. population) is due to tritium and carbon-14. Since tritium and carbon-14 tend to be dispersed uniformly throughout the body, the staff did not include estimates of the doses to other organs in the Draft Supplement. However, the doses to all of the other organs with the exception of bone would be approximately the same as the dose to the total body (see the last page of printout "B"). Note that the dose to the bone that is listed on the last page of printout "B" is a very conservative estimate because it is based on the use of an n-factor (i.e., a relative damage factor) of 5 for carbon-14. Since the ICRP no longer recommends the use of an n-factor, the dos to the bone would be less than the value derived from printout "B" (i.e., about 400 person-rems to the U.S. population). See response to I.16.a concerning doses to the population within 50 miles of the plant.

(b) The principal documents are: (1) Draft Supplementa, p. D-30; (2) computer printout "B"; and (3) ICRP Publication 2 (1959).

(c) The principal documents are: (1) ICRP Publication 30, Limits for Intakes of Radionuclides by Workers; and (2) Killough, G. G., et al, Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities, NUREG/CR-0150, Vol. 1 (1978).

Interrogatory I.17

Provide the answers to questions 14 a.-i. above with respect to the dose commitment calculations for fuel reprocessing rather than blanket and core fuel fabrications.

Response to I.17

a. - b. See response to Interrogatory I.16.m.

c. Dose conversion factors are listed on pp. 8-12 of computer printout "A" (Enclosure A). Since the specific activity model was used to estimate the dose from exposure to C-14, a dose conversion factor for C-14 is not listed in printout "A". In estimating doses from exposure to C-14 a body burden of 400 μCi was assumed to correspond to a dose of 5 rem.

d. The bases for the dose conversion factors is described in Chapter IV, Section J, Appendix A of NUREG-0002, Volume III.

e. See response to Interrogatory I.14.e.

f. See response to Interrogatory I.14.f.

g. The principal documentation for the population dose estimates for exposure to radioactive effluents from the fuel reprocessing plant is printout "B" (Enclosure B).

h. See response to Interrogatory I.14.h.

i. See computer printout "B" (Enclosure B).

Interrogatory I.18

a. Identify by page number and location on the page where the dose estimates cited in D.2.4.3 are found in the RABGAD computer print-out.

b. Supply the computer printout for this purpose.

Response to I.18

(a) Only one dose estimate is cited in §D.2.4.3. The dose estimate (140 person-rems to the total body of the U.S. population) is derived from the last page of printout "B". (Enclosure B).

(b) Printout "B".

Interrogatory I.19

At page D-30, the staff states that CRBR high-level wastes are projected to occupy less than 1% of the total inventory of a typical high-level waste repository. Identify the total inventory of a typical high-level waste repository for which this estimate was made.

Response to I.19

See response to I.12.a.

Interrogatory I.20

At page D-31, the staff estimates that the cumulative radiation dose to transportation workers and the general population would be approximately 24 person-rems per year for the CRBR and its related fuel cycle. Table D.16 on page D-32 provides a breakdown of the person-rems that, when summed, leads to 24 person-rems per year. With respect to each entry in Table D.16 that exceeds one person-rem per year, provide the underlying analysis, including all input assumptions that were used to estimate these person-rem exposures."

Response to I.20

As stated in Supplement 1 to the FES, the dose calculations were made using the methodology and assumptions set forth in NUREG-0170, Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, December 1977. Details of the calculations and underlying assumptions are contained in the attachment for each entry in Table D.16 that exceeds one person-rem per year.

TRANSPORTATION DOSE CALCULATIONS

The following calculations describe dose values given in Table D.16 of NUREG-0139, Supplement No. 1, for those values of one person-rem/year or greater. The calculation model is that used in NUREG-0170, Vol. 1, Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes.

Assumptions made regarding the dose to transportation workers:

- 2 crew members per truck or 2 guards per rail shipment
- 2 mrem/hr maximum dose rate
- crew exposed only during actual travel
- duration of exposure = distance/average speed
- shipment distance = 2500 miles except for plutonium dioxide
- shipment distance = 3000 miles for plutonium dioxide
- traffic conditions as follows:

	Population density		
	High	Medium	Low
Fraction distance traveled	0.05	0.05	0.90
Average truck speed (miles/hr)	30	50	55
Average train speed (miles/hr)	15	25	25

Using these assumptions, the time required was calculated as follows:

Truck shipments (except plutonium dioxide)

$$\left[\frac{0.05}{30} + \frac{0.05}{50} + \frac{0.90}{55}\right] 2500 = 47.6 \text{ hr}$$

Truck shipments (for plutonium dioxide)

$$\left[\frac{0.05}{30} + \frac{0.05}{50} + \frac{0.90}{55}\right] 3000 = 57.1 \text{ hr}$$

Rail Shipments

$$\left[\frac{0.05}{15} + \frac{0.05}{25} + \frac{0.90}{25}\right] 2500 = 103.3 \text{ hr}$$

The dose to crew members (person-rem/year)

$$= 0.002 \text{ rem/hr} \times \Delta T \ \frac{\text{hr}}{\text{shipment}} \ \times 2 \text{ persons} \times \text{SPY} \ \frac{\text{shipments}}{\text{year}}$$

The number of shipments per year (SPY) is given in Table D.14 and D.15.

For the shipments of interest, the doses to transport workers is obtained from the above equation as follows:

Type of shipment	SPY	ΔT (hr)	Annual dose (person-rem)
Plutonium dioxide	14	57.1	3.2
Fresh fuel assemblies	14	47.6	2.7
Spent Fuel assemblies	14	103.3	5.8
Spent blanket assemblies	12	103.3	4.9*
CRBRP solid radwaste	8	47.6	1.5
Reprocessing TRU waste/metal scrape	24	47.6	4.6
HLW	3	103.3	1.2*

*Corrected values (see concluding paragraph of this response).

The annual population dose to persons surrounding the transportation link while the shipment is moving is given by

$$\underset{\text{(person-rem/yr)}}{\text{Dose}} = 3.47 \times 10^{-10}(K)\left[\frac{f_r PD_r}{V_r} + \frac{f_s PD_s}{V_s} + \frac{f_u PD_u}{V_u}(f_o + 1.636\ f_1)\right]$$

$$\times\ PPS \times SPY \times FMPS$$

where f_r, f_s, f_u = fraction of distance traveled in rural, suburban, and urban areas, respectively

PD_r, PD_s, PD_u = population density in rural, suburan, and urban areas, respectively (persons per square mile)

V_r, V_s, V_u = average speed in rural, sururban, and urban areas, respectively (miles/hr)

f_o = fraction of urban travel on freeways or four-lane roads

f_1 = fraction of urban travel on city streets

PPS = average number of packages per shipment

SPY = number of shipments per year

FMPS = distance of shipment (miles)

K = dose rate factor

The above equation is taken from NUREG-0170, page D-6.

The only population dose to members of the general population greater than 1 person-rem/year is from truck shipments of TRU waste including metal scrap. For these shipments, the same assumptions were made concerning traffic conditions as were used for calculating doses to transportation workers (see above).

Then $f_r = 0.90$ $\quad$ $f_s + 0.05$ $\quad$ $f_u = 0.05$

$V_r = 55$ $\quad$ $V_x = 50$ $\quad$ $V_u = 30$

FMPS = 2500 miles

From Table D.15, SPY = 24 shipments per year.

The following assumptions were made: (Note: These population densities were taken from Table 4-6, page 4-16, of NUREG-0170)

PD_r = 15 persons/square mile

PD_s = 2,000 persons/square mile

PD_u = 10,000 persons/square mile

$f_o = 0.02$

$f_1 = 0.98$

$K = 10^3$ (This assumption, as made on page D-6 of NUREG-0170, treats the shipment as a single package point source. Therefore PPS = 1 by definition.)

Substituting the above values in the equation, the dose in person-rem/yr

$$= 3.47 \times 10^{-10} \times 10^3 \times \quad \frac{0.9 \times 15}{55} + \frac{0.05 \times 2000}{50} + \frac{0.05 \times 10{,}000}{30}$$

$$(0.02 + 1.636 \times 0.98) \times 1 \times 24 \times 2500$$

$= 0.61$ person-rem/yr

From Equation D-10 on page D-7 of NUREG-0170, the dose to the population during shipment stops, in person-rem/yr, is given by:

Dose (person-rem/yr) $= 2.54 \times 10^{-9} K (SPY) \times (\Delta T_r PD_r + \Delta T_s PD_s + \Delta T_u PD_u)$

where $\Delta T_r, \Delta T_s, \Delta T_u$ = total stop times in rural, suburban, and urban areas, respectively (hr)

and K, SPY, PD_r, PD_s, and PD_u are as used in the previous equation.

Assume shipments each take five days and four nights. Further assume that food/fuel stops occur for 1 hr/day in suburban areas and rest stops occur 14 hr/day in rural areas. Assume no stops in urban areas. Then

$$\Delta T_r = 14 \times 4 = 56 \text{ hr}$$

$$\Delta T_s = 1 \times 5 = 5 \text{ hr}$$

$$\Delta T_u = 0$$

Then the population dose during shipment stops is

Dose (person-rem/yr) $= 2.54 \times 10^{-9} \times 10^3 \times 24 \times \left[(56 \times 15) + (5 \times 2000)\right]$

$= 0.66$ person-rem/yr

The dose to persons in vehicles sharing the transport link with the shipment is calculated as follows:

<u>The dose to persons traveling in the opposite direction from the shipment</u>. Using Equation D-17 on page D-11 of NUREG-0170.

$(Dose)_{opp} = Q(K)(SPY)(FMPS)(P)(F)$

where $Q = 1.89 \times 10^{-7}$

P = number of persons per vehicle (average)

F = traffic factor

$$= f_r \frac{N_r' I_{fwy}}{V_{Tr}^2} + f_s \left(\frac{f_{rh} 2N_s' I_{fwy}}{(V_{Ts}/2)^2} + \frac{f_n N_s' I_{fwy}}{(V_{Ts})^2} \right)$$

$$+ f_u \left[f_{fwy} \left(\frac{f_{rh} 2N_u' I_{fwy}}{(V_{Ts}/2)^2} + \frac{f_n N_u' I_{fwy}}{(V_{Tr})^2} \right) \right.$$

$$+ f_{4\ell} \left(\frac{f_{rh} 2N_u' I_{4\ell}}{(V_{Ts}/2)^2} + \frac{f_n N_u' I_{4\ell}}{(V_{Ts})^2} \right)$$

$$\left. + f_{cs} \left(\frac{f_{rh} 2N_u' I_{cs}}{(V_{Tu}/2)^2} + \frac{f_n N_u' I_{cs}}{(V_{Tu})^2} \right) \right]$$

where f_r, f_s, f_u are as previously defined, and

f_{rh} = fraction of distance traveled in rush-hour traffic

f_n = fraction of distance traveled in normal traffic

f_{fwy} = fraction of travel on freeways or interstates

$f_{4\ell}$ = fraction of travel on four-lane roads

f_{cs} = fraction of travel on city streets

V_{Tr} = average velocity on freeways (miles/hr)

V_{Ts} = average velocity on freeways in suburban population density zones and on all four-lane roads (miles/hr)

V_{Tu} = average velocity on city streets (miles/hr)

$I_{fwy} = 2.9 \times 10^{-2} ft^{-1}$

$I_{4\ell} = 4.8 \times 10^{-2} ft^{-1}$

$I_{cs} = 1.5 \times 10^{-1} ft^{-1}$

N_r', N_s', N_u', = traffic count (average number of cars per hour traveling in one direction) in rural, suburban, and urban areas, respectively

In solving the equation, the simplifying assumption was made that all travel was in normal traffic. Thus, $f_{rh} = f_{cs} = 0$ and $f_n = 1.0$. It was further assumed that

$f_{fwy} = 0.98$

$f_{4\ell} = 0.02$

$V_{Tr} = 55$ miles/hr

$V_{Ts} = 50$ miles/hr

The one-way traffic count per hour (normal traffic) was assumed to be as given in Table 4-6 of NUREG-0170, page 4-16, rounded up to the next whole number. Thus,

$N_r' = 500$

$N_s' = 800$

$N_u' = 3000$

The average number persons per vehicle was assumed to be 2 in vehicles going in the direction opposite to that of the shipment; therefore P = 2. Then

$$(\text{Dose})_{opp} = 1.89 \times 10^{-7} \times 10^{3} \times 24 \times 2500 \times 2 \times \left[\frac{0.9 \times 500 \times 2.9 \times 10^{-2}}{(55)^2} + \frac{0.05 \times 800 \times 2.9 \times 10^{-2}}{(50)^2} + 0.05\left(\frac{0.98 \times 3000 \times 2.9 \times 10^{-2}}{(55)^2} + \frac{0.02 \times 3000 \times 4.8 \times 10^{-2}}{(50)^2}\right)\right]$$

$$= 0.14 \text{ person-rem/yr}$$

<u>The dose to persons traveling in the same direction as the shipment</u>.

Using Equation D-22 on page D-13 of NUREG-0170,

$$(\text{Dose})_{\text{same dir}} = 3.79 \times 10^{-7}\ (K)(FMPS)(SPY)(P)(F)$$

where the traffic factor, F, is the same as that used in the calculation of $(\text{Dose})_{\text{opp}}$ except that:

$I_{fwy} = 0.008$

$I_{4\ell} = 0.031$

$I_{cs} = 0.097$

The same assumptions apply as in the preceding case. Then

$$(\text{Dose})_{\text{same dir}} = 3.79 \times 10^{-7} \times 10^{3} \times 24 \times 2500 \times 2 \times \left[\frac{0.9 \times 500 \times 0.008}{(55)^2} + \frac{0.05 \times 500 \times 0.008}{(50)^2} + 0.05\left(\frac{0.98 \times 3000 \times 0.008}{(55)^2} + \frac{0.02 \times 3000 \times 0.031}{(50)^2}\right)\right]$$

$$= 0.079 \text{ person-rem/yr}$$

Summing the four components to the population gives:

Persons surrounding link while shipment is moving (off-link)	0.61 person-rem/year
Persons exposed during shipment stops	0.66 person-rem/yr
Persons on-link moving in opposite direction	0.14 person-rem/yr
Persons on-link moving in same direction	<u>0.08</u> person-rem/yr
TOTAL	1.49 person-rem/yr

The staff has continued its review of Draft Supplement to NUREG-0139. Errors in the dose calculation to transport workers were found for transport of spent blanket assemblies and high-level waste from reprocessing as noted above. These corrections result in an increase in the total doses to transport workers and general population from transport of radioactive material to approximately 30 person-rems per year. This correction will be made in the final supplement.

Interrogatory 21

On page J-1, paragraph 4, the statement is made that

> The results of the Staff's analyses of the realistic consequences of design-basis accidents were presented in the FES Table 7.2. The reported values appear to the Staff to be reasonable. This conclusion is based upon comparison of realistic dose consequences of the CRBRP design-basis accidents with the corresponding doses for some recently evaluated LWRs such as the Comanche Peak, Callaway, and Palo Verde plants, as shown in Table J.1. (emphasis added.)

Whereas, in the Staff's Response to Interrogatory 45 of Intervenors' Twenty-Fifth Set of Interrogatories, dated June 18, 1982, the Staff stated:

> ... the Staff is currently not depending upon the numerical values of calculated doses presented in Table 7.2 of the FES for its conclusions regarding CRBR accidents:
>
> a) Is the Staff relying on the calculated doses presented in Table 7.2 of the FES for its conclusions regarding CRBR accidents?
>
> b) Is the response to Interrogatory 45 still current? If not, please update it.

Response

a) No, not for the conclusions contained in FES Supplement. The Staff's conclusions are based on the risk estimates presented in the Draft FES Supplement rather than on the doses in Table 7.2. As stated in Appendix J the risks from Class 9 accidents dominate those from DBAs.

b) Yes.

Interrogatory 22

How can the doses from CRBR design-basis accidents be validated by comparing them against LWR accidents?

Response

The doses in FES Table 7.2 appear to the Staff to be reasonable and are similar in magnitude to the doses for some LWRs as illiustrated in Table J.1. in the Draft Supplement to the FES (DSFES). The DSFES does not say that the CRBRP design basis accident doses can be validated by comparing against LWR accidents. The Staff's independent analysis of design basis accidents will be performed as a part of the safety review of the CRBRP.

Interrogatory 23

a) With respect to each CRBR dose calculation in Table 7.2, explain in detail the nature of the similarities between LWR accidents and CRBR accidents that support using the doses from a LWR accident to validate the dose for the corresponding CRBR accident.

b) With respect to each CRBR dose calculation, identify each difference between the corresponding CRBR and LWR accident scenarios, i.e., each input assumption. Here we are seeking quantitative data, not qualitative responses.

c) Explain why each of the differences in "b." would not significantly affect the conclusion that "the reported values (for CRBR in Table 7.2) appear to the Staff to be reasonable." Here we are particularly interested in the comparison between Class 8 accidents and the similarities and differences between "large break LOCA and site suitability source term accidents."

d) Identify and provide all input data, computer codes (if applicable), formulas, notebooks, calculations, details of calculations, and other documentation used by the Staff to calculate the doses appearing in Table J.1 (p. J-2) under the column identified as CRBRP FES.

e) In your answer to "d." above, display (i) all the arthmetic used in the calculations, (ii) each computer code, (iii) each input to computer codes, (iv) each hand calculation, (v) each algebraic equation, and (vi) the value of each parameter of each equation.

The purpose of this interrogatory is to determine if the results in Table J.1 can be reproduced and to validate the results.

If the data requested under "d." and "e." above are made available for inspection and copying, provide a detailed guide (a "road map") that identifies the various pieces of data so that one can readily follow the calculations that are not relevant to the interrogatory. For example, if "microfiches of all computer runs" are made available, we would like to know which microfiches and which computer runs go with which calculations.

Response to Interrogatory 23 (a-c)

As explained in the answer to other interrogatories, there is no direct CRBR analogy to a light water reactor LOCA (due to design criteria). However for the general category of Class 8 accidents (as defined in the withdrawn proposed Annex to Appendix D, 10 CFR Part 50, that remains as Appendix A to Environmental Standard Review

Plan (ESRP) 7.1, NUREG-0555, dated February 1979) a number of fission product releases and attenuation assumptions are made that differ from those made for considerations of design bases accidents in Safety Evaluation Reports (SERs). The Staff has not conducted a parameter-by-parameter evaluation of the changes in expected offsite doses that would be estimated by using the assumptions suggested in ESRP 7.1 and those used in SER reviews for either light water reactors or CRBRP. However, the Staff has established guidelines for such dose estimates in SRP 7.1 for light water reactors. In comparing the dose estimates for Class 8 accidents in Table 7.1 of the FES for the CRBR, the Staff has found no bases for concluding that the environmental risk of accidents is not acceptably low. Specifically, the design basis accident doses presented in Table 7.2 of the FES are equal to or less than the 10 CFR Part 20.105(a), individual whole body dose limit of 0.5 rem per calendar year.

d) The information in Table J.1 on CRBR DBAs was taken from Table 7.2 of the FES. Therefore our response to Interrogatory 45 of the NRDC's Twenty-Fifth Set is applicable to this interrogatory.

e) See above response to 23d.

Interrogatory 24

The results of the Staff's FES accident analyses are said to be "realistic consequences." Presumably this implies that different assumptions were used to calculate the doses than the assumptions used to perform the SSST analysis in the SSR.

a) Identify quantitatively each specific difference in the assumptions in the two cases.

b) Where possible, identify the sensitivity of the results to the change in the input assumptions.

Response

The Staff's realistic analysis presented in the DSFES is based on the realistic consequence analysis methodology of the Reactor Safety Study (RSS). The conservative analysis of the Site Suitability Report (SSR) is a deterministic analysis and is based on a postulated Site Suitability Source Term (SSST). The assumptions for the probabilistic anaylsis are presented in the DSFES and the SSST dose calculation assumptions are discussed in the SSR. However, the probabilistic methodology assumptions of the DSFES are not comparable to the assumptions of the deterministic calculations of the SSR. Specifically there are differences in source terms, probability considerations, source term attenuation assumptions in the reactor complex, accident sequences, meteorology assumptions and dose estimating procedures. No quantitative comparison for each difference has been made.

Interrogatory 25

On page J-2, paragraph 2, the statement is made that:

> ... accidents of the types represented by those described in FES Table 7.2 for Classes 2-8 have a finite and relatively larger likelihood of occurrence during the operating lifetime of the CRBRP than the occurrence of Class 9 accidents.

On page J-3, paragraph 2, the Staff states:

> The Class 9 accident discussed in the FES involved a sequence and release representative of possible core disruptive accidents.... The frequencies of severe (Class 9) accidents at the CRBRP involving potential core disruption and containment failure...."

a) What is the basis for the Staff's view that the accidents identified as "site suitability source term" in Table 7.2 of the FES are Class 8 5 (design basis) rather than Class 9?

b) Define "core disruption" as used in Appendix J and elsewhere in the DES.

c) Define "core disruption accident (CDA)" as used in Appendix J and elsewhere in the DES.

d) Define "Class 8 accident" as used in Appendix J and elsewhere in the DES.

e) Define "Class 9 accident" as used in Appendix J and elsewhere in the DES.

f) Define "design basis accident" as used in Appendix J and elsewhere in the DES.

g) If any definition given in response to questions b)-f) above is different from the definition of these terms as used by the Staff in other Staff documentation, responses to Intervenor interrogatories or in the licensing hearing, explain in precise detail (i) any and all differences, (ii) the significance of these differences.

Response

a) The relation between the site suitability source term, DBAs, and Class 9 accidents is described in the Staff's prefiled testimony for the site suitability portion of hearings. In addition, the accident progression numbering system is in Appendix A to the Environmental Standard Review Plan, Section 7.1, NUREG-0555, dated Feb. 1979.

b) The term "core disruption" is considered to involve a change in fuel assembly geometry from its design configuration of such a degree that undercooling or reactivity increase might occur. The Staff views the extent of core disruption in different ways for different purposes. In reviewing the adequacy of design measures to prevent initial fuel failure from propagating we require that such failures not affect additional failures beyond a limited region, such as a few fuel pins. However, in evaluating the risks from CDAs we consider more wide spread disruption which would have radiological consequences.

c) The term "core disruptive accident" applies to very unlikely event(s) which lead to core disruption. This is not a single accident, but rather a spectrum of accidents (CDAs) involving, with decreasing likelihood an increasing degree of degradation in core geometry.

d) From Appendix A to Environmental Standard Review Plan 7.1, NUREG-0555, dated Feb. 1979, "Class 8 accidents are those considered in safety analysis reports and AEC (sic NRC) Staff safety evaluations."

e) From Appendix A to Environmental Standard Review Plan 7.1, NUREG-0555, dated Feb. 1979, "The occurrences in Class 9 involve sequences of postulated successive failures more severe than those postulated for establishing the design basis for protective systems and engineered safety features."

f) The meaning of "DBA" is as discussed in the Staff prefiled site suitability testimony.

g) The Staff intent is that the above terms have a constant meaning. However, the discussion accompanying the use of these terms may differ for different applications.

Interrogatory 26

On page J-3, paragraph 4, the statement is made that:

> Core disruption could be initiated by...
> 3) core-wide fuel failure as exemplified by propagation of local fuel faults (FFP).

a. Does the Staff view core disruption as requiring "core-wide fuel failure"?

b. Would partial core fuel failure constitute core disruption in the Staff's view?

c. How many fuel pins or assemblies would have to fail to meet the Staff's definition of (i) core disruption; (ii) core disruptive accident?

d. If core disruption is initiated:

   i) Would it be reasonable to assume that full core involvement is a likely outcome?

   ii) Would it be prudent to assume that full core involvement is a likely outcome for purposes of the environmental and site suitability review of a reactor of the general size and type as the CRBR?

   iii) If your answer to i) or ii) above is no, explain the reason for your answer.

Response

a) Yes, but we would consider the term "core-wide" to include a range of conditions and not necessarily only a condition in which the whole core is disrupted.

b) Possibly, it would depend on the situation. We have no specific percentage of the core in mind when we use the term core disruption.

c) No specific number of fuel failures is associated with the Staff's definition of core disruption.

d) (i) Not necessarily, the disruption might terminate depending on conditions.

(ii) The Staff has assumed full core involvement in its Appendix J analysis.

Interrogatory 27

Identify and produce every document that was relied upon for Staff's (and Staff consultants') judgement that "there is sufficient inherent redundancy, diversity, and independence in the SGAHRS and DHRS systems to achieve a core degradation frequency due to LOHS events of less than 10^{-4} per reactor" (p. J-3, paragraph 6).

Response

The Staff did not rely on any specific documents for its judgement of the estimated bounding frequency of LOHS events. Instead the basis of this judgement was the cumulative knowledge and experience of the Staff and its consultants.

As described in Appendix J the frequency of LOHS is based in part on the redundancy and diversity of the CRBR decay heat removal systems and in part on the reliability of PWRs, which have similar redundancy and diversity in their auxiliary feedwater system (AFWS) to the CRBR SGAHRS. Evaluations of PWR AFWS reliabilities including that in WASH-1400 and more recent studies suggest that failure frequencies in the range of 10^{-5} to 10^{-4} per demand may be achieved. The general trend of these studies rather than a specific case is the basis for the conclusion that the CRBR SGAHRS can achieve similar reliability. Because CRBR also has a DHRS to back up the SGAHRS we believe the LOHS failure frequency will be below 10^{-4} per demand. The formal reliability program at CRBR adds further assurance that this will be the case.

Interrogatory 28

Identify quantitatively the reliabilities "typical(ly) achievable for PWR auxiliary reactor-year systems" (J-3, paragraph 6).

Response

We believe that it is possible to achieve an unreliability in the range of 10^{-5} to 10^{-4} per demand for a PWR auxiliary feedwater system.

Interrogatory 29

a) Identify quantitatively the reliability of each of the steam generator auxiliary heat removal system component and each component of the direct heat removal system of the CRBRP and PWR, respectively.

b) Identify separately the combined reliability of all of these systems taken together for CRBR and separately for PWRs in order to show the margin in terms of overall systems reliability that has been applied to account for common cause and multiple failures.

Response

The Staff has not relied on specific reliability data for components or specific reliability studies for either CRBR or PWRs, for its Appendix J analysis. Reliability analyses of CRBR which have been completed or which may be undertaken in the future will be considered as part of the Staff's review of the CRBR reliability program most probably at the OL stage of licensing. Reliability data and analyses for PWRs do exist but as stated above they were not relied on in any detail for Appendix J.

Interrogatory 30

a) At the top of page J-4, identify each and every component of an "effective reliability program."

b) Identify and produce any and all documents describing such a program.

c) Identify and produce each document that the Staff has relied upon as a basis for its conclusion that high reliability in the final design and operation of the CRBR can be achieved through an effective reliability program.

Response

No specific documents have been generated by the Staff to describe the elements of an acceptable reliability program. The applicants' proposed reliability program (Appendix C of the PSAR) is currently under review to determine what changes, if any, will be needed in the program. Consequently we have not relied on the applicants' proposed program or any documents describing it. However, the Staff believes that an effective reliability program for CRBR can be developed. Some important general elements of such a program are: (1) formal documentation of reliability procedures including those related to operation, testing, surveillance, and maintenance; (2) utilization of appropriate reliability techniques such as fault trees, event trees, failure modes and effects analyses, and probabilistic in risk assessment; (3) performance of tests on components and systems to establish a quantitative data base; (4) systematic elimination of common cause failure modes.

Interrogatory 31

Quantify and give the uncertainty values for the LOHS probability contribution for CRBR from simultaneous loss of offsite and onsite AC electric power and the steam-driven auxiliary feedwater trains.

Response

The Staff has not performed a quantitative uncertainty analysis of the LOHS frequency used in Appendix J for CRBR. The Appendix J frequencies are estimates of upper bounds of frequencies of events which could lead to CDAs. The estimates are based on the cumulative experience and judgement of the Staff and its consultants rather than on quantitative reliability analyses.

Interrogatory 32

Identify and produce each and every document that the Staff relied upon for its conclusion that a significant contributor to the LOHS probability for the CRBR would be from simultaneous loss of offsite and onsite AC electrical power and the steam-driven auxiliary feedwater train (J-4).

Response

The response to this is the same as for #27 in that the possibility of the subject sequence was considered in arriving at the LOHS frequency estimate.

Interrogatory 33

Identify and produce each and every document relied upon by the Staff for its conclusion that "for these reasons LOCAs are not considered credible (i.e., design basis) events at CRBRP" (J-4). Cite the appropriate pages.

Response

The cumulative knowledge of the Staff and its consultants rather than a specific document were relied upon by the Staff for its conclusions in Appendix J regarding whether LOCAs are DBAs for CRBR. This issue was also discussed in the SSR and the Staff's prefiled testimony for the site suitability hearings.

Interrogatory 34

Identify and produce each and every document the Staff relies upon for its conclusion that "the frequencies assumed for LOHS adequately bounds the LOCA contributions to core disruption frequency" (J-4). Cite the appropriate pages.

Interrogatory 35

Identify and produce each and every document the Staff relied upon for its conclusion that "the frequency assumed for LOHS core degradation sequences adequately bounds the flow blockage contribution to core disruptive frequency" (J-4). Cite the appropriate pages.

Response to Interrogatories 34-35

As indicated in the response to Interrogatory 33 the Staff has explained why LOCAs and flow blockage at CRBR can be made very unlikely, in its site suitability testimony. No specific documents were relied on for these conclusions. Instead, the Staff's general experience and knowledge formed the basis for the judgement. In general the fact that passive design features are primary in preventing LOCAs and flow blockage for an LMFBR, whereas active systems are primary in preventing LOHS, led us to the conclusion that the assumed LOHS frequency would be large enough to take into account the LOCA and flow blockage frequencies.

Interrogatory 36

Identify and produce all documentation used as a basis for the Staff conclusion that "although the Staff review of these systems is not complete, it is the judgement of the Staff that there are sufficient inherent redundancy, diversity, and independence in the overall shutdown system designs to expect an unavailability of less than 10^{-5} per demand" (J-4). Cite the appropriate pages.

Interrogatory 37

In the following sentence on page J-4, what did the Staff assume was the unavailability rate for light water reactor shutdown systems? What is the basis for this estimate?

Interrogatory 38

a) Identify and produce all documentation relied upon by the Staff for the estimate of the unavailability rates for the light water reactor shutdown systems. Cite the appropriate pages.

b) How does the Staff reconcile this estimate with the estimate appearing in the proposed ATWS rule?

Response to Interrogatories 36-37-38

In NUREG-460, "Anticipated Transients Without Scram for Light Water Reactors, " Vol. I, Section 4.3, an estimate of the frequency of ATWS for typical LWRs was given as 2×10^{-4} per year. Estimates in this same range were subsequently quoted by the Commission in its statement regarding ATWS rulemaking. The currently proposed design of the CRBR shutdown system includes two independent and diverse systems, each of which is comparable to an LWR shutdown system. Any modifications to this design or to the applicant's reliability program needed to assure high reliability will be identified in the SER. Because the design and the reliability program are not final they have not been definitive in making the reliability estimate.

Because of the potential for common mode failure it is not appropraite to attribute ATWS frequencies to CRBR as low as might be obtained by multiplication of the unreliabilities possible for the primary and secondary shutdown systems. Instead, to be conservative, a range of 10^{-5} to 10^{-4} per year has been selected as a reasonable preliminary estimate for CRBR. Although we believe the most likely CRBR ATWS frequency to be on the low end of this spectrum we have used 10^{-4} per year as the bounding value for purpose of risk estimates in Appendix J.

Interrogatory 39

At page J-4, the Staff states "the CRBRP fuel design will be required to have an inherent capability to prevent rapid propagation of fuel failure from local faults."

a) Identify fully each requirement that will be imposed by the Staff on the CRBR fuel design to provide this inherent capability.

b) Describe fully the basis for the Staff's view that this inherent capability will in fact prevent rapid propagation of fuel failure from local faults.

c) Identify and produce all documentation relied upon by the Staff for its conclusion that the CRBR fuel design will have the required inherent capability to prevent rapid propagation of fuel failure from local faults.

d) Identify each and every system the Staff relies upon for its statement that systems to detect more slowly developing faults will also be required.

Response

a) Although the general intent of the requirement is known the final wording and specific details of the requirement or any associated criteria or confirmatory programs will not be developed until the SER is prepared.

b) The Staff and its consultants have extensive knowledge and experience related to fuel design. Based on this knowledge and experience we have concluded that fuel cladding can be fabricated with sufficient strength and ruggedness that a local failure in one pin will not cause a rapid failure in adjacent pins.

c) No specific documentation has been relied on for this conclusion. Instead the general knowledge and experience of the Staff and its consultants is the basis for the conclusion.

d) The details of the systems to prevent propagation of slowly developing faults are not final at this time. The criteria for such systems will be reported in the SER. Based on the Staff's general knowledge of the feasible design of such systems we are confident that it is possible to install a sufficiently reliable system for detection of fuel faults at CRBR. It is anticipated that a relatively simple detection system design meeting NRC standards for accident monitoring instrumentation will suffice.

Interrogatory 40

a) Quantify the frequency of fuel failure propagation referred to by the Staff in their statement at the top of page J-5: "Therefore, the frequency of fuel failure propagation is considered very low." What is meant by the term "very low" in this regard?

b) What is the uncertainty in this estimate?

Interrogatory 41

At the top of page J-5, the Staff states, "...the frequencies attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruptive frequency from fuel failure propagation."

a) What is the analytical basis for this conclusion?

b) Identify and produce all documents utilized by the Staff that form the basis for this conclusion.

Response to Interrogatories 40(a) and (b), and 41

Because prevention of fuel failure propagation is primarily achieved by passive design measures, i.e., mechanical strength and ruggedness, supplemented by a relatively simple detection system related to such propagation, we believe the frequency of CDAs related to fuel failure propagation will be only a fraction of the bounding CDA frequency which includes failure of active systems. The Staff's general judgement and experience rather than specific documents is the basis for this conclusion.

Interrogatory 42

In the summary in the first full paragraph on J-5, the Staff has summed the frequencies of core disruption and estimates a combined or net frequency of 10^{-4} per reactor year or less. Since 10^{-4} per reactor year or less was the estimated frequency of each of the classes of initiators identified above, explain how the Staff arrived at the conclusion that the sum of these is no larger than each of the individual contributions.

Response

The presented initiator class frequencies represent, in each case, a judgement that each frequency is no greater than 1×10^{-4} per reactor year and is expected to be appreciably smaller. Further, the scoping nature of this analysis is consistant with order of magnitude estimates of individual contributors to core disruption. In each case, frequencies are rounded off to the next largest order of magnitude to obtain bounding estimates. Thus it is from the viewpoint that each class frequency is expected to be appreciably smaller than 1×10^{-4} per reactor year that the judgement is made that the sum of these frequencies is no greater than 1×10^{-4} per year.

Interrogatory 43

On page J-6, the Staff has assigned conditional probabilities to the primary system failure for Categories I, II, and III, and separately for Category IV.

a) Explain fully the basis for the Staff's quantification of these failure rates.

b) Identify the uncertainty in each estimate.

c) Identify and produce all documents relied upon by the Staff for its assessment of these failure rates. Cite the appropriate pages.

We are not seeking a response that speaks in generalities. We wish to know the specific documentation that the Staff is relying upon for the basis of these estimates.

d) Did the Staff, for example, consider CRBRP-1 as one of the documents that it relied upon for these estimates?

e) Did the Staff examine and rely in any way on any probabilistic risk assessment, such as risk assessments performed for SNR-300, in reaching its conclusions with regard to these conditional probabilities?

Response

The Staff's estimate that the conditional probability of primary system failure Category IV is 0.1 was based on two points. First, for simplicity a single general CDA initiation frequency of approximately 10^{-4}/year which included the combined frequencies of various specific CDA initiators was used. However, the specific CDA initiators do not have equal potential for resulting in an energetic CDA. The fraction, 0.1, was therefore in part employed to compensate for this simplification. Second, the Staff's general knowledge of and experience with the extensive research on the phenomena that may occur in a core disruptive accident has led to the conclusion that energetics large enough to cause a Category IV type failure are relatively unlikely to occur even if a

CDA is initiated. Factors which have been a consideration in this conclusion are that (a) incoherent fuel failures and material rearrangement are more likely than the coherent behavior associated with high energetics, (b) small criticalities which disperse fissionable material without energetics are more likely than large energetic criticalities, (c) the heterogeneous core design slows down power escalations due to voiding and minimizes the potential for rapid reactivity insertion due to fuel motion and (d) the effect of the upper internals structures in mitigating CDA generated forces. The Staff did not rely in any specific way on CRBRP-1 or on the risk assessments performed for SNR-300.

Interrogatory 44

a) On page J-7, what is the containment isolation unavailability for operating pressurized water reactors?

b) What is the unavailability for boiling water reactors?

c) Identify and produce all documents relied upon by the Staff for these estimates.

Response

With regard to LWRs containment isolation unavailability is typically in the range 10^{-4} to 10^{-2} per demand. This estimate is based upon experience and knowledge in general; no documents were specifically relied upon.

Interrogatory 45

a) What is the basis for the Staff's estimate that failure to recover AC power before containment failure occurs is estimated to have a frequency of about 10^{-2} per demand?

b) Identify and produce all documents relied upon by the Staff for this estimate, citing the appropriate pages.

Response

Typical data for restoration of transmission line outages indicates that the chance of failure to recover for roughly a 24 hour period from loss of offsite power is 10^{-2} or less per demand. It is the Staff's judgement that the same value can be used as a conservative estimate for CRBRP. This judgement was based on general knowledge and experience; however, a point of reference can be found in Figure III 6-4 on page III-87/88 of Appendix III, Reactor Safety Study, WASH-1400 (NUREG-75/014), October 1975, a copy of which is located at the Commission's H Street Public Document Room.

Interrogatory 46

a) What is the basis for the Staff's conclusion that an unavailability of less than 10^{-2} per demand is feasible for containment isolation at the Clinch River Breeder Reactor?

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

Based on the Staff's general knowledge and experience with LWRs and LMFBRs, it is judged that failure of containment isolation is not more likely at CRBRP than at an LWR, and that an unavilability of less than 10^{-2} per demand is feasible. This judgement is also supported by the similarity in electrical controls and other equipment used to isolate containment.

Interrogatory 47

a) What is the basis for the Staff's assumption (on page J-9) that "because there are more than one million pounds of primary coolant sodium a dense aerosol (10-100 micrograms/cc) could be airborne in the RCB."

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

Repeated experiments with sodium fires have failed to produce aerosols of density as high as 100 micrograms/cc. At these concentrations the deposition rate becomes so high that concentrations cannot be further increased. The behavoir of an aerosol resulting from a large sodium fire is, for example, illustrated in BMI-NUREG-1989, Figure 8 (p. 41). Thus 100 $\mu g/cm^3$ (micrograms/cc) is conservative for the characterization of the upper limit of aerosol concentration resulting from a large sodium fire. During the period in which the pool is boiling, on the order of 5000 kg/hr of sodium will be released into the RCB atmosphere. This source rate is consistant with an airborne concentration of 10 $\mu g/cm^3$ or greater. For example, the steady state airborne concentration is simply, $S/V\alpha$, where S = average source rate in $\mu g/hr$, V = containment volume $\sim 10^{11} cm^3$, and α = removal rate. Using a value of S equal to 5×10^{12} $\mu g/hr$ yields a steady state airborne concentration of 50 $\mu g/cm^3$. The Staff does not rely on any specific documents for its judgement of the aerosol density. NUREG-1989 mentioned above is one example of a document that supports this range.

Interrogatory 48

What is the basis for each of the bounding frequency estimates of containment release identified in Table J.2 and page J-8 given in units of per reactor year for each of the four CDA classes?

Response

The bounding frequency estimates of containment release are given in Table J.2 on page J-8. Their basis is provided in Appendix J, for example, multiplying the generic frequency of CDA initiator occurrence (1×10^{-4} CDA per reactor year) times the conditional probability that the initial energy release is large enough to be in primary system failure Category IV (0.1 per CDA) times the probability of failure of the containment isolation function upon demand (1×10^{-2} per demand) gives the bounding frequency estimate for CDA Class 4 in Table J.2, 10^{-7} per reactor year.

Interrogatory 49

a) On page J-9, display the supporting analysis for the conclusion that "leakage from the RCB considering CDA Class I involves design leakage at rates of 10^{-4} to 10^{-5} per hour and filtered venting which is 97-99% efficient."

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

The design basis leakage rate for the CRBRP containment is 0.1% of the containment volume per day at the design pressure of 10 psig; 0.1% per day is about 4.2×10^{-5} per hour. If the leakage rate varies as the square root of the pressure differential, it will be generally in the range of 10^{-4} to 10^{-5} per hour.

Although the design of the filtering system has not been finalized, the applicant indicates in CRBRP-3, Volume 2, page 4-9, that the filtering system is expected to be able to achieve 97-99% efficiencies. It is the Staff's opinion that such efficiencies are feasible.

Interrogatory 50

a) What is the basis for the Staff's estimate on page J-9 that "in CDA Class II, approximately 57% of the RCB atmosphere will be released soon after failure by overpressurization because the RCB pressure drops from about 2.3 atmospheres (abs) to one atmosphere (abs)."

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

Considering normal design margin required by code requirements it was estimated that, conservatively, the containment vessel should hold at least twice its design basis pressure of 10 psig; this is about 2.3 atmospheres (abs). If overpressurization failure produced a leak of sufficient size, the containment vessel atmosphere would vent down to atmospheric pressure, releasing about 57% of its gaseous contents. These are rough estimates; a larger fraction could be so vented if the containment held to higher pressure, and a smaller fraction might be so vented if the failure produced a more limited leak.

Interrogatory 51

What is the basis for the Staff's estimate that "the leakage rate to the environment considering failure of the containment to isolate a ventilation supply or exhaust line (CDA Classes III and IV) is estimated to be on the order of 10^{-1} to 10^{-2} per hour, similar to the rates after overpressurization failure" (J-9)?

Response

The estimate is based on the condition that since free communication will exist between the Reactor Containment Building (RCB) and the environment, the RCB will be close to atmospheric pressure considering failure of the containment to isolate a ventilation supply or exhaust line (or after the overpressurization failure has reduced the RCB pressure to near atmospheric). Therefore in this condition, the leakage rate from the RCB to the environment will depend on the volume and thermal energy input into the RCB from the reactor cavity and head release (containment isolation failure case). Leakage rates in the range of 10^{-1} per hour adequately bound those anticipated for the head release and leakage rates around 10^{-2} per hour adequately bound leakage rates anticipated for longer term pool releases within the failure to isolate containment scenario. While no documents were specifically relied upon for these estimates, CRBRP-3, Volume 2 (page 3-103 and 3-173) indicate that these estimates are conservative.

Interrogatory 52

a) What is the basis for the Staff's estimate that sodium boiling will occur in a 100-200 hour period and not a longer or shorter period?

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

This comes from considerations of a simple heat balance taking account of the decay heat and other input energy sources. While no specific documents were relied upon to make this estimate, the Staff is not aware of analyses that contradict this estimate. CRBRP-3 (pages 3-21 and 3-25) for example indicates a boiling period of about 120 hours.

Interrogatory 53

a) What is the basis for the Staff's estimates of the head release fractions that were selected in Table J.3 at page J-9?

b) Present all analytical calculations that were used to provide these estimated releases.

c) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

The head release fractions (Table J.3) were selected on the basis of judgement from consideration of general LMFBR research of energetic CDAs involving a bubble of vaporized fuel material rising against the reactor vessel head, giving consideration also to the relative volatilities of different types of fission products and other materials. The selections were therefore not based on a set of analytical calculations or on any specific documents.

Interrogatory 54

On page J-10, the Staff states that "typical analysis for similar sodium aerosol conditions indicate deposition rates in a single chamber of between 0.5 and 1.0 per hour."

a) What analyses are referred to by the Staff here?

b) Identify and produce all documents relied upon by the Staff for this estimate.

c) What size chamber is involved here, and what are the environmental conditions assumed in the chamber for which these estimates are appropriate?

Response

a) In the concentration range under discussion, many calculations have led to deposition rates in the range 0.5 to 1.0 per hour. A typical example is given in Figure 6 of BMI-NUREG-1989 (attached to response Number 47 of this set of responses). In this figure, the deposition rate is greater than 0.5 for the concentrations under discussion.

b) No specific documents were relied on. However the report BMI-NUREG-1989, pages 36-41, provides a typical example of this sort of calculation.

c) CRBR geometry, 310^{0}K, in the example cited.

Interrogatory 55

Display the subsequent calculations that form the basis for the Staff's conclusions that "considering leakage rates between 10^{-2} and 10^{-1} per hour, therefore indicate that between 1% and 20% of the particulate airborne fission products may eventually be released to the environment."

Response

Consideration of differential equations for the removal rates, involving terms of the e^{-Rt} type, leads to the (leaked) release fractions being the result of ratios $R_1/(R_1 + R_2)$ where R_1 is "leakage rates between 10^{-2} 10^{-1} per hour" and R_2 is "deposition rates in a single chamber of between 0.5 and 1.0 per hour" (page J-10); the results range from 1% to 20%.

Interrogatory 57

a) What is the basis for the Staff's assumption that this release would not occur until about 24 hours after the head release and about 14 hours after pool boiling begins (J-10)?

b) Identify and produce all documents relied upon by the Staff for this estimate, citing appropriate pages.

Response

Heat balance estimates indicate that boiling begins at about 9 hours. Pressurized hydrogen would increase in the containment building at rates dependent on the rate of sodium boiloff and sodium concrete reactions. The Applicants' analysis indicates that venting, purging, and cooling should begin at about 36 hours (CRBRP-3). Based on the Staff's knowledge of the possibility of sodium concrete reaction rates greater than assumed by the Applicant we have selected 24 hours as a reasonable estimate of the time at which venting, purging and cooling would be necessary. It was assumed that one of the active systems would fail to function, causing immediate containment failure at 24 hours.

Interrogatory 58

a) Identify and produce all documentation and calculations related to the consequence model referred to on page J-10.

b) Provide all model calculations related to the CRBR, including all sensitivity studies, performed by the Staff using this consequence model and identify for each the principal changes and assumptions that were made in order that the outputs for the various sensitivity studies could be distinguished from one another and from the primary calcluation used to provide the data in Table J.2 at page J-8.

Response

a) The consequence model referred to on page J-10 is discussed in Section J.1.2 and illustrated in Figure J-1.

b) The consequence model referred to on page J-10 was not used to generate data in Table J.2. On the contrary, the data in Table J.2 were used as input to consequence model referred on page J-10.

Interrogatory 59

To the extent that each of the estimates of percent of core inventory released to the environment for each of the 4 CDA clases presented in Table J.2 and page J-8 are based on analyses other than that provided in response to Interrogatory 37 above, provide the analysis displaying explicitly all calculations, computer models, and input and output assumptions. If these estimates were performed by computer calculations, identify and produce a copy of the computer code, the computer input data, and the output, and identify for each input and output data sets where the calculated release numbers for core inventory release to the environment are found. That is, on what page and line of the output are these estimates found?

Response

The values of the percent of core inventory released to the environment for each of the 4 CDA classes presented in Table J.2 on page J-8 are based on contributions from three sources: vessel head releases, pool releases and dry cavity releases. The head releases to the Reactor Containment Building (RCB) are specified in Table J.3 on page J-9. The head releases in primary system failure Category III are conservatively used for CDA Class 3 of Table J.2. CDA Classes 1, 2 and 4 conservatively use primary system failure Category IV head releases.

Pool releases to the RCB depend on the relative volatility of the specific isotopes compared to that of the sodium. All I and Cs-Rb remaining in the pool is assumed to be released to the RCB. About 50% of the remaining Te-Sb, and Ba-Sr isotope groups are assumed to be released and none of the solid fission product groups (Ru and La) are assumed to be released to RCB during the pool boiloff process.

After cavity dryout, about 12% of the remaining Te-Sb, and Ba-Sr isotope groups (about 5% of their total inventory) and about 5% of the remaining Ru and La groups (nearly 5% of their inventory) are estimated to be released to the RCB.

Once the input of sodium and fission products into the RCB are determined, the releases to the environment can be estimated. For each CDA Class and RCB source term (head, pool, dry cavity releases) the containment leakage mode (filtered or unfiltered) and rate, as well as the approximate sodium aerosol concentration in the RCB, are estimated.

Thus the ratio of leakage rate to leakage plus fallout rates, as discussed on page J-10 of the FES Supplement, are estimated for each CDA Class and RCB source term. This ratio, when multiplied by the fraction of each isotope in the RCB, results in an estimate of the fraction of each isotope released out of the RCB. If filtering is operative, the filtering inefficiency (1 minus filter efficiency) is also multiplied with the release fraction to obtain the environmental release fraction. Once the release fractions to the environment are calculated, for each isotope group of each RCB source term of each CDA class, they are combined to form a total release fraction for each isotope group for each CDA class. The releases represented by a set of isotope group release fractions are then used as input into the consequence model. The consequence model requires a constant rate environmental release; thus the environmental releases for CDA Classes 1 through 4 were assigned start and duration times. These times are assigned such that the input environmental releases to the consequence model occur earlier and for a shorter duration than best estimates would indicate. This is done to insure that a conservative bias is applied with regard to the use of this data. No computer calculations or specific documents were relied on for the analysis.

Interrogatory 60

With regard to WASH-1400 and NUREG-75/014, cited in footnote 1 of Table J.2 at page J-8, identify the page number(s) where these estimates are made. Identify and produce any other documents relied upon by the Staff for this estimate.

Response

The footnote 1 of Table J.2 at page J-8 of the Appendix J of Draft FES Supplement does not refer to any specific release estimates. It refers only to the background of the groups of the isotopes and release mechanisms presented in Appendix VII of WASH-1400, NUREG-75/014.

Interrogatory 61

What is the core inventory for each of the isotopes of each element identified in Table J.2?

Response

The CRBRP core inventories of the significant isotopes used in the consequence model are listed in Draft FES Supplement Table J.4.

Interrogatory 62

a) What is the basis for the STaff's assumption that filtered venting begins at about 24 hours after CDA initiation?

b) Identify and produce all documents relied upon by the Staff for this estimate.

Response

See response to Interrogatory 57 of this set.

Interrogatory 63

a) Identify and produce the analytical support for the conclusion on page J-11 that "the doses associated with this accident class are not expected to exceed 10 CFR 100 guidelines." Explicitly display all dose calculations and results.

b) Did the Staff calculate the bone surface dose?

c) If so, what dose conversion factor did the Staff use?

d) Over what period of time did the Staff integrate the releases to the environment for purposes of this calculation?

e) What is the basis for the Staff's assumption that this integration period was adequate and covered the entire period of the passage of the cloud?

f) Who performed the radiological consequence modelling? That is , identify the principal experts who were responsible for providing the input assumptions and exercising the codes?

g) Were these calculations performed in-house by the NRC Staff, or by an outside contractor?

h) If outside, produce all documentation between the Staff and the contractor related to these calculations.

Response

a) A conservative analysis of the accident resulting in the referenced class accident was presented in the Section III.D of the Site Suitability Report (SSR). The calculated doses presented in Table IV of that report are significantly lower than the 10 CFR 100 dose guidelines discussed in SSR Section III.D. The realistic consequences of the referenced accident class will be even lower than those presented in SSR, Table IV. Also see the response to Interrogatory 59.

b) No. The Staff presented the bone doses in the DSFES.

c) The transcript of the Site Suitability Hearing held in Oak Ridge, Tenn., for August 25, 1982, identifies the dose conversion factors for the various calculations. See transcript page 2390.

d) The Staff calculations presented in the SSR were based on integration of the two hour releases for doses at the Exclusion Area Boundary and the integration of the thirty day releases at the outer boundary of the Low Population Zone.

e) The periods of integration are specified in 10 CFR 100.11.

f) The calculations of the doses presented in the revision to the SSR were performed by Larry Bell of the Accident Evlauation Branch. The Core Disruptive Accident (CDA) and the release fractions presented in the Draft FES Supplement were estimated by the NRC contractor, Science Applications Inc. (see Response No. 71). The probabilistic risk estimates, using the data in Table J.2 as an input to the CRAC Code, were made by the NRC Staff, Mohan Thadani, Accident Evaluation Branch. The inputs for a year's data of site meteorology used in the CRAC analysis, was provided by the NRC Staff Irwin Spickler, Accident Evaluation Branch. The population distribution projected to the year 2010 over a radius of 500 miles around the CRBRP, used in the CRAC analysis, was provided by Charles Ferrell, Site Evaluation Branch.

g) The calculations were performed in-house.

Interrogatory 64

a) Identify and produce all output data for the consequences modelling discussed on page J-13, unless it has already been provided in response to other Interrogatories above.

b) Identify those calculations that give the peak number of consequences, in terms of both early fatalities and latent cancers.

c) By what factor does early evacuation reduce

i) the peak early fatalities, and

ii) the peak number of latent cancers?

d) What assumptions were made with regard to the dose required to produce an early fatality?

e) Display the number of early fatalities as a function of probability for various assumptions of evacuation.

f) Display the number of latent cancers as a function of probability for various assumptions regarding evacuation.

Response

a) The output data of the CRAC calculations are in the form of computer printouts, and are available in the Staff files. The computer printouts can be examined and/or copied in the NRC Staff offices.

b) The computer printouts appropriately identify the requested information. See above response to 64a.

c) The Staff has not performed any sensitivity analysis of the early evacuation assumptions.

d) The CRAC analysis is based on the Reactor Safety Study consequence model. The dose response relationship "B" in Figure 2 of NUREG-0340 is used in the CRAC code to estimate the early fatalities.

e) The Staff has not performed an analysis of the sensitivity of the health effects to the evacuation assumptions.

f) The Staff has not performed an analysis of the sensitivity of the latent cancers to the evacuation assumptions.

Interrogatory 65

Identify and produce the computer code and code users manuals and other relevant documents for the consequence model described at pp. J-13 and J-14.

Response

The CRAC code is the only code used for the referenced consequence model. The CRAC computer manual was previously made available to you for inspection and copying.

Interrogatory 66

On page J-18, quantify what is meant by the statement "compliance with current NRC siting, structural, and seismic design criteria and with 10 CFR 73 for physical security provides assurance that reactor-related risks from external events and sabotage are adequately low." In other words, quantify, either by a bounding estimate or a range of estimates, what is meant by the term "adequately low."

Response

As stated in the FES it is difficult to quantify risks from sabotage or external events beyond the design basis. However, it is the Staff's judgement that compliance with regulatory criteria related to external events and sabotage reduces the risks from such events to approximately the same low level as those presently attributed to LWRs.

Interrogatory 67

a) What CDA energetic level is required to produce a containment failure mode caused by either spray fire or missile?

b) What is the basis for the Staff estimate?

c) Identify and produce all documents relied upon by the Staff for this estimate.

d) What is the conditional probability of production of such a spray fire or missile, given the occurrence of a core disruptive accident?

e) How does the Staff reconcile its statement at page J-18 that "quantification of the frequency of this very improbable nonmechanistic event at this time would involve such large uncertainties that the results would have no real meaning" with the Staff's estimate at page J-6 of a conditional probability of .1 for primary system failure Category IV, assuming that a CDA accident occurs?

Response

a) The CDA energetic level required to produce a missile or a spray fire, assuming the current design of the primary coolant system, has not been determined.

b)-c) The basis of the Staff evaluation of CDA energetics and primary system capability, including documents relied upon, will be discussed in the SER.

d) If, during the safety review, it is discovered that physically reasonable rearrangements of fuel, coolant, or cladding could lead to CDA energetics in excess of the current design capability of the primary coolant system, the strength of the coolant system will be enhanced, or head restraints and sodium spray deflectors will be added as needed to prevent containment failure from missiles or spray fires. The measures taken, if any are needed, to prevent missiles or spray fires involve passive rather than active features and are therfore very reliable. We believe the probability of their failure would be only a small fraction of the probability of containment isolation or mitigating system failure.

e) The basis for the conditional frequency of 0.1 for primary system failure Category IV is given in response to Interrogatory 43. The statement regarding the quantification of the frequency of the extreme energetic CDA on page J-18 of the DES was included in recognition of the remote possibility that the highly coherent behavior among other factors (see response to No. 43) needed for high energetics could occur. The Staff is not currently aware of any way this could happen through a natural course of events, but it is possible to speculate or hypothesize such behavior. Because of the speculative and hypothetical conditions needed for high energetics it is not meaningful to make a judgement of the conditional frequency of such conditions. However, we are confident that the conditional frequency is much smaller than 0.1.

Interrogatory 68

At the bottom of page J-18, the Staff concludes "in summary, from the limited quantitative analysis discussed above, it is the best estimate of the Staff that the frequency of individual classes of severe accidents resulting in fatalities or even doses exceeding 10 CFR 100 guidelines is less than 10^{-6} per reactor year."

a) What is the Staff's quantitative estimate of the uncertainty in this best estimate result? In other words, what is the standard deviation (1,2 or 3 sigma) of this best estimate?

b) What is the basis for this Staff estimate of uncertainty?

Response

The use of the term "limited quantitative analysis" was not intended to imply that an uncertainty analysis had been performed. Appendix J involves estimates of frequencies of a range of Class 9 accidents based on judgement rather than a detailed probabilistic risk assessment. Therefore, a standard deviation (which is a statistical parameter) for the analysis would not have any significance.

Interrogatory 69

a) At the top of page J-19, what is meant by the phrase "to gain perspective regarding representative system unreliabilities"? In other words, what kind of conclusions does the Staff imply should be drawn from these estimates?

b) How reliable are these conclusions in the Staff's view?

c) What is the basis for the Staff's estimates of the reliability of the Staff's conclusions?

Response

The phrase "to gain perspective" refers to use as a general comparison; the implication is that the estimates are not strongly dependent on the material in the Reactor Safety Study. The conclusions are at the bottom of page J-19; they are the judgement of the Staff, and the Staff judges them to be highly reliable.

Interrogatory 70

At the bottom of page J-19, the Staff concludes "the analysis confirms the FES conclusion that accident risks at CRBR can be made acceptably low."

a) What is meant be the phrase "acceptably low" in this regard?

b) Can the Staff quantify what is meant by acceptably low? If so, provide the quantified result.

c) What is meant by the phrase "the analysis confirms"?

d) Should the reader attach a different meaning to the phrase "the analysis confirms" in the sentence and the phrase "to gain perspective regarding" in the sentence at the top of the page?

e) Does "confirm" mean "prove," or does "confirm" mean "gain perspective?" Please explain.

f) Is the perspective gained from these analyses meant to be any different from the perspectives on light water reactor risk that was gained by the Lewis Committee?

Response

a) As stated by the Staff on page J-19 of the Draft FES Supplement, the conclusion that "the accident risks at CRBRP can be made acceptably low" follows from the previous statement in the same paragraph that, "the assessment of environmental risk of accidents, assuming reasonable protective action, provides perspective on the overall risk from CRBRP accidents in comparison with those from LWRs." The comparison of risks in Table J.5 shows that CRBRP risks are not significantly different from Midland Risks. Furthermore, it is the Staff judgement that the uncertainty bounds of both CRBRP and LWRs are comparable.

The Staff's conclusion on page J-19, therefore, arises from the comparability of the CRBRP risks with the LWR risks with some margin, and within comparable uncertainty bounds.

b) See response to 70 a) above.

c),d),e),f) As stated in the Appendix J of the Draft FES Supplement on page J-8, the Staff indicated that further calculations were performed to provide "additional perspective" on risk associated with the Class 9 accidents. By "additional perspective", the Staff meant that the results presented in Appendix J are additional to those presented in the 1977 FES. The results of the additional analysis were used to gain a perspective of risk to support the conclusion reached in the FES regarding accident risks.

Interrogatory 71

a) Identify each Staff person and Staff consultant responsible for, or providing technical assistance in, the preparation of Appendix J.

b) Identify the affiliation of each Staff consultant and their place of work.

c) Identify by subsection each part of Appendix J that each Staff person and consultant worked on and describe the nature of his or her work.

d) Approximately how many hours were spent by each Staff member and consultant in preparing Appendix J?

Response

a) Those principally involved in the preparation of Appendix J were Bill Morris, Paul Leech, and Jerry Swift of NRC's CRBRPO; Mohan Thadani and Gerry Hulman of NRC's Accident Evaluation Branch; and Ed Rumble of Science Applications, Inc. Other NRC personnel who contributed were John Long, Irwin Spickler, Argil Toalston, Charles Ferrell and Richard Codell. Other technical experts on the NRC Staff were occasionally consulted.

b) Dr. Edmund T. Rumble is employed in the Palo Alto, California, office of Science Applications, Inc.

c) Appendix J was prepared using input from several sources. Bill Morris, Ed Rumble, and Jerry Swift provided discussions of core disruptive accident initiators, primary system failure modes, containment failure modes, frequency estimates for the various failure modes, and release fractions to the environment. Mohan Thadani performed the consequence analysis using the CRAC code and provided discussions of relative risks. Jerry Swift provided input on the reactor radionuclide inventory. Richard Codell contributed the section on liquid pathways, and Argil Toalston contributed the material on economic impacts. Irwin Spickler provided the meteorological information, and Charles Ferrell provided population information. Bill Morris, Gerry Hulman, Jerry Swift, Mohan Thadani, Ed Rumble and Paul Leech reviewed the final version for accuracy and completeness. Paul Leech coordinated the effort.

d) In the preparation of Appendix J and revisions to Section 7.1, Bill Morris spent about 2 weeks, Paul Leech about a week, Jerry Swift and Mohan Thadani several weeks each, Gerry Hulman, Irwin Spickler, Charles Ferrell, Richard Codell and Argil Toalston a few days each. Ed Rumble spent about 40 hours on Appendix J. Other spent a few hours on it.

The above times indicate the period spent actually writing Appendix J. The time spent obtaining the information which formed the background and perspectives of the contributors to Appendix J, upon which the Staff relied, was a total of approximately 100 hours. However the total background in LMFBRs for this group of individuals is about 50 man-years.

Next page is 104

III. Appendix L and Chapter 9

Interrogatory 72

Identify in detail the floodplain area (lowland and relatively flat areas) adjoining the Clinch River in and around the proposed site that is subject to a 0.2 percent chance of flooding in any given year.

a) Provide a detailed map delineating

i. the boundaries of such floodplain;

ii. all CRBR facilities proposed for construction in such floodplain.

Response

The Staff has not identified the areas at the Clinch River site which are subject to a 0.2 percent chance of flooding (500-year flood) in any given year. The Staff does not need such information since all of the plant structures except the river water intake and discharge structures would be located above the probable maximum flood level, which would be higher than the 500-year flood level.

Interrogatory 73

Provide a detailed map showing the current proposed location of the CRBR barge unloading facility to the nearest tenth of a river mile.

Response

Enclosed is a copy of a sketch provided to the Corps of Engineers by DOE in 1981 which shows the proposed barge unloading facility at Clinch River mile 14.8.

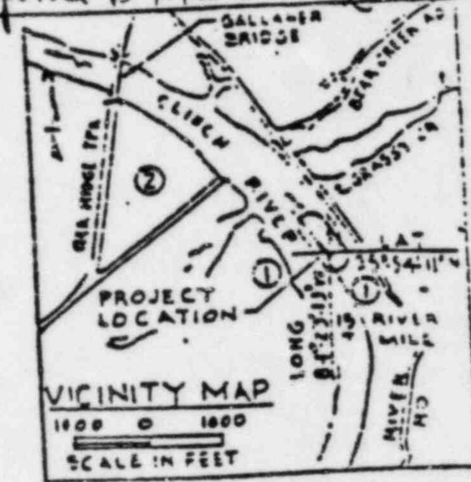
Interrogatory 74

Describe in detail the current status of plans for the construction of a coal gasification plant at the Murphy Hill alternative site.

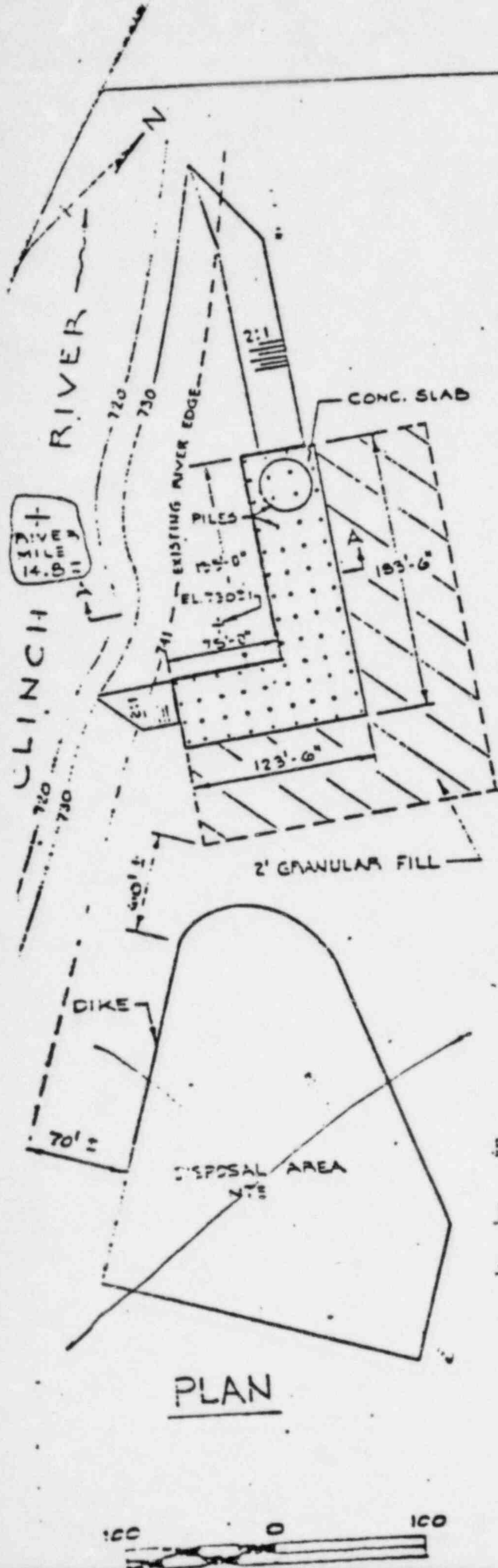
Response

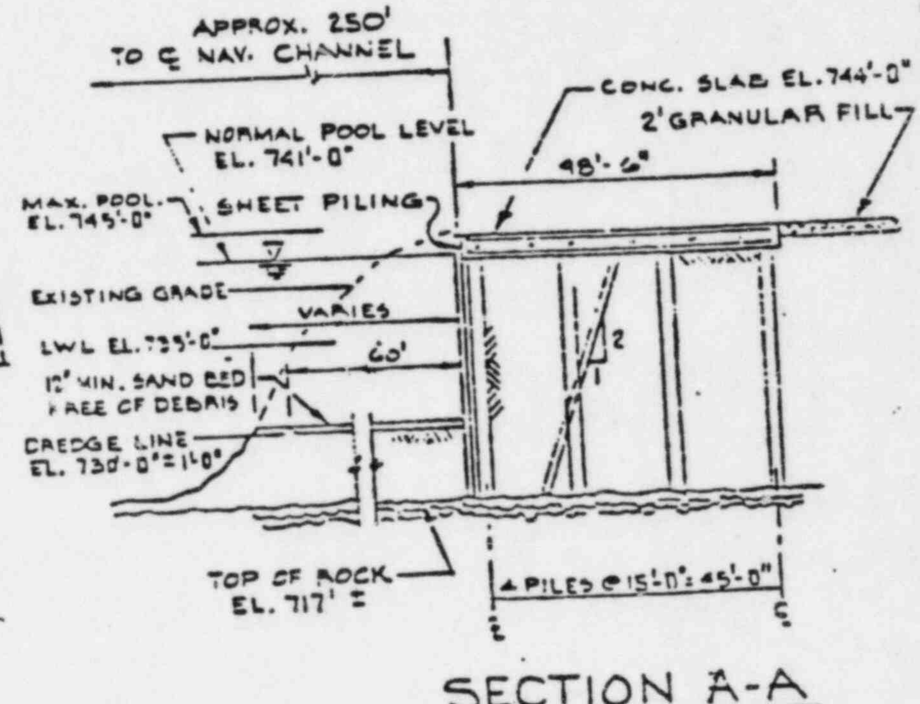
It is the Staff's understanding that the U. S. Synthetic Fuels Corporation is currently considering a request from TVA and/or private interests for financial assistance to construct the proposed coal gasification plant at TVA's Murphy Hill site. Clearing of the site was nearing completion when NRC Staff members inspected it in Feb. 1982. The Staff has no further details relative to the current status of this proposed plant.

For response to NRDC #73 (37th Set Part II)



REFERENCE:
TVA NAVIGATION CHART
TENNESSEE RIVER
WATTS BAR RESERVOIR
EDITION OF 1963





TOTAL EXCAVATION APPROX. 11,000 CY.
(APPROX. 5,000 CY. BELOW L.W.L. EL 735)
EXCAVATE SAND/SILT BY CLAMSHELL
AND/OR DRAG LINE AND DEPOSIT IN
PROPOSED ADJACENT DISPOSAL AREA
TEMPORARY SAND FILL 700 CY.

PURPOSE: CONSTRUCT BARGE UNLOADING FACILITY

DATUM: MEAN SEA LEVEL

ADJACENT PROPERTY OWNERS:
① FEDERAL GOV'T. (TVA)
② TENNESSEE STATE ELLINGTON AGRICULTURAL CENTER

PROPOSED BARGE UNLOADING
FACILITY
IN CLINCH RIVER
NEAR RIVER MILE 14.5
AT OAK RIDGE
COUNTY OF ROANE
STATE OF TENNESSEE

APPLICATION BY:
D.O.E. - CRBRP PROJECT OFFIC
P.O. BOX U

Interrogatory 75

Describe in detail for each of the following alternative sites:

a) Hartsville (the sites of the two cancelled LWR units);

b) Phipps Bend (the sites of the cancelled LWR units);

c) Yellow Creek (the site of deferred LWR construction); and

d) Murphy Hill.

i. The extent to which the CRBR could utilize any water intake facilities that have already been constructed;

ii. the extent to which the CRBR could utilize any water discharge facilities that have already been constructed;

iii. the extent to which the CRBR could utilize any other existing or partially constructed facility at the site. Describe such facility in detail;

iv. the cost of constructing each facility discussed in response to (i)-(iii) above at the alternative sites;

v. the cost of constructing each such facility from scratch at the proposed Clinch River site;

vi. if any of the facilities in (i)-(iii) could be utilized by the CRBRP with some modification, the cost of such modification;

viii. if the above answer is yes to (vii), provide a map of the alternative site and indicate in detail the portions of the already cleared site that could be utilized for CRBR construction;

ix. the water quality impacts that would be attributable to the breeder plant if no other LWR reactor or coal gasification plant were located there;

x. the extent to which any terrestrial resources would be disturbed at each alternative site if the CRBR were constructed on already cleared portions of the site;

xi. the boundaries of the floodplain (lowland and relatively flat) area in and around the site that is subject to a 1 percent or greater chance of flooding in any given year;

xii. the breeder reactor facilities that would most probably be located in the 1 percent elevation floodplain;

xiii. the boundaries of the floodplain in and around the site that is subject to a 0.2 percent chance of flooding in any given year;

xiv. the breeder reactor facilities that would most probably be located in the 0.2 percent elevation floodplain;

xv. the boundaries of any wetlands located on the site;

xvi. any mitigating measures that would probably be necessary to avoid adverse impacts to the 1 percent elevation floodplain;

xvii. any mitigating measures that would probably be necessary to avoid adverse impacts to wetlands;

xviii. the cost of each mitigating measure described in response to xvi and xvii above.

Response

i.-ii. The extent to which existing water intake and discharge facilities at the alternative sites could probably be utilized is addressed generally in the Aquatic Ecology sections of Appendix L in the Draft Supplement (DSFES) to the FES (NUREG-0139). No analysis has been made of the structures, pumps, pipes, etc., to determine what changes would be required to accomodate the breeder demonstration plant.

iii. The Staff does not believe that the existing LWR plant structures at the alternative sites are likely to be usable because of CRBRP design differences. Supporting facilities such as roads, offices, shops and storage facilities probably would be usable.

iv.-viii. The staff does not have this information.

ix. No significant water quality impacts would be attributable to the breeder plant under the circumstances stated.

x. The Staff does not know precisely to what extent the CRBR could be constructed on portion(s) of the alternative sites that have already been cleared. However, it can be generally stated that the less clearing required for construction of the CRBR facility, the less disturbance of terrestrial resources would occur.

xi.-xiv.,xvi. Reconnaisance-level information on the four alternative sites did not include information on floodplains. Without such information and drawings showing how the demonstration would be laid out on each site, the Staff cannot determine which facilities would be in the floodplain and what mitigation measures would be required. In general, the water intake and discharge facilities must be located in the floodplains, while the main plant structures are rot so located since NRC criteria for safety-related features of nuclear power plants require that such features be adequately protected from effects of the design basis flood.

xv. The boundaries of any wetlands located on the site.

a) Hartsville - As indicated in the Hartsville FES-CP (NUREG-75/039, June 1975, Figure 2.9), the site contained several small ponds and intermittent streams capable of supporting wetland-type vegetation. Upon commencement of construction activities at this site, drainage ponds and sediment basins were built (FES-CP, Section 4.5.1) to protect the terrestrial and aquatic environment. These resultant man-made basins in the vicinity of the Cumberland River have provided some additional wetland-type habitat.

b) Phipps Bend - As indicated in the Draft Supplement (Appendix L, p. L-22), holding ponds and sediment basins constructed to provide run-off and erosion control at the Phipps Bend site are also providing productive wetland areas. These erosion control ponds and basins are located primarily adjacent to the Holston River.

c) Yellow Creek - As indicated in the Draft Supplement (Appendix L, p. L-28) and the Yellow Creek FES (NUREG-0365, 1977, Section 4.3.1.1), onsite construction activities have already affected two small wetland areas associated with the intake facilities and the barge-unloading facilities (Yellow Creek FES, Figures 2.1 and 3.4).

d) Murphy Hill - As indicated in the Draft Supplement (Appendix L, p. L-15), the Staff indicated no onsite wetland areas. Since the publication of the Draft Supplement, the Staff has learned that the Murphy Hill site contains two small wetland areas which would be protected from Construction.[1]

In addition, all of the shoreline shallow embayments, and inlets that surround or border the above sites, as well as the Clinch River site, could be classified as shoreline wetlands. All of these areas have been or could be protected from significant construction and operational activities.

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[1] Coal Gasification Project. TVA Final Environmental Impact Statement. July 1981 (TVA/ONR/PCS-81/3).

xvii. The Staff believes that avoidance of shoreline wetlands would be the normally required mitigation actions on the above sites. Location of all major plant facilities away from the shoreline would significantly protect these wetlands.

xviii. The Staff has not calculated specific costs of such mitigation measures. It is the Staff's judgement that such costs would be comparable for all of the sites named by the interrogatory.

Interrogatory 76

In evaluating the preferability of alternative sites regarding overall socioeconomic impact, explain in detail the relative weight, if any, the Staff gives to each of the following factors:

a) estimated size of the available labor pool;

b) potential impacts on historical areas;

c) potential impacts on archaeological areas;

d) potential impacts on scenic areas;

e) potential impact on recreation areas;

f) potential impact on other protected areas;

g) potential impact on cultural areas;

h) potential displacement of residential activities;

i) potential displacement of economic activities;

j) potential traffic disruption;

k) potential visual intrusion.

Response

The Staff used the following five categories to evaluate the alternative sites:

a) Displacement or disruption of onsite archeological, historic, scenic, recreational, and cultural resources (corresponds to items b, c, d, e, f and g of list);

b) Displacement of residential and economic activities (items h and i);

c) Anticipated points of vehicular congestion casued by construction worker or truck traffic to and from site (item j);

d) Visual intrusion of station structures in offsite areas (item k); and

e) Size and availability of labor pool (item a).

Numerical weights were not applied to these categories in the abstract. Rather, the Staff used a two-step process to evaluate the site alternatives. The first step involved a comparison of each site individually with the CRBR site with respect to the magnitude of each category. A three-point scale was used to indicate judgementally whether an alternative site was preferred to, comparable with, or less desirable than Clinch River on the basis of each individual category. The second step involved the determination of each alternative site's preferability with respect to the Clinch River site. This step required that the importance of each category at a site be evaluated in terms of its potential impact -- judegmentally determined -- to the community. In this context, labor size and availability were determined to have most weight because of its potential to adversely affect baseline congestion. The remaining factors were judgementally determined to have the same weight relative to each other. In developing this response the Staff did not rely on any document or study.

Interrogatory 77

Provide the Staff's best estimates of:

a) the types of modifications to the CRBRP design that would be necessary if applicants were required to restrict their thermal discharges to the Clinch River during periods when the river water temperature is high and zero flow conditions exist (see DES p. L-21);

b) the cost of such modifications.

Response

The Applicants have not identified any CRBRP design modifications that would be necessary under the stated conditions and, at this time, the Staff does not foresee the need for such modifications.

Interrogatory 78

On page 9-10, the Staff states that "an attempt was made to apply CONCEPT to the CRBRP...." Identify and provide:

a) the CONCEPT computer code used by the Staff;

b) the CONCEPT program manual;

c) all input data to the CONCEPT code used by the Staff;

d) all CONCEPT output data for the CRBRP;

e) the results of any sensitivity studies performed on these results.

Response

The following documents will be made available with those assembled in response to NRDC's Third Request to Staff for Production of Documents:

I. CONCEPT Documentation

1. ORNL-5470. CONCEPT-5 User's Manual, C. R. Hudson III, January 1979.

2. Internal memo to H. I. Bowers for C. R. Hudson, Summary of work performed on the CONCEPT computer code, September 2, 1981. (Summarizes modifications made to the CONCEPT code during the summer of 1981.)

II. CONCEPT Data Base

1. "Phase IV Final Report and Fourth Update of the Energy Economic Data Base (EEDB) Program," prepared for the U. S. Department of Energy (Argonne National Laboratory) under Contract Number 31-109-38-6411 by United Engineers and Construction, Inc., September 1981. (Updates reference PWR plant cost models to January 1982 regulatory status.)

2. The cost-index data base in CONCEPT is updated semi-annually. The most recent update for the enclosed runs utilized July 1981 data on construction labor rates and site-related materials from Ref. 16 of ORNL-5470, cost indexes for equipment from Ref. 12, 13 and 14, and cost indexes for engineering services from Ref. 15.

III. CONCEPT Results for CRBR

1. Computer runs for 400mw-800mw-1200mw PWR (only sensitivity analysis performed).

2. Letter report from H. I. Bowers to Argil Toalston, June 15, 1982.

Interrogatory 79

At page 9-14, the Staff presents Table A9.5, which estimates the costs of locating the breeder reactor at a site other than Clinch River. This table contains three entry lines of estimated costs for relocating at TVA alternative sites, including a 4-year delay (high range), a 4-year delay (low range), and a 3-year delay.

a. For each of these three cases, provide a breakdown of estimated delay costs for each of the four TVA alternative sites considered by the Staff.

b. For the 3-year delay case, indicate whether such estimate is considered in the high range or low range.

c. Provide both the high range and the low range estimates for the 3-year delay case.

d Identify each and every assumption used by the Staff to differentiate its high range and low range estimates for the 4-year delay case.

e. For each of the TVA alternative sites, provide the estimated costs of relocation of the breeder reactor at that site, assuming that:

i. no other LWR or coal gasification plant is to be built at that site; and

ii. the CRBR utilizes the existing facilities and cleared site to the fullest extent practicable.

Response

a. The information requested cannot be provided as the Staff does not possess the requested data.

b. For the 3 year delay case, estimate is for the low range.

c. Low range estimates are given in the last row in Table A9.5, page 9-14. High range is as follows:

Year of Expenditure		1982 Present Worth	
$Million	% of Base	$Million	% of Base
4667.6	132.4	3738.8	109.2

d. The only assumption that differs between the high range and low range estimates for the 4-year delay case is the Applicants' estimate of the incremental cost associated with a 43 month delay for items 3 thru 16 of Table 9.4 in the FES Supplement. The low range estimate assumes $44 million whereas the high range estimate assumes $445 million for these items. All adjustments made to these estimates by the NRC have been consistent as between the low and high range.

e. The Staff's estimated cost of relocation to a TVA alternative site is as reported in Table A9.5, page 9-14 of the FES Supplement. The Staff is unable to discern cost differentials resulting from the assumptions employed in part e, items i. and ii. of this interrogatory.

Interrogatory 80

In deriving Table A9.5, did the Staff use the Applicants' estimates in Table A9.4 of 45-month delay costs for specific items such as excavation and additional studies?

a. If the Staff revised any of the Applicants' estimates (other than the reduced revenue from sale of electricity), provide the Staff's revised estimates for that factor, and give the reason for such revision.

Response

a. With the exception of reduced revenue from sale of electricity the Staff used Applicants' estimates as presented in Table A9.4. These values were adjusted to reflect 36 and 48-month delays. Items 1 and 2, escalation and Staff support stretchout were impacted by these adjustments. Also, the values were assumed to be distributed over the future time-period as discussed in the FES Supplement p. 9-14.

Interrogatory 81

Explain in detail the bases for the Staff's conclusion that revenue from the sale of LMFBR electricity at Hanford would equal $1097 million.

Response

First, the Staff adjusted the DOE estimate of revenues for a TVA site. The Applicants used a replacement energy cost of $27.53/MWh in 1981 dollars. Assuming 8%/year escalation, revenues over the 1990-1995 time frame were estimated at $679.2 million in year of expenditure (YOE) dollars. The Staff assumed the CRBR will displace average cost coal fired generation in the East South Central region. In 1981, the average coal fuel cost was 170.4¢/10^6 BTU. Assuming an average plant heat rate of 11,000 BTU/kWh, Staff estimated a cost for displaced energy of $18.74/MWh in 1981 dollars. Based on this difference, the Staff re-calculated the TVA revenues at $350.5 million in YOE dollars for the 1990-1995 time period.

At the Hanford site, the Staff assumed the breeder's electrical output would displace equal portions of coal and oil fired energy. The average fuel cost of coal and oil for 1981 in the Pacific region was 121.0¢/10^6 BTU and 662.6 ¢/10^6 BTU, respectively. Assuming a 50-50 mix, the average 1981 fuel cost is 391.8¢/10^6 BTU. This average fuel cost is about 2.3 times what the staff calculated for a TVA site. Thus, Staff applied a factor of 2.3 to its TVA revenue estimate of $350.5 million to arrive at its estimate of $806.2 million (YOE$) for the 1990-1995 revenues. For the delay scenario, Staff escalated the $806.2 million by 8%/year escalation. Thus, for a 4 year delay, Staff's estimate is $1096.8 million ($806.2 million x 1.08^4).

Interrogatory 82

Regarding the occurrence of periods of zero flow into Clinch River, describe in detail:

a) the estimated number of days per year of zero flow;

b) the estimated duration of each period of zero flow;

c) the maximum possible number of days of zero flow;

d) the maximum possible duration of any period of zero flow;

e) the stretch of the Clinch River that would be affected by any period of zero flow described in response to (a) - (d) above.

Response

The flow into the Clinch River will never be zero, even if there were no discharge from Melton Hill Dam. Tributaries to the Clinch River upstream and downstream of the site will provide water to the Clinch River. In addition, there will be some direct infiltration of groundwater from the river banks directly to the Clinch River.

Assuming that the interrogatory is intended to relate to periods of zero release from Melton Hill Dam, the Staff responds as follows:

a) Section 2.5.1.3 of the ER states : "Since the closure of the dam in 1963, there has been an average of about 31 days per year on which no water was released from Melton Hill Dam."

b) ER Table 2.5-2 shows the periods of zero release from Melton Hill Dam during May 1963 through December 1979.

c), d) The Staff does not know this information. In ER Section 2.5.1.3 the Applicants state that "extended periods of zero flow from Melton Hill Dam are not anticipated in the future."

e) The Clinch River would be affected from Melton Hill Dam downstream by zero releases from the dam.

Interrogatory 83

Explain in detail the licensing conditions that would have to be imposed in order to:

a) avoid impingement and entrainment losses to aquatic biota at the Hartsville site;

b) avoid disruption of archeological sites at the Yellow Creek site.

Response

a) "Licensing conditions" to protect aquatic biota would appropriately be specified in the NPDES Permit, not in NRC's Construction Permit. In the FES for the Hartsville CP review the Staff indicated that perforated pipes might be considered as an alternative for reducing intake losses.

b) Until a specific location for the LMFBR is chosen on the Yellow Creek site, the Staff cannot determine specifically what license conditions would be imposed. However, the Yellow Creek FES (NUREG-0365) at page 4-20 contains the following statements concerning Staff requirements for avoiding potential impacts to archeological resources:

> The applicant has developed a mitigation plan for the archeological sites that are to be disturbed or destroyed during the construction of the plant. Because of the importance of the kinds and locations of sites that have been identified, the Staff believes that an archeologist must be present when initial earth-moving activities take place. Thus, buried sites or sites not located during the survey phase can be recognized and proper action can be taken to record and collect salvage data.

Interrogatory 84

For each of the following sites:

a) Hanford,

b) INEL,

c) Savannah River:

i. Explain in detail what X/Q values would be utilized for evaluating the impacts of routine and accidental releases.

ii. Identify with particularity the additional licensing costs, if any, that would be required to design the LMFBR against tornadoes at the Clinch River site as compared to these sites.

iii. Identify with particularity the additional licensing costs, if any, that would be required at Clinch River because of less favorable diffusion conditions as compared to these sites.

iv. Identify with particularity the additional costs, if any, that would be required to protect against construction-related impacts to aquatic biota at the Clinch River site as compared to these sites.

v. Identify with particularity the additional costs, if any, that would be required to protect against operation-related impacts at the Clinch River site as compared to these sites.

vi. Identify with particularity the additional costs that would likely be associated with siting the plant at these sites due to "uncertainties about the tectonic regime."

Response

i. The Staff does not know precisely what X/Q values would be utilized by the Staff in a licensing proceeding to evaluate the impacts of routine and accidental release at the Hanford, INEL, and Savannah River sites because the Staff analysis is dependent on the plant configuration and the exclusion area and low population zone boundaries. Because all three sites are extremely large and the population distributions for each site is different no determination can be made of the applicable exclusion area and low population zone boundary distances.

ii. The Staff has not attempted to quantify the cost differential for CRBRP that would result from designing the facility to withstand Region 3 specified tornado parameters instead of Region 1 tornadoes. Tornado design costs cannot be examined by themselves because the structural design for a facility includes the capability to withstand accident conditions, seismic, and tornado effects. In many cases designing structures and components to withstand seismic and/or accident conditions will inherently reduce tornado design costs. In order to determine the added tornado design costs, a total structural and component design cost analysis would have to be performed for all of the appropriate design parameters for each of three sites.

iii. The Staff has not attempted to quantify the potential cost savings that could result from relocating the CRBRP facility to any of these three sites. The plants as designed appear to meet licensing criteria at any of the three sites.

iv. The Staff did not look explicitly at the cost of protecting aquatic biota at the Clinch River site. However it is expected that the erosion and sediment control cost will represent a small part of the construction cost. Each site would need an erosion and sediment control plan. Therefore, a comparison of differences in costs for such mitigation among sites without also considering the total site preparation, excavation and foundation costs would not be particularly helpful in comparing sites.

v. The Staff did not consider in detail the relative costs of protecting against aquatic biota impacts from operation in its alternative sites review. Hanford and Savannah River would require that these impacts be considered in designing intake and discharge structures. Both sites now have some degree of protection. Cost differences for protecting the aquatic environment would be small relative to total cost differences in developing a cooling water system.

INEL would use groundwater as a source and return water to the ground. Thus, protection of aquatic biota is not a consideration at INEL. Costs of aquatic impact avoidance may be less at INEL. However, it is anticipated that the cost of developing and operating the cooling water supply would be greater. To consider only the dollar cost of protection of the aquatic environment without considering total project cost in comparing alternatives is not particularly helpful.

vi. Cost relative to geologic and seismic considerations at each site are addressed below:

Hanford

The Hanford Reservation is in a region that is extremely complex from a tectonic point of view. The relatively short historic record (on the order of 100 years) indicates low to moderate seismicity, but there is evidence of fairly recent tectonic activity. Therefore, each proposed nuclear power plant site must be extensively investigated to define site specific characteristics and to determine whether or not there is a significant relationship between the site geology and the regional geology. Applicants for nuclear power plant sites on the Hanford Reservation (Washington Public Service Supply System and Puget Power) have spent or are spending up to several tens of million dollars validating their sites. In the eastern U. S., in areas where tectonic structures are known to be very ancient (tens to hundreds of million years old) and seismicity is relatively low, geological and seismological investigations are generally less costly. In general, structural design and construction costs of a plant are also expected to be lower in an area with a lower earthquake design basis.

Savannah River

During the time that the Savannah River site was being considered as an alternate site for the LMFBR, the U. S. Geological Survey released an open file report (Faye and Prowell, 1982) that postulates two high angle northeast striking faults in the subsurface, the northernmost of which crosses the sourthern third of the SRP. Evidence for an upper age limit of last movement was not given in the report but relatively geologically young (40 million years old) strata were reported to be offset.

To ensure that the LMFBR site at SRP had the proper seismic design the NRC would require that the area around the site be investigated to ascertain that similar faults were not present closer to the site. During the summer of 1982, Georgia Power Company conducted a multimillion dollar investigation of these faults (particularly the northernmost one) as part of its studies for the Vogtle Nuclear Power Plant. These studies demonstrated that the postulated faults did not offset strata younger than 40 million years old. With this new information the LMFBR applicants would not be required to conduct an extensive investigation of these faults, but they would still be required to investigate the site in detail to determine the site specific characteristics of soils underlying the SRP site, and the potential effects on these foundation soils of a recurrence of the Charleston South Carolina 1886 Earthquake (Maximum Modified Mercalli Intensity X).

INEL

The INEL is located on the Snake River Plain of southern Idaho. As indicated in the FES, little is known about the seismic potential of this area. Like the Hanford Reservations, seismicity within the plain is low, but the historical record is short. The Snake River Plain is bounded by regions of relatively high seismicity which also contain yound gaults. Two of these faults, the Howe and Arco, are located in the mountains north of INEL. Alignments of volcanic vents and domes lie along projections of the faults out onto the Sanke River Plain suggesting a genetic relationship with the faults although the basalt lavas on the Plain are not known to be offset. It is anticipated that if the LMFBR is to be constructed at INEL, considerable regional and site geological and seismological investigations would have to be carried out to determine if there were hazardous geologic structures at the site or whether or not there was a relationship between geologic features at the site and regional tectonic structures. Investigations of that type in western U. S. usually cost several million dollars.

The following references were used in formulating this response:

1. U. S. Nuclear Regulatory Commission, 1982, Safety Evaluation Report related to the operation of WPPSS Nuclear Project No. 2, Docket 50-537; NUREG-0892, March 1982.
2. Niccum, M. R., 1971, General Geologic and Seismic Information for the NRTS and Vicinity; Aerojet Nuclear Company, response to AEC Question 11A.
3. Faye, Robert E., and David C. Prowell, 1982, Effects of Late Cretaceous and Cenoxoic Faulting on the Geology and Hydrology of the Costal Plain near the Savannah River, Georgia and South Carolina; USGS Open File Report 82-156.

Interrogatory 85

Explain in detail:

a) the expected amount of radioactivity that would be released to the INEL site groundwater;

b) the estimated amount of radioactivity that would be released to the INEL site groundwater during a Class 9 accident;

c) any mitigating measures that could reasonably be implemented to avoid or reduce radioactive groundwater contamination;

d) the likely licensing costs that would be necessary to ensure water availability at the INEL site;

e) the extent of any projected future use of the INEL groundwater as a source of public drinking water, including the probable size of the future population affected;

f) the costs of designing a system for waste stream disposal in the groundwater at INEL.

Response

a) Generally, releases of radioactivity to groundwater are a relatively insignificant impact of power plant operations and are not evaluated in analyses of alternative sites. As a first approximation, the quantities presented in Table 3.1, page 3-14 of the FES, could be assumed for the INEL site. However, waste treatment system designs take site characteristics into consideration and a design for the INEL site could be quite different from that for the Clinch River site.

The amount of radioactivity released to the INEL site groundwater from normal plant operation would depend on the options chosen by the plant designers. If liquid radwaste were disposed to a lined pond, and allowed to evaporate, virtually no radioactivity would reach groundwater. If the pond depended on seepage to dispose of its waste water, a fraction of the radionuclides could enter the groundwater, depending on a number of factors such as quantity of water disposed, chemical characteristics of the waste, sorption onto soil, and halflife of the radionuclide. The Staff has no such information on the disposal practices which would be in effect at the test site.

b) The Staff has not performed an specific evaluation of the radioactivity that would be released to the INEL site groundwater from Class 9 accidents. However, in Appendix J (page J-18) of the FES Supplement, it was noted that these releases would probably be considerably smaller than those for postulated core melt accidents at light water reactors. It is also less likely that a core melt accident at the INEL site would contaminate either surface or groundwater because of the semi-arid nature of the INEL site, and the fact that the water table there is very deep (200-900 feet below grade), while the water table at the Clinch River site is relatively near the surface.

However, there is no substantial advantage in locating CRBR at any of the alternate sites, including INEL. This conclusion was based on the consideration that the risk from Class 9 accidents will be acceptably small and that there would be considerable expense associated with relocation. Therefore, the radioactivity release to groundwater at INEL is of no significance. If the release were to be smaller than at CRBR, that would only mean that an already acceptably small consequence of Class 9 accidents would be reduced even further.

c) No detailed study of mitigation for potential groundwater contamination at the INEL site has been performed.

d), e) f), Responses to those questions cannot be provided without a detailed engineering study of the site.

Interrogatory 86

Identify with particularity the amount of additional costs that might be required to validate the Savannah River safe shutdown earthquake and operating basis earthquake.

Response

See the Response to Interrogatory 84.

IV. Contention 5

Interrogatory 87

How many days of heavy fog annually would the Staff expect to occur at the Clinch River site, particularly given the increased heat rejection from the current CRBR design?

Response

As stated in Section 5.3.3 of the FES, it is expected that fogging conditions due to cooling tower operations at the Clinch River site could occur on approximately 40 days per year. Fogging would only occur for a few hours during these 40 days. Most of these fogging occurrences would coincide with natural fog because the atmospheric conditions conducive to natural fog occurrence are also conducive to the form tion of cooling tower fogging.

Interrogatory 88

Discuss in detail the extent to which the latest Clinch River meteorological data collected by applicants differs from the meteorological data collected during the period from February 17, 1977, to February 17, 1978.

Response

The Staff understands that the Applicants did not collect meteorological data onsite after February 1978 until they resumed doing so in April 1982. The Staff has not seen the meteorological data collected since April 1982, and therefore cannot discuss it.

Interrogatory 89

Explain the basis for the Staff's finding in Section 10.2.4.3 that DES estimates of environmental effects are not significantly different from those discussed in the FES.

a) Indicate all sections where decommissioning effects are discussed in the FES, and discuss fully any differences, whether or not considered significant by the Staff.

Response

Environmental impacts of decommissioning, which were discussed in the FES Supplement but not in the 1977 FES, are not significantly different than those known and expected by the Staff through reactor decommissioning experience. Differences in the two reports are discussed below.

In the FES Supplement the Staff discussed the commitment of a few acres of land at the reactor site for the SAFSTOR or ENTOMB modes. Although land commitment was not specifically listed in the 1977 FES, the SAFSTOR mode used at FERMI-I, and the ENTOMB mode used at Hallam were discussed in the 1977 FES. Thus, the Staff knew in 1977 that the SAFESTOR and ENTOMB alternatives would require a commitment of a few acres of land.

The commitment of land at the low level waste burial grounds was not discussed in the 1977 FES. However, reports on the decommissioning of FERMI-I, Hallam and Elk River discussed the quantities of decommissioned reactor components which were shipped to low level waste burial grounds. These reports were available and known to the Staff in 1977. Therefore, the estimates of space needed as stated in the FES Supplement were not unexpected. Similarly, the possibility that some components may have to go to deep geologic of high level disposal areas was not unexpected since the Staff recognized in the 1977 FES that the reactor vessel and internal neutron shields may contain long lived isotopes such as Nickel-59 and Nickel-63.

The disposal of sodium, although not listed as an environmental impact in the 1977 FES, was not an unexpected impact as its disposition at FIRMI-I was discussed. Accordingly, the Staff was aware that the decommissioning of sodium cooled reactors would require disposal of the sodium used in the reactor.

Exposure to workers was not estimated in the 1977 FES. However, the Staff was aware of the person rem exposure in dismantling the Elk River reactor as well as the exposure control methods that were used during the decommissioning of Elk River and a number of other reactors. The requirements for maintaining occupational exposure as low as reasonably achievable during decommissioning were in effect before 1977.

Interrogatory 90

Explain the Staff's response to Interrogatory 74 of Intervenors' Twenty-Fifth Set of Interrogatories that the CRBR will either be dismantled shortly after final shutdown or dismantled after 50 to 100 years in a SAFSTOR status.

a) Discuss the Staff's reliance on Cobalt 60 rather than Ni-59 or other radionuclides in explaining the safe storage period in paragraph 5, Section 10.2.4.2 of the DES. Provide a full and complete explanation, including all documents, data, analyses, etc., relied on by the Staff for its choice of Cobalt 60.

b) Explain in detail all possible economic, societal, and environmental costs "being considered in the ongoing developement of NRC rules" regarding disposal of components containing long-lived radionuclides.

c) Explain fully the reasons the Staff has deferred the evaluation of these long-lived radionuclides to "the end of the Safe Storage period" (paragraph 6, Section 10.2.4.2).

Response

a) The Applicants project the exposure rate from Cobalt 60 in the reactor vessel and the fixed radial shield to be more than 10,000 times higher than the Niobium-94 exposure rate and more than one million times higher than the exposure rate from Nickel-59 at reactor shutdown. This data is included in the supplemental response to the Eighteenth Set of Interrogatories from the applicants (August 6, 1982). See Figures 1, 2 and 3 enclosed. The dominance of Cobalt 60 with respect to exposure rate at shutdown is also shown in NUREG/CR 0130. Since Cobalt 60 has a 5.2 year half life it remains dominant for 75 to 100 years, until levels of radioactivity are dominated by Niobium-94 which has a 20,000 year half life. Therefore, the safe storage period is determined by Cobalt 60 with respect to potential occupational exposure.

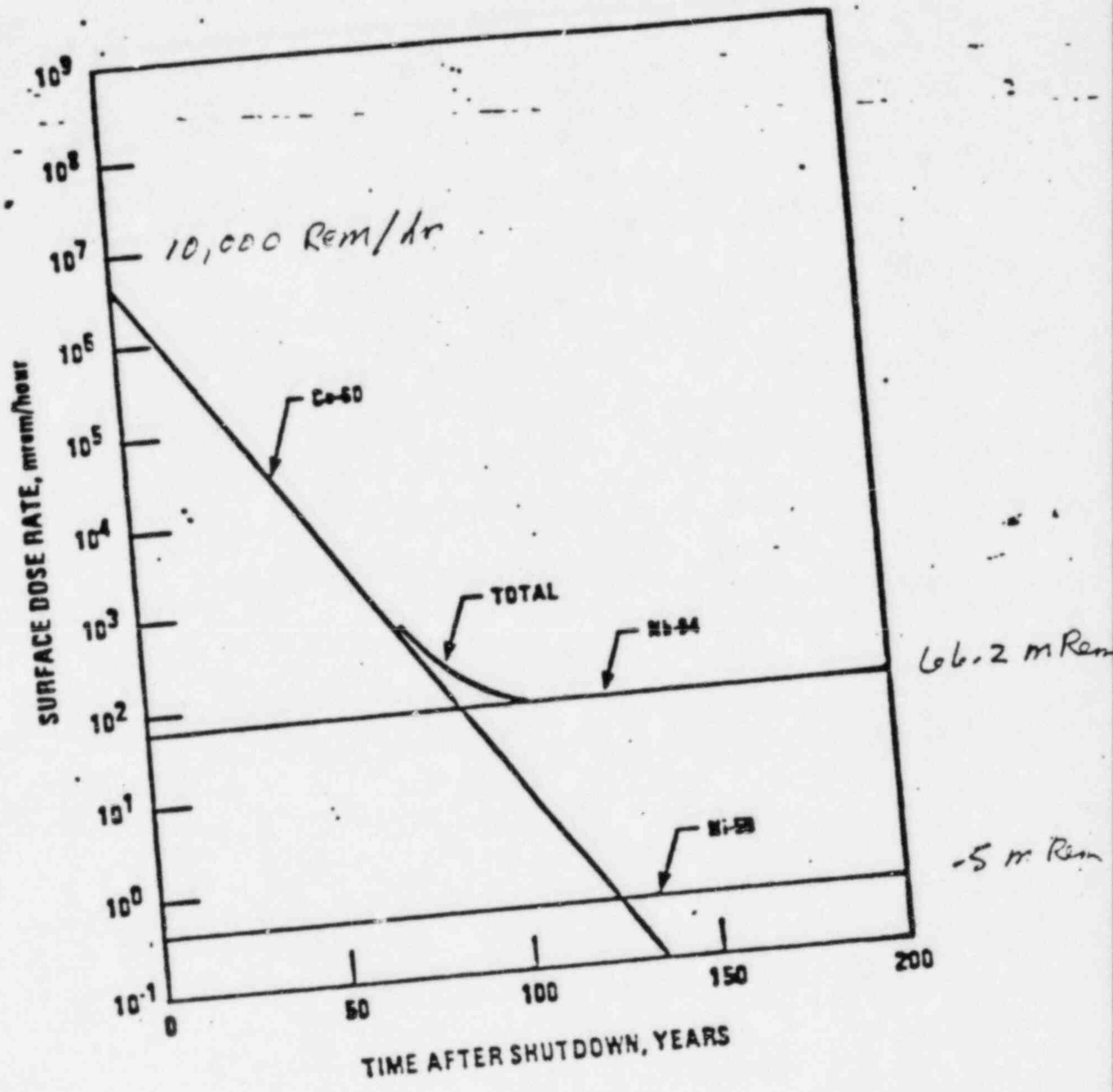

Figure 3. Time Dependence of Dose Rate at Inner Surface of CRBRP Reactor Vessel at — Core Midplane

7434-3

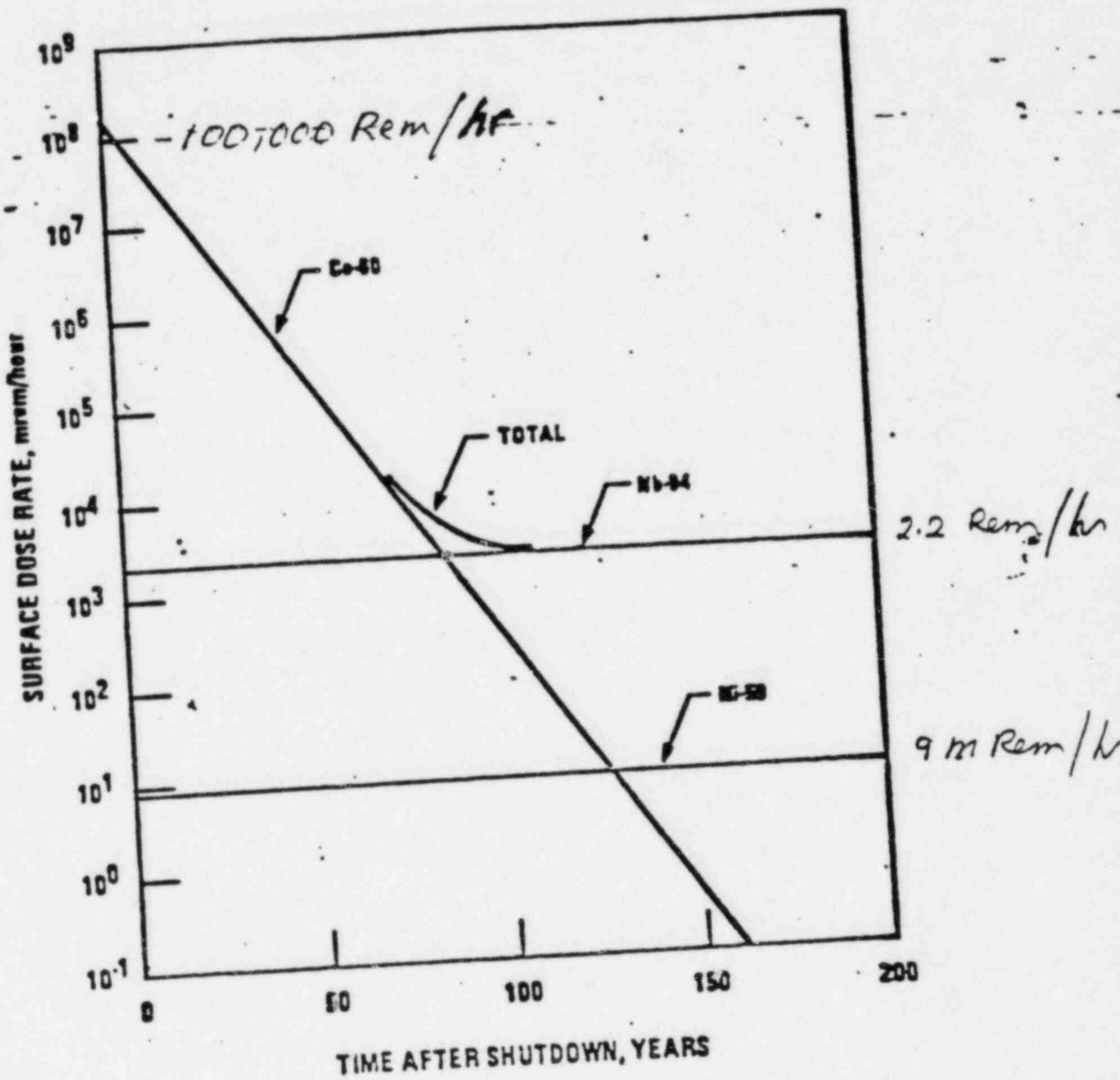

Figure 2. Time Dependence of Dose Rate at Inner Surface of CRBRP Fixed Radial Shield at Core Midplane

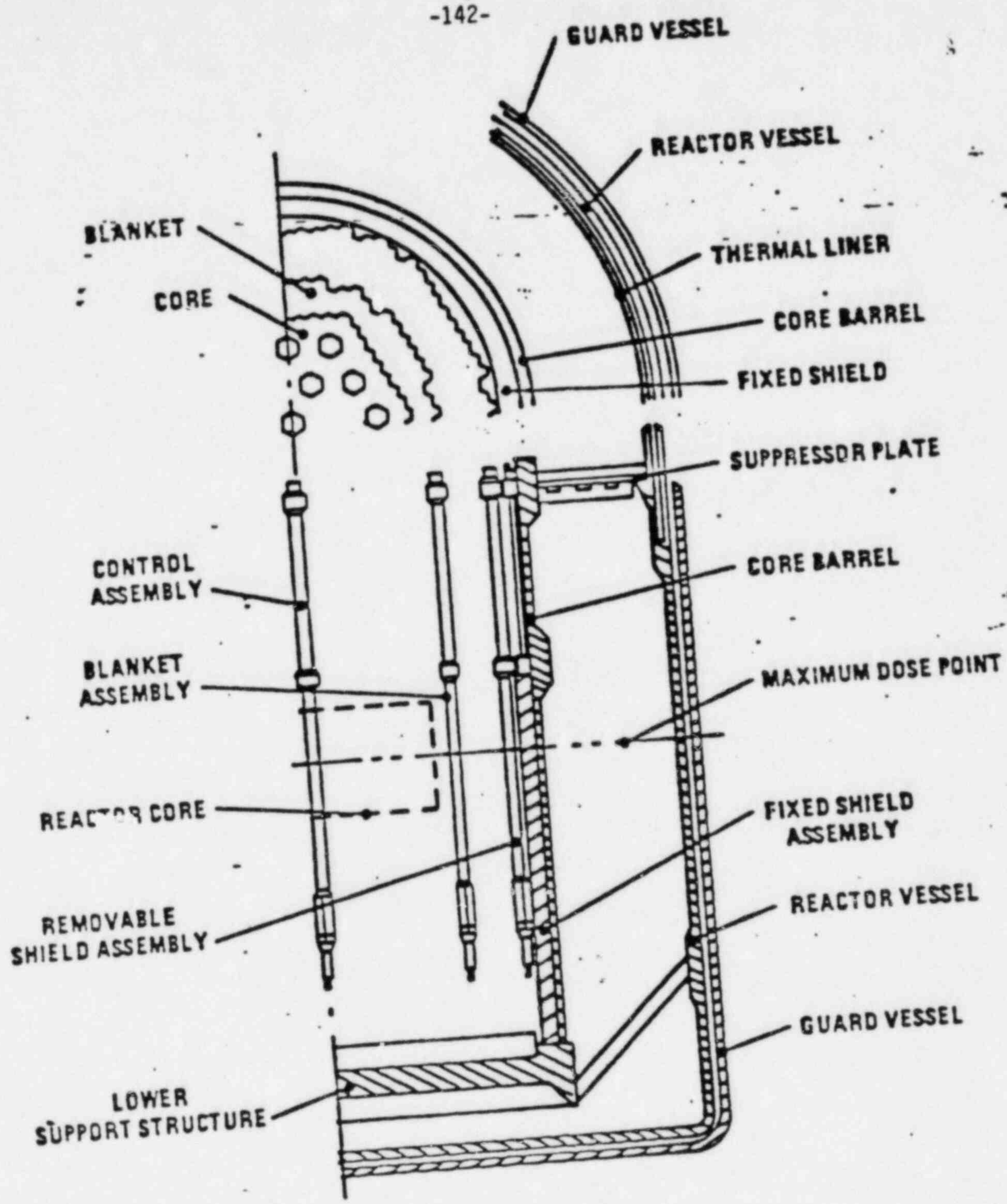

Figure 1. CRBRP Reactor System Schematic – Permanent Components (Hashed Regions) – Considered in Analysis of Ni-59, Nb-94, and Co-60 Dose Rates

7434-1

b) NRC rules now under developement, "Licensing Requirements for Land Disposal of Radioactive Waste" (10 CFR Part 61), specify limits on the disposal of lived isotopes such as NI-59 and Ni-94 in a near surface disposal facility.

Copies of the proposed rule, 10 CFR Part 61, the Draft Environmental Impact Statemen on 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," (NUREG-0782), and the commission paper on SECY-82-204, 10 CFR Part 61, dated May 19, 1982, are available at 1717 H Street, Washington, D.C., in the NRC Public Document Room.

If long lived isotopes in parts of the reactor vessel or radoshield exceed levels permitted at near surface disposal facilities these parts may have to be sent to a deep geologic disposal site or to interrim storage at a DOE facility untill the deep disposal site is established. Since a deep geologic disposal site has not been selected yet, an environmental statement on an approved deep geologic disposal site is not available.

c) Maximum exposure rates from long lived isotopes such as Niobium-94 and Nickel-59 have been estimated by the Applicants to be 2.2 rem/hr and 9 mrem/hr respectively (supplemental responses to Eighteenth Set of Interrogatories, August 6, 1982). These values are consistant with values calculated for PWR reactors in NUREG/CR 0130. It may be possible to directly measure, or indirectly determine by radiochemistry analysis, the amount of each of these radionuclides at the end of a long term safe storage if the gamma emmissions from Cobalt 60 are sufficiently reduced. However, the gamma exposure from Cobalt 60 in the reactor vessel immediately after reactor shutdown is more than 10,000 times higher than the exposure from the above long-lived isotopes. This prevents any direct measurement or accurate radio chemistry determination of the concentrations of these isotopes immediately after reactor shutdown, and the beginning of the SAFESTOR period. The amount of long-lived isotopes could be re-evaluated at the end of the safe storage period either on the basis of direct measurements, or radiochemical analysis of core samples.

Interrogatory 91

Discuss the "commitment of resources to ensure continued security at the licensed low-level waste burial grounds" mentioned in the DES, paragraph 4, Section 10.2.4.3.

a) Explain fully and identify all documents, analyses, memoranda, calculations, evaluations, assessments, etc., relied on by the Staff in this discussion or in paragraph 4.

Response

The cost of security or licensed low-level waste burial grounds is included in the fee that a burial ground charges for taking delivery of radioactive waste. The fee for burial is included in the estimated costs given in Section 10.2.4.5 of the Draft Supplement and NUREG/CR 0130. A discussion of security resources required at a licensed low-level waste burial ground is included in references listed in our response to Interrogatory 90 a.

Interrogatory 92

Specify the ALARA exposure parameters (paragraph 7, Section 10.2.4.3) for decommissioning workers. Explain and provide specific data or reports relied upon in:

a) arriving at the ALARA level;

b) applying ALARA to each of the three decommissioning alternatives.

Response

The Staff does not utilize a specific level of exposure designated as ALARA. The Staff will review the procedures, equipment and techniques proposed by a licensee in the decommissioning plan and concur or require alternate methods to assure exposure are maintained as low as reasonably achievable (ALARA). Improvements in technology, remote handling, and instrumentation may result in changes in what we consider to be ALARA.

Interrogatory 93

Given that the Fermi I reactor core melt experience on or about October 5, 1966, is now providing information directly relevant to CRBRP decommissioning, explain in detail why this event was not discussed in the DES, including how this event could affect the decommissioning of sodium-cooled breeder reactors.

Response

The accident at Fermi-I was not the cause of decommissioning. The accident which involved the melting of two fuel assemblies occurred on October 5, 1966. Damaged fuel was replaced, some modifications to the facility were made, the primary sodium decontaminated and the reactor returned to operation. The reactor resumed low power operation on July 18, 1970, and returned to full power (200 mwt) on October 16, 1970. Fermi-I was shutdown on December 1, 1971, and subsequently decommissioned due to lack of funding for continued operation.

Accident recovery is not considered to be part of the cost of decommissioning. Reactor accidents are covered in Section 7 of the CRBR FES and the FES Draft Supplement, Appendix J.

Interrogatory 94

Describe in detail the additional costs, change in environmental effects, and uncertainties involved in an early decommissioning at CRBR as opposed to decommissioning after about 30 years of operation.

Response

If the CRBR was decommissioned early in its life the costs may be less, not more. Early decommissioning of CRBR will result in a reduction of the radioisotope inventory in the reactor vessel, radial shield, and biological shield. If the inventory of radioisotopes are less, less shielding for workers and waste shipment would be required.

Interrogatory 95

Given that no deep geologic disposal site has yet been determined, discuss fully what alternative plans exist for early waste disposal or management vis-a-vis decommissioning.

a) What additional costs, environmental effects, or uncertainties are involved in alternate plans for waste disposal management?

Response

Proposed rules "for land disposal of radioactivity waste" discussed in the Staff's response to Interrogatory 90 (b) ma, allow the disposal of most if not all of the reactor components in those licensed burial grounds, commonly referred to as low-level waste burial grounds. If parts of the CRBR reactor vessel or radial shield with higher concentration of long lived isotopes are not permitted at low level waste burial grounds they would then go to interim storage or a DOE site until a deep geologic disposal site is available.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF JAMES E. AYER

I, James E. Ayer, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Chemical Engineer, Division of Fuel Cycle and Material Safety, Office of Nuclear Material Safety and Safeguards.

2. I am duly authorized to participate in answering Interrogatory 14 of the 27th Set filed on September 17, 1982 and I hereby certify that the answer given is true to the best of my knowledge.

James E. Ayer

Subscribed and sworn to before me
this day of 1982

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF LARRY W. BELL

I, Larry W. Bell, being duly sworn, state as follows:

1. I am employed by the Nuclear Regulatory Commission as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatory 55 of the 27th Set filed on September 17, 1982 and I hereby certify that the answer given is true to the best of my knowledge.

Larry W. Bell

Subscribed and sworn to before me
this day of 1982

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF LEWIS G. HULMAN

I, Lewis G. Hulman, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as Chief of the Accident Evaluation Branch, Division of Systems Integration, in the Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 25, a, d, and e of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Lewis G. Hulman

Subscribed and sworn to before me
this day of 1982

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF SIDNEY FELD

I, Sidney Feld, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Regional Environmental Economist, Antitrust and Economic Analysis Branch, Division of Licensing, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 78 to 81 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Sidney Feld

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF GERALD E. GEARS

I, Gerald E. Gears, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Land Use Analyst, Terrestrial Resources Section, Environmental Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 75x, xv, xvii, and xviii of the 27th Set filed on September 17, 1982 and I certify that the answers given are true to the best of my knowledge.

Gerald E. Gears

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF JOHN K. LONG

I, John K. Long, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer, Research Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 47 and 54 of the 27th Set filed on September 17, 1982, and I hereby certify that theanswers given are true to the best of my knowledge.

John K. Long

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF RICHARD B. CODELL

I, Richard B. Codell, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Hydrologic Engineer, Hydrologic and Geotechnical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 72, 75xi-xiv, xvi, xviii, 82, and 85a-f of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Richard B. Codell

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF ROBERT B. SAMWORTH

I, Robert B. Samworth, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Section Leader of Aquatic Resources Section, Environmental Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 75ix and 77 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Robert B. Samworth

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF MICHAEL T. MASNIK

I, Michael T. Masnik, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Fisheries Biologist, Environmental Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 75i, ii, 83a, and 84iv, v of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Michael T. Masnik

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF PAUL H. LEECH

I, Paul H. Leech, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Project Manager, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 73, 74, and 75iii-viii of the 27th Set filed on September 17, 1982, and I hereby certify that theanswers given are true to the best of my knowledge.

____
Paul H. Leech

Subscribed and sworn to before me
this   day of   1982.

____
Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF JERRY J. SWIFT

I, Jerry J. Swift, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 3, 30, 42-46, 48-53, 55-57, 62, 67, 69, 71 and 85a., b. of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

____
Jerry J. Swift

Subscribed and sworn to before me
this day of 1982.

____
Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF MOHAN C. THADANI

I, Mohan C. Thadani, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer, Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 21-24, 58, 60, 61, 63-65, and 70 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Mohan C. Thadani

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF EDWARD F. BRANAGAN, JR.

I, Edward F. Branagan, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Radiological Physicist, Radiological Assessment Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 14 through 18 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Edward F. Branagan, Jr.

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF BILL M. MORRIS

I, Bill M. Morris, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Section Leader of the Technical Review Section, Clinch River Breeder Reactor Program Offfice, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 25b, c, f, g, 26-41, 66, 67, 68, and 71 of the 27th Set filed on September 17, 1982, and I hereby certify that the answers given are true to the best of my knowledge.

Bill M. Morris

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF IRWIN SPICKLER

I, Irwin Spickler, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Chief of Section C of the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Rea tor Regulation.

2. I am duly authorized to participate in answering Interrogatories 84i, ii, iii, 87 and 88 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Irwin Spickler

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF HOMER LOWENBERG

I, Homer Lowenberg, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Chief Engineer, Office of Nuclear Material Safety and Safeguards.

2. I am duly authorized to participate in answering Interrogatories #1 through #13, #19 and #20 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Homer Lowenberg

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF MICHAEL KALTMAN

I, Michael Kaltman, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Regional Planning Analyst, Siting Analysis Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 76 and 83b, of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Michael Kaltman

Subscribed and sworn to before me
this   day of   1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF PETER B. ERICKSON

I, Peter B. Erickson, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Project Manager, Standardization and Special Projects Branch, Division of Licensing, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 89 to 95 of the 27th Set filed on September 17, 1982, and I hereby certify that theanswers given are true to the best of my knowledge.

Peter B. Erickson

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF RICHARD B. McMULLEN

I, Richard B. McMullen being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Geologist, Geosciences Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 84vi and 86 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Richard B. McMullen

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF EDMUND T. RUMBLE, III

I, Edmund T. Rumble, III, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a consultant to the Office of Nuclear Reactor Regulation on safety matters related to the proposed Clinch River Breeder Reactor Plant.

2. I am employed as the Corporate Vice President of Science Applications, Inc.

3. I am duly authorized to participate in answering Interrogatories 27-29, 31, 36-38, 40-45, 48, 51-54, 56, 57, 59, 63, and 67e of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Edmund T. Rumble, III

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF ROBERT L. ROTHMAN

I, Robert L. Rothman, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Seismologist, Geosciences Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 84vi and 86 of the 27th Set filed on September 17, 1982 and I hereby certify that the answers given are true to the best of my knowledge.

Robert L. Rothman

Subscribed and sworn to before me
this day of 1982.

Notary Public

My Commission Expires:

ENCLOSURES A, B, AND C

Included only for:

Chairman Miller

George Edgar, Esq.

Barbara Finamore, Esq.

William E. Lantrip, Esq.

William M. Leech, Esq.

DESIGNATED ORIGINAL
Certified By [signature]

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S ANSWERS TO NATURAL RESOURCES DEFENSE COUNCIL, INC. AND THE SIERRA CLUB TWENTY-SEVENTH SET OF INTERROGATORIES TO STAFF" dated October 1, 1982 in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system or, as indicated by a double asterisk, by hand delivery, this 1st day of October, 1982:

Marshall Miller, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Mr. Gustave A. Linenberger
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Dr. Cadet H. Hand, Jr., Director
Administrative Judge
Bodega Marine Laboratory
University of California
P.O. Box 247
Bodega Bay, California 94923

Alan Rosenthal, Esq., Chairman
Atomic Safety and Licensing Appeal
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Dr. John H. Buck
Atomic Safety and Licensing Appeal
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

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Tennessee Valley Authority
819 Power Building
Chattanooga, Tennessee 37401

Director
Clinch River Breeder Reactor Plant
Project
U.S. Department of Energy
Washington, D.C. 20585

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Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Daniel T. Swanson

Daniel T. Swanson
Counsel for NRC Staff