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	)	Docket No. 50-537
TENNESSEE VALLEY AUTHORITY	)	
	)	
(Clinch River Breeder Reactor Plant)	)	

NRC STAFF TESTIMONY CONCERNING ADEQUACY OF THE DESIGN BASIS ACCIDENT SPECTRUM

Q1. Please state your names and affiliations.

A1. My name is Richard A. Becker. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, Clinch River Breeder Reactor Program Office, in the Office of Nuclear Reactor Regulation. My involvement with the Clinch River Breeder Reactor (CRBR) review has been with the design basis accident analysis, heat transport system and steam generator system.

My name is Hukam C. Garg. I am employed by the U.S. Nuclear Regulatory Commission as an Electrical Engineer in the Equipment Qualification Branch in the Office of Nuclear Reactor Regulation. My involvement in the CRBR review is that I am responsible for review of the environmental qualification of electrical equipment important to safety and safety-related mechanical equipment.

My name is Shou-nien Hou. I am employed by the U.S. Nuclear Regulatory Commission as Senior Mechanical Engineer in the Mechanical Engineering Branch in the Office of Nuclear Reactor

Regulation. My involvement in the CRBR review was in the mechanical engineering design aspects of piping integrity, including postulated break criteria and seismic design.

My name is Thomas L. King. I am employed by the U.S. Nuclear Regulatory Commission as Chief of the Technical Review Branch, Clinch River Breeder Reactor Program Office, in the Office of Nuclear Reactor Regulation. My involvement in the CRBR review is that I am responsible for direction of the Technical Review Branch's review of the fast sodium-cooled-related aspects of the CRBRP safety review.

My name is Dr. Bill Morris. I am employed by the U.S. Nuclear Regulatory Commission. I am Branch Chief, Electrical Engineering Branch in the Office of Reactor Research. During the construction permit review until March 1983, I supervised the review as Section Leader, Technical Review Section, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, and I participated extensively in the issuance of the Safety Evaluation Report for CRBR.

My name is Dr. Charles E. Rossi. I am employed by the U.S. Nuclear Regulatory Commission. I am a Section Leader in the Instrumentation and Control Systems Branch in the Office of Nuclear Reactor Regulation. As a Section Leader, I supervised the review of the CRBR instrumentation and control systems. I was also a member of an interoffice, interdisciplinary NRC Task Force established to determine the generic implications of the two events at Salem which resulted in the publication of NUREG-1000, Vol. 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," dated April 1983.

My name is Robert Schemel. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Human Factors Engineer in the Human Factors Engineering Branch of the Division of Human Factors Safety, Office of Nuclear Reactor Regulation. My involvement with the CRBR review has been to review the control room design with respect to human factors engineering.

My name is Dr. Jerry J. Swift. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, Clinch River Breeder Reactor Program Office, in the Office of Nuclear Reactor Regulation. My involvement with the CRBR review has been to coordinate the review of the radioactive source term analyses and the review of the Probabilistic Risk Assessment of CRBR.

My name is Ashok K. Agrawal. I am employed by the Brookhaven National Laboratory as a Nuclear Engineer in the Department of Nuclear Energy. The Brookhaven National Laboratory has provided technical assistance under contract with the U.S. Nuclear Regulatory Commission in connection with the licensing review of the CRBR. As part of that effort, I have been involved in the study of DBA initiators and related analyses.

My name is John E. Hanson. Presently, I am employed by Los Alamos National Laboratory. Prior to publication of the CRBR Safety Evaluation Report (SER), NUREG-0968, I was employed by the Idaho National Engineering Laboratory under contract with the U.S. Nuclear Regulatory Commission, assigned to the CRBR review in the areas of design basis accident delineation and evaluation.

My name is Edmund T. Rumble, III. I am employed as a Corporate Vice President of Science Applications, Inc. (SAI). Presently, I am providing consultant services to the Nuclear Regulatory Commission. My involvement with the CRBR review has been as a member of an SAI team providing technical assistance to the Office of Nuclear Reactor Regulation on safety matters related to the proposed CRBR.

Q2. Have you prepared statements of professional qualifications?

A2. (Panel) Yes. Copies are attached to this testimony.

I. INTRODUCTION AND OVERVIEW

Q3. What subject matter does this testimony address?

A3. (Panel) This testimony addresses whether a reliable basis exists for excluding the Core Disruptive Accident (CDA) as a Design Basis Accident (DBA) for CRBR.

Q4. What are Design Basis Accidents (DBAs)?

A4. (Becker) Design Basis Accidents are a set of events and associated conditions which are used to assess the way specific nuclear power plant systems respond to abnormal conditions. As such, Design Basis Accidents fall within the "Design Basis", which is defined in 10 CFR § 50.2(u), as "that information which identifies the specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design." The regulation further states that "these values may be (1) restraints derived from generally accepted 'state of the art' practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system or component must meet its functional goals."

Q5. Does the Staff require that DBAs be defined and evaluated at the CP stage?

A5. (Becker) Yes.

Q6. What is the purpose of requiring a definition and evaluation of DBAs?

A6. (Becker) While nuclear power plants, including CRBR, must be designed to minimize the occurrence of accidents, the plants must nonetheless be designed to cope with such accidents should they occur. Generically, nuclear power plants must be capable of performing three basic safety functions: (1) control or stop the nuclear fission reaction; (2) remove the generated heat; and (3) prevent the uncontrolled release of radioactivity.

Nuclear power plant safety designs, systems and structures are intended either directly or indirectly through the "defense in depth" concept to assure that these functions will be performed. As such, DBAs are selected and analyzed to determine if installed or proposed safety features can cope adequately with postulated accident events by performing the basic safety functions noted above.

The Staff requires that conservative margins be demonstrated by applicants in their analyses of the postulated events. This is accomplished by the use of conservative DBA analyses and assumptions, as well as the use of conservative acceptance criteria. Postulated events must be satisfactorily mitigated, i.e., the plant must meet all specific acceptance criteria, even if single failures are also postulated to have occurred in the safety systems under evaluation.

Q7. How are DBAs normally selected?

A7. (Becker) Design basis accidents for nuclear power plants are selected to represent a reasonable envelope of the credible events which might occur and which require mitigation by active systems or passive structures. The choice of the specific events typically depends on the type of reactor, with different sets of events being selected for BWRS, HTGRs, PWRs and LMFBRs. No regulatory criteria have been established for making these choices. Instead, engineering judgement regarding the kinds of faults or phenomena which might occur for a given type of nuclear reactor is employed. The selected events may range from those which may occur one or more times per year, to those events which may never occur during the life of the plant.

When the occurrence of an event has been judged to be so improbable that it is not "credible", the event is excluded from the design basis envelope. For example, accidents involving an initiating event and simultaneous multiple failures of the mitigating safety systems have been judged to be so improbable that they need not be included as design basis accidents for nuclear power reactors. Such accidents have instead been designated as "beyond the design basis accidents." Because such accidents typically involve some degradation of the reactor core, the term "core disruptive accidents" (CDAs) is also used by the Staff to describe such severe accidents.

Although probability is a consideration in distinguishing DBAs from accidents which are beyond the design basis, the Staff does not employ specific numerical probability thresholds. Rather, engineering judgement, based on such deterministic criteria as quality assurance, compliance with regulatory standards, redundancy, independence, and diversity is generally employed in determining that multiple failures of safety systems need not be postulated as part of the design basis for nuclear power plants. Applicable experience is also employed in selecting DBAs.

Q8. Please describe the means by which the DBAs for CRBR were initially selected?

A8. (Becker) The CRBR DBAs were initially selected and submitted to the Staff by the Applicants in Chapter 15 of the PSAR.

Q9. Does the Staff believe the DBA spectrum for CRBR selected by the Applicants is sufficiently comprehensive such that all credible accidents are enveloped?

A9. (Becker) Yes. As set forth in Section 15.1 of the CRBR SER, the Staff has determined that the DBA spectrum for CRBR is sufficiently comprehensive so as to envelope all credible accidents for CRBR.

Q10. What is the basis for the Staff's judgement as to the comprehensiveness of the CRBR DBA spectrum?

A10. (Becker) The Staff based its judgement upon an independent evaluation of the CRBR DBA spectrum, involving the following elements:

1) A thorough review of the systems proposed for the CRBR design to perform the necessary safety functions.
2) A thorough review of engineered safety features which mitigate the resulting accident should the primary system fail;
3) A thorough review of the DBAs proposed by the Applicants compared to the proposed CRBR design;

4) A comparison of CRBR DBAs to LWR DBAs;
5) A comparison of CRBR DBAs to DBAs of other domestic liquid metal-cooled fast reactors (LMFRs) that have been operated, are operating or are under design;
6) A comparison of CRBR DBAs to DBAs of foreign LMFBRs; and
7) An examination of available Failure Modes and Effects Analyses (FMEAs) and initiator studies.

The detailed bases which underlie the Staff's conclusions as to item 1 above is set forth throughout the CRBR SER, NUREG-0968. The detailed bases for the Staff's conclusions for items 2 and 3 are set forth in Sections 6 and 15 of the SER, respectively. In view of the comprehensiveness of the Staff's review and evaluation with regard to items 1-3 in the SER, this testimony does not further address these items.

The comparisons referred to in items 4-6 set forth above are contained in "Comparison of Clinch River Breeder Reactor Design Basis Accidents with Those for Light Water Reactors and Liquid-Metal Cooled Fast Reactors," EGG-NTAP-6152 (J. Hanson, Idaho National Engineering Laboratory) (January 1983) and in "Comparison of CRBR Design Basis Events with Those of Foreign LMFBR Plants," NUREG/CR-3240 (A. Agrawal, Brookhaven National Laboratory) (March 1983). The FMEA and initiator studies that were examined are listed in response to Question 53 below. A summary of the Staff's analyses and conclusions with regard to items 4-7 are contained in the CRBR SER and are further described in Part IV of this testimony.

A reference to the studies performed with respect to items 4-7 above is contained in SER Section 15.1.1.1 (p. 15-5). A summary of the Staff's analyses and conclusions with regard to items 4-7 is contained in Part IV of this testimony.

Q11. What are core disruptive accidents (CDAs)?

A11. (Becker, Morris) A core disruptive accident is an accident so severe that the reactor core or more specifically the fuel geometry is significantly modified over a substantial region of the core. Among the variations in the subsequent behavior are (1) successful in-core cooling of the disrupted core, (2) reactor vessel thermal failure because of inability to cool the disrupted core, and (3) mechanical reactor vessel failure because of power bursts from reactivity excursions or fuel coolant interactions which might occur as a result of fuel and coolant relocation. These variations have been analyzed and discussed in Appendix A to the "Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant" (NUREG-0968, March 1983) ("SER").

Q12. Under what conditions might CDAs occur?

A12. (Becker, Morris) A core disruptive accident could occur if either (1) a failure to remove heat from the fuel at a sufficient rate occurs so that fuel integrity is lost, or (2) a local failure in a fuel assembly propagates beyond that assembly to adjacent regions of the core.

Failure to remove sufficient heat from the fuel could occur if any, or a combination, of the following should occur:

(a) Failure to shut down the nuclear chain reaction when necessary during an over-power or a flow reduction transient,

(b) Failure to maintain sufficient primary coolant inventory to keep the fuel covered with coolant,

(c) Failure to maintain sufficient coolant flow to provide a heat removal path from the fuel, or

(d) Failure to extract sufficient heat from the coolant to maintain fuel integrity.

Propagation of local fuel faults to large regions of the core could occur if there is a failure to shut down the nuclear chain reaction prior to the point at which sufficient local damage has spread to other regions of the core.

Q13. Are CDAs a DBA for light water reactors (LWRs)?

A13. (Becker, Morris, King) No. CDAs are considered incredible in LWRs and are, therefore, outside the design basis.

Q14. On what basis are CDAs considered to be incredible for LWRs?

A14. (Becker, Morris, King) The evolution of LWR licensing has produced a body of design criteria and practices applied in the Staff's review of a nuclear power plant design which form the basis for the judgement that key safety functions can be performed with high reliability. Key aspects of these criteria and practices are the general reliability concepts of redundancy, diversity and independence as supplemented by testing and inspection. In addition, requirements for quality assurance at all levels of design, construction and operation provides confidence that structures, systems and components will perform satisfactorily. Fundamentally, these practices grew out of the Staff's use of engineering judgement informed by engineering assessment of the performance characteristics of the various systems and components in a nuclear power reactor and of the kinds of system or component failures that may occur. Application of these criteria and practices has become known as the deterministic approach and has become formalized by the requirements and guidelines embodied in 10 C.F.R. Part 50 and the "Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (SRP) (NUREG-0800, July 1981), and in the various Regulatory Guides and NUREG documents used by the Staff in the review process. An additional factor in this approach is the "defense in depth" concept, which recognizes the importance of accident prevention as well as mitigation and termination, and encourages the design approach of multiple protective systems and features. Also, the deterministic approach includes consideration of generic concerns identified as TMI Action Items and/or unresolved generic safety issues.

Probability-based methodologies such as risk assessment and reliability estimates are being assimilated into the licensing process and may be useful in some fashion in making decisions regarding the DBA spectrum. However, it is the Staff's judgement that these methodologies have not matured sufficiently at this time to be used as a decisive basis for determining whether CDAs should be excluded from the DBA spectrum. Presently, engineering judgment based upon the deterministic approach is considered to be the most tested, mature and comprehensive licensing methodology available for making such decisions.

It is the Staff's judgement that a nuclear power plant design which has undergone a thorough review using the deterministic approach and which has been found to be acceptable can exclude CDAs from the design basis as being improbable, because multiple failures of subsystems or systems of high reliability and high construction quality must occur to produce a CDA, and such multiple failures are not likely to occur.

Q15. Does the Staff consider CDAs to be beyond the design basis accident spectrum for CRBR and, if so, what is the basis for that judgement?

A15. (Becker, King, Morris) The Staff considers CDAs to be beyond the design basis spectrum for CRBR. We base that judgement on the utilization of the LWR deterministic methodology, appropriately modified to account for the salient differences between CRBR and LWRs. Although most commonly utilized for the licensing of LWRs, this approach is sufficiently universal and flexible that it is applicable to the licensing of all reactor concepts currently under consideration for commercial application, with appropriate modifications, including the CRBR.

One objective of the Staff's review of CRBR is to assure a level of safety comparable to that of LWRs. Application of the deterministic approach by the Staff in its construction permit review for CRBR is expected to result in the achievement of this comparable level of safety. In LWRs, as discussed above, CDAs are excluded from the design basis by application of the deterministic approach. Since the safety functions which must be performed to prevent CDAs are not fundamentally different in CRBR and LWRs, CDAs can be excluded as a DBA in both reactor types. In sum, it is the Staff's judgment that the application of this deterministic methodology, appropriately modified to account for the salient differences between LWRs and CRBR, provides a sufficient basis to exclude CDAs from the CRBR design basis as well.

Q16. How is the remainder of this testimony organized?

A16. (Becker, King, Morris) In the remainder of this testimony, the Staff summarizes (1) the significant differences between CRBR and LWRs and the resulting modifications to LWR deterministic criteria; (2) the safety functions which must be performed in CRBR to prevent CDAs, and the means by which the performance of these functions is accomplished; (3) the method by which the design basis accident spectrum for CRBR was verified; and (4) the role of the CRBR probabilistic risk assessment (PRA) and reliability assurance program.

II. MODIFICATION OF LWR DETERMINISTIC CRITERIA FOR APPLICATION TO THE CRBR

Q17. Please describe the significant differences between CRBR and LWRs?

A17. (King, Morris) The significant differences between CRBR and LWRs are associated with:

(1) the use of liquid sodium as a coolant in CRBR, which requires that the plant be designed to accommodate sodium leaks, that equipment be qualified for the sodium and sodium combustion product environment, and that a means to control the sodium inventory be established;

(2) the higher temperature operating conditions, which requires certain different design considerations and analyses;

(3) the fast neutron spectrum, higher power density, and higher enrichment of the core, which require added assurance of adequate shutdown and heat removal capability; and

(4) the limited experience with LMFBRs as compared to LWRs.

Q18. Have these differences necessitated a modification of the LWR deterministic criteria as applied to the review of CRBR?

A18. (Becker, King, Morris) Yes. In its safety review for CRBR, the Staff performed a comprehensive review of all safety criteria included in an LWR review, identifying instances where modifications were needed as a result of the differences between CRBR and LWRs. In many areas, no modification of the LWR safety criteria was found to be necessary. In other areas, these differences resulted in significant modifications to the LWR general design criteria. These modifications are identified in SER § 3.1, in the discussion concerning the CRBR Principal Design Criteria (PDCs). In view of the differences between the CRBR and LWRs, as discussed in response to Question 17 above, these modifications resulted in the following major requirements for CRBR which differ from LWR requirements:

(1) Two fast acting redundant, independent, diverse reactor shutdown systems (RSSs);

(2) Redundant, independent, diverse decay heat removal systems such that after an initiating event and a single failure, at least two heat removal paths remain available;

(3) A means to prevent or detect conditions which could lead to fuel failure propagations;

(4) Provision in the design for sufficient sodium coolant flow and maintenance of sodium coolant inventory;

(5) Features in the design to accommodate sodium leaks and sodium fires; and

(6) A commitment to implement a formal reliability assurance program.

A detailed description of modifications to the LWR criteria that have been made for the CRBR can be found in SER § 3.1, "Principal Design Criteria."

Q19. Please describe the function and requirements of the Principal Design Criteria.

A19. (King) The Principal Design Criteria represent the Staff's general design requirements with which CRBR must comply. More detailed criteria and guidelines are then used by the Staff and Applicants in implementing the PDCs, such as those contained in the Standard Review Plan and the various Regulatory Guides used by the Staff in its review process. The PDCs address requirements for those systems and features necessary to accommodate design basis accidents. In general, the PDCs require that the plant:

a) be designed to high standards of quality,

b) be designed for external events such as floods and earthquakes,

c) be designed to prevent and withstand fires,

d) be designed for equipment and systems to perform their functions in the DBA environment, and

e) include systems and features to prevent accidents, control the nuclear reaction, remove heat, monitor system parameters and performance, and contain radioactive material.

In some key areas, more specific requirements on system redundancy, diversity, independence, assumed failures, testing, and inspection are also included.

Q20. How were the CRBR Principal Design Criteria developed?

A20. (King) As stated in the Introduction to Appendix A to 10 C.F.R. Part 50, the General Design Criteria for LWRs are considered to be applicable to other types of nuclear power plants and are to be used as guidance in developing PDCs for a new plant. Accordingly, the PDCs were based on the general design criteria for LWRs, contained in 10 CFR Part 50, Appendix A. In the development of the PDCs for CRBR, the Staff considered the guidance in Appendix A to 10 CFR Part 50, in the following manner:

(a) Where there was no substantial difference between CRBR and LWRs, the Staff considered the LWR criteria to be applicable and adopted the appropriate criteria.

(b) For those LWR criteria considered generally applicable to CRBR, the Staff adopted, to the maximum extent practicable, the LWR criteria with appropriate modifications to adapt them to CRBR.

(c) On the basis of its review, the Staff identified and developed additional criteria for CRBR where there were significant differences between LWRs and the CRBR.

The criteria in Appendix A to 10 CFR Part 50 were used to the maximum extent possible. Wording changes were made only to adapt the criteria to CRBR terminology, for completeness, or to include additional requirements or conservatisms deemed appropriate because of the inherent differences between LWRs and CRBR or the more limited operating experience with LMFBRs compared to that of LWRs. Adhering as closely as possible to the wording and requirements of the LWR criteria is considered appropriate because (1) the Staff's goal has been to assure a level of safety equivalent to that of an LWR, and (2) the LWR criteria are implemented by existing NRC guides and technical positions whose effects could be impacted by changing the criteria.

Additionally, the PDCs of the FFTF and SEFOR reactors were reviewed, along with draft ANS Standard 54.1, "General Safety Design Criteria for an LMFBR Nuclear Power Plant" (July 1981), to determine their applicability. The results of these reviews were factored into the development of the CRBR criteria.

Q21. Do the CRBR Principal Design Criteria contribute to the Staff's conclusion that CDAs do not have to be included in the DBA spectrum?

A21. (King) Yes. The CRBR PDCs, like the deterministic LWR General Design Criteria upon which they are based, require sufficient redundancy, diversity and independence in safety systems so that failure to perform the basic safety functions which prevent CDAs is considered incredible. Since these same requirements (as modified to account for the differences between CRBR and LWRs) are applied to CRBR, the Staff has concluded that CDAs can be considered incredible at CRBR as at LWRs.

III. PERFORMANCE OF PRINCIPAL SAFETY FUNCTIONS

Q22. What are the fundamental safety functions, in LWRs and CRBR, that are necessary to prevent CDAs from occurring?

A22. (Becker, Morris) The safety functions which must be achieved to prevent CDAs are as follows:

(1) Shut down the nuclear chain reaction, when necessary,

(2) Maintain sufficient coolant flow,

(3) Remove sufficient heat from the fuel, and

(4) Maintain sufficient primary coolant inventory.

In addition, in LMFBRs, special attention is given to a further safety function:

(5) Avoid propagation of local fuel faults beyond an assembly.

Q23. How are these functions accomplished in CRBR?

A23. (Becker, Morris) In our review of CRBR, the Staff has determined that these safety functions can be accomplished if the design includes the features listed in Answer 18 above.

In addition to features to accomplish these fundamental safety functions, the deterministic methodology requires that attention be focused as well on ancillary functions. A concern in this regard for CRBR that is not present for LWRs is the ability of the plant to accommodate a sodium leak. Accordingly, the Staff believes that for the above features to function properly, the CRBR design must include measures to protect against damage to equipment, structures, and components from chemical reactions involving sodium (sodium fires, sodium-water reactions, sodium-concrete reactions).

A. Shutdown of the Nuclear Chain Reaction

Q24. Please describe the Staff's review of systems necessary to shut down the nuclear fission chain reaction?

A24. (Rossi) The CRBR RSS review was performed in accordance with Section 7.2 of the Standard Review Plan. The review criteria for LWRs were used as the basis for the review of the CRBR instrumentation and controls. The acceptance criteria used were the CRBR Principal Design Criteria and IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." The Staff's evaluation of the CRBR instrumentation and control systems, including the RSS, is discussed in Chapter 7 of the SER.

Q25. Please describe the CRBR systems which are necessary to shutdown the nuclear fission chain reaction?

A25. (Rossi) For shutting down the fission chain reaction in CRBR, there are two independent and diverse shutdown systems, referred to as the Primary and Secondary Reactor Shutdown Systems (RSS). Each system, by itself, is capable of terminating the design basis events without exceeding specified limits even if the most reactive control rod in that system cannot be inserted. One of these systems relies upon scram breakers to initiate shutdown; the other system relies upon a different component principle to initiate shutdown. The Staff's evaluation of these systems is documented in Chapter 7 of the CRBR SER.

The Primary RSS consists of three redundant and physically separate instrument channels for each measured parameter. The three channels are used in a two-out-of three coincidence logic to generate reactor trip signals. Three redundant logic trains are provided. The five scram breakers of the Primary RSS are arranged in a manner such that trip signals from two of three logic trains will open a sufficient number of the scram breakers to interrupt power to the Primary RSS control rods. Interruption of power to the control rods causes the rods to be inserted into the core.

The Secondary RSS uses types of equipment different from that in the Primary RSS and, in general, monitors a set of parameters diverse from those monitored by the Primary RSS (neutron flux, however, is measured by both RSSs). Neutron flux is measured by both the Primary and Secondary RSSs; it is sensed with compensated ionization chambers in the Primary RSS, and with fission chambers in the Secondary RSS. Three redundant and physically separate instrument channels are used to sense each measured parameter. Three redundant logic trains are used in the Secondary RSS such that two out of three trip demand signals will result in insertion of the Secondary RSS control rods. The Secondary RSS control rods are tripped by venting pneumatic pressure, which releases a latch on each control rod. The pneumatic pressure is vented by scram solenoid valves actuated by the Secondary RSS in a two-out-of-three configuration. Scram breakers are not utilized. As with the Primary RSS, the Secondary RSS control rods will scram on loss of power.

Since both Reactor Shutdown Systems consist of three redundant channels and three redundant sets of logic, each system, by itself, is capable of performing the safety function of shutting down the reactor even if a single failure has occurred within that system. The shutdown system designs include provisions such as the use of physical separation and isolation devices to insure that malfunctions in a channel or set of logic of one shutdown system cannot propagate to a redundant channel or set of logic of the same shutdown system, or to any channel or set of logic of the second shutdown system. All RSS equipment required to shutdown the reactor is designed to remain functional following either an operating basis earthquake or a safe shutdown earthquake.

Q26. With respect to design features to preclude scram failure, does the CRBR differ from reactor shutdown systems used on current LWRs?

A26. (Rossi) Yes. As discussed above, CRBR has two independent and diverse systems for detecting abnormal events and automatically initiating control rod insertion. Each system measures a variety of plant parameters to determine the need for automatic rod insertion. Current light water reactors have only one system (which measures a variety of plant parameters) for initiating automatic rod insertion following an abnormal event.

Q27. How does the availability of two independent and diverse reactor shutdown systems at CRBR impact upon the likelihood of occurrence of an anticipated transient without scram (ATWS) which could lead to a CDA?

A27. (Rossi) The Staff has concluded that the probability of a CDA initiated by an ATWS in CRBR is sufficiently low and may be considered to be incredible. The events occurring at Salem 1 in February 1983 (involving failure of two reactor trip circuit breakers to open automatically upon receipt of a valid trip demand signal) have not changed this conclusion, because the Salem plant (like other LWRs) has only one fast acting RSS. Further, generic actions are being developed with respect to LWRs, to address the circumstances leading to the specific type of failures at Salem, as well as to improve the reliability of reactor shutdown systems in general. If appropriate, these generic RSS actions will be applied to the CRBR reactor shutdown systems to reduce the ATWS probability even further.

B. Maintenance of Sufficient Coolant Flow, Heat Removal and Coolant Inventory

Q28. Please describe the Staff's review of the features needed to maintain sufficient coolant flow, heat removal and coolant inventory?

A28. (Becker, King, Hou) The Staff's review was performed in accordance with the Standard Review Plan (Chapter 4, Reactor; Chapter 5, Reactor Coolant Systems and Connected Systems; and Chapter 7, Instrumentation and Control), modified as necessary to account for the unique aspects of CRBR. Generic external hazards such as seismic events which could affect the heat removal systems were also reviewed. The results of the Staff's review are set forth in Sections 4, 5, 7 and 15 of the SER.

Q29. What features of CRBR provide for sufficient coolant flow, heat removal from the fuel and coolant inventory?

A29. (King, Becker) The features of CRBR which are proposed to provide for sufficient coolant flow, heat removal from the fuel and coolant inventory are (1) the CRBR heat removal systems (including their associated features to prevent flow blockages and gas bubbles in the core); (2) guard vessels for the reactor vessel, primary pump and IHX, along with elevated piping; and (3) systems to prevent, detect and accommodate sodium leaks.

Q30. Please describe the CRBR heat removal systems and their role in the maintenance of coolant flow, heat removal and coolant inventory?

A30. (Becker) The CRBR heat removal systems consist of the main heat transport system (HTS) and the Shutdown Heat Removal System (SHRS). The main HTS is comprised of three identical heat transport loops used to carry heat from the reactor core through a primary loop, isolated from an intermediate loop by means of an intermediate heat exchanger. The heat transported by the secondary loop generates steam in two identical evaporator modules. The generated steam is passed through and superheated in a third module. The superheated steam then passes to a turbine generator to generate electricity, and the waste heat is rejected to the atmosphere. A pump in each primary and intermediate loop provides motive power to circulate the coolant. These three main HTS loops are designed to remove the full power generation of the core.

The SHRS consists of subsystems utilized for removing heat after the reactor has been shutdown (decay heat). The SHRS consists of the three main HTS loops, plus a diverse heat removal system called the Direct Heat Removal Service (DHRS). Decay heat is normally removed through the main Heat Transport System (HTS), steam, condenser, and feedwater systems. Each HTS loop is also provided with a safety grade backup decay heat removal system called the Steam Generator Auxiliary Heat Removal System (SGAHRS). The SGAHRS utilizes steam vent valves, and a steam-to-air heat exchanger to dump heat to the atmosphere. Feedwater is supplied by an auxiliary feedwater system similar to that utilized in LWRs. No offsite or onsite power (other than batteries) is required for decay heat removal through the SGAHRS. This is accomplished via natural convection in the sodium loops, and via steam venting and the steam-to-air heat exhanger in the SGAHRS. Further information on the Staff's position concerning the ability of CRBR to remove decay heat via natural convection is provided in response to Board Question 4.

If for any reason, all three HTS loops are lost or unusable beyond the Intermediate Heat Exchanger, operation of the Direct Heat

Removal Service (DHRS) can be initiated, utilizing the reactor overflow path through a heat exchanger to reject the decay heat through the air coolers used to cool the ex-vessel storage tank (EVST). The DHRS requires AC power (either from offsite or onsite sources) and can accomplish its function even with a single failure of any active component.

When the reactor is operating at power, the main HTS provides sufficient heat removal capacity. The RSS is tripped and the reactor is shutdown if the parameters of normal operation indicate that the ability to remove heat is being impaired. The HTS and reactor vessel also contain features to prevent local flow blockages and gas bubbles from passing through the core. These are described in Section 4.4 of the SER.

Q31. What roles do the guard vessels and elevated piping perform in providing for sufficient coolant flow, heat removal and coolant inventory?

A31. (Becker, King) Although CRBR does not operate above the sodium saturation temperature, a small leak (if left unattended) has the potential of draining large quantities of sodium from a system. The proposed CRBR design acknowledges this possibility and has incorporated guard vessels around the reactor vessel, primary pump and IHX to maintain an adequate inventory. The guard vessels accomplish this function by being sized and designed such that a leak occurring within the guard vessel envelope will fill the annulus between the guard vessel and the component so that sufficient sodium remains in the reactor vessel to cover the core and exit nozzles, provided the primary pumps are reduced to pony motor speed. This allows any loops without a leak to continue to remove the decay heat. In addition, the piping between guard vessels is elevated so that the possibility of syphoning is avoided. To ensure that the primary pumps are reduced to pony motor speed, procedures will require the plant to be shutdown upon indication of a sodium leak. In addition, the Primary Reactor Shutdown System has an automatic trip on low reactor vessel level to provide automatic protection (i.e., reactor shutdown and pump trip) against a large leak. Sodium inventory in the intermediate loops is maintained by physical separation and independence of the three intermediate loops, so that a leak in one loop is not able to affect the other two loops.

Q32. How is the prevention, detection and accommodation of a sodium leak accomplished?

A32. (Becker, Hou) CRBR is to be designed with a sensitive sodium leak detection system for the primary and intermediate HTS loops. This system is to be redundant and diverse and is to be capable of detecting leaks down to about 100 grams/hr of sodium. The piping system is to be constructed of stainless steel in accordance with ASME Boiler and Pressure Vessel Code Section VIII, to accommodate normal and off-normal loads and to provide high quality control. The low pressure-high temperature operating conditions of the sodium cooling system require a design utilizing thin-walled piping. The seismic response of the thin-walled sodium system is quite different from the heavy, thick-walled high pressure LWR cooling systems. In addition, high temperature material properties must be considered for both short term strength and long term phenomena such as thermal aging and creep. Current design rules have taken these effects into consideration and applicable design margins for the sodium systems under seismic loads were analytically demonstrated. The Staff's concerns related to the adequacy of current high temperature design rules have been identified and resolved by a combination of additional analytical effort and further confirmatory materials testing. Section 3.9.9 of the SER provides the Staff's evaluation in this area. Further information is to be presented in a Staff Exhibit in this proceeding.

The HTS sodium is not expected to leak, but even if an undetected flaw were present and a leak occurred, the material properties of the stainless steel are such that a small leak would develop in lieu of a sudden large leak. In addition, Staff calculations have shown that large leaks in the primary system, which are several times larger than the Design Basis Leak (DBL), may be sustained during operation before core cooling is impaired to the point of violation of the DBA acceptance criteria of no sodium boiling in the core, as discussed in SER § 15.3.2. Leaks larger than the DBL are considered highly improbable based on (a) attention to quality control at all stages of design and construction; (b) comprehensive piping design for anticipated loadings such as seismic, thermal transients and other factors; (c) the expected presence of detectable small leaks prior to large leaks; and (d) a sensitive leak detection system.

The ways in which CRBR accommodates the chemical reactions resulting from a sodium leak are discussed in Section III.D of this testimony.

Q33. What has the Staff concluded with respect to the CRBR heat removal systems?

A33. (Panel) The Staff has concluded, based on its review of the CRBR heat removal systems, that the proposed design provides features sufficient to prevent flow blockage, and that there is sufficient redundancy, diversity and independence in the systems to ensure adequate coolant flow and heat removal from the fuel. In addition, these systems have been sufficiently protected from loss of sodium inventory such that the probability of a CDA resulting from an inability to cool the core is sufficiently low. Accordingly, the Staff has determined that CDAs resulting from a failure to maintain sufficient coolant flow, heat removal and/or coolant inventory may be excluded from the CRBR design basis accident spectrum.

C. Avoiding Propagation of Local Fuel Faults

Q34. Please describe the Staff's requirements related to the propagation of local fuel failures.

A34. (King) The Staff's concern with regard to local in-core fuel or blanket failures at CRBR is that the design must be adequate to prevent or detect such failures prior to their propagating beyond the initating assembly. Accordingly, the CRBR Principal Design Criteria include requirements on the prevention and detection of fuel failure progagation events.

Q35. Are the effects of local in-core failures and the potential for their propagation to other parts of the core considered in the CRBR safety analysis?

A35. (King) Yes. Such events have been considered by both the Applicants (in PSAR Section 15.4) and the Staff (in SER Section 15.4).

Q36. Which local in-core failures were considered?

A36. (King) The Applicants and Staff considered in-core failures of fuel, blanket and control pins due to: (a) random or stochastic clad failure; (b) local overpower; (c) local flow blockage; and (d) gas bubble passage through the core.

Q37. Please describe the Staff's review of local fuel failures?

A37. (King) The Staff, assisted by Los Alamos National Laboratory, conducted a review of the Applicants' analyses, presented and referenced in PSAR Section 15.4. In addition, an independent review was conducted of the available literature concerning failure propagation and detection; the results of a recent in-reactor test designed specifically to investigate failure propagation was also reviewed.

Q38. What has the Staff concluded with respect to the capability of CRBR to avoid propagation of local fuel faults?

A38. (King) The Staff has concluded that the potential for failure propagation in CRBR is very low and that even if initiated, means will be provided to detect the propagation in sufficient time to shutdown the reactor prior to significant propagation. These conclusions are based upon the following considerations:

a) Failure propagation has not been observed in any operating LMFBR for which data are available (data from communist countries are limited and, therefore, no conclusion can be drawn with regard to failure propagation in their LMFBRs).

b) CRBR is to contain features to minimize the potential for local flow blockage, overpower assemblies and gas bubbles in the core.

c) The results of extensive analysis and testing performed to investigate various types of local failures and their potential for propagation indicate that only certain conditions of local heat-generating blockages (i.e., fuel expulsion from one pin causes local blockages around its neighbors) cause failure propagation, and that the speed at which this propagation occurs is slow enough to allow detection and reactor shutdown prior to significant propagation.

d) The proposed CRBR design includes a delayed neutron monitoring system capable of rapidly detecting any local failure events which could result in failure propagation.

e) Until such time as operation with failed fuel is well understood, restrictions (as discussed in SER Section 4.2.1.3.2.6) will be placed on CRBR operation with failed fuel. These restrictions will minimize the potential for failure propagation due to unanticipated failed fuel behavior.

Therefore, based upon the above, the Staff considers that CDAs initiated by fuel failure propagation are of sufficiently low likelihood that they may be excluded from the CRBR design basis accident spectrum.

D. Accommodation of Sodium Leaks and Sodium Fires

Q39. Please describe the differences between CRBR and LWRs with respect to the environmental qualification of equipment.

A39. (Garg) The significant difference between CRBR and LWRs in the area of environmental qualification of equipment is that the equipment in CRBR may be exposed to sodium aerosols and combustion products, whereas equipment in the LWR environment may become exposed to chemical or demineralized water spray.

Q40. Have the Staff's concerns with respect to this matter been resolved to the Staff's satisfaction?

A40. (Garg) Yes. As discussed in Section 3.11 of the SER, the Applicants have committed to qualify all Class 1E electrical equipment for the sodium environment to which they can be exposed during any DBA. Mechanical equipment will be qualified for anticipated environments or will be protected by enclosures. Based on the fact that it is feasible either to qualify or to protect the equipment, the Staff finds that the environmental qualification criteria can be satisfied for the CRBR.

Q41. Is it important that the CRBR have the capability to accommodate sodium leaks which could result in sodium fires?

A41. (Becker, King) Yes. One of the primary differences between CRBR and LWRs is that in CRBR, the liquid used to cool the reactor and transport heat reacts chemically with both air and water. When exposed to even small amounts of oxygen or water, sodium will combine readily with the oxygen (burn) and give up heat. When exposed to concrete, the sodium reacts readily with the water in the concrete, releasing hydrogen and energy.

Q42. Has the Staff determined that the CRBR will be able to accommodate sodium leaks which could result in sodium fires?

A42. (King) Yes. It has been common practice in LMFBRs to surround the radioactive primary heat transport system with steel-lined cells which can be inerted; this design approach has been continued in CRBR. Each primary loop has its own steel-lined cell which will be purged with nitrogen until the oxygen content is about 2%. This reduces the impact of a leak and any subsequent fire. The steel liners in the cell remove the possibility of sodium-concrete reactions which could cause hydrogen generation and possible overpressure, and failure of the cells. The Staff's evaluation of sodium fires appears in SER §§ 9.13.2 and 15.6.2. The cell liners are considered to be Engineered Safety Features, as noted in SER Chapter 6; the cell liner evaluations appear in SER §§ 3.8.3 and 9.13.

Sodium fires outside the primary system, where the sodium is not radioactive, are controlled with catch pans and fire suppression decks. These are shallow steel basins at the bottom of air-filled concrete cells which allow the sodium to drain away while limiting the surface area that is exposed to oxygen, thus permitting the bulk of the sodium to cool and solidify.

CRBR has been designed so that the potential for sodium-water reactions realistically exist only at the steam generators (SGs). Relief systems are to be provided on the SGs to dump reaction products from a sodium-water reaction to inerted dump tanks so as to prevent the sodium-water reaction from damaging the primary system. In addition, the water is to be rapidly dumped from the affected SG to terminate the reaction.

Finally, the design is to be reviewed with respect to the environmental qualification of safety-related instrumentation and equipment, to limit systems interaction common mode failures and to ensure the ability to monitor operations during accident-generated environments.

IV. VERIFICATION OF THE CRBR DBA SPECTRUM

Q43. Has the Staff verified the comprehensiveness of the CRBR DBA spectrum?

A43. (Becker) Yes. The ways in which the Staff verified the comprehensiveness of the CRBR DBA spectrum are summarized in response to Question 10 above.

Q44. Has the Staff considered the DBAs of LWRs, LMFRs and other LMFBRs in assessing the comprehensiveness of the CRBR DBA spectrum?

A44. (Hanson, Agrawal) Yes. The objective of the comparisons of DBAs is to utilize the disciplined thought processes of other plant designers and engineers to discover DBAs which might have been overlooked for CRBR. With these comparisons, it is possible to be confident that the CRBR DBAs comprise a sufficiently complete set. DBAs do tend to be design specific and the importance of types of DBAs may vary between plants of different design. The general safety functions which must be performed, the systems required to perform these functions, and the required engineered safety features are similar for different types of plants. Accordingly, each event must be examined to determine if it represents a DBA applicable to CRBR or simply an accident unique to the plant (or system) for which it was considered.

Q45. Are there additional reasons why CRBR DBAs have been compared with LWR DBAs?

A45. (Hanson) The Staff has stated that the review goal for CRBR is to ensure a level of safety comparable to that for a LWR. Also, there are generally similar functions that must be performed in both reactor systems. It is, therefore, appropriate that the CRBR DBAs be compared with those of LWRs, as referenced in the Standard Review Plan, NUREG-0800.

Q46. Did your review of LWR DBAs result in the identification of any DBAs which should be but were not included in the DBA spectrum for CRBR?

A46. (Hanson) No. This review and comparison provides additional assurance that the CRBR DBA spectrum is complete.

Q47. For which domestic liquid metal-cooled fast reactors were DBAs compared with the CRBR DBAs?

A47. (Hanson) The review of domestic LMFRs covered FERMI-1, SEFOR, EBR-II, FFTF and the proposed Large Demonstration Plant (LDP).

Q48. Did the Staff's review of DBAs for domestic liquid metal-cooled reactors result in the identification of any DBAs which should be but were not included in the DBA spectrum for CRBR?

A48. (Hanson) No. This review and comparison provides additional assurance that the CRBR DBA sprectrum is complete.

Q49. For which foreign LMFBRs were DBAs compared with the CRBR DBAs?

A49. (Agrawal) The review of foreign LMFBRs was based on LMFBR plants having power levels comparable to or higher than that of the CRBR, which are either currently operating or are in an advanced stage of design or construction. The plants reviewed are the PHENIX and SUPER PHENIX plants in France, the SNR-300 in the Federal Republic of Germany, the MONJU reactor in Japan, and the PFR in the United Kingdom. Not included in my study are a number of smaller plants and plants for which data were not available for review.

Q50. Please summarize your findings on the comparison of CRBR DBAs and the DBAs of foreign LMFBRs?

A50. (Agrawal) There are two key findings. First, the methodology used in the selection of the design basis accidents for foreign LMFBR plants is similar to that used by the Applicants in CRBR. Secondly, the list of DBA events considered by the Applicants is comparable to those considered in the foreign LMFBR plants.

Q51. Did your review of foreign LMFBR DBAs result in the identification of any DBAs which should be but were not included in the DBA spectrum for CRBR?

A51. (Agrawal, Becker) No. However, the Staff has determined that further analysis should be provided in the Applicants' FSAR concerning the potential for freezing of sodium in the steam generator, in the event that a sufficient quantity of very cold water were allowed to pass through the steam generator. In the CRBR, if such an event were to occur, protection may be provided by a safety function dependent upon a primary-to-intermediate sodium flow mismatch trip. Further, the direct heat removal service (DHRS) would be available to remove decay heat if all three heat transport system (HTS) loops were disabled. Notwithstanding these considerations, in the event that sodium freezing is determined to be a concern upon the conclusion of Applicants' analysis, a plant protection system (PPS) trip could be added during the OL review stage to resolve this matter.

Q52. What conclusions do you draw from your comparison of foreign LMFBR DBAs with those of CRBR?

A52. (Agrawal) The comparison of DBAs for CRBR and for foreign LMFBRs provides additional assurance that the CRBR DBA spectrum is complete.

Q53. Which FMEAs and accident initiator studies were examined by the Staff?

A53. (Becker, Rumble) The FMEAs considered by the Staff were those in Supplement 1 to Appendix C of the CRBR PSAR. The initiator studies considered by the Staff were those contained in "CRBRP Safety Study" (CRBRP, March 1977) ("CRBRP-1"); "Estimated Recurrence Frequencies for Initiating Accident Categories Associated with the Clinch River Breeder Reactor Plant Design," NUREG/CR-2681 (E. Copus, Sandia National Laboratory) (April 1982); "LMFBR Accident Delineation Study, Phase I Final Report," NUREG/CR-1507 (D. Williams, Sandia National Laboratory) (November 1980); and "Risk Reduction Feasibility Study of Selected Modifications to CRBRP Safety

Systems," SAI-83-959-WA (B. Atefi and R. Liner, SAI) (September 1982).

Q54. Were any additional accident initiators or accidents found in examining the available FMEA and accident initiator studies?

A54. (Becker, Rumble) No additional accident initiators or accidents were found from the examination of available FMEA and accident initiator studies. The examination of these sources provides additional assurance that the CRBR DBA sprectrum is complete.

Q55. Has the Staff's review of the selection of DBAs for CRBR taken into consideration the role of human error?

A55. (King, Morris, Becker, Schemel) Yes. The Staff's review at the CP stage concentrates on ensuring that the design criteria applied to the plant include requirements directed toward minimizing the potential for human error. These requirements are embodied in the deterministic criteria applied to CRBR (Principal Design Criteria, Standard Review Plan (NUREG-0800), and the other Regulatory Guides and NUREG documents applied in the CP review), and minimize the potential for human error by specifying requirements as to (a) redundancy, diversity and independence of safety systems (see Chapter 3.1 of the SER); (b) application of human factors engineering principles (see Chapter 18 of the SER); and (c) application of requirements resulting from the TMI accident listed in 10 C.F.R. § 50.34(f), many of which are directed toward reducing the potential for human error.

The plant design at the CP stage is only reviewed with respect to its potential for meeting the criteria; actual design approval and a more detailed review of additional considerations with respect to human factors are scheduled at the operating license (OL) review stage.

In addition to specifying requirements as to safety system redundancy, diversity and independence, the deterministic criteria listed above specify: (1) when single failures must be assumed, (2) what initial conditions must be assumed in the safety analysis, and (3) where the design must include human factors requirements.

The requirements as to redundancy, diversity, independence, single failures and initial conditions, in effect, require that the plant be designed for failures (whether human failure or equipment failure). For example, multiple human errors would be required to reach the initial condition assumptions of certain DBAs because the conditions imply multiple violations of procedures. Also, the failures assumed during DBAs could be caused by either human error or hardware failure, and the response of the plant is not affected by whether the source of the failure is human or hardware. Therefore, in effect, human error is considered in the selection and course of DBAs.

Q56. Would operation of CRBR present any more difficult challenge to the operator or present any more opportunity for human error than is presented by operation of an LWR?

A56. (King, Rumble, Morris, Schemel) CRBR is comparable to a modern LWR with respect to the general ways in which human errors may potentially occur, and the steps used in LWRs to minimize human errors are directly applicable to CRBR. The CRBR is not considered to present any more difficult challenge or opportunity for error than is presented by operation of an LWR. Some features of CRBR, such as the two independent, diverse, redundant shutdown systems, provide additional margin to accommodate human error beyond that which may be available in LWRs.

Q57. What does the Staff conclude about the comprehensiveness of the potential accident initiators and design basis accidents considered for CRBR?

A57. (Becker) In defining and analyzing the design basis accident for CRBR, the potential initiators have been either described and studied or generically bounded and studied. Based upon (1) a careful evaluation of the CRBR design basis accident spectrum, (2) a comparison of CRBR DBAs with the DBAs of LWRs, domestic LMFR and LMFBRs, and foreign LMFBRs, and (3) an examination of available FMEAs and accident initiator studies, the Staff has concluded that the CRBR DBA spectrum is complete, and that the entire spectrum of credible accident initiators has been enveloped.

V. THE ROLE OF THE CRBR PROBABILISTIC RISK ASSESSMENT AND RELIABILITY ASSURANCE PROGRAM

A. The Probabilistic Risk Assessment (PRA)

Q58. Has the Staff performed any analyses at the CP stage to gain perspective of CRBR reliability and risk?

A58. (Rumble) The Staff and its contractors (NUS Corporation, Brookhaven National Laboratory, Sandia National Laboratories, UCLA, and Science Applications, Inc.) have performed numerous analyses to investigate the reliability of CRBR systems and to delineate important accidents which, if led to conclusion, could potentially cause core degradation and releases to the environment. The main purpose of these analyses was to study the potential for system weaknesses which could degrade the inherent diversity, redundancy and independence in the CRBR design. Consideration of this reliability-oriented information has led to enhancement of safety system designs; the most recent example of this is the upgrading of the Direct Heat Removal Service (DHRS) to conform to the single failure criterion. A list of the NRC-sponsored reliability oriented studies for CRBR is attached to this testimony (DBA Testimony Attachment 1).

Q59. What is a Probabilistic Risk Assessment?

A59. (Rumble) Probabilistic Risk Assessment (PRA) in the context discussed here is a methodology for estimating the public health risk due to the operation of a particular nuclear power plant. In order to perform this estimate, accident sequences are delineated and quantified to obtain their likelihood of occurrence and public health consequences. By systematically and comprehensively identifying accident sequences which contribute significantly to the total risk, a risk envelope can be established which bounds the public risks of plant operation. The quantification of accident sequence likelihoods involves the estimation of accident initiator frequencies and the unavailability of possible mitigating systems, functions and features. The quantification of consequences entails the analyses of physical processes during accident progression, of containment system response, of radionuclide release and transport, and of environmental consequences.

Q60. Are PRAs used to identify Design Basis Accidents for CRBR or LWRs?

A60. (King) No. The regulatory framework has not developed a means of using PRAs to establish the DBA spectrum. However, information generated by PRAs can be useful in gaining insights into the comprehensiveness of the DBA spectrum.

Q61. Is a PRA for CRBR relied upon by the Staff to demonstrate that core disruptive accidents ("CDAs") need not be included in the DBA spectrum?

A61. (King) No. CDAs are excluded from the CRBR DBA spectrum by application of deterministic methodology, as for light water reactors. This involves adherence to the general principles of redundancy, diversity and independence, use of the Standard Review Plan, conformance to regulations, regulatory guides and branch technical positions, and application of adequate Principal Design Criteria. Further confidence in the reliability and safety obtained through this methodology is derived from the formal Reliability Assurance Program which the Staff has required CRBR to have. The Probabilistic Risk Assessment of CRBR will provide a further review of plant design and operations for potential weaknesses, but is not required to support the determination that CDAs may be excluded from the DBA spectrum.

Q62. Is there any regulatory requirement that a PRA be performed?

A62. (Swift) 10 CFR § 50.34(f)(1)(i) states:

> (1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license.
>
> (i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8) (Footnote omitted).

In this regard, it should be noted that the requirements of 10 CFR § 50.34(f)(1) are applicable only to specific LWRs and a manufacturing facility. Nonetheless, the Staff has chosen to apply the requirement for a PRA to CRBR, to provide additional assurance that the risks from CRBR operation will be equivalent to LWRs.

Q63. Is there any requirement that a PRA be completed before a construction permit can be issued?

A63. (Swift) No. Completion of a PRA before a CP is issued is not required by regulation, nor is it the Staff's practice to so require.

Q64. Has a PRA for CRBR been completed?

A64. (Swift) No. The Applicants commenced work on the PRA in or about June 1981, and the effort is continuing.

Q65. Is it anticipated that a PRA for CRBR will be completed prior to the issuance of a construction permit?

A65. (Swift) No. The Applicants have committed in Appendix J of the PSAR to a schedule showing delivery of the completed PRA for CRBR in December 1984; based upon this schedule, the Applicants are expected to satisfy the requirement of 10 CFR § 50.34(f)(1)(i) that a PRA be completed within two years of issuance of the construction permit.

Q66. Is the CRBR PRA to be comparable in comprehensiveness to the "Reactor Safety Study" (WASH-1400)?

A66. (Rumble) Yes. The Applicants have, in Appendix J of the PSAR, committed to a PRA comparable to the Reactor Safety Study.

Q67. What is the primary objective of the CRBR PRA?

A67. (Rumble, Swift) The primary objective of the CRBR PRA is to determine the relative importance of individual systems and components to plant reliability and safety, so as to identify potential weaknesses. It can also aid in identifying specific preventive and mitigative actions to reduce risks.

Q68. What is the Staff's role in the CRBR PRA?

A68. (Swift) As described in the SER, the Staff is conducting an ongoing review of the Applicants' PRA effort. The review effort is being conducted with assistance from Science Applications, Inc. Activities of the review effort include detailed review of specific major elements of the study, integrated review of the overall PRA, and continued monitoring by review of the PRA products and by participation in interactive meetings with the Applicants. The Applicants have committed to interactive meetings to convey early information on methodology and interim results to facilitate the Staff's review.

Q69. Does the Applicants' PRA appear to be of sufficient scope, depth, and quality to meet the NRC requirements?

A69. Yes. The work completed to date and the plans to which the Applicants have committed for the remaining effort indicate that the final products will be of satisfactory scope, depth and quality to meet NRC requirements.

This judgement is based upon the following considerations. The Staff has been reviewing the plans and products of the PRA since the summer of 1982. In addition, several meetings with the Applicants and their contractors have been held to discuss the PRA activities. The IEEE/ANS PRA Procedures Guide (NUREG/CR-2300) and the Interim Reliability Evaluation Program Procedures Guide (NUREG/CR-2728), as well as recently completed and ongoing PRAs, provide a basis for defining the specific methods to be used in the Applicants' PRA. The work which has been completed, the Applicants' plans for the remainder of the PRA effort, and the experience level of their PRA personnel indicate that the desired elements will be performed in sufficient depth and quality to provide a satisfactory product. As the work proceeds, the Staff will continue to review interactively the Applicants' progress to help assure that the final products will be satisfactorily completed.

Q70. Could the CRBR PRA identify other CRBR accident possibilities of greater frequency or consequence than the accident scenarios analyzed by Applicants and Staff?

A70. (Rumble, Swift) Yes, it could, although whether it will cannot be known at this time. As indicated in response to Question 67 above, the primary objective of the PRA is to determine the relative importance of individual systems and components to plant reliability and safety so as to identify potential weaknesses. Whatever unanticipated weaknesses the PRA uncovers will be evaluated, and if the Staff judges it necessary, the Staff will require modifications to upgrade or install appropriate safety systems, rather than consider CDAs to be Design Basis Accidents. Based upon experience with LWR PRAs, these improvements are not expected to be of such a magnitude as to cause a total redesign of the safety systems, since the deterministic criteria applied to the design are expected to result in highly reliable systems.

Q71. Is completion of the CRBR PRA on a schedule such that it can realistically provide timely feedback for risk reduction?

A71. (King) Yes. The schedule, milestones and resource allocation for the CRBR PRA are set forth in Section 4 of Appendix J of the PSAR. The scheduling reflected in the PSAR permits the PRA to influence system designs as they are developed, in conformance with 10 CFR § 50.34(f)(1)(i). This scheduling will also permit the PRA to provide input to operating procedures early in their development.

Q72. Can the PRA influence those aspects of CRBR design that the Applicants consider to be already completed?

A72. (King) At the construction permit (CP) stage, a set of criteria are developed which the Applicants must commit to. The Staff's review at the CP stage only determines that the proposed design appears to have the potential to meet the criteria necessary for safety. At this stage, the Applicants proceed with construction of their designs at their own risk.

While some portions of the CRBR design have been completed by Applicants, not all of the CRBR design is completed. Designs for long-lead-time components must be completed early, but are still subject to change. Even the major components that are already fabricated can be changed if that is found to be necessary; an example is the possible re-machining of major parts of the reactor vessel head which might be required if that is determined to be necessary for the head to achieve its required potential to accommodate a highly energetic CDA. Furthermore, it is a requirement that the results of the PRA be factored into the design. Thus, the fact that the Applicants may consider some portions of the design to be final will not preclude the Staff from requiring at the OL stage that design modifications be made, if such changes are determined by the Staff to be necessary.

B. Role of the CRBR Reliability Assurance Program

Q73. Please describe the CRBR Reliability Assurance Program?

A73. (King) The CRBR Reliability Assurance Program (henceforth called the Program) is a program to be performed by the Applicants throughout the life of CRBR which will qualitatively and quantitatively assess the reliability of those CRBR systems and features which prevent CDAs and limit releases within 10 CFR Part 100 guidelines. The reliability information will be utilized by Applicants to enhance the design and operation of CRBR. Elements of the Program are currently in place and underway. The full Program will be in place subsequent to the issuance of the construction permit.

Q74. Are there any existing NRC regulations or Staff guidance that require a reliability program similar to the CRBR Reliability Assurance Program?

A74. (King) No.

Q75. Has the Staff determined it is important to implement such a program for CRBR?

A75. (King) Yes. Due to the limited operating experience with LMFBRs as compared to LWRs, the Staff concluded that such a program would provide an additional conservatism to account for this factor. Accordingly, the Staff has required the Applicants to implement this type of reliability program.

Q76. How was the CRBR Reliability Assurance Program developed?

A76. (King) The criteria for the Program were developed by the Staff. These are given in Appendix C of the CRBR SER, and summarized in response to Questions 87, 88 and 92. The Applicants have committed to comply with these criteria (see letter J. R. Longenecker (DOE) to J. N. Grace (NRC), dated March 2, 1983). The details of the Program are to be developed and documented by the Applicants subsequent to the issuance of the CP, with periodic reviews and audits by the Staff during the operating license review.

Q77. What is the overall objective of the Program?

A77. (King) In general terms, the activities under the Program are to be performed to help ensure that the risk to the public from CRBR is comparable to that from a current LWR. The overall objective of the Program is to evaluate and enhance the safety-related reliability inherent in the application of 10 CFR Part 50, the CRBR Principal Design Criteria and the other standards and guides applied to the design. This evaluation and enhancement will provide further assurance that the CRBR design is capable of providing for accident prevention, termination, and mitigation so that the likelihood of a core disruptive accident or of exceeding 10 CFR Part 100 guidelines is acceptably low.

Q78. Is the CRBR Reliability Assurance Program relied upon by the Staff to demonstrate that CDAs need not be included as Design Basis Accidents?

A78. (King) The Staff has based its decision regarding the exclusion of CDAs from the DBA spectrum on the feasibility of achieving high system reliability by implementing the deterministic criteria contained in 10 CFR Part 50, in the Standard Review Plan (NUREG-0800), and in the Staff's Regulatory Guides and other NUREG documents which the Staff utilized in its licensing review of CRBR. As stated in response to Question 75 above, the Reliability Assurance Program is required in order to account for the limited operating experience with LMFBRs as compared to LWRs. It is the Staff's judgment that the commitment by the Applicants to perform such a program in accordance with the Staff's criteria (as seth forth in Appendix C of the SER) will compensate for this difference in experience. Reliance upon this commitment at this stage of consequences. Equipment testing is used in a developmental program to verify design, to explore failure modes, equipment performance, and extended limits of operation in a qualitative reliability sense. Equipment qualification is a standard requirement for nuclear reactors to ensure performance under required environmental conditions. Failure evaluation serves to ensure appropriate design and operational feedback and corrective action. Section C.2.1 of the SER (NUREG-0968) provides a more detailed description of each of the above activities.

Q82. Can fault tree or event tree analyses be used in the Program?

A82. (Rumble) Yes, both fault tree and event tree analyses can be used in the Program.

Q83. Is the use of fault tree and event tree analysis an accepted analytic method?

A83. (Rumble) Fault tree and event tree analysis is a widely used and accepted analytical technique in performing reliability analysis and risk assessments. The Reactor Safety Study (WASH-1400) used such techniques. The IEEE/ANS PRA Procedures Guide (NUREG/CR-2300) and the Interim Reliability Evaluation Program Procedures Guide (NUREG/CR-2728) also rely on fault tree and event tree analysis. The use of fault tree and event tree analyses in WASH-1400 and its continued use over the past eight years has provided sufficient insight into the capabilities and limitations of these techniques so that they can be employed in a systematic and standardized fashion to make meaningful, rational decisions.

Q84. Is a reliability data base required for event tree and fault tree analyses?

A84. (Rumble) Yes. Reliability data are needed for input to reliability analysis such as fault tree and event tree analysis. When directly applicable data do not exist in sufficient quantities to allow a statistical determination of failure rates, estimates of failure rates are made using data from components of similar design and application. If this is not possible, engineering judgement is used. In instances where data uncertainties are large, sensitivity and uncertainty studies are conducted to determine the impact of these uncertainties. In formulating decisions regarding feedback to design and operation, these uncertainties (and their impact upon the reliability estimates) are considered along with the best estimate information. Thus using this type of procedure, variations in the supporting data base can be accounted for and meaningful reliability information can be obtained.

Q85. Do sufficient data exist to support a reliablity program for CRBR?

A85. (Rumble) Yes. There are sufficient data available to support the generation of reliability information for identifying and making decisions regarding feedback to design and operation. This judgement is based upon the fact that most of the components and subcomponents used in CRBR are identical or similar to equipment used in LWRs, other LMFBRs, and other industries. Additionally, the ways in which human error can occur in CRBR are similar to the ways in which human error can occur in LWRs. The Applicants also have underway test programs for those components unique to CRBR which will help support reliability estimates. In any event, the Staff's criteria for the Program require that the reliability data used have a well documented basis.

Q86. How are uncertainties in reliability estimates taken into consideration?

A86. (Rumble) It is a standard practice to perform reliability assessments using "best estimate" reliability data. As stated above in response to Question 84, the effect of uncertainties in these data can then be assessed by sensitivity and uncertainty analyses to determine to what extent uncertainties affect the overall results. Based upon the results of sensitivity and uncertainty studies, action can be taken to improve the data base or change the design to reduce the effect of uncertainties which significantly impact the frequency or consequences of potential accidents. The Staff will ensure that the Program adequately accounts for uncertainties during its review of the Applicants' implementation of the Program.

Q87. For which CRBR systems and features do the Staff's criteria require that reliability information be gathered?

A87. (King) Reliability information is required to be gathered for those systems and features whose functions are necessary to prevent core disruptive accidents, or to mitigate CDAs such that the likelihood of exceeding 10 CFR Part 100 dose guidelines is acceptably low. The extent of the reliability activities performed for each system or feature depends upon: (1) the response time required for these systems or features to perform their safety function; (2) whether or not the system has active components or features; (3) whether or not these active components or features are accessible for repair; (4) the accumulated base of directly applicable exprience in LWRs or other LMFBRs; (5) whether the system is designed for prevention or mitigation; and (6) the judged importance to the protection of public health and safety.

In ranking the safety functions, reactor shutdown and shutdown heat removal are considered of primary importance, and thus those systems utilized in fulfilling these two functions will receive emphasis in the Program. Furthermore, it was concluded that both the front line and support systems (electric power, cooling, etc.) necessary to perform each function will be included in the Program. Eight safety functions have been identified for which reliability information will be gathered in the Program. These are:

(1) reactor shutdown

(2) shutdown heat removal

(3) coolant system boundary integrity

(4) features to prevent core flow blockage

(5) features to prevent failed fuel propagation

(6) containment

(7) spent fuel cooling

(8) active features to mitigate core disruptive accidents.

The specific information gathering activities (described in response to Question 81 above) to be applied to each of the eight safety functions is summarized in Figure C.1 of the CRBR SER.

Q88. Describe the Staff's criteria for the Program's second element, feedback to design, operation, surveillance and maintenance?

A88. (Rumble, King) The criteria for the second element of the Program require that the reliability information be fed back into the CRBR design process in time to affect final design. The second element also requires that the Program remain in place during the lifetime of the facility so that reliability information will continue to be generated and utilized in the operation, surveillance, and maintenance of CRBR. Thus, the various activities conducted as part of the Program must provide a mechanism to accomplish this. Many activities in support of this Program are currently underway, as documented in a letter from J. R. Longenecker (DOE) to P. S. Check (NRC), dated January 11, 1983; this letter also describes how the feedback is to be accomplished for each of these activities. It is the intent of the Staff to review the adequacy of this feedback process during the OL review.

Q89. How is it envisioned that the numerical reliability results generated by the Program will be utilized?

A89. (Rumble) The Staff envisions that the use of quantitative data will be limited to relative comparisons among CRBR components, systems and accident sequences and may include some comparison to NRC Safety Goals or other LWR risk assessments. However, it is not a requirement of this Program to employ a numerical goal or criteria for specifically determining the nature and extent of the feedback to the design. This is because the analytical techniques and data used to predict numerical reliability values are not sufficiently advanced to allow precise calculations. Qualitative and quantitative reliability information will, however, be generated as part of this Program and will be used in a qualitative manner to help make engineering judgements regarding feedback to the design.

Q90. How is it expected that a decision will be made regarding whether or not a design or procedure change is required?

A90. (Rumble) The Applicants have committed to develop the process by which the information is fed back into the design and procedures and the criteria or rationale used to control this process. As set forth in Appendix C (p. C-7) of the CRBR SER, the Staff believes that generic criteria are preferred for controlling the information feedback process, with additional criteria applied on a case-by-case basis. For example, the criteria may require that specific reliability information be compared against the CRBR Principal Design Criteria, against comparable performance in modern LWRs, or against the NRC's safety goals. The criteria may also require that the reliability information itself be analyzed to identify specific large contributions to risk. Regardless of the specific considerations utilized, the Applicants have committed to provide clear documentation to assist the Staff in understanding these considerations and how they are applied in determining the extent of feedback on the design, operation, surveillance, and maintenance of CRBR. The Staff intends to audit the implementation and results of the Program during the construction of CRBR to ensure that it is being accomplished in a fashion consistent with its goals.

Q91. Must the Reliability Assurance Program for CRBR be completed prior to issuance of a construction permit?

A91. (King) No. If the Program is conducted on a time frame such that information generated by the Program is evaluated and incorporated, as necessary, into the CRBR design, then the Program's purpose will be satisfied. The Staff has required in the SER (p. C-7) that the Program be implemented in such a fashion, and the Applicants have committed to implement the Program consistent with this requirement.

Q92. Describe the Staff's criteria for the Program's third element, traceability and auditability.

A92. (Rumble) The criteria for the third element of the Program require clear documentation from the Applicants which will allow the Staff to make a determination regarding the completeness and adequacy of the Program. Documentation should be generated for all Program activities, including Program schedule, results, criteria used in assessing the need for a design or procedure change, and the changes resulting from the Program.

Q93. Will the Staff assess whether or not the Program is properly implemented?

A93. (King) Yes. The Staff intends to assess the implementation of the Program throughout the construction of CRBR. The Staff's evaluations and conclusions in this regard will be set forth in the operating license ("OL") SER for CRBR.

Q94. How will the Staff determine whether or not the Program is implemented properly and meets its objectives?

A94. (King) The Staff intends to maintain an an ongoing review of the Applicants' Program through audits and reviews of the Program to ensure that the requirements and overall objectives are being met.

Q95. What will be the Staff's criteria for judging whether or not the Program is implemented properly?

A95. (King) The Staff's criteria for judging whether or not the Program is implemented properly and meets its objectives will be to:

a) assess whether or not the Program is on a schedule to impact final design,

b) assess the adequacy of the Applicants' criteria for implementing design or procedural changes,

c) assess whether or not these criteria are in fact being applied in a consistent and comprehensive fashion,

d) assess the adequacy of proposed design or procedural changes, and

e) assess whether uncertainties in the data and models are adequately accounted for.

In all cases, the Staff will require adequate documentation of the Program in order for the Staff to perform its assessments.

Q96. What is the interrelationship between the CRBR Reliability Assurance Program and the CRBR Probabilistic Risk Assessment (PRA)?

A96. (Rumble) The CRBR PRA described in Appendix J of the PSAR will provide an estimate of and identify contributors to public risk from operation of CRBR using the information and design details available at the time the PRA is completed. The PRA will evaluate risk from all sources of the design, including external events. Results of the PRA will then be used to study specific safety issues and to identify areas for improvement in the design and procedures. Once these activities are completed, the final report issued and the NRC review satisfactorily completed, the PRA will have served it purpose with regard to licensing and no further formal interaction with the NRC regarding the PRA is contemplated thereafter.

The CRBR Reliability Assurance Program as set forth by the criteria described in Appendix C of the CRBR SER (NUREG-0968) is a program to be conducted during final design but also to be continued throughout the plant lifetime. The criteria for the program specify a variety of reliability information gathering activities which concentrate on those safety functions associated with prevention of CDAs and limiting releases within 10 CFR Part 100 guidelines. These activities will provide more detailed evaluations of specific CRBR safety functions than the PRA and provide more detailed feedback to CRBR design and operation in these areas. The purpose of generating models that will remain in place during the lifetime of CRBR is to provide up-to-date tools to evaluate the impact of operating experience and future changes.

While there is no direct or formal link between the reliability information gathering activities mentioned above and the PRA, the Program's criteria do not exclude using models and information developed in the PRA.

VI. CONCLUSION

Q97. Please summarize the Staff's conclusions with respect to whether CDAs may be properly excluded from the Design Basis Accident spectrum for CRBR?

A97. (Panel) It is the Staff's conclusion that by relying upon the deterministic approach applicable to LWRs, as modified to account for the characteristics of LMFBRs, CDAs need not be included in the DBA spectrum for CRBR. With only a few exceptions, the Staff's review procedures, the applicable criteria and standards in Title 10 of the Code of Federal Regulations and the Standard Review Plan normally applied to LWRs, are applicable to LMFBRs. These criteria are the basis for excluding core disruptive accidents from the design basis accident spectrum for LWRs, and when supplemented with the special criteria necessary to account for the characteristics of LMFBRs, contribute to the Staff's confidence that CDAs can be made very improbable for CRBR. In addition, it is the Staff's judgment that the safety functions which must be fulfilled to make CDAs very improbable can be implemented for CRBR. This confidence is based on two points. First, those safety functions which must be achieved for an LMFBR such as CRBR are not fundamentally different from the safety functions successfully implemented for LWRs. Second, the special characteristics associated with design and operation of an LMFBR and the ways they could impact these safety functions are well understood because of the general knowledge and experience gained from design and operation of fast-sodium-cooled reactors such as FERMI-1, EBR-I and II, SEFOR, FFTF, and foreign LMFBRs.

In reaching these conclusions, the Staff has also taken into account the analyses of other data in its review of the completeness of the DBA spectrum, human factors, and the role of the PRA and the reliability program.

The Staff considered the completeness of the DBA spectrum to be important, because it assures that the fundamental safety functions are being performed and that the effect of some lesser failure or systems interaction with a potential for serious complications will not go unconsidered and/or uncorrected (defense in depth).

By applying deterministic criteria similar to those applicable to LWRs, the Staff has also taken into account the possibility that human error could initiate accidents or complicate or defeat the mitigation of accidents by safety systems.

Further, although it is not expected that the probabilistic risk assessment or the reliability program will produce any substantial modifications in the CRBR design, these elements will provide further systematic evaluations of the plant design and operation capable of identifying a need for corrective action and ensuring that such corrective action is implemented.

Based upon the above, the Staff considers that CDAs may be excluded from the DBA spectrum for CRBR.

Richard A. Becker

PROFESSIONAL QUALIFICATIONS

I am presently an LMFBR Engineer in the Technical Review Section, CRBR Program Office of NRC responsible for systems review for Design Bases Accident Analyses and Safety Systems for CRBR Licensing.

I have 25 years experience in the nuclear industry. Prior to coming with the NRC, I was Manager, Energy Projects Development, a technical liaison position between the General Electric Co. and DOE in the Fission and Fusion technoligies. Prior to that position, I was associated with the Southwest Experimental Fast Oxide Reactor (SEFOR). I held two positions at that facility. First, Manager, Program and Analysis with responsibility for acceptance testing, start-up testing, Operations support in physics and engineering and the experimental program; and later as site manager. SEFOR was a liquid metal cooled fast reactor dedicated to measuring the doppler coefficient in a mixed oxide fueled system characteristic of commercial LMFBRs.

Prior to SEFOR, I was a Thermionic Systems Engineer and subsequently Project Engineer on a space power system. In addition, I have held engineering positions in experimental physics, thermal hydraulics and heat transfer on a variety of LWR and gas cooled reactor sytsems.

I am a graduate of the University of Colorado with BS degrees in Engineering Physics and Business Management. I have completed the General Electric Co.'s three year Advanced Engineering Program.

Publications

Becker, R. A., "Runaway Analysis For A Gas-Cooled Reactor," General Electric Company, (TID 4500), APEX 585, July 1960

Johnson, M. L. and Becker, R. A., "Fuel Handling For SEFOR." Proceedings of 17th Conference on Remote Systems Technology, ANS, 1669

Becker, R. A. and Russell, J. L., "Relative Effectiveness of Reactor Control Materials," General Electric Company, GEAP 3201, July 1959

Becker, R. A., "State-of-the-Art In Thermionics," Journal of Spacecraft and Rockets, July 1967

Becker, R. A., "Thermal Stress: A Computer User's Guide," General Electric Company, DCL 60-11-710, November 1960

Editor and contributor to three final design reports for STAR-R Project (Classified SECRET)

Professional Qualification
of
Hukam C. Garg

My name is Hukam C. Garg. I am employed as an electrical engineer in the Division of Engineering, U.S. Nuclear Regulatory Commission, Washington, D.C. I joined the NRC in January, 1981. My duties and responsibilities include the review of licensee and license applicant environmental qualification programs for safety-related equipment. This review encompases the methods used for establishing environmental conditions, the systems and components selected for qualification, the analyses and test procedures employed, audits of qualification documentation, and inspection of installed equipment at the plant sites.

Prior to my present position, I was employed by Gilbert/Commonwealth Associates from 1973 - 1980. My most recent position was Supervising Engineer for the Instrumentation and Control Section. In this position I was responsible for the instrumentation and control aspects, including the equipment qualification, ~~of~~ for nuclear power plants. I had previously worked for Fluor Power Inc., formerly Pioneer Service and Engineering Company (1969 - 1973), in the design of electrical systems for nuclear power plants.

In 1967, I received a Bachelor of Science Degree in Electrical Engineering from the G.S. Technological Institute in India. In 1969, I received a Master of Science Degree in Electrical Engineer from Illinois Institute of Technology. I am also a registered Professional Engineer in the States of Ohio and Illinois. I have also taken training courses in BWR Technology and equipment qualification.

List of Publications

1.) "Engineered Safety Features Actuation System Follow-Up System Logic," H.C. Garg, S.V. Athavale, J.R. Oranchek. Presented at the 1979 IEEE Symposium on Nuclear Power Systems.

2.) "RFI Effects on Nuclear Power Plant I&C Equipment," H.C. Garg, S.V. Athavale, J.R. Oranchek. Presented at the 1979 IEEE Symposium on Nuclear Power Systems.

PROFESSIONAL QUALIFICATIONS

DR. SHOU-NIEN HOU

Dr. Shou-nien Hou is a Senior Mechnical Engineer, and is assistant to the Chief of the Mechanical Engineering Branch (MEB) for performing independent reviews of generic matters and coordinating technical positions among the Staff. He is also the reviewer and the technical contract monitor related to the Clinch River Breeder Reactor in areas of mechanical design. Working for the NRC and the former AEC since 1972, he has participated in technical reviews of nuclear power and plant design criteria, operating problems, safety issues, probabilistic risk assessment, and dynamic analysis and testing of piping, equipment, reactor internals and nuclear safety features. He also served as the monitor of several technical contracts, as the leader of the Seismic Qualification Review Team for conducting plant seismic audits, and as the Task Manager for developing Staff positions on plant safety for the postulated pipe rupture event. In addition, he is on several National Standards Committees and has participated in the development of several Regulatory Guides.

Born in 1934 in China, he came to the USA in 1957. He received his B.S. in Civil Engineering from Taiwan University in 1955, his M.S. in Structure Dynamics from Virginia Tech in 1958, and his Ph.D in Structural Mechanics from M.I.T. in 1968. After graduation, he had various working experiences in structural design, stress analysis, research and development in space vehicle dynamics, and technical review in nuclear power plant safety. He was a visiting lecturer to universities in England, Chile, and to the government in Taiwan, China. He is the author of a dozen technical papers, a recipient of the Apollo Achievement Award from NASA, and a member of Sigma Xi, Tau Beta Pi, Chi Epsilon, AIAA and ANS.

Thomas L. King

PROFESSIONAL QUALIFICATIONS

I am presently Chief, Technical Review Branch in the CRBR Program Office, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. In this capacity, I am responsible for the direction of the Branch's Review of those aspects of CRBRP related to a fast, sodium cooled reactor. This includes direction of the Branch's review of CRBRP sodium systems, fuel handling systems, CDA analysis, support systems, reliability program, safety criteria and analysis.

I received a Bachelor of Science degree in Mechanical Engineering from Drexel University. I also received a Master of Science degree in Mechanical Engineering from Stanford University.

I have over fourteen years of professional experience in the nuclear field. While I worked for the Department of Energy (DOE), I held various positions in the Division of Reactor Research and Technology. These included positions as a Reactor and Nuclear Engineer in the Core Design Branch, the Liquid Metal Systems Branch, and the Components Branch where I worked on the FFTF Project, the EBR-2 project and Facilities at the Engineering Technology Center in Santa Susana, California. In 1975 I was assigned to the DOE FFTF Project Office in Richland, Washington where I held positions as a Reactor Engineer in the Operational & Experimental Safety Division and Branch Chief for FFTF Engineering until April 1982 at which time I joined the NRC as a Reactor Engineer.

List of Publications

1) "FFTF Reactor Characterization Program" T. L. King (DOE) & J. Rawlins (HEDL)
ANS invited paper - 1981 Winter Meeting - San Francisco

2) "Reactor and Plant Performance During FFTF Nuclear Startup"
T. L. King & C. E. Moore - DOE
Ans Topical Meeting - September 1981 - Newport, RI
(Technical Basis for Nuclear Fuel Cycle Policy)

B. M. MORRIS

PROFESSIONAL QUALIFICATIONS

I am currently, Chief of the Electrical Engineering Branch, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission. During the construction permit review and safety evaluation report preparation, I was Section Leader, Technical Review Section, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. In this capacity, I am responsible for direction of the technical review section's review of the fast sodium-cooled CRBRP safety review.

I received a Bachelor of Science, M.S., and Ph.D. degrees in physics from the University of Tennessee.

I spent five years teaching engineering and physics at Worcester Polytechnical Institute. I also spent five years doing research in engineering and nuclear physics at Savannah River and Oak Ridge National Laboratory. In 1977, I joined the Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation in the Reactor Safety Branch. I then worked in the Instrumentation and Control Systems Branch. I then became a Section Leader in the Reactor Systems Branch.

I have published several Journal papers in the fields of physics and nuclear engineering.

STATEMENT OF PROFESSIONAL QUALIFICATIONS

CHARLES E. ROSSI

I have been with the U. S. Nuclear Regulatory Commission (NRC) since October 1980. Since August 1981 I have been a Section Leader in the Instrumentation and Control Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. I am responsible for supervising the review of nuclear power plant instrumentation and control system designs for compliance with regulatory criteria. From October 1980 to August 1981 I was a Principal Reactor Engineer in the Instrumentation and Control Systems Branch. I performed the operating license review of the Callaway and Wolf Creek instrumentation and control system designs, the review of construction permit applicant responses to Three Mile Island Lessons Learned Items related to instrumentation and control systems, and the review of licensee responses to recommendations made by Babcock and Wilcox resulting from failure modes and effects analyses of the Integrated Control System.

I have a Ph.D degree (1969) and M.E degree (1967) in Applied Physics from Harvard University, a M.S degree (1962) in Physics from George Washington University and a B.A degree Magna cum Laude Highest Honors (1958) in Engineering and Applied Physics from Harvard University. I have a certificate from a six month reactor engineering course given by the Bettis Atomic Power Laboratory (1960). I was elected to Phi Beta Kappa in 1958 and Sigma Xi in 1962.

From June 1958 to July 1962 I served as a commissioned officer in the United States Navy. I was assigned to Naval Reactors, U. S. Atomic Energy Commission, where I reviewed and approved test and operating procedures for submarine nuclear power plant fluid systems and reactor instrumentation and control systems designs for the pressurized water reactor at Shippingport, PA.

From September 1966 to November 1977 I held professional and management positions in the Nuclear Energy Systems division of the Westinghouse Electric Corporation. As a manager I supervised the preparation of system functional design requirements for nuclear reactor plant systems which affect plant control, protection, and transient performance. In addition to reactor control and protection systems, these systems included emergency feedwater systems, emergency boration systems, and steam dump systems. For four years I was the lead engineer responsible for establishing functional requirements for reactor control and protection systems used in the Westinghouse 3 loop nuclear reactor plants and for performing transient and accident analyses of these plants for safety analysis reports submitted to the Atomic Energy Commission.

From November 1977 to October 1980 I was Systems and Civilian Applications Program Manager in the Office of Inertial Fusion at the U. S. Department of Energy. In this position, I provided technical and administrative direction for studies of the commercial applications of inertial confinement fusion.

I am a member of the American Nuclear Society and past member of the IEEE Nuclear Power Engineering Committee Standards Subcommittee (SC-6) on Safety Related Systems. I have authored or co-authored over ten technical articles for presentation at conferences or publication in journals.

I was a co-inventor for U.S. Patent 4,222,822 "Method for Operating a Nuclear Reactor to Accommodate Load Follow While Maintaining a Substantially Constant Axial Power Distribution."

ROBERT J. SCHEMEL

PROFESSIONAL QUALIFICATIONS

HUMAN FACTORS ENGINEERING BRANCH

DIVISION OF HUMAN FACTORS SAFETY

I am a Senior Human Factors Engineer in the Human Factors Engineering Branch in the Division of Human Factors Safety. In this position I plan, coordinate, and conduct the review and evaluation of assigned nuclear power plant designs and operations from the standpoint of human factors and man-machine systems interfaces to enhance the functional effectiveness of operator interaction with plant operation and plant shutdown following normal operation, transients, and accidents. I participate in studies and analysis of human factors in man-machine interface problems as they pertain to plant operations and to control room design and operations.

I studied Electrical Engineering at Drexel Evening School, Philadelphia, Pennsylvania, from 1938 to 1941. I served in the U.S. Army Air Corps from 1942 to 1945 where I carried out surface and upper air meteorological observations and assembled and operated upper air observation stations in the European Theater. I received a Bachelor of Science degree in physics and mathematics from the University of Scranton, Scranton, Pennsylvania, in 1950, and a Master of Science degree in physics from Union College, Schenectady, New York, in 1953. I studied Human Factors Engineering at the University of Southern California in the spring of 1980 and in a course given by Oak Ridge Associated Universities in the Summer of 1980.

In 1950, I was employed by the General Electric Company, Knolls, Atomic Power Laboratory, where I conducted experiments on intermediate spectrum reactor cores which aided in the design of the reactor used in the nuclear submarine "Seawolf". This involved basic nuclear physics as well as dynamic behavior of reactors. I acted as a physics advisor to the operating crew during startup and initial tests of the submarine. I also conducted studies of the physics of reactor and plant control with emphasis on safety system design requirements and control schemes.

In 1958 I was employed by Allis-Chalmers Atomic Energy Division. In my first assignment, I directed a group involved in studying the physical characteristics of gas cooled reactors, the associated energy transfer systems, and the control and safety systems. I formulated equations, devised electronic analog circuits for these systems and components, and conducted a dynamic analysis which resulted in the conceptual design of plant safety and control systems. In my second assignment, I had complete responsibility for startup, testing and delivery of the 30 Mw R-2 Research Reactor, Studsvik, Sweden. I specified and directed the test program required to demonstrate the performance of the plant from startup through full power operation.

In January 1962 I returned to the home office after completing this foreign assignment. At this time, I was Section Head, Reactor Operations Section for startup and operation of all Allis-Chalmers built reactors. This responsibility included review and direction of the preparation of preoperational tests, reactor startup tests, plant operating manuals, technical specifications, revisions to technical specifications, and radiological physics procedures.

In March 1967 I joined the Atomic Energy Commission on the staff of the Division of Reactor Licensing. From 1967 to 1974 I performed technical reviews, analysis and evaluations of the nuclear safety aspects of (1) applications for license amendments and technical specification changes for power, test and research reactors and critical facilities, and (2) applications for construction permits and operating authorizations for research reactors and critical facilities. In addition, I was responsible for continuous review of all operating reactors assigned to Operating Reactor Branch No. 1. From 1974 to 1980 I performed technical reviews, analysis, and evaluations in the area of core performance. This work was in reactor physics concerning such things as; core power distribution and reactivity effects in steady state and transient conditions; reactor physics measurements, accuracy of core measurements, and core technical specifications.

From April 1980 to the present: I am a Senior Human Factors Engineer in the Human Factors Engineering Branch in the Division of Human Factors Safety. In this position I have conducted control room design reviews for three near term operating licenses and for six construction permits. I have participated in the preparation of guideline documents for Detailed Control Room Design Reviews and Safety Parameter Display Systems. I am at present in the process of reviewing Program Plans and schedules for accomplishing the control room reviews required by the TMI Action Plan.

Jerry J. Swift

PROFESSIONAL QUALIFICATIONS

I am employed by the Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The title of my position is Reactor Engineer.

From 1980 to 1982, I worked at the U.S. Department of Energy on the environmental and safety aspects of a variety of nuclear technologies, including fusion devices, accelerators, transportation, waste management, and sewage irradiation.

From 1972 to 1980, I was employed in the Office of Radiation Programs of the U.S. Environmental Protection Agency, as a Nuclear Engineer and Environmental Protection Engineer. During this period, I was involved in evaluating the potential radiation doses and radioactive contamination of the environment that might result from severe reactor accidents, with application to the development of Protective Action Guides. I was also involved in the evaluation of normal operating releases from nuclear fuel cycle facilities in support of the development of 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." I assisted in development work for standards and guidance by EPA on management of high level wastes and uranium mill tailings. I aided EPA's participation in nuclear policy reviews by Presidents Ford and Carter. I managed EPA's NEPA reviews of environmental statements for two LWRs. I managed EPA's lengthy review of the Reactor Safety Study, WASH-1400.

From 1970 to 1972, I held a position as Physicist in the Institut fur Reaktorsicherheit,, (now the Gesellschaft fur Reaktorsicherheit mbH) in Cologne, Germany; in this position, I was primarily concerned with evaluating the nature and quantities of radoactive materials that might by released in postulated nuclear reactor accidents, and the resulting radiation doses that might be experienced. I was also involved inevaluating siting conditions.

While a graduate student at the Catholic University of America, from 1966 to 1970, I held the position of Assistant University Radiological Safety Official, performing health physics functions throughout much of the University.

I received the degree of Geological Engineer from the Colorado School of Mines in 1955, a Master of Science degree in Nuclear Engineering from Iowa State University in 1965, and a Ph.D. degree in Nuclear Engineering from the Catholic University of America, Washington, D.C. in 1971. I have current certification in Health Physics from the American Board of Health Physics.

In my present position, I am primarily involved in the review of those potential accident event sequences which may lead to radiological consequences. This includes both sequences within the design basis and sequences beyond the design basis. I also coordinate the efforts of other technical reviewers on these topics.

ASHOK K. AGRAWAL

PROFESSIONAL QUALIFICATIONS

I am an employee of the Brookhaven National Laboratory, Upton, New York which is operated by the Associated Universities, Inc. for the U.S. Department of Energy. Brookhaven National Laboratory is a recognized center for scientific research and development. BNL provides, under a contract, consulting services to the U.S. Nuclear Regulatory Commission in many areas of light water reactors as well as the Clinch River Breeder Reactor. My own current involvement in the CRBR technical review is in the areas of design basis accidents review.

I hold an Sc.D. (Doctor of Science) degree in Nuclear Engineering from the Massachusetts Institute of Technology, Cambridge, Massachusetts. Subsequent working experience for the last fourteen years have all been in the general field of LMFBR safety. I spent more than four years at the Argonne National Labortaory, after a little over one year working experience in the Fast Flux Test Facility (FFTF) with the Westinghouse Electric Corporation, I joined BNL in 1974 and have since been involved with two key areas: the development of a major thermohydraulics computer code (SSC) for LMFBR and the technical review of CRBR.

I have published extensively in the field of reactor safety. Some of the publications are noted below.

Publications

1. Agrawal, A.K. and Khatib-Rahbar, M., "Dynamic Simulation LMFBR Systems", Atomic Energy Review 18, 329-552 (1980).

2. Agrawal, A.K. et al., "An Advanced Thermohydraulic Simulation Code for Transients in LMFBRs (SSC-L Code)," Brookhaven National Laboratory, BNL-NUREG-50773 (1978).

3. Decay Heat Removal and Natural Convection in Fast Breeder Reactors, Agrawal, A.K. and Guppy, J.G., Editors, Hemisphere Publishing Co., Washington (1981).

JOHN E. HANSON

PROFESSIONAL QUALIFICATIONS

I am presently the Program Manager for Space and Military Reactor Programs at the Los Alamos National Laboratory responsible for development, planning, and implementation of space and military reactor programs at the Laboratory.

Prior to this position which I took in January 1983, I was employed for three years with EG&G Idaho as Principal Engineer. In that time I was involved in the formulation of the TMI-2 core examination program and the NRC Severe Fuel Damage Program. I also participated in the new production reactor concept evaluation, Clinch River Breeder Reactor licensing and served on a DOE Fact Finding Group for Public Law 96-567.

From 1966 to 1979 while with the Westinghouse Hanford Company, I managed the Fast Flux Test Facility fuel development and safety research programs; programs of major national scope.

From 1956 to 1966 I was employed by the General Electric Company's Hanford Atomic Products Operation in Richland, Washington; the Vallecitos Nuclear Laboratory in Pleasanton, California; and the Advanced Reactors Division in San Jose, California.

I hold a MS degree in mechanical engineering and professional engineer licenses in mechanical and nuclear engineering.

Edmund T. Rumble, III

PROFESSIONAL QUALIFICATIONS

I am an employee and Corporate Vice President of Science Applications, Inc. (SAI), a nationwide research and consulting firm. In this capacity, I perform contract research on energy-related projects. Presently, I am a member of an SAI team providing technical assistance to the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission on safety matters related to the proposed Clinch River Breeder Reactor Plant.

I received a commission in the U.S. Navy and a Bachelor of Science degree from the U.S. Naval Academy. After graduation, I was qualified for and served as a U.S. Naval submarine officer responsible for operational and administrative aspects of a submarine nuclear power plant. Following my honorable discharge from the Navy, I received my Master of Science and Ph.D. degrees in Nuclear Engineering from UCLA. I am a Professional Engineer registed in the State of California and am listed in Who's Who in Technology Today, American Men and Women of Science, and Who's Who in California (14th ed).

I have been involved in LWR and LMFBR safety research at UCLA and SAI for the past ten years. My doctoral thesis involved modeling LMFBR core accidents. I have made technical contributions, managed, and acted as an advisor in deterministic and probabilistic safety analyses and assessments of LWRs and LMFBRs. Recently, I managed a major portion of, and technically participated in, a two-year, multi-organizational risk-oriented study of the SNR-3000, an LMFBR under construction in West Germany.

DBA TESTIMONY ATTACHMENT 1

NRC SPONSORED RELIABILITY ORIENTED STUDIES FOR CRBR

1. NUS-2001, "Assessment of CRBR Reliability Program and Safety Systems," December 1976.

2. NUREG/CR-0405, "Markovian Reliability Analysis Under Uncertainty With an Application on the Shutdown System of the CRBR," September 1978.

3. NUREG/CR-0013, "LMFBR Fuel Analysis Task C: Reliability Aspects of LMFBRs," February 1978.

4. UCLA-ENG-7682, "An Analysis of the Reliability of the Shutdown Heat Removal System for the CRBR," August 1976.

5. NUREG/21962, "Reliability of the Shutdown Heat Removal System of the Clinch River Breeder Reactor," October 1976.

6. BNL-NUREG-31297, "Review of the Status of CRBR Licensing Technical Issues Related to Heat Removal System and Severe Accident Analysis," April 1982.

7. NUREG/CR-1507, "LMFBR Accident Delineation Study - Phase I," November 1980.

8. NUREG/CR-2681, "Estimated Recurrence Frequencies for Initiating Accident Categories Associated With the Clinch River Breeder Reactor Plant," April 1982.

9. SAI-348-83-PA, "An Estimate of Release Frequencies for CRBRP Potential Core Disruptive Accidents," January 1983.

10. SAI-83-959-WA, "Risk Reduction Feasibility Study of Selected Modifications to CRBR Safety Systems," September 1982.