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R. Stark
C. Thomas
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March 23, 1982

Docket No.: 50-537

Mr. John R. Longenecker
Licensing and Environmental Coordination
Clinch River Breeder Reactor Plant
U. S. Department of Energy, NE-561
Washington, D.C. 20545

Dear Mr. Longenecker:

SUBJECT: CLINCH RIVER BREEDER REACTOR PLANT, REQUEST FOR ADDITIONAL INFORMATION

As a result of our review of your application for a construction permit for the Clinch River Breeder Reactor Plant, we find that we need additional information in the area of Core Performance. Please provide your final responses by May 15, 1982.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you desire any discussion or clarification of the information requested, please contact R. M. Stark, Project Manager (301) 492-9732.

Sincerely,

Original Signed by
Paul S. Check

Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: Service List

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OFFICE	CRBRPO:NRR	CRBRPO:NRR	CRBRPO:NRR	CRBRPO:NRR			
SURNAME	RStark/bm	CThomas	WFoster	PSCheck			
DATE	3/23/82	3/23/82	3/23/82	3/ /82			

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

Daniel Swanson
Office of the Executive
Legal Director
U. S. Nuclear Regulatory
Commission
Washington, D.C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
450 James Robertson Parkway
Nashville, TN 37219

William E. Lantrip, Esq.
City Attorney
Municipal Building
P. O. Box 1
Oak Ridge, TN 37830

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, D.C. 20036

Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
Knoxville, TN 37902

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

Raymond L. Copeland
Project Management Corp.
P. O. Box U
Oak Ridge, Tennessee 37830

Barbara A. Finamore
S. Jacob Scherr
Ellyn R. Weiss
Dr. Thomas B. Cochran
Natural Resources Defense
Council, Inc.
1725 I Street, N.W.
Suite 600
Washington, D.C. 20006

Eldon V. C. Greenberg
Tuttle & Taylor
1901 L Street, N.W.
Suite 805
Washington, D.C. 20036

L. Ribb
LNR Associates
Nuclear Power Safety Consultants
8605 Grimsby Court
Potomac, MD 20854

QUESTIONS TO THE APPLICANT ON PSAR NUCLEAR DESIGN

CS 491.1 (3.1) In Section 3.1, "Conformance with General Design Criteria," there is no design criterion comparable to 10 CFR 50, Part A, Criterion 28, "Reactivity Limits." Why is this general design criterion not a part of Section 3.1? Are appropriate limits on the potential amount and rate of reactivity increase discussed in this criterion going to be quantitatively specified?

CS 491.2 (3.1) Please explain why Criterion 29, "Protection against Anticipated Operational Occurrences," of 10 CFR 50, Part A is not a criterion in Section3.1 of the PSAR.

CS 491.3 (4.2.2.1) The specific speed of response requirements does not seem to be presented in Section 4.2.2.1.3. Where is it presented?

CS 491.4 (4.2.3.3) Why aren't there sufficient calculational uncertainities listed to enable one to judge the fragility of the PCRS and SCRS scram conclusions?

CS 491.5 (4.2.3.3) In one case you claim one dollar of reactivity is inserted within .3 seconds after rod motion begins. Why doesn't this agree with the presentation in Figure 4.2-122?

Why aren't the treatments of PCRS and SCRS scrams parallel?

CS 491.6 (4.2.3.4) Are the PCA position indicators and dampers also being tested?

Which of the tests mentioned in Section 4.2.3.4.1 have been completed and documented?

CS 491.7 (4.3.2.1) The Source Range Flux Monitoring (SRFM) System is described in Section 4.3.2.1.5. The applicant is requested to provide the minimum acceptable neutron flux at the detector, the maximum acceptable gamma-ray dose rate at the detector, and the calculated neutron flux and gamma-ray dose rate at the detector. Also provide a description (method, models, etc) of how these calculations were made.

In this same section it is indicated that SRFM detector operating characteristics will be experimentally verified in ZPPR critical experiments which will mock up the actual CRBR installations as close as practicable. In addition transport calculations will be employed to account for neutron streaming effects in the cavity which cannot be mocked up in the ZPPR. Please respond to the following:

(1) Describe the transport calculations (method, models, etc) to be performed.

(2) What plans are being made to verify that the planned ZPPR experiment is an accurate test of the SRFM detectors for the actual CRBR?

(3) Are transport calculations being planned to determine the radiation environment at the SRFM detectors for the actual CRBR configuration as well as the ZPPR configurations? If so, describe these planned calculations, e.g., method, model, etc.

CS 491.8 (4.3.2.2) The applicant indicates that power distribution limits are derived from maximum allowable peak heat generation rates for nominal and anticipated operational conditions, which combined with the rod mechanical and thermal parameters, assure that incipient fuel melting does not occur in the fuel pellet with peak power. What are these specific, quantitative, power distribution limits? What are these specific, quantitative, power distribution limits? What are the maximum (quantitative) allowable peak heat generation rates (linear power) for nominal and anticipated operational conditions? What clad and coolant temperatures correspond to these maximum peak heat generation rates?

CS 491.9 (4.3.2.2) What type of instrumentation is planned to allow detection of flux (power) tilts in the core at operational levels?

491.10 (4.3.2.2) In Section 4.3.2.2.9c clarify the expression "3σ equivalent uncertainties".

CS 491.11 (4.3.2.3) The Doppler reactivity constant has been computed for temperatures above 2100 degrees K (e.g., 3000, 4000 and 5000 degrees K). What assumptions were made, or how uncertain are these high temperature coefficients when your basic 30-group library probably only gives temperature dependent cross sections to 2100 degrees K? Can you argue that any safety considerations only very weakly depend on accurate high temperature Doppler coefficients?

CS 491.12 (4.3.2.3) Uncertainty in the Doppler constant has been based principally on the analysis of the SEFOR Core I and II experiments performed by GE. Is the extrapolation from SEFOR to CRBR simply from one reactor to another or between reactors and methods? If the PSAR method for calculating the Doppler constant is different from that used by GE to calculate the SEFOR Doppler constant, please provide a comparison of the two methods. Also provide justification as to why the uncertainty of the GE method should be accepted as that for the CRBR method. The PSAR also indicates that Ref. 9 provides data for extrapolation from SEFOR to LMFBR power reactors accounting for differences in core composition, core spectrum, etc. Provide justification that the SEFOR to LMFBR power reactors (1973 type) extrapolation should be identical to the SEFOR to CRBR extrapolation.

CS 491.13 (4.3.2.3) It appears that Expansion and Bowing Reactivity Coefficients are computed by integrating core movements over axial and radial worth curves. Has the reactivity of the small core reconfiguration been accounted for?

As the core heats up the structural and fuel materials expand increasing the size of the core. Does the mass of sodium necessarily increase in the expanded core? That is, is it possible that sodium expands enough with higher temperature that its mass in the core stays the same or even decreases?

CS 491.14 (4.3.2.4) There appears to be insufficient control rod neutronics. Provide descriptions of calculations and data that characterize control rod burnup, management, and flux and power distributions.

CS 491.15 (4.3.2.7) In order to establish the criticality of the hot-full power CRBRP, why not perform a direct K calculation at hot-full power conditions?

CS 491.16 (4.3.2.7) Why were the minimum critical configuration calculations performed with only P_0 cross sections?

CS 491.17 (4.3.2.9) Section 4.3.2.9 and Table 4.3-35 report neutron flux and fluence data at locations within the core and in structural components outside the core, i.e., core barrel and reactor vessel. Please respond to the following question regarding the ex-core calculations:

1. What was the calculational method used?

2. What was the geometrical model used and what modeling approximations were made?

3. What was the neutron source used in the calculations? What was the basis for this source and what approximations were made in incorporating it into the ex-core calculations?

4. What procedure was used to insure that the ex-core neutron fluences, over plant life, are conservative and representative of the worst points on the core barrel and pressure vessel?

5. What is the accuracy of the calculated fluxes?

6. What are the limiting flux (fluence) values for the core barrel and pressure vessel?

CS 491.18 (4.3.3) Cite references where your procedures (methods, codes, models, and data) have been clearly compared with some other laboratory's procedures for the calculation of Doppler coefficients, sodium void coefficients, control rod worths, power and flux distributions, material worths, burnup, bowing reactivity coefficients, power coefficients, temperature defects, startup coefficients, etc.

Some of the fundamental neutronics parameters of the CRBR are Doppler coefficients, sodium void coefficients, control rod worhts, power and flux distribution, burnups, and bowing reactivity coefficients. Identify the particular safety consideration that you feel is the most impacted, limited, or made uncertain by each of the above parameters. Then, identify the most uncertain link in the calculational chain for each of the parameters.

CS 491.19 (4.3.3) The applicant's method of calculating the CRBR and applying biases derived from critical assembly investigations is similar to that performed for the FFTF. Since startup measurements on the FFTF have been completed, what investigations have been performed to discover which methods and calculations did not stand up well for the FFTF and hence may be suspect for the CRBR?

Does the 30-group neutron cross section library (your basic starting point) have a reference? Can this exact library be obtained in order to reproduce any of your calculations?

CS 491.20 (4.3.3.7) In this section you state that ZPPR-4 control rod bank worth C/E values range from 0.95 to 1.04. Why do these numbers differ from those presented in Table 4.3-40?

CS 491.21 (15.1.2) Section 15.1.2 describes qualitative core limits for normal operations, transients, and accidents. Are specific, quantitative, design limits going to be specified? If not, please justify why qualitative limits are preferrable to quantitative limits.

CS 491.22 (15.1.2) No sodium boiling is used as a limit for extremely unlikely faults in Table 15.1.2-2. This limit does not appear to have a specific value (temperature) as it depends on the coolant pressure. If this criterion results in a variable quantitative temperature limit for the various events considered why is the corresponding design limiting coolant temperature (and its basis) not specified for each event?