NUREG-0968
Supplement No. 1

Safety Evaluation Report
related to the construction of the Clinch River Breeder Reactor Plant

Docket No. 50-537

U. S. Department of Energy
Tennessee Valley Authority
Project Management Corporation

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

May 1983

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In March 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0968) regarding the application for a license related to the construction of the Clinch River Breeder Reactor Plant.

Since the preparation of the Safety Evaluation Report the Advisory Committee on Reactor Safeguards considered the Clinch River construction permit license application at its 276th meeting and subsequently issued a favorable report, dated April 19, 1983 to the Commission (See Appendix I of this report). In addition, we have received and reviewed additional documents associated with the application, and held a number of meetings with the applicants. These events and documents are identified in Appendix E to this supplement.

This supplement, SSER-1, to the Safety Evaluation Report, provides our evaluation of additional information received from the applicants since preparation of the SER regarding previously identified outstanding review items, and our response to the comments made by the Advisory Committee on Reactor Safeguards in its report.

Each section of this supplement is numbered and titled to correspond to the sections of the Safety Evaluation Report (SER) that have been affected by our additional evaluation and does not replace the corresponding section of the SER. Appendix E is a continuation of the chronology and lists additional documents used in the supplemental review. Appendix J is an errata sheet for the SER.

The NRC Licensing Project Manager for the Clinch River Breeder Reactor Plant is Richard M. Stark. Mr. Stark may be contacted by calling (301) 492-9732 or by writing to: CRBR Program Office, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.6 Summary of Outstanding Construction Permit Issues

The staff had identified certain outstanding issues in its review which had not been resolved with the applicants at the time the SER was issued. The current status of all open items is discussed below:

Item and Section	Status
(1) Review of RDT Standards F9-4T and F9-5T (3.9.9.2.3)	Closed
(2) Compliance with Regulatory Guide 1.75 (7.2.2.6)	Open
(3) Plant Protection System Monitor (7.2.2.7)	Closed
(4) Solid-State Programmable Logic System (7.3.2.4)	Closed
(5) Emergency Planning, 10 CFR 50, Appendix E, Part II, Requirements A and B (13.3.2.1)	Closed
(6) Quality Assurance (17.3)	Open

3.9.9.2.3 Finding No. 3 - Design Analysis Methods, Codes and Standards

(Resolved, based on applicants' commitment to evaluate the safety significance of material property variations and future technical developments, and to provide verification and qualification of computer programs used in the elevated temperature design analysis.)

The CRBRP Principal Design Criteria were used as the basis for this review. The PSAR for the CRBRP has been written following the Standard Review Plan (SRP) for light-water reactors (LWRs). The SRP, however, contains no review procedures and acceptance criteria that are applicable for components in elevated temperature service where creep is governing. The only national consensus or NRC-approved codes and standards are ASME Code Cases 1592-3, -4, -5, and -6 for components in elevated temperature service and RG 1.87, "Guidance for Construction of Class 1 Components in Elevated Temperature Reactors." However, numerous revisions to Code Case 1592 for Class 1 components in elevated temperature service have been made and are included in the current version of Code Case N-47, which is the successor to Code Case 1592 which was used by the applicants.

Creep-rupture damage at stress raisers was evaluated by the ratios of the time at stress to the minimum time to rupture at the stress. Since the elastically calculated thermal stresses at stress raisers are well above yield, the yield strength properties were used to calculate local stresses. Average rather than minimum yield strength values were used to evaluate creep-rupture damage according to RDT F9-5T so as not to underestimate the stresses and damage. However, cyclic hardening can more than double and the yield strengths of austenitic materials, thereby increasing the local stresses and creep-rupture damage. Since creep-rupture damage is such a highly nonlinear function of stress, the damage occurring after cyclic hardening can be orders of magnitude higher. At all locations where the local stress exceed yield, these effects should be included in the creep-rupture damage evaluation.

The applicants in the PSAR and in the report, WARD-D-0185, indicate that full inelastic analysis will be used for locations where elastic analysis results do not meet Code limits.

The NRC has reviewed the inelastic design methods for elevated temperature components described in RDT F9-4 and RDT F9-5. Acceptance criteria are based on 10 CFR 50, which requires demonstration of the verification and qualification of the finite element computer programs used for performing design analysis.

Several large computer programs including ANSYS, WECAN, MARC and ABAOUS have been used by the applicants for performing the inelastic finite element design analyses. Verification of these programs is accomplished primarily by comparing calculated results with exact analytical solutions or experimental data available from the literature or specially designed benchmark problems. Qualification is accomplished by showing that the solution is applicable and adequately represents the actual component behavior for the anticipated operating conditions.

RDT Standard F9-ST provides guidelines for verification and qualification of the finite element computer programs as well as recommended constitutive equations describing the plastic and creep response of the structural materials. Existing validation of the inelastic analysis portions of the computer programs is largely limited to the ORNL benchmark calculations and comparisons with test results with very simple geometries and loading conditions. The computer techniques have been shown to model at least the qualitative behavior of simple structures. Further qualification is in progress and planned for the next several years in the DOE Base Technology Program. For licensing purposes a commitment is required to complete the Verification and Qualification of materials models and their use in computerized design analysis methods prior to receiving an operating license.

The preliminary code evaluation of WARD-D-0185 Report, "Integrity of Primary and Intermediate Heat Transport System Piping in Containments," is based on elastic analyses. For some locations the results of elastic analyses given in the report do not satisfy code limits. Moreover, the code does not have any applicable elastic analysis criteria for discontinuities. In some cases accumulated inelastic strains are evaluated using the simplified method given in Item 6.2 of RDT-ST Standard for insignificant creep. This RDT Standard limits application of this method only by the condition that the maximum metal temperature is always below the value corresponding to the point where $S_m > S_t$ for 10^5 hours. For hot-leg piping this condition is satisfied. However, this condition is not as limiting as ASME Code Case N-47 wherein primary membrane plus bending stresses are allowed to reach 1.5 S_m but are limited to 1.25 S_t.

Resolution

The resolution consists of the following actions:

(1) The applicants commit to keep abreast of the developing design technology for operation at elevated temperatures and to assess the potential CRBRP safety implications of new developments.

(2) For those elevated temperature components containing radioactive sodium where inelastic design analyses are used, the applicants have committed to a confirmatory program to evaluate the significance of material property variations. The program should be completed prior to submitting an operating license application. This requires that minimum yield strength and minimum creep strength (80 percent of the average isochronous curves) properties be used to evaluate the fatigue damage,

$$\sum_{j=1}^{p} \left(\frac{n}{N_d}\right)_j$$

and the accumulated inelastic strains. These damage actions and the creep rupture damage,

$$\sum_{k=1}^{q} \left(\frac{\Delta t}{T_d}\right)_k$$

shall be presented for both minimum and average material properties using the method provided by the ASME Code Case for Class 1 components in elevated temperature service and reported.

The creep portion of the total accumulated inelastic strains (membrane, pending, peak) shall be presented using the method provided by the ASME code case for Class 1 components in elevated temperature service and reported.

The applicants shall demonstrate the structural adequacy of the components with the above values of damage and inelastic strain.

As a result of the staff review of materials properties variations, the applicants are required to consider minimum and average properties in performing the Confirmatory Programs associated with Findings 1, 5 and 9 in Para. 3.9.9 of the SER.

(3) The applicants have committed to provide formal verification and qualification of the computer programs and elements thereof which were or are to be used in their elevated temperature design analyses. Verification will include references to benchmark problems documented in the open literature, sample problem solutions supplied by the computer program developed and submission of problem solutions by the applicants. Qualification will include formal reports on comparisons with test results and other work done by the applicants to qualify those portions of the computer programs which are being used for the first time in a formal nuclear power plant licensing process.

These computer design analysis methods include inelastic material behavior models. Validation of design methods will include verification that the materials models represent the actual material behavior to the extent needed in engineering design. Constitutive relations for the complex directional dependence of the plastic and creep hardening mechanisms are included in computerized finite element methods. Kinematic (as well as isotropic) strain hardening, cyclic hardening and creep hardening (including relaxation with reversed loading) are built into subroutines in these computer programs. The directions and biaxialities of the strain increments caused by the two most severe types of loading (thermal transients and seismic loads) are different. In the creep regime, the sequence of loading also affects the resulting stresses (which produce creep rupture damage), strain ranges (which produce creep fatigue damage), and the accumulated strains which are limited by the Code.

NRC also recognizes that end-of-life strains are obtained by extrapolating finite element creep ratcheting results for a few cycles (typically 3 to 10) to the total number of operating cycles anticipated during the life of the plant. Because of the large number of parameters, variables and combinations thereof, the applicants cannot feasibly provide formal documented verification and qualification of all the relevant cases. Therefore, NRC will complete the Verification and Qualification of the finite element programs by independently auditing materials models, and constitutive relations. Computer solutions will be checked for selected geometries and loading sequences chosen to exercise key elements of the methods which are important in the design analyses of CRBR. All verification and qualification requirements will be completed prior to granting of an Operating License.

(4) The applicants have committed that the method of item 6.2 in RDT Standard F9-5T will not be used in the final Stress Reports.

The staff finds that the resolution is adequate and acceptable for the CP review.

7.2.2.6 Regulatory Guide 1.75

The staff's evaluation of the applicants' conformance with Regulatory Guide 1.75 is provided in Section 7.2.2.6 of the Clinch River Breeder Reactor Plant (CRBRP) SER (NUREG-0968, dated March 1983). The staff identified this item as an unresolved item. The staff considers this item (Regulatory Guide 1.75) to be a complex concern that involves coordination with the CRBR Program Office and review branches outside the ICSB. We are continuing to work with the applicants and other NRC review branches to resolve this open item before the issuance of the construction permit and will report our findings in a future supplement to the SER.

7.2.2.7 PPS Monitor

On the basis of its review of the information furnished by the applicants regarding the Plant Protection System (PPS) monitor which was reported in Section 7.2.2.7 of the SER, the staff expressed concerns regarding certain design provisions of the monitor. Among these were the common tie-in point for the three Primary RSS subsystem channels, the use of a non-Class 1E test device, the lack of provisions for an optional or redundant test method and the lack of a defined method for periodically verifying the correct operation of the monitor.

In a letter dated April 8, 1983, the applicants made commitments to:

(1) Provide two independent on-line PPS monitors such that a failure in one PPS monitor would not propagate to the second PPS monitor. The test results from the second PPS monitor would be used to check the test results from the first PPS monitor. Self-test features will be provided for each monitor.

(2) Perform a Failure Modes and Effects Analysis (FMEA) on the PPS monitor to provide assurance that common mode failure mechanisms do not exist that would incapacitate the RSS. The PPS monitor design will be modified to prevent any common mode failures that are discovered during the FMEA.

(3) Make both PPS monitors Class 1E.

(4) Use isolation devices in the PPS monitor design that are qualified to the same criteria discussed in the CRBRP SER Section 7.2.2.2.

The staff has concluded after reviewing the PPS monitor design and the applicants' commitments noted above that our previous concerns have been resolved. However, as a confirmatory item, the applicants are required to revise the CRBRP PSAR discussion regarding the PPS monitor to reflect the final design criteria as noted above. The PSAR should also indicate that Appendix B quality assurance requirements will be applied to the PPS monitor.

7.3.2.4 Solid State Programmable Logic System (SSPLS)

The applicants were asked to identify and document where microprocessors, multiplexers, or computer systems may be used in or interface with safety-related systems. The applicants in a meeting have stated that the above systems are used in several non-Class 1E applications and where a microprocessor, multiplexer, or computer interfaces with a Class 1E signal, that signal will be isolated by a qualified Class 1E isolator prior to being utilized by a non-Class 1E system. The two systems which use microprocessors, multiplexers, or computers for Class 1E application are the Solid State Programmable Logic System (SSPLS) and the Radiation Monitoring System (Section 7.6 of the SER).

The SSPLS controls and actuates safety-related (Class 1E) equipment. It contains control logic, signal conditioners, isolation devices, and auxiliary circuitry. The SSPLS will be qualified to IEEE 279, 323, 344, and 383 as required for all Class 1E devices in order to minimize the possibility of failures. In addition, the SSPLS is comprised of three separate and redundant safety-related divisions so that a failure in one division will not affect any component or device in the other divisions.

The staff questioned the applicants with regard to the SSPLS manual initiation capability. The applicants stated that all motor operated or pneumatically actuated valves controlled by the SSPLS can be operated or actuated manually, however, pumps, fans, and dampers require operability of the SSPLS in order to be manually initiated.

It was not apparent from the PSAR which CRBR systems utilized the SSPLS and which of the equipment controlled by the SSPLS was safety-related. In a letter dated February 15, 1983, from John R. Longenecker to J. Nelson Grace, the applicants provided a list of systems and the circuitry that will utilize the SSPLS. This list included the Power Distribution System, Emergency Chilled Water System, HVAC System, Fire Protection System, Recirculating Gas Cooling System, Reactor Heat Transport System, Plant Service Water System and Auxiliary Liquid Metal System.

The applicants stated that the SSPLS will perform the necessary logic operations and interlocking functions and provide final outputs to each piece of equipment to be controlled.

Microprocessor based circuitry will be dedicated to control only one device so that failure of one microprocessor will not affect the operation of any other device or controlled component. In addition, the applicants have stated that failures will be reduced by having the microprocessor based systems meet the following design criteria:

(1) modules using microprocessors shall be capable of being tested on a discrete basis.

(2) Each microprocessor shall be furnished with continuous self-diagnostic capability to interrogate its function.

(3) SSPLS will be designed for maximum reliability and availability. The design goal availability for each device shall be approximately 99.9955%.

(4) The software used to implement the microprocessor logic will be testable and subjected to verification and validation and will meet the requirements of IEEE 739-1981, "Standards for Software Quality Assurance Plans."

(5) The features provided for periodic testing will also be used to operate the equipment manually.

The staff has reviewed the SSPLS and the design criteria that will be utilized for the microprocessors. The present staff position is that incorporating this new technology (i.e., microprocessors) into safety-related systems at CRBRP involves both the potential for improving system performance and the potential for introducing new system failure modes. Microprocessors offer potential advantages over discrete circuitry for the proposed application. Some of these advantages are the increased checks and testability which may lead to greater reliability, improved equipment response time and minimization of equipment and cabling.

Regarding the manual initiation concern (pumps, fans and dampers requiring SSPLS to operate), the applicants have stated that all of the systems can be manually initiated (albeit with the use of the microprocessors) and that if a single failure of the manual actuation control circuitry occurs for any device controlled by one SSPLS safety division, the ability to manually initiate the redundant device in the other SSPLS safety division will not be affected.

The intent of Section 4.17 of IEEE-279 is to provide manual initiation of each protective action at the system level such that no single failure with the manual, automatic, or common portions of the protection system shall prevent initiation of protective action by manual or automatic means. Furthermore, manual initiation should depend upon the operation of a minimum of equipment. Regulatory Guide 1.62, which describes a method acceptable to the staff for complying with the requirements of Section 4.17 of IEEE-279, states in part that the amount of equipment common to both manual and automatic initiation should be kept to a minimum. It is preferable to limit such common equipment to the final actuation devices and the actuated equipment. The intention of the above regulation and guidance is to provide highly reliable manual initiation designs for protective actions.

The staff has concluded that the CRBRP SSPLS design includes an acceptable method for meeting Section 4.17 of IEEE-279. This conclusion is based on the fact that each redundant microprocessor will afford a high reliability factor (design goal is approximately 99.9955% availability), will be supported by redundant divisions, and, in fact, will only replace discrete components that would lead to the same manual initiation design (manual initiation would be dependent on these discrete components). Furthermore, the microprocessor will be furnished with continuous self-programmatic capability. The applicants have provided an acceptable description for the construction permit stage of review including design criteria and information regarding the systems that will utilize the SSPLS. The staff has reviewed the applicable sections of the PSAR and the responses to our questions and concluded that the SSPLS design is acceptable. However, the staff wishes to perform a detailed review of unique common mode failures that have the potential of disabling microprocessors and the potential lack of the proper technical information regarding microprocessors that will be documented in the FSAR.

The staff requires the applicants to provide the following additional information during the OL review:

(1) A system description of the SSPLS including a description of the hardware and software utilized by the SSPLS. The overall hardware description should include the sensors (ranges and rates of change) bypass capabilities, interlocks, redundancy, diversity, and outputs (displays, trips ranges and actuated devices). Block diagrams should be included to provide an integrated presentation of the hardware functions. All interfaces, both internal and external to the SSPLS, should be delineated.

(2) A description of the overall computer organization. This description should include the programming philosophy and conventions to be utilized; the execution philosophy; the interrupt structure and philosophy; the utilization and relationships between various types of storage; the data base structure and philosophy; and the programming language and conventions.

(3) A discussion of the program operating conditions such as interruptability, re-locatability and protected memory.

(4) A discussion of the scaling techniques utilized and the accuracy and time response requirements for the SSPLS.

(5) A discussion of design qualification testing.

(6) A discussion of the operating environmental requirements (normal, abnormal, and accident).

(7) A discussion of the control of changes due to fuel cycles, set points, maintenance, and future design improvements.

(8) A discussion of test features (on-line and periodic), including auto-diagnostic capability.

(9) A discussion of the functional independence of the hardware and software.

(10) A discussion of the design techniques utilized to prevent elevated DC control voltages in the SSPLS circuitry from causing a common mode failure mechanism for premature degradation or component failures within the SSPLS (IE Information Notice No. 83-08).

(11) A summary description of the test program including the acceptance criteria which will be implemented to assure that the effects of Electromagnetic Interference (EMI) will not cause unique failure modes which would prevent the required safety system(s) from functioning.

In summary, the staff considers that the applicants have provided adequate information for the C.P. review to demonstrate that the final design can fully meet all applicable regulations including IEEE-279-1971. Additional review addressing the specific areas discussed above will be performed by the staff at the time of the OL review.

13 CONDUCT OF OPERATIONS

13.3 Emergency Preparedness Evaluation

13.3.1 Background

The staff's evaluation of the applicants' preliminary emergency plan is provided in Section 13.3 of the Clinch River Breeder Reactor Plant (CRBRP) SER (NUREG-0968, dated March 1983). The preliminary emergency plan for CRBRP (CRBRP plan) was reviewed against the requirements of 10 CFR 50, Appendix E, Part II. In the SER the staff specifically identified three items for which additional information/clarification was required. In addition, a working meeting with the applicants was held on March 21, 1983, in order to discuss clarification of additional items concerning: emergency planning zone (EPZ) boundary determination; impediments to evacuation (e.g., use of buses to evacuate Edgewood School); and the statutory nature of the agreement between CRBRP and the Tennessee Emergency Management Agency.

On April 8, 1983, the applicants submitted additional information/ clarification for those items identified in the SER and those discussed at the working meeting, as discussed above. The applicants committed to include the new information in the next scheduled PSAR amendment to the CRBRP plan. The applicants' responses have been evaluated and are discussed in Section 13.3.2 of this supplement. The order of presentation corresponds to the listing of items that appear in Section 13.3.2 of the SER, followed by a discussion of those items addressed at the working meeting on March 21, 1983.

13.3.2.1 Discussion - Requirements A and B, Items 1 and 2

The applicants have identified the Tennessee Emergency Management Agency (TEMA) as the state agency responsible for coordination of the efforts of all state agencies and local governments in the development of response plans that have an impact beyond the capability of a single agency or local government to control. The actual agreements and arrangements involved with such state agency and local government will be specifically defined in the State of Tennessee CRBRP Radiological Emergency Response Plan which will be provided in the CRBRP FSAR.

The state of North Carolina, Department of Crime Control and Safety has been identified by the applicants as the principal agency in North Carolina responsible for coping with emergencies.

Based on our review of their plan and submittal as discussed above, the staff finds that the applicants have provided an acceptable response to these two items and that the requirements of Appendix E, Part II, Items A and B, have been met.

13.3.2.6 Discussion - Requirement H, Capability for Real Time Meteorology

The proposed update to the CRBRP PSAR confirms that real time meteorology will be used in dose assessment related to actual and potential releases of radioactivity. Real time meteorological data will be available in the Control Room, Technical Support Center, Central Emergency Control Center, Muscle Shoals Emergency Control Center and State of Tennessee Emergency Operations Center.

Based on our review of their plan and submittal as discussed above, the staff finds that the applicants have provided an acceptable response to this item and that the requirements of Appendix E, Part II, Item H, have been met.

Items Clarified as a Result of the Working Meeting on March 21, 1983

Determination of EPZ Boundary

The preliminary evacuation time estimates (PSAR Appendix 13.3A) assumed a circular, 10 mile plume EPZ boundary. The final EPZ boundary will be determined by the State of Tennessee following coordination of the planning efforts with local government agencies. The applicants will provide CRBRP specific information to the State to supplement, as necessary, NUREG-0396 EPZ size guidance.

This final EPZ boundary will also consider such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

Impediments to Evacuation

The applicants specify that there are no known impediments that have been identified which would significantly affect effective evacuation of the assumed EPZ. The proposed update of the PSAR Section 13.3A, paragraph C(4)d will reflect the means to evacuate schools and other institutions. This procedure will be specifically addressed in the CRBRP FSAR.

Agreement Between CRBRP and TEMA

The applicants submitted, by reference, the agreement with TEMA (letter, E. P. Tanner, Director, TEMA, to H. J. Green, TVA, July 6, 1982) which describes TEMA's statutory authority and responsibility in the preparation, coordination and updating of emergency response plans, and the conduct of emergency operations by all participating agencies.

Conclusion

On the basis of its review of the applicants' preliminary plans for coping with emergencies (including the April 8, 1983 submittal), the staff concludes that, provided the items identified in Section 13.3.2 for which the applicants have made commitments are accomplished, the preliminary plans are acceptable and meet the requirements of 10 CFR 50, Appendix E, Part II.

17 QUALITY ASSURANCE

17.3 Q. A. Program

The staff's evaluation of the applicants' Q.A. program is provided in Section 17.3 of the Clinch River Breeder Reactor Plant (CRBRP) SER (NUREG-0968, dated March 1983). The program was reviewed against the applicable Q.A. criteria of 10 CFR 50 Appendix B and TMI Action Plan (NUREG-0660) Item I.F. In the SER the staff indicated that it was still reviewing the list of structures, systems and components controlled by the Clinch River Breeder Reactor Quality Assurance Program. The staff asked several questions in order to clarify and complete the list. The applicants have since provided a response (Longenecker to Grace letter dated April 8, 1983) which addressed the staff questions. The staff review of the response finds that most staff questions are now resolved. However, several items still require further discussion and clarification. The staff has requested that the applicants meet with the staff in May to resolve the remaining items. Therefore, this issue remains open and will be addressed in a future supplement to the SER. The staff requires resolution of this item before the issuance of the construction permit.

19. Report of the Advisory Committee on Reactor Safeguards

The ACRS full committee completed its review of the Clinch River Breeder Reactor Plant at its 276th meeting. A copy of the committee report dated April 19, 1983 is attached as Appendix I of this report. A discussion of the current status of each item on which the committee commented or made recommendations in the report is included in the following paragraphs.

1) The committee recommended that additional seismic studies, similar to those done for large piping, be done for small piping. The staff has requested that this study be performed by the applicants and submitted to the staff. The staff will review this report prior to the issuance of a construction permit.

2) The applicant has agreed to perform a PRA which includes assessment of shutdown heat removal systems, confinement, hydrogen combustion, and sodium concrete attack. The applicants' schedule for preparing the PRA is in Section D.3 of the SER. The schedule calls for a final report in December 1984. The staff will review the PRA in 1985 and will be prepared to present its preliminary findings to the ACRS in late 1985. The staff will consider this to be a condition of the license.

3) The committee concurs with the staff position on CDAs.

4) The committee recommends more confirmatory work in a number of areas related to containment cleanup and venting including more work in aerosol chemistry and behavior monitoring of the containment atmosphere, hydrogen distribution and the development of procedures for venting and purging.

   The staff agrees and, while we found the applicants' analyses in these areas to be sufficient for the construction permit, the staff is requiring some additional confirmatory work in several areas, and the applicants have committed to pursue these matters. These additional areas are discussed in the staff SER and include more detailed studies of the deposition of aerosols on the containment wall and heat sinks, and the subsequent effects on heat transfer. The qualification of vital equipment in this environment is also required (see SER page A.4-23).

The staff is also requiring additional studies for the operating license review of the requirements for monitoring the composition of the containment atmosphere and environment (see SER page A.4-15 and A.4-20) and on development of specific requirements for the vent-purge operation (see SER page A.4-15).

5) The committee concurs with the staff position on natural circulation testing.

6) The committee concurs with the staff position on the reliability assurance program.

7) The committee concurs with the staff position on the materials confirmatory program.

8) The staff agrees the close monitoring of pertinent foreign experience of steam generators is prudent. The staff commits to monitoring foreign experience and will require that the applicants monitor all relevant foreign experience.

9) The staff can assist the committee during the operating license review should the committee wish to complete its review of the principal design criteria. The staff has modified the current principal design criteria as a result of the past meetings where the PDC were presented to and discussed with the committee. The staff has concluded in Section 3.1 of the SER that these criteria are adequate for CRBRP.

10) The staff agrees that features to reduce sabotage are very important. The staff will conduct a detailed sabotage study as a part of its operating license review. Refer to SER 13.7 for a discussion of the applicants plans and OL requirements.

11) The status of all outstanding issues has been revised and is reported in Section 1.6 of this report.

APPENDIX E

CHRONOLOGY

March 2, 1983	Applicants submitted Amendment 76 to the PSAR which includes: revisions to Section 3.1, "Conformance with General Design Criteria"; Section 4.2, "Reactor Mechanical Design"; Chapter 7, "Instrumentation and Controls"; Chapter 9, "Auxiliary Systems"; and Appendix A, "Computer Codes."
March 4, 1983	Applicants submitted responses to questions raised in recent discussions regarding thermal margin beyond the design base (TMBDB).
March 4, 1983	Applicants submitted a description of the CRBRP Cell Liner Design Validation Program and Fallback Plan.
March 4, 1983	Applicants response to ACRS questions regarding the capability for shutdown heat removal on natural circulation, with a loss of bulk AC power, for greater than two hours.
March 4, 1983	Applicants submitted a report entitled "CRBRP Heat Transport System Incontainment Piping Reserve Seismic Margin," in responses to questions raised by the ACRS during the February 11, 1983, committee meeting.
March 7, 1983	Applicants submitted a corrected PSAR Table 5.6-12 that was in error in Amendment 76.
March 8, 1983	Applicants responded to NRC concerns as to the possibility that a fission-gas-driven compaction of the core might lead to energetics during a Hypothetical Core Disruptive Accident (HCDA).
March 8, 1983	Notice of meeting with the applicants for March 21, 1983, to discuss emergency planning open items.
March 10-11, 1983	Full ACRS Committee meeting on CRBRP (Final photo-copies of SER distributed to ACRS).
March 11, 1983	SAI submitted their report entitled "Risk Reduction Feasibility Study of Selected Modifications to CRBRP Safety Systems, SAI-84-123-WA."
March 11, 1983	Letter to ACRS providing 16 final photo-copies of the SER
March 11, 1983	Letter to applicants providing final copies of the SER.

March 14, 1983	Applicants submitted a report entitled "The Adequacy of the CRBRP Reactor Vessel NDE Inspections."
March 14, 1983	Notice of meeting with applicants for March 23, 1983, to discuss mechanical engineering open items. (Meeting rescheduled for March 31, 1983.)
March 15, 1983	Letter to ACRS responding to a question which the staff requested that the applicants provide a report which discusses the analysis, approach, and methodology for determining piping margins in excess of the SSE.
March 23, 1983	Letter to applicants providing printed copies of the SER.
March 23, 1983	Applicants submitted new information on the CRBRP Cell Liner Design Validation Program.
April 1, 1983	Memo to ACRS CRBR Working Group on Systems Integration and Instrumentation Control providing eight documents related to CRBRP.
April 1, 1983	Applicants submitted additional information on instrumentation and controls.
April 1, 1983	Summary of the February 9, 1983 meeting with applicants on probabilistic risk assessment.
April 4, 1983	Applicants submitted new information on CRBRP Cell Liner Design Validation Program; specifically, analysis of the wall liner at square penetrations.
April 7, 1983	Applicants submitted response to Open Item No. 1, "Review of RDT Standards F9-4T and F9-5T."
April 8, 1983	Applicants submitted response to Open Item No. 3, "Plant Protection System (PPS) Monitor."
April 8, 1983	Applicants submitted response to Open Item No. 5, "Emergency Planning."
April 8, 1983	Applicants submitted response to Open Item No. 6, "Quality Assurance."
April 12, 1983	Applicants submitted response to Open Item No. 4, "Solid State Programmable Logic System (SSPLS)."

April 18, 1983	Applicants submitted final report from Stanford Research Institute (SRI) on CRBRP scale model tests.
April 22, 1983	Notice of meeting with the applicants for April 28, 1983 on the status of cell liner confirmatory test plans.

APPENDIX G

REFERENCES

The following reference was inadvertently omitted in the March SER.

Hanson, J.E., "Comparison of Clinch River Breeder Reactor Design Bases Accidents with those for Light Water Reactors and Liquid-Metal-Cooled Fast Reactors," EGG-NTAP-6152, Idaho National Engineering Laboratory, Idaho Falls, ID, January 1983.

APPENDIX I

ACRS REPORT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 19, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE CLINCH RIVER BREEDER REACTOR PLANT

During its 276th meeting, April 14-16, 1983, the Advisory Committee on Reactor Safeguards (ACRS) completed its review of the application of the U. S. Department of Energy, the Tennessee Valley Authority, and the Project Management Corporation (the Applicants) for a permit to construct the Clinch River Breeder Reactor Plant (CRBRP). Previous consideration had been given to this project during the Committee's 267th meeting, July 8-10, 1982; 271st meeting, November 4-5, 1982; 272nd meeting, December 9-11, 1982; 273rd meeting, January 6-8, 1983; 274th meeting, February 10-12, 1983; and 275th meeting, March 10-12, 1983. Subcommittee and Working Group meetings were held in Washington, D. C. on February 2-3, 1982; March 30-31, 1982; May 4-5 and 24-25, 1982; June 1-2 and 24-25, 1982; August 18-19, 1982; September 30, 1982; October 26 and 27, 1982; November 19, 1982; December 1 and 10, 1982; January 7, 1983; February 3-4, and 24, 1983; and March 16-17, 1983. During this review, the Committee had the benefit of discussions with representatives of the Applicants and their consultants. We also had the benefit of the documents listed.

The CRBRP will be a liquid-sodium-cooled, mixed-oxide-fueled, fast-breeder reactor demonstration power plant. Design power is 975 MWt (350 MWe). This is the only fast breeder power plant which the ACRS has reviewed for formal licensing purposes within the past decade, although the Committee offered advice on the Fast Flux Test Facility design which is similar to that of the the CRBRP.

The proposed CRBRP site is located in Roane County in east central Tennessee approximately 25 miles west of Knoxville, Tennessee. The site consists of approximately 1300 acres on a peninsula formed by a meander in the Clinch River. The site property is owned by the U. S. Government and is currently in the custody of the Tennessee Valley Authority. The minimum distance to the exclusion area boundary is 2200 feet, and the population center distance, based on the actual population distribution, is 7 miles north-northeast of the plant. The ACRS reported on the suitability of the proposed CRBRP site in its report to you dated July 13, 1982.

In its report of July 13, 1982 on the suitability of the CRBRP site, the ACRS said, "With regard to the seismic design of this plant, we believe it is important that the combination of seismic design basis and margins in the seismic design be such that this accident source represents an acceptably low contribution to the overall risk from the plant. We believe this matter will warrant detailed examination at the construction permit stage to assure that necessary margins are available for all important systems and components." The NRC Staff has accepted an SSE and OBE for the CRBRP of 0.25g and 0.12g, respectively. The U.S. Geological Survey has raised a concern regarding a postulated local seismic zone and, while the Staff does not accept the arguments for the local zone, its existence would significantly increase the probability of exceeding the SSE. In any case, we believe that a considerable seismic margin for no loss of function of the shutdown heat removal system should be shown for low probability earthquakes larger than the proposed SSE. Ongoing studies by both the Applicants and the NRC Staff have indicated that appreciable margins exist for the large piping in the primary heat transport system of the plant; however, similar studies have not been made for small piping. Since a common mode loss of piping integrity in all three heat transport loops could disable the entire heat removal system, which in turn could lead to core melt, it is important to assure the integrity of small piping as well as of all other components needed to accomplish shutdown heat removal.

The Applicants are conducting a full-scope probabilistic risk analysis (PRA) on the current CRBRP design. The PRA should be completed soon enough that its review by the NRC Staff and the ACRS, and any resulting recommendations or additional requirements, can be considered in the design of the plant. We recommend that careful detailed attention be given in the PRA to the following topics, among others:

- The adequacy of means for shutdown heat removal, including scenarios involving earthquakes more severe than the safe shutdown earthquake. Among other things, the significance of the vulnerability of the direct heat removal system, as designed, to leaks in the primary system should be examined, as well as the effects of possible steam generator tube degradation.

- The adequacy of the secondary containment, the filter system, and other features important to limiting the uncontrolled release of radioactive material following postulated accidents involving core melt. Scenarios which might lead to overpressure of the containment should be systematically identified, and examination should be made of possible design changes to reduce the likelihood of uncontrolled releases of radioactive materials in terms of their efficacy and costs. The possible merit of a dedicated emergency power supply for the filter system should be included in such studies.

- A careful search for scenarios which have the potential for a loss of containment integrity due to hydrogen combustion. This should include any potential for confusion by the operator as to the proper course of action as well as operator errors, including those of commission.

- An examination of the merits and costs of means of delaying attack of the concrete by sodium and hot fuel.

The ACRS believes that timely completion of the PRA by the Applicants, to permit its review and evaluation by the NRC Staff and the ACRS, should be a condition of the construction permit.

An historical liquid-metal fast-breeder-reactor safety concern has been the potential for large reactivity excursions caused by, for example, a combination of failure to scram and either a loss of coolant flow or an insertion of reactivity. It is sometimes postulated that such an excursion could lead to vaporization of coolant and fuel and to rupture of the primary containment (i.e. reactor vessel, etc.) and possibly secondary containment (i.e., the steel containment shell) due to the pressures resulting from the vaporization. This event is termed an energetic core disruptive accident (CDA). Both the Applicants and the NRC Staff have independently reviewed this potential and have concluded that the probability of such an accident is quite low. Further, both conclude that, even if such a combination of events did occur, the magnitude of the resulting mechanical forces in the CRBRP design would be well below the capability of the primary containment system to withstand such forces without rupture. We concur in the NRC Staff position.

Both the Applicants and the NRC Staff have also concluded that the probability of core melt from a nonenergetic event is low. However, the NRC Staff has required, and the Applicants have provided, means to mitigate the consequences of such a core melt, should one occur. The Committee concurs in this approach and recommends that these mitigative features must be designed so as to afford a very high likelihood of successful function. The Applicants are placing considerable reliance on an air cleaning system to control releases, should the outer containment have to be vented following a major accident in the CRBRP. To confirm the anticipated performance of this system, however, more work needs to be done in many areas, including the following:

- Establishment of the physical and chemical nature, the concentrations as a function of time, and the ultimate fate of the aerosols.

- Establishment of a better basic understanding of the thermal, mechanical and chemical interactions between concrete and sodium or hot fuel.

- Establishment of reliable, unambiguous means of monitoring hydrogen and oxygen concentration in the secondary containment.

. Potential for plutonium criticality within the scrubber proposed as part of the secondary containment venting clean-up system.

. Assessment of hydrogen buildup in the secondary containment under various scenarios including the potential for nonuniform concentrations.

. Assurance of the availability of clear, simple, safe procedures for venting and purging in the unlikely event of such an accident.

In the area of shutdown heat removal reliability, the Applicants will rely on natural circulation capability to remove decay heat should there be a loss of all AC power. The NRC Staff will require demonstration of this capability prior to operation.

The reliability assurance program described by the Applicants is appropriate for this stage of plant development. Most of the emphasis is on what are considered to be safety or protection systems. The Applicants have committed to a more comprehensive reliability analysis. We recommend that, insofar as feasible, this analysis give particular attention to nonsafety systems, the malfunction of which may challenge protection systems.

The Applicants have several materials programs in progress, some of a confirmatory nature, on topics such as creep fatigue and creep rupture damage and the thermal aging of piping. We believe these programs are important and should continue.

Safety knowledge in the area of sodium-water interaction at the steam generator interface seems well in hand. However, it is recommended that both the Applicants and the NRC Staff closely monitor pertinent developments and experience in other countries.

The ACRS has not completely reviewed the proposed CRBRP Principal Design Criteria and is not prepared to endorse them in this letter.

As for any new type of plant, it is recommended that further thought be given to providing design features to reduce, as far as practical, both the feasibility and consequences of sabotage.

The issues discussed above as well as many described in the SER are ones for which more work must be done prior to their resolution. As further information is acquired, we wish to be kept informed and will recommend safety modifications to the existing design as appropriate. We expect to follow the ongoing CRBRP design, development, and construction programs more closely than would be the case for a typical LWR plant.

The Advisory Committee on Reactor Safeguards believes that, if the matters noted above and the open items described in the SER are resolved in a satisfactory manner, the CRBRP can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Additional comments by ACRS Member Robert C. Axtmann are presented below.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

Additional Comments by ACRS Member Robert C. Axtmann

Many of its strongest proponents agree that CRBR is an archaic design for a technology that will not be needed until well into the 21st century. On the other hand, far too many LMFBR experiments and reactors have ended disastrously. If fusion proves feasible and environmentally acceptable within the next fifty years, the breeder will share a niche in technological history with the hydrogen-filled dirigible. Whatever the risks of CRBR (and no one claims there are none), I find no rational basis for this project.

References:

1. Project Management Corporation, Clinch River Breeder Reactor Project, "Preliminary Safety Analysis Report," Volumes 1-27 and Amendments 1-75.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant," NUREG-0968, Volumes 1 and 2, dated March 1983.
3. EG&G, Idaho, Inc., Wood-Leaver and Associates, Inc., and Fauske and Associates, Inc., "Clinch River Breeder Reactor Plant Probabilistic Risk Assessment - Phase I," Main Report and Appendices A-G, EGG-EA-6162, dated January 1983.

APPENDIX J

ERRATA TO MARCH SER

Page 4-85, 2nd paragraph, 3rd line change "unlikely" to "likely" (that the proposed).

Page 7-26, 2nd full paragraph, last sentence, change 5.3.2.3 to 5.7.3

Page 9-119, 4th paragraph, 2nd line change "two" to "one" (of which)

Page 9-100, 4th paragraph, 1st line change Section 9.5.1 to Section 9.13.1

Page 9-115, last paragraph, last line change Section 9.5.1 to Section 9.13.1

Page 13-4, first paragraph under Section 13.3 "Emergency Planning" should be placed after the words "...DOE's Oak Ridge Reservation" in the second paragraph (line 33)

Page 13-4, move Section 13.3.1 "Introduction" up to the first paragraph under Section 13.3 heading

Page 13-11, line 2, after the word additional, insert "information that was requested by the staff concerning: the location of key plant"

Page C-9, paragraph on Assessment of Applicants Program, 6th line change March 2, 1981 to March 2, 1983.

Page F-2, add H. Hummel, Argonne National Lab. to the list of Consultants.

Page F-3, add E. Schwegler, Los Alamos National Laboratory to list of Consultants.

Page E-30 insert: June 22, 1982	Letter to applicants providing copies of the revised Site Suitability Report for the CRBRP (NUREG-0786) reaffirming the conclusions in the 1977 SSR.
Page E-30 insert: June 28, 1982	Letter to ACRS providing copies of the revised Site Suitability Report for the CRBRP (NUREG-0786) reaffirming the conclusions in the 1977 SSR.
Page E-38 insert: December 14, 1982	Applicants submitted a summary of the December 9, meeting on structural margin beyond the design base.

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0968 Supplement No. 1
4 TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant Supplement No. 1 Docket No. 50-537	2 (Leave blank)
	3 RECIPIENT'S ACCESSION NO
7 AUTHOR(S) Clinch River Breeder Reactor Program Office	5. DATE REPORT COMPLETED MONTH May YEAR 1983
9 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Clinch River Breeder Reactor Program Office Office of Nuclear Reactor Regulation Washington, D. C. 20555	DATE REPORT ISSUED MONTH May YEAR 1983
	6 (Leave blank)
	8 (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Clinch River Breeder Reactor Program Office Office of Nuclear Reactor Regulation Washington, D. C. 20555	10 PROJECT/TASK/WORK UNIT NO
	11 FIN NO.
13 TYPE OF REPORT Supplement to SER for a CP	PERIOD COVERED (Inclusive dates) March 11, 1983 through May 2, 1983
15. SUPPLEMENTARY NOTES	14 (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 1 to the Safety Evaluation Report for the application by the United States Department of Energy, Tennessee Valley Authority, and the Project Management Corporation, as applicants and owners, for a license to construct the Clinch River Breeder Reactor Plant (Docket No. 50-537) has been prepared by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. This supplement provides an evaluation of additional information received from the applicants on previously identified outstanding review items since the preparation of the SER. Also included is a response to the comments made by the Advisory Committee on Reactor Safeguards in its April 19, 1983, report to the Commission.

17 KEY WORDS AND DOCUMENT ANALYSIS

Clinch River Breeder Reactor Plant
(CRBRP)
Safety Evaluation Report

17a DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19 SECURITY CLASS (This report)	21 NO OF PAGES
Unlimited	Unclassified	
	20 SECURITY CLASS (This page)	22 PRICE S

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