NUREG-0968
Vol. 1
Main Report

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

Docket No. 50-537

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

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1983

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ABSTRACT

The Safety Evaluation Report for the application by the United States Department of Energy, Tennessee Valley Authority, and the Project Management Corporation, as applicants and owners, for a license to construct the Clinch River Breeder Reactor Plant (Docket No. 50-537) has been prepared by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. The facility will be located on the Clinch River approximately 12 miles southwest of downtown Oak Ridge and 25 miles west of Knoxville, Tennessee. Subject to resolution of the items discussed in this report, the staff concludes that the construction permit requested by the applicants should be issued.

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ABBREVIATIONS

ABHX	-	air blast heat exchanger
ACI	-	American Concrete Institute
ACLP	-	above core load pad
ACRS	-	Advisory Committee on Reactor Safeguards
AFS	-	annulus filtration system
AFW	-	auxiliary feedwater
AFWS	-	auxiliary feedwater subsystem
AFWS	-	auxiliary feedwater system
AHM	-	auxiliary handling machine
AI	-	Atomics International
AIME	-	American Institute for Mining, Metallurgical, and Petroleum Engineers
AI-RM	-	Atomics International - reactor manufacturers
AISC	-	American Institute of Steel Construction
ALARA	-	as low as is reasonably achievable
ALMS	-	auxiliary liquid metal system
AM	-	accident monitoring
ANI	-	American Nuclear Insurers
ANL	-	Argonne National Laboratory
ANS	-	American Nuclear Society
ANSI	-	American National Standards Institute
ARD	-	Advanced Reactor Division, Westinghouse Electric Corporation
ARPI	-	absolute rod position indication
ARSD	-	Advanced Reactors Systems Dept., GE
ASB	-	Auxiliary Systems Branch
ASCE	-	American Society of Civil Engineers
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing and Materials
ATWS	-	anticipated transients without scram
AVT	-	all-volatile treatment
BNL	-	Brookhaven National Laboratory
BOC	-	beginning of cycle
BOEC	-	beginning of equilibrium cycle
BOL	-	beginning of life
BOP	-	balance of plant
BRD	-	Breeder Reactor Division, Burns and Roe, Inc.
BTP	-	branch technical position
BWR	-	boiling-water reactor
CAM	-	continuous air monitor
CAPS	-	cell atmosphere processing system
CCFA	-	common cause failure analysis
CCP	-	core component pot
CDA	-	core-disruptive accident
CDF	-	cumulative damage function

CECC	-	Central Emergency Control Center
CFR	-	Code of Federal Regulations
CIS	-	containment isolation system
CLCV	-	cold-leg check valve
CMP	-	core midplane
COCORP	-	Consortium for Continental Reflection Profiling
CP	-	construction permit
CRBR	-	Clinch River Breeder Reactor
CRBRP	-	Clinch River Breeder Reactor Plant
CRBRP-REP	-	Clinch River Breeder Reactor Plant Radiological Emergency Plan
CRCIP	-	Clinch River Consolidated Industrial Park
CRDL	-	control rod drive line
CRDM	-	control rod drive mechanism
CRDS	-	control rod drive system
CRT	-	cathode-ray tube
CS	-	core support
CST	-	condensate storage tank
CW	-	cold worked
D/G	-	diesel generator
DBA	-	design-basis accident
DBE	-	design-basis event
DBF	-	design-basis flood
DBL	-	design-basis leak
DEG	-	double-ended guillotine
DEMA	-	Diesel Engine Manufacturers Association
DFR	-	Dounreay fast reactor
DHRS	-	direct heat removal service
DLS	-	ductility limited strain
DNB	-	departure from nucleate boiling
DND	-	delayed neutron detection
DND	-	delayed neutron detector
DNPEC	-	Division of Nuclear Power Emergency Center
DOE	-	Department of Energy, U.S.
EAL	-	emergency action level
EAS	-	essential auxiliary support
EAS	-	essential auxiliary support (system)
EBR-II	-	experimental breeder reactor No. II, INEL
ECCS	-	emergency core cooling system
ECWS	-	emergency chilled water system
EDEG	-	equivalent double-ended guillotine
EEB	-	electrical equipment building
EHC	-	electrothydraulic control
EM	-	electromagnetic
EMC	-	engineering mockup critical
EMI	-	electromagnetic interference
EOC	-	end of cycle
EOF	-	emergency operations facility
EOL	-	end of life
EPSW	-	emergency plant service water
EQ	-	environmental qualification
ER	-	Environmental Report
ESF	-	engineered safety feature(s)

ESFAS	-	engineered safety features actuation system
ESG	-	Energy Systems Group, Rockwell International Corp.
ETEC	-	Energy Technology Engineering Center
ETSB	-	Effluent Treatment Systems Branch
EVS	-	ex-vessel storage
EVS	-	ex-vessel system
EVST	-	ex-vessel storage tank
EVTM	-	ex-vessel transfer machine
FCCI	-	fuel cladding chemical interaction
FCMI	-	fuel-cladding mechanical interactions
FCTT	-	fuel cladding transient tester
FEMA	-	Federal Emergency Management Agency
FERMI-1	-	Enrico-Fermi LMFBR Plant
FES	-	Final Environmental Statement
FFTF	-	fast flux test facility
FHC	-	fuel-handling cell
FMEA	-	failure mode effects analysis
FPD	-	full-power day
FR	-	Federal Register
FSAR	-	final safety analysis report
FV	-	floor valve
GDC	-	general design criterion (criteria)
GE	-	General Electric Company
GE-RM	-	General Electric - reactor manufacturer
GETR	-	General Electric Test Reactor
GFI	-	ground fault interrupt
GM	-	Geiger Mueller
H&V	-	heating and ventilating
HAA	-	head access area
HAZ	-	heat-affected zone
HCDA	-	hypothetical core disruptive accident
HCF	-	hot channel factor
HEDL	-	Hanford Engineering Development Laboratory
HEPA	-	high-efficiency particulate air
HTGR	-	high-temperature gas reactors
HTS	-	heat transfer system
HTS	-	heat transport system
HVAC	-	heating, ventilating, and air conditioning
HVAC	-	heating, ventilation and air conditioning
I&C	-	instrumentation and control
IALL	-	intermediate activity liquid level
ICRP	-	International Commission on Radiological Protection
ICSB	-	instrumentation and control systems branch
IEEE	-	Institute of Electrical and Electronics Engineers
IES	-	Illuminating Engineering Society
IGR&P	-	inert gas receiving and processing
IHTS	-	intermediate heat transport system
IHTS	-	intermediate heat transfer system
IHX	-	intermediate heat exchanger

IMAS	-	impurity monitoring and analysis system
INEL	-	Idaho Nuclear Engineering Laboratory
INPO	-	Institute for Nuclear Power Operations
IRP	-	intermediate rotating plug
ISI	-	inservice inspection
ISMS	-	inoperable status monitoring system
ISR	-	intermediate sodium removal
IVTM	-	in-vessel transfer machine
KECC	-	Knoxville Emergency Control Center
KSR	-	key systems review
LALL	-	low activity liquid level
LANL	-	Los Alamos National Laboratory
LCCV	-	large component cleaning vessel
LCT	-	large component transporter
LDP	-	Large Development Plant
LED	-	light-emitting diode
LIM	-	lower inlet module
LLTR	-	long leg test rig
LMBFR	-	liquid metal fast breeder reactor
LN_2	-	liquid nitrogen
LOCA	-	loss-of-coolant accident
LOHS	-	loss of heat sink
LOOP	-	loss of offsite power
LPMS	-	loose parts monitoring system
LPZ	-	low population zone
LRM	-	lead reactor manufacturer
LTC	-	long-term cooldown
LWR	-	light-water reactor
MAELU	-	Mutual Atomic Energy Liability Underwriters
MCJ	-	maintenance communications jacking (system)
MEB	-	Mechanical Engineering Branch
MG	-	motor generator
MIT	-	Massachusetts Institute of Technology
MMI	-	modified Mercalli intensity
MPS	-	multipurpose sampler
MSECC	-	Muscle Shoals Emergency Control Center
MSIV	-	main steam isolation valve
MSL	-	mean sea level
MSL	-	minimum safe level
MSV	-	main steam valve
MSV	-	mean square voltage
NCWS	-	normal chilled water system
NDE	-	nondestructive examination
NEPA	-	National Envrionmental Policy Act of 1969
NFPA	-	National Fire Protection Association
NFSC	-	new fuel shipping container
NFUS	-	new fuel unloading station
NI	-	nuclear island
NRC	-	U.S. Nuclear Regulatory Commission
NRL	-	Naval Research Laboratory
NSSS	-	nuclear steam supply system

OBE	-	operating basis earthquake
OD	-	outside diameter
OL	-	operating license
ORGDP	-	Oak Ridge Gaseous Diffusion Plant
ORNL	-	Oak Ridge National Laboratory
ORO	-	Oak Ridge Operations Office
OSC	-	operations support center
OSIS	-	outlet steam isolation system (subsystem)
P&ID	-	piping and instrumentation diagram
P/F	-	power-to-flow ratios
PA-IC	-	public address intraplant communication (system)
PACC	-	protected air-cooled condenser
PAS	-	plant annunication system
PAX	-	private automatic exchange (system)
PCI	-	pellet cladding interaction
PCRS	-	primary control rod system
PDC	-	principal design criterion (criteria)
PDH&DS	-	plant data handling and display system
PEL	-	proportional elastic limits
PEOC	-	plant expected operating conditions
PFAD	-	plugging filter aerosol detector
PFR	-	prototype fast reactor
Phenix	-	French LMFBR demonstration plant
PHTS	-	primary heat transport system
PIE	-	postirradiation examination
PLLMS	-	postulated large liquid-metal spill
PMC	-	Project Management Corporation
PMF	-	probable maximum flood
PMP	-	probable maximum precipitation
PO	-	Project Office
POTC	-	Power Operations Training Center
PPS	-	plant protection sytem
PRA	-	probabilistic risk assessment
PSAR	-	preliminary safety analysis report
PSI	-	preservice inspection
PSR&D	-	primary sodium removal and decontamination
PSSP	-	primary sodium-sampling package
PTI	-	plugging temperature indicator
PWR	-	pressurized-water reactor
PWST	-	protected water storage tank
QA	-	quality assurance
QAC	-	quality assurance criteria
RAPS	-	radioactive argon processing system
RBCB	-	run beyond cladding break
RCB	-	reactor control building
RCB	-	reactor containment building
RCBP	-	Reactor coolant pressure boundary
RDT	-	(DOE Standard)
RDT	-	Reactor Development and Technology
RG	-	regulatory guide

RGCS	-	recirculating gas cooling system
RGT	-	rotating guide tube
RHR	-	residual heat removal
RIA	-	reactivity insertion accident
RL	-	Richland Operations Office, DOE
RM	-	reactor manufacturers
RPIS	-	rod position indication system
RQD	-	rock quality designation
RRHES	-	reactor residual heat extraction system
RRHRS	-	reactor residual heat removal system
RRP	-	reactor to rotating plug
RRPI	-	relative rod position indication
RRS	-	reactor refueling system
RRS	-	removable radial shielding
RRS	-	required response system
RSB	-	reactor service building
RSMP	-	remote shutdown monitoring panel
RSS	-	reactor shutdown system
RT	-	room temperature
RT	-	radiography testing
RTD	-	resistance temperature detector
RV	-	reactor vessel
S&W	-	Stone & Webster Engineering Corporation
SAFDL	-	specified acceptable fuel design limits
SAI	-	Science Applications, Inc.
SAR	-	safety analysis report
SCRS	-	secondary control rod system
SEFOR	-	Southwest Experimental Fast Oxide Reactor
SER	-	safety evaluation report
SFSC	-	spent fuel shipping cask
SG	-	steam generator
SGAHRS	-	steam generator auxiliary heat removal system
SGB	-	steam generator building
SGBFP	-	steam generator building flooding protection
SGS	-	steam generation system
SGS	-	steam generator system
SHRS	-	shutdown heat removal system
SID	-	sodium ionization detector
SLMS	-	small liquid-metal spill
SLSF	-	sodium loop safety facility
SMBDB	-	structural margins beyond the design basis
SR&D	-	sodium removal and decontamination
SRP	-	Standard Review Plan
SRSS	-	square root of the sum of the squares
SS	-	steady state
SSC	-	Super System Code
SSCCW	-	secondary service closed cooling water
SSE	-	safe shutdown earthquake
SSP	-	sodium sampling package
SSPLS	-	solid-state programmable logic system

SSR	-	site suitability report
STP	-	standard temperature and pressure
Super-Phenix	-	slow neutron reactor
SWRPRS	-	sodium water reaction pressure relief system
SWRPRS	-	sodium water reaction pressure relief subsystem
SWRPS	-	sodium water reaction product system
T/H	-	thermal and hydraulic
TEC	-	Technology for Energy Corp., Knoxville, TN
TEMA	-	Tennessee Emergency Management Agency
TGB	-	turbine generator building
TGS	-	turbine generator system
TGSS	-	turbine gland sealing system
T/H	-	thermal and hydraulic
THD	-	thermal hydraulic design
THDV	-	thermal hydraulic design value
TLC	-	transient limit curves
TLP	-	top load pad
TMBDB	-	thermal margin beyond design basis
TMI	-	Three Mile Island
TOP	-	transient overpower
TREAT	-	transient reactor test
TRS	-	test response spectrum
TS	-	Technical Specifications
TSC	-	technical support center
TVA	-	Tennessee Valley Authority
UIS	-	upper internals structure
USGS	-	United States Geological Survey
USI	-	unresolved safety issue
UT	-	ultrasonic
VHF	-	very high frequency
VRS	-	vacuum relief system
VWE	-	vanadium wire equilibration
W	-	Westinghouse Electric Corporation
W-ARD	-	Westinghouse Advanced Reactors Division
W-LRM	-	Westinghouse-lead reactor manufacturer
WG	-	water gage
WM	-	Waltz Mill
WVN	-	water vapor nitrogen
ZPPR	-	zero power plutonium reactor

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This report is the Safety Evaluation Report (SER) on the application for a construction permit (CP) for the Clinch River Breeder Reactor Plant (CRBRP) based on the application filed by the U.S. Department of Energy (DOE), Project Management Corporation (PMC), and Tennessee Valley Authority (TVA) (hereafter called the applicants). This report was prepared by the U.S. Nuclear Regulatory Commission (the staff) and summarizes the results to date of the staff's safety review of the proposed facility. The review has been largely limited to those subjects that could have an impact on the decision to issue a construction permit. The NRC Licensing Project Manager for the Clinch River Breeder Reactor Plant is Richard M. Stark. Mr. Stark may be contacted by calling (301) 492-9732 or by writing to: CRBRP Program Office, U.S. Nuclear Regulatory Commission, Washington, DC 20555. The NRC staff principal reviewers and their contractors for this project are listed in Appendix F.

The application for a construction permit for the CRBRP was filed with the U.S. Atomic Energy Commission (now the U.S. Nuclear Regulatory Commission) on April 11, 1975. The Preliminary Safety Analysis Report (PSAR) was docketed on June 13, 1975 under Docket No. 50-537.

Before issuing a CP for a nuclear power plant, the NRC staff is required to review the CRBRP proposed design criteria, as described in the PSAR, and the potential effect on the health and safety of the public. The safety review of the CRBRP has been based on the PSAR and Amendments 1 through 75 thereto. These documents and copies of this report are available to the public for inspection or copying for a fee at the NRC Public Document Room at 1717 H Street, NW, Washington, DC 20555; the Oak Ridge Public Library, Civic Center, Oak Ridge, TN 37830; and the Lawson McGhee Public Library, 500 West Church Street, Knoxville, TN 37902. Copies of this report may be obtained as indicated on the inside front cover.

During the course of its review, the staff has met a number of times with the applicants and their consultants to discuss the proposed design, construction, and operation of the facility. Responses to questions raised since the issuance of the PSAR are provided in Amendments 1 through 75 of that document.

This SER summarizes the results to date of the staff's safety review of the CRBRP and identifies, where applicable for a construction permit review, the technical details considered in evaluating the safety aspects of the plant. The proposed design of CRBRP was reviewed against the applicable Federal regulations, CP criteria, and the intent of NRC "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), NUREG-75/087 and NUREG-0800, dated December 1977 and July 1981, respectively (includes branch technical positions).

The general design criteria for a nuclear power plant are specified in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50), Appendix A. The principal design criteria for the CRBRP were developed using Appendix A of 10 CFR 50 as a basis and modified and expanded to reflect the unique aspects of CRBRP. The principal design criteria used as a basis for this review are identified in Section 3.1 of this SER. The Standard Review Plan, NUREG-0800, was written to cover a variety of site conditions and plant designs. The staff used the Standard Review Plan where it was applicable to the CRBRP application and supplemented the review with criteria that apply only to the CRBRP application where required. Exceptions and supplements to the Standard Review Plan are noted in each section of this SER.

The review is based on the defense-in-depth safety philosophy, commonly known as a three-levels-of-safety approach.

The first level of safety provides criteria for reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic features of the design, such as quality assurance, redundancy, diversity, independence, maintainability, testability, inspectability, and fail-safe characteristics. The plant design criteria must not only accommodate steady-state power conditions, but also have adequate tolerance for normal operating transients, such as startup, shutdown, and load following.

The second level of safety provides criteria for protection against anticipated and unlikely faults, such as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel-handling errors, that might occur in spite of the care taken in design, construction, and operation of the plant. This level of safety for the public is provided by redundancy of critical components as well as by protection devices and systems designed to ensure that such events will be arrested. The requirements for these protection systems must be based on a spectrum of occurrences that the plant design must safely accommodate. Conservative design practices, including the provision of redundant detecting and actuating equipment, must be incorporated in the protection systems to ensure both the effectiveness and reliability of this second level of design.

The third level of safety criteria supplements the first two levels by providing acceptable plant response to extremely unlikely faults such as pipe leaks, sodium fires, or sodium-water reactions. Although occurrence of these faults is of low probability, appropriate engineered safety features must be incorporated into the CRBRP design to safely accommodate such events. Conservative assumptions and evaluation methods are used to develop adequate designs. In addition, conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant. These include such low-probability events as severe earthquakes, tornadoes, and floods. These faults and natural phenomena combine to define the design-basis envelope.

As discussed above, the safety design philosophy of the plant must provide for mitigation of the full range of events, from relatively trivial events to postulated design-basis accidents.

Because of the extremely conservative safety criteria for the plant and the extensive safety features required by the principal design criteria, the staff concludes that core disruptive accidents can and must be excluded from the design-basis accidents for the plant. However, to provide additional safety margins, because of the difference in experience between a light-water reactor (LWR) and a liquid metal fast breeder reactor (LMFBR), certain hypothetical core disruptive accidents (HCDAs) have been analyzed. As a result, additional design margins are required to accommodate the consequences of postulated HCDAs. These include both structural and thermal margins. Structural margins beyond the design basis (SMBDBs) ensure that extra margins will exist to accommodate structural loadings on the reactor vessel system and the components of the primary heat transport system from postulated HCDAs. Thermal margins beyond the design basis (TMBDBs) will ensure that radiological consequences of HCDAs will be mitigated to acceptable levels. Results of the beyond-the-design-basis evaluations are described in Appendix A of this SER.

Review and evaluation of compliance by the applicants to NUREG-0718, Revision 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," have been incorporated into the appropriate sections of this report. Table 1.1 provides cross references relating the lessons learned from the accident at Three Mile Island (TMI) as they apply to construction permits to the sections in this report where the items are discussed.

Sections 2 through 18 of this SER contain the staff's review and evaluation of safety-related issues that have been considered during the review of the CP application. Section 19 is reserved for the report of the Advisory Committee on Reactor Safeguards. Section 20 contains an evaluation of whether the operation of the facility will be detrimental to the common defense and security. Section 21 provides an assessment of the financial qualifications of the applicants. Section 22 provides the financial protection and indemnity requirements for preoperational storage of nuclear fuel and operation of the facility. Section 23 contains the conclusions of this report.

Appendix A contains the staff's evaluation of the events considered to be beyond the design basis. Appendix B is a discussion of the applicability of the generic unresolved safety issues relating to the CRBRP. Appendix C contains the staff's assessment of the Reliability Program. Appendix D contains the current assessment of the applicants' Probabilistic Risks Assessment Plan. Appendix E is a chronology of NRC's principal actions related to the review of the application. Appendix F is a list of principal staff reviewers and staff consultants. Appendix G is a list of references that are used in this SER.* Appendix H reproduces the U.S. Geological Survey's final seismological review of the CRBR site.

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, the staff prepared the Draft Environmental Statement (NUREG-0024) that set forth the conditions related to the proposed construction and operation of the CRBRP. The Final Environmental Statement for CRBRP (NUREG-0139) was issued in February 1977, a draft supplement to the Final Environmental Statement was issued in July 1982, and Supplement No. 1 was issued in October 1982.

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*Availability of documents listed in Appendix G is given on the inside front cover of this SER.

The "Site Suitability Report in the Matter of the Clinch River Breeder Reactor Plant" was originally published on March 4, 1977. A revision to this document was published in June 1982 (NUREG-0768).

The review and evaluation of the CRBRP for a construction permit is only one stage in the continuing review by the staff of the design, construction, and operating features of the plant. The construction of the facility will be monitored in accordance with the inspection program of the staff. During the operating license review stage, the NRC staff will review the final design to determine that the safety requirements, as outlined in this report, are met.

It should be noted that this review is based on the following key assumptions regarding plant operation which, if changed, will require further staff review:

(1) peak fuel burnup of 80,000 MWD/MT

(2) no load following operation

(3) three-loop operation only

(4) restricted operation with failed fuel, as described in Section 4.2 of this SER

(5) 33% Pu-enriched fuel

(6) 12% Pu-240 plutonium isotopic composition

1.2 General Plant Description

The proposed CRBRP will be a liquid-sodium-cooled fast breeder reactor nuclear power plant. The major systems of the proposed plant are the reactor, heat transport and related systems, steam generator and related systems, turbine generator and related systems, fuel-handling system, power transmission and plant electrical system, auxiliary systems, and instrumentation and control system. The major plant operational parameters are given in Table 1.2.

The thermal hydraulic design parameters for the proposed plant are based on a thermal power rating of 975 MW.

The proposed site for the CRBRP is on a peninsula bounded on the south by the Clinch River and on the north by DOE's Oak Ridge Reservation and is within the city limits of Oak Ridge, Tennessee. The 1,364-acre site is owned by the U.S. Government and is in the custody of the Tennessee Valley Authority (TVA). The proposed site is 12 mi southwest of downtown Oak Ridge and 25 mi west of Knoxville, Tennessee.

A system of three piped circuits will transport heat from the reactor, through primary and intermediate sodium loops, to steam generator modules that will produce steam for the turbine. The primary heat transport system (PHTS) will remove the heat generated in the fuel assemblies, blanket assemblies, control rod assemblies, and structural elements. Each of the three intermediate heat transport systems (IHTSs) will receive heat from the PHTS through an intermediate heat exchanger. The IHTS will transfer heat outside containment with nonradioactive sodium. The intermediate heat exchanger (IHX) will act as a barrier for the transfer of radioactive materials between the PHTS and IHTS. The IHTS will be maintained at higher pressure than the PHTS to inhibit leakage from the radioactive PHTS into the nonradioactive IHTS. Each primary loop will contain a hot-leg (>800°F normally) pump, an intermediate heat exchanger, a cold-leg (>800°F normally) check valve, and interconnecting piping between the abovementioned components and the reactor vessel inlet and outlet nozzels. Each intermediate loop will have a cold-leg pump, intermediate sodium expansion tank, and interconnecting piping to transport the sodium from the tube side of the IHX to the superheater inlet and from the evaporator outlets back to the IHX tube-side inlet.

The PHTS piping and components will be located in cells within the containment building. The components and piping for each loop will be located within three vaults (cells) in the containment building. The cells will be separated from each other by concrete shielding walls and will be inerted with nitrogen, which will be circulated for cooling. Those parts of the PHTS equipment that will come in contact with sodium will be located in a nitrogen atmosphere below the level of the containment building operating floor. Each heat transport system (HTS) cell will have a separate atmosphere, and the reactor cavity and the HTS pipeways will have a common atmosphere. The pump drive systems (motors, speed controllers, and heating and seal assemblies) will be located in an environment of air above the operating floor.

The reactor will have 156 fuel assemblies, 82 inner blanket assemblies (6 of which are to be interchangeable with fuel assemblies), 126 radial blanket assemblies, and 15 control assemblies (9 primary assemblies and 6 secondary assemblies). The reactor fuel assemblies will be about 14 ft long, with an active core height of 36 in., 14-in. upper and lower axial blankets, and a 48-in. fission gas plenum. Each fuel assembly will contain 217 stainless steel cladding fuel pins. Each blanket assembly will contain 61 stainless steel cladding blanket pins. The fuel in the active core will be oxides of plutonium and uranium (PuO_2/UO_2). The blanket rods will be 116.5 in. long with 64-in. depleted uranium oxide pellets and a 48-in.-long plenum. The control rod absorber material will be enriched boron carbide (B_4C). Each primary control assembly will contain 37 absorber pins and each secondary assembly will contain 31 pins. The core will be designed for annual refueling. The coolant flow will be upward through the core. The free sodium surface in the upper plenum will be covered by argon.

The reactor will be located in a stainless steel reactor vessel that will have a nominal inside diameter of 20 ft 3 in., and will be 58 ft 8 in. high, from the bottom of the vessel to the top of the support ring. The vessel will be provided with a closure head designed to accommodate through-the-head refueling. The reactor vessel, IHX, and primary sodium pumps will be enclosed by freestanding guard vessels.

The offsite power systems will deliver the power to and from the site and will include transformers, switchgear, structures, and overhead and underground conduits. Included will be devices by which the main generator will be connected or isolated from the TVA distribution grid. In addition, this system will provide the unit station service transformers, reserve transformers, and related primary-side switchgear through which the station auxiliary loads will be supplied power from the power grid.

The onsite power systems will distribute and control the electrical energy for the site and will interface with the offsite power system at the secondary terminals of the unit station service and reserve transformers. The onsite power systems will provide the following functions:

(1) receive power from the offsite power supplies and transform the voltage to the utilization levels of the nuclear island building, lighting, and site-service loads

(2) provide diesel generators for standby power, and batteries and inverters for vital ac and dc power

(3) provide cable, conduit, raceway, and shielded penetration systems for interconnecting wiring for electrical power and control, instrumentation, lighting, and communication

(4) provide control and interlocking operations when these functions will not be provided by other systems.

The plant control system will integrate the various plant instrumentation and control systems. The system will also provide the control and instrumentation circuits and devices required to provide overall plant control, which automatically will maintain essentially constant steam temperature and pressure at the turbine inlet over the 40-100% load range of operation. The plant control system will be designed to provide load-follow capability as required in practical utility operation.

The plant protection system (PPS) will include the shutdown system and the containment isolation system and will interface with the shutdown heat removal system. The PPS will not require the reactor operator or control system to implement a protective action.

The shutdown system will consist of two systems, either of which will be capable of shutting down the reactor.

The reactor confinement/containment system will provide a protective boundary between the plant and the surrounding environment in the event of a serious radioactive release. The containment will consist of a steel pressure vessel with inner and outer concrete shields below the operating floor. The free-standing steel vessel will be cylindrical in shape with an ellipsoidal spherical top and flat bottom. The cylindrical steel vessel will be approximately 1.5 in. thick with an inner diameter of 186 ft; the ellipsoidal spherical top will be 158.3 ft above the operating floor. The containment will house the reactor vessel and the primary heat transport system components, including the intermediate heat exchanger, primary pumps, primary piping, sodium overflow tank, portions of the compartment inerting and primary cover gas systems, one primary sodium storage tank, and some of the fuel-handling equipment. The reactor containment building will be surrounded by a concrete confinement structure, with an annulus space separating the two structures. The annular space between the containment and confinement will be maintained at a negative pressure relative to atmospheric pressure during normal operation and will be exhausted through high-efficiency filters should an accident occur. The concrete confinement will be designed to meet tornado missile and seismic Category I criteria.

The nuclear steam supply system and the steam and power conversion system for CRBRP will be supervised and controlled from the main control room.

1.3 Comparisons With Similar Facility Designs

A comparison of the principal features of other loop-type liquid fast reactors is given in Table 1.3.

Comparison With the Fast Flux Test Facility

Although the fast flux test facility (FFTF) is a test reactor and does not generate electricity, several similarities exist that allow important comparisons to be made. Both FFTF and the proposed CRBRP are loop-type plants (as compared with a pool type) that use the primary and intermediate heat transport concept. The plants' mechanical systems, core, instrument and electrical systems, auxiliary liquid metal, radioactive waste system, fire protection, and high-temperature design are very similar in design and materials. Vast amounts of research and development information, test program data, and design experience from the FFTF have been used in the design of the CRBRP. (See Table 1.3-2 of the PSAR.)

In addition to the hardware similarities, the organizational structure for the design and construction of the CRBR and the FFTF are very similar.

1.4 Identification of Agents and Contractors

The Department of Energy is the responsible Federal agency and lead agency. The applicants also include the Project Management Corporation (PMC) and the Tennessee Valley Authority (TVA). Westinghouse Electric Corporation is the lead reactor designer and manufacturer and is responsible for the overall nuclear island, reactor system, and primary heat transport system. The General Electric Company (GE) is responsible for the intermediate heat transport system and the steam generator systems. Atomics International (AI) is responsible for the fuel-handling system, maintenance, and auxiliary systems. GE is also the supplier of the turbine generator. Burns and Roe, Inc., is the architect-engineer for the project, and Stone & Webster Engineering Corporation will manage the construction.

1.5 Summary of Principal Review Matters

NRC technical review and evaluation of the information submitted by the applicants considered and will continue to consider the principal matters summarized below:

(1) The population density and land-use characteristics of the site environs and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology) have been identified to establish that these characteristics have been determined adequately and have been given appropriate consideration in the plant design, and that the site characteristics

are in accordance with NRC's siting criteria of 10 CFR 100, taking into consideration the design of the facility, including the engineered safety features provided.

(2) The design, fabrication, construction, and testing criteria, and the expected performance characteristics of the plant structures, systems, and components important to safety have been identified. The NRC review also determines that the principal design criteria (PDC), quality assurance criteria (QAC), regulatory guides (RGs) and other appropriate rules, codes, and standards have been identified and are justified.

(3) The expected response of the Clinch River Breeder Reactor Plant to various anticipated operating transients and to a broad spectrum of postulated accidents has been evaluated. On the basis of this evaluation, NRC determined that the potential consequences of a few highly unlikely postulated accidents (design-basis accidents (DBAs)) would exceed those of all other accidents considered. The staff performed conservative analyses of these DBAs to determine if the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence, would not exceed the Commission's guidelines for site acceptability given in 10 CFR 100. The staff determined that potential offsite doses of these highly unlikely DBAs would not exceed the 10 CFR 100 guideline doses. In addition the staff reviewed those events that are classified as being beyond the design basis to ensure that (a) their probabilities of occurrence are sufficiently low and (b) the reactor vessel and containment would limit the radiological consequences of such an event, should it occur, to an acceptable level.

(4) The applicants' engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the plans for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, have been reviewed to determine the applicants' technical qualifications to operate the facility safely.

(5) The design criteria of the systems provided for control of the radiological effluents from the CRBRP have been reviewed to determine the capability of these systems to control the release of radioactive wastes from the facility within the limits of the Commission's regulation in 10 CFR 20 and the operation of equipment provided in such a manner as to reduce radioactive release to levels that are as low as reasonably achievable (ALARA) within the context of the Commission's regulations in 10 CFR 50 and to meet the dose design objectives of Appendix I to 10 CFR 50.

(6) The applicants' quality assurance program for the construction of the facility has been reviewed to ensure that the program complies with the Commission's regulations in 10 CFR 50, and that the applicants will have proper controls over the CRBR construction so that there is reasonable assurance that the facility can be constructed safely and reliably.

1.6 Summary of Outstanding Construction Permit Issues

The staff has identified certain outstanding issues in its review which have not been resolved with the applicants at the time this report was issued. The staff plans to resolve these open items or to generate a schedule for their resolution. The schedule and/or resolution will be presented to the Advisory Committee on Reactor Safeguards in the April full-committee session.

(1) Review of RDT Standards F9-4T and F9-5T (3.9.9.2.3)

(2) Compliance with Regulatory Guide 1.75 (7.2.2.6)

(3) Plant Protection System Monitor (7.2.2.7)

(4) Solid-State Programmable Logic System (7.3.2.4)

(5) Emergency Planning, 10 CFR 50, Appendex E, Part II, Requirements A and B (13.3.2.1)

(6) Quality Assurance (17.3)

Table 1.1 Cross-reference table for TMI-2 Action Plan items

Item	Shortened title	SER section(s)
I.A.4.2	Long-term training simulator	13.2
I.C.5	Procedures for operating equipment feedback	13.6.1
I.C.9	Program plan for procedure development	13.5.2
I.D.1	Control room design reviews	18
I.D.2	Plant safety parameter display console	18
I.D.3	Safety system status monitoring	18
I.F.1	Expand QA list	17
I.F.2	Detailed QA criteria development	17
II.B.1	Reactor coolant systems vents	4.4 and 5.3
II.B.2	Plant shielding for postaccident access	12.1
II.B.3	Postaccident sampling	9.8
II.B.8.1	Probabilistic risk assessment	Appendix D
II.B.8.2	Dedicated containment penetration rulemaking	N/A*
II.B.8.3	Hydrogen control	Appendix A
II.B.8.4	Containment integrity	Appendix A
II.B.8.5	Evaluation of alternative hydrogen control systems	N/A**
II.D.1	Testing requirement for reactor coolant system relief valves	N/A***
II.D.3	Reactor coolant system relief valve position indication	7.5.2.11
II.E.1.1	Auxiliary feedwater system evaluation	5.6.2.2 and 15.6.2.1
II.E.1.2	Auxiliary feedwater system automatic initiation and flow indication	5.6.2.2 and 7.4.2.2
II.E.3.1	Reliability of power supplies for natural circulation	8.3.3.5
II.E.4.1	Containment design dedicated penetration	N/A†
II.E.4.2	Containment design isolation dependability	6.2
II.E.4.4	Containment design purging	6.2

See footnotes at end of table.

Table 1.1 (Continued)

Item	Shortened title	SER section(s)
II.E.5	Design sensitivity of B&W reactors	N/A††
II.F.1	Accident monitoring instrumentation	11.5
II.F.2	Recovery from conditions leading to inadequate core cooling	5.6.3
II.F.3	Instrumentation for monitoring accident conditions	7.5.2.8
II.G.1	Power supplies for pressurized relief valves, block valves, and level indicators	8.3.3.5
II.J.3.1	Organization and staffing to oversee design and construction	13.6.2
III.A.1.2	Upgrade emergency support facilities	13.3
III.D.1	Primary coolant sources outside containment	11.5.3
III.D.3.3	In-plant radiation monitoring	12.2.4
III.D.3.4	Control room habitability	6.3

* CRBR design includes systems for containment vent, purge, filtering, and heat removal; therefore, this item is not considered applicable.

** CRBR will not require an alternate hydrogen control system (see Appendix A of this SER).

***CRBR reactor coolant system relief valves will relieve cover gas only and will not affect coolant inventory; therefore, this item is not considered applicable.

† Applies only to plants with ex-containment hydrogen recombiners. Because CRBR will not have any, this item is not applicable.

†† Applies to Babcock & Wilcox light-water reactors only.

Table 1.2 CRBRP operational parameters

Thermal output (rated)	975 MW
Electrical output	
Gross (rated)	380 MW
Net	350 MW
Steam production	3.34×10^6 lb/hour
Steam temperature at turbine	900°F
Steam pressure at turbine	1,450 psi
Total primary coolant flow rate	41.5×10^6 lb/hour
Breeeding ratio	
Initial cycle	1.29
Equilibrium core	1.24
Average burnup	
Initial core	50,000 MWd/ton
Future core	80,000 MWd/ton

Table 1.3 Comparison of the proposed CRBRP with similarly sized fast reactors

Features	CRBRP	Fast flux test facility	SNR 300	MONJU	BN 350
Thermal output, MW	975	400	736	714	1,000
Electrical output, MW	350	0	282	248	150 + freshwater
Primary loops, number	3	3	3	3	5 operational 1 standby
Fuel	UO_2-PuO_2	UO_2-PuO_2	UO_2-PuO_2	UO_2-PuO_2	UO_2
Core size, liter	2,900	1,040	2,300	2,300	1,900
Reactor total flow, kg/sec	5,200	2,200	3,500	4,260	4,460
Reactor inlet temperature, °F	730	680	710	747	572
Reactor outlet temperature, °F	995	937	1,015	984	932
Average core power density, kW/liter	380	460	330	295	430
Max. fuel burnup, MWd/kg	80	80	83	100	50
Shutdown heat removal scheme	Short-term--rejects heat to atmosphere via direct steam dump from steam drum. Heat rejecting capability about 180 MWt. Long-term--condenses steam from steam drum in air-cooled condenser. Long-term capacity, about 4.5% of rated power. Backup system cools sodium from reactor overflow vessel via	Pony motor provides flow in event of primary/intermediate pump outage. Heat transport system designed to naturally circulate in event of total pump loss. Heat rejected to atmosphere through dump heat exchanger.	Emergency cooling system operates on natural circulation. 6 coolers submerged between shield tank and reactor vessel. Each cooler has separate secondary cooling circuit that has EM pumps. Total capacity, 6 Mw.	3 auxiliary systems extract heat from IHXs. Secondary loops of auxiliary coolant systems have EM pumps to maintain circulation.	Total decay heat removal capability relies on natural circulation in reactor and steam generator. Design philosphy promotes natural circulation. IHX is horizontal and above reactor core. Normal water levels in steam generator constitute 1 hour of evaporative water. Additional water added later. Calculated heat removal capability, 4-5% of heat in primary circuit.

Table 1.3 (Continued)

Features	CRBRP	Fast flux test facility	SNR 300	MONJU	BN 350
Shutdown heat removal scheme (Continued)	Na/NaK heat exchanger. NaK rejects heat to atmosphere via NaK/air heat exchanger.				
Containment concept	Confinement/containment. Concrete confinement with annulus air space surround-steel cylindrical pressure vessel with hemispherical dome and flat bottom. 186-ft diameter 160 ft bottom to spring-line. Steel shell about 1-1/2 in. thick. Concrete shielding below grade. Leak rate of steel containment at design pressure of 10 psig, 0.1% volume per day. Annulus will be maintained at negative pressure during normal operation and exhausted through filters in the event of an accident.	Single containment. Steel cylindrical pressure vessel with hemi-ellipsoidal top and bottom heads. 135-ft diameter. 187-ft--top to bottom. Steel shell 1-3/8 in. thick. Concrete shielding below grade. Leak rate at design pressure of 10 psig, 0.1% volume per day.	Double containment. Inner containment: each radioactive system contained in a concrete vault with steel liner, on inside. Vaults nitrogen inerted. Outer containment; all primary vaults contained in rectangular concrete building with steel liner on outside. Close-loop ventilated gap between the two maintained at -0.15 psig. Allows 0 radioactive release in immediate postaccident period. Design pressure, 4 psig.	Double containment configuration. Inner containment: all radioactive elements except spent fuel contained in steel-lined nitrogen-filled concrete vaults. Outer containment: steel cylinder with a hemispherical dome on top, 155-ft. diameter Con-concrete cylinder surrounds entire containment. Design pressure, 4 psig.	No containment building. Reliance on fuel element, reactor vessel, and plug. Secondary containment around reactor vessel and piping. Reactor and plug, IHX, and pumps in sealed steel-lined, inert gas-filled cells.

Table 1.3 (Continued)

Features	CRBRP	Fast flux test facility	SNR 300	MONJU	BN 350
Shutdown and control systems	2 systems. Primary system will have 9 rods, collapsible roller nut drives, spring-assisted gravity insertion. Alternate shutdown system will have 6 rods. Will have separate sensors, transmitter comparators and logic, ball nut screw drive, hydraulic-assisted insertion, below-the-head release. Primary rods will release at roller nut. Either system will be capable of reactor shutdown with most reactive rod stuck.	2 systems. 3 safety rods in inner region. 6 control rods outside safety rods. All have roller nut drive mechanisms. Either system capable of reactor scram. Rod disconnect at the roller nut.	2 systems including sensors and trip points. Primary rods lowered from top by a roller nut mechanism. 3 secondary rods inserted from bottom by release of magnet, which holds down a spring. Absorbers are flexible and can be inserted into slightly deformed channel.	3 regulatory rods (natural B_4C). 9 shim rods (slightly enriched B_4C). 4 safety rods (slightly enriched B_4C). 3 backup shutdown rods (highly enriched B_4C).	Normal operational control accomplished by 6 burnup rods, 2 regulator rods, and 1 temperature compensator rod. Scram capability provided by 3 safety rods. Each of 3 rods has independent and separate electronic circuits and sensors. Any two rods capable of reactor shutdown.

NOTE: SNR 300 is located in Germany; MONJU is located in Japan; BN 350 is located in the U.S.S.R.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

The July 1981 edition of the Standard Review Plan (SRP) (NUREG-0800) includes Section 2, "Site Characteristics." The Clinch River Breeder Reactor Plant (CRBRP) was reviewed in accordance with Sections 2.1.1, 2.1.2, and 2.1.3 of the SRP. The results of this review are contained in Section 2.1 of this safety evaluation report (SER).

2.1.1 Site Location and Description

The proposed CRBRP site is located in Roane County in east-central Tennessee approximately 25 mi west of Knoxville and within the city limits of Oak Ridge, Tennessee. The CRBRP site consists of approximately 1,364 land acres on a peninsula formed by a meander in the Clinch River. The site is bounded on the east, south, and west by the Clinch River and on the north by the U.S. Department of Energy's (DOE's) Oak Ridge Reservation. The site is shown on a general map of the region in Figure 2.1 and on a local area map in Figure 2.2.

The topography of the site area is characterized by a series of parallel ridges generally oriented in a northeast to southwest direction. Chestnut Ridge, running across the northern part of the site, is the dominant topographic feature, reaching an elevation of 1,100 ft above mean sea level (MSL). The grade elevation in the southern part of the site where the proposed plant structures will be located is 815 ft above MSL. The coordinates of the CRBRP are 35° 53' 24" north latitude and 84° 22' 57" west longitude. The universal Mercator coordinates are 39_{74} 709 north and 7_{36} 262 east.

2.1.2 Exclusion Area Authority and Control

The applicants have defined the exclusion area as shown in Figure 2.3 with a minimum distance to the exclusion area boundary of 670 m (2,200 ft) from the center of the reactor containment building.

The site property is owned by the U.S. Government and is currently in the custody of the Tennessee Valley Authority (TVA). TVA will transfer custody of 630 acres (out of 1,364 acres) to DOE for the purpose of designing, constructing, and operating the CRBRP; TVA will retain custody of the remainder of the exclusion area. After 5 years of operation, TVA may exercise an option to acquire control of the entire plant, including the exclusion area. Since both TVA and DOE (together with Project Management Corporation) are joint applicants for the CRBRP, the staff concludes that the applicants will have the authority to determine all activities within the exclusion area will not be affected by any internal changes of plant control or exclusion area custody.

The proposed exclusion area will not be traversed by any public highways or railroads; however, the Clinch River along the eastern, southern, and western boundary is included within the exclusion area. Movement on the Clinch River will be controlled in the event of an emergency by the applicants in coordination with other appropriate agencies as specified in the radiological emergency plan. The river bank on the plant site will be posted to inform river users of the nearby nuclear plant. A small family cemetery is located in the southern part of the site. Access to this cemetery will be controlled by the applicants.

The staff concludes that the applicants will have the authority to determine all activities within the exclusion area, as required by 10 CFR 100, based on the applicants' custody of the site property and the commitment to make arrangements to control traffic on the Clinch River in the event of an emergency.

2.1.3 Population Distribution

Approximately 4,440 people resided within 5 mi of the Clinch River site in 1980. This represents an increase of 1,700 persons in this area since the 1970 Census. Kingston, Tennessee, located 7 mi away in the west direction, is the largest nearby town and had a 1980 population of 4,367. Other major nearby communities are Oak Ridge, Tennessee (1980 population 27,522), located 9 mi northeast, and Knoxville, Tennessee (1980 population 182,249), located 22 mi east-northeast of the reactor site. Table 2.1 shows the current and projected populations within 30 mi of the site. Approximately one-third of the area within 5 mi of the site consists of land owned by the U.S. Government and in custody of either TVA or DOE. Although the Clinch River site is within the city limits of Oak Ridge, the residential area is located between 7 and 14 mi northeast of the site.

Transient population in the vicinity of the site, other than travelers on local roads and highways, consists primarily of workers at three large industrial activities on the Oak Ridge reservation. There are approximately 5,600 employees at the Oak Ridge Gaseous Diffusion Plant, located about 3 mi north-northwest of the site; 5,000 employees at the Oak Ridge National Laboratory, located about 4 mi east-northeast of the site; and 6,300 employees at the Y-12 plant, located about 9 mi northeast of the site. Recreational facilities within 10 mi of the site consist of numerous small camping and water access areas. The nearest recreational activities of significance are a 100-unit commercial camping site about 1 mi southeast of the site. The Atomic Speedway, a stock car track, is 3 mi east of the site. The applicants estimate that the peak hour use of the recreational facilities within 10 mi of the site totals about 10,000 persons, based on 1980 information, and is projected to increase to about 11,000 by 1990. Over 50% of these recreational visitors are attributed to spectators at the stock car track.

The staff compared the projected population density in the CRBRP site vicinity with the criteria given in RG 4.7 and Section 2.1.3. The resident plus weighted transient population density within 30 mi of the site at the projected time of plant startup (taken to be year 1990) was compared with a density of 500 persons per square mile. Figure 2.4 shows the 1990 population as a function of distance compared with a site having a uniform density of 500 persons per square mile. Similarly, the resident plus weighted transient population density within 30 mi of the site at projected end of plant life (taken to be year 2030) was compared with 1,000 persons per square mile. Figure 2.5 shows the projected year 2030 population as a function of distance compared with a site having a uniform density of 1,000 persons per square mile. From the figures it can be seen that the population density in the vicinity of the CRBR site will be well within 500 persons per square mile at time of plant startup and well within 1,000 persons per square mile at end of plant life. The weighted transient population within 10 mi was estimated to be about 7,000 persons. This was added to the 52,000 residents in generating Figures 2.4 and 2.5. Most of the transients are employees of local industries and local school population. Since a significant number are probably also people who reside within 10 mi of the plant, the staff estimates that simply adding the weighted transients to the residents introduces some double counting and that the numbers are therefore conservative. The weighting factors used were based on annual averge occupancy factors for the various groups.

In order to verify the applicants' population data, the staff obtained an independent estimate of the 1980 population within 50 mi of the site from U.S. Bureau of the Census data and compared this to the applicants' 50-mi population value for 1980. The U.S. Bureau of the Census value of 837,300 was in good agreement with the applicants' value of 830,800. The staff also compared the applicants' projected population growth rate to the year 2030 for the area within 50 mi of the site to the population projections of the U.S. Bureau of Economic Analysis for Economic Area 50, an area comprising east central Tennessee and southeastern Kentucky. This comparison showed that the applicants' population growth projection of 2.5% per decade for the area within 50 mi of the site is less than the regional growth projection of 5.6% per decade for Economic Area 50 made by the U.S. Bureau of Economic Analysis. Using these higher growth factors, the staff has also determined that the projected population is lower than the trip levels of RG 4.7. The applicants have specified a low population zone (LPZ) with an outer boundary distance of 4,025 m (2.5 mi) measured from the center of the proposed reactor location. Approximately one-third of this area consists of land within the Oak Ridge reservation and the remainder is characterized by low-density rural development with no large concentrations of population. On the basis of data presented by the applicants, the staff estimates that less than 1,500 people resided within 3 mi of the site in 1980, and the applicants project virtually no change in the LPZ population over the lifetime of the plant.

The nearest population center, as defined in 10 CFR 100, is Oak Ridge, Tennessee, which had a population of 27,552 in 1980. The population center distance, based on the actual population distribution, is 7 mi in the north-northeast direction. This distance is greater than one and one-third times the distance from the proposed reactor location to the outer boundary of the LPZ, as required by 10 CFR 100. The applicants state that future residential development of Oak Ridge will not result in population growth closer than 5 mi from the site because of present zoning restrictions. The staff concludes that the present LPZ meets the requirements of 10 CFR 100, even if the population center distance is reduced to 5 mi.

2.1.4 Conclusion

On the basis of (1) the 10 CFR 100 definitions of the exclusion area, low population zone, and population center distance; (2) the staff analysis of the onsite meteorological data from which the relative concentration factors (χ/Q) were calculated (see Section 2.3 of this SER), and (3) calculated potential radiological dose consequences of design-basis accidents (see Section 15 of this SER), the staff concludes that the exclusion area, LPZ, and the population center distance meet the criteria of 10 CFR 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

The CRBRP was reviewed in accordance with Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, and 3.5.1.6 of the SRP (NUREG-0800). The results of this review are contained in Section 2.2 of this SER.

2.2.1 Transportation Routes

The major highway in the vicinity of the site is Interstate 40, which passes approximately 1.25 mi to the south. State Route 58 is about 1.5 mi to the northwest and State Route 95 is about 3 mi east at their closest points of approach. These distances are sufficient to protect the plant from accidental detonation of explosive materials on these roads. Hazardous materials for the nearby Oak Ridge National Laboratory (ORNL) and Oak Ridge Gaseous Diffusion Plant (ORGDP) facilities are transported over these highways. An accident involving a tank truck carrying anhydrous hydrofluoric acid (AHF) has been postulated for evaluation purposes. The installation of hydrogen fluoride (HF) detectors in the control room air intakes, and the distances of these routes from the site, ensure that highway accidents involving AHF will not preclude the suitability of the site. The staff asked that the applicants examine other toxic materials being transported on I-40 near the CRBRP. Based on 1982 information obtained from the Tennessee Public Service Commission, it was determined that hydrogen bromine, hydrofluoric acid, chloropicrin, and acetic anhydride were toxic materials being transported one or more times a week past the site. Because of the quantities involved, only the first two materials listed would result in concentrations at the CRBRP control room in excess of threshold limit values. The applicants are currently evaluating the need for adding a hydrogen bromide detector to the control room air intake monitoring system. The staff concludes that appropriate detectors can be provided, as required, in the control room to safely mitigate accidental toxic gases releases, and that such accidents, therefore, will not preclude the safe operation of the plant.

The closest major rail line is approximately 10 mi northwest of the site. This distance is sufficient to eliminate potential railroad accidents as a factor in the plant design.

There is some commercial barge traffic on the Clinch River past the site. Lock records at the Melton Hill Dam, approximately 5 mi upstream from the site, indicate that over a 10-year period (1966-1975) an average of four barges per year went through the locks. The barges carried steel products primarily, and the applicants state that no explosive, toxic, or hazardous materials have been transported by barge past the site. There is a potential for increased barge traffic because of the proposed construction of coal barge loading facilities above the site. However, no hazardous materials are expected to be included in this increased barge traffic. The staff concludes that postulated accidents involving hazardous material on barges need not be considered in the design of the plant. In addition, collision of a barge with the cooling water intake structure will not affect the ability of the plant to operate safely, since water intake from the Clinch River is not essential for a safe shutdown.

2.2.2 Nearby Facilities

Industrial facilities near the proposed CRBRP consist primarily of the nuclear-related facilities on the Oak Ridge reservation. The Oak Ridge Gaseous Diffusion Plant, located about 3 mi north-northwest of the site, produces enriched uranium. Accidental release of anhydrous hydrofluoric acid stored at ORGDP has been identified as a hazard that could possibly impact on the safe operation of a nuclear plant at the Clinch River site by affecting plant operators in the control room. To evaluate this accident, it was postulated that an AHF storage tank failed and 2,000 lb of the AHF evolved as a gas over a 15-min period. Using conservative assumptions regarding meteorology and not taking credit for the buffer effect of the intervening ridges between the ORGDP release point and the Clinch River site, the concentration of HF gas at the site was determined. The analysis indicated that on the basis of the assumptions noted above, the concentration of HF gas at the Clinch River site could exceed the recommended exposure limits.

The applicants have committed to evaluate the habitability of the control room in accordance with the guidelines contained in RG 1.78 and to install HF detectors in the control room air intakes, which will alarm and automatically isolate the control room upon detection of the gas. The applicants also state that the transit time of the gas between the release point and the site would be of sufficient length to allow for communication between ORGDP and CRBRP and subsequent isolation of the control room. Communication procedures between ORGDP and CRBRP are to be included in the site emergency plan. The staff has reviewed the applicants' analysis and concludes that the potential accidental release of a large quantity of HF gas at ORGDP will not preclude the acceptability of the Clinch River site on the basis that the installation of HF detectors in the control room air intakes and adequate communication procedures will ensure the timely isolation of the control room. Section 6.4 of this SER describes the design of the toxic gas detectors in the ventilation system of the control room used to provide protection to the personnel working in the control room.

The Oak Ridge National Laboratory is located about 4 mi east-northeast of the site. The approximately 5,000 employees at ORNL are engaged in basic and applied research in nuclear and other technologies. The Y-12 Plant, which employs about 6,300 persons, is located 9 mi northeast of the site. Production as well as research and development facilities are provided at Y-12 for DOE. No activities have been identified at either ORNL or Y-12 which constitute a hazard to the safe operation of a nuclear plant at the Clinch River site.

One small industrial facility is located on a 33-acre tract in the 112-acre Clinch River Consolidated Industrial Park (CRCIP) along the northern boundary of the site approximately 1.5 mi from the proposed location of the plant structures. This industry employing 30 people, fabricates neutron absorbers for power reactors and fuel elements for test reactors. This activity is not considered to be hazardous with respect to the construction and operation of the CRBR.

The closest airports to the site are two light-plane facilities located about 10 mi from the site. McGhee-Tyson (Knoxville), located 28 mi east-southeast of the site, is the closest major airport with scheduled commercial flights. The nearest flight path is V16 between Knoxville and Hinch Mountain, which passes about 10 mi south of the site. The staff concludes these aviation facilities are far enough away to ensure a low likelihood of an aircraft crash adversely affecting the safe operation of the plant.

The nearest pipeline is a 6-in. natural gas pipeline which runs in a north-south direction and passes about 1.3 mi east of the proposed location of the plant structures. Because of the relatively small size of the pipeline and its distance from the site, the staff concludes that this pipeline will not affect the safe operation of the plant even if in the future a more hazardous gas such as propane were added to the natural gas in the pipeline.

The applicants state that there are no oil refineries or storage facilities, quarries, or mineral extraction operations in the vicinity of the site. There are no military bases or facilities within 10 mi of the site.

In order to evaluate the potential impact on the CRBR of the possible future expansion of existing facilities and the development of new DOE programs, the applicants asked the Oak Ridge Operations Office to survey the present activities and to establish the basis for a long-range land-use plan for the Oak Ridge reservation. The results of this survey, and DOE site selection and impact evaluation requirements, provide reasonable assurance that potential new activities on DOE-controlled land will not impose an undue risk on the safe operation of the CRBRP.

In addition to the projected DOE programs, the Exxon Nuclear Company had requested a 2,500-acre site on the Oak Ridge reservation for storing and reprocessing spent fuel and had submitted an application to the NRC to construct this facility. The Exxon site was to be located approximately 2.5 mi north-northeast of the Clinch River site. Plans for this facility have been terminated and the application was withdrawn.

2.2.3 Conclusions

The staff review has been conducted against the criteria given in GDC 4 and SRP Section 2.2.3. The staff concludes that the plant is adequately protected and can be operated with an acceptable degree of safety taking into consideration activities at nearby facilities.

2.3 Meteorology

Evaluation of regional and local climatological information, including extremes of climate and severe weather occurrences which may affect the design and siting of a nuclear plant, is required to ensure that the plant can be designed and operated within the requirements of Commission regulations. Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required for a determination that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the Standard Review Plan (NUREG-0800), utilizing information presented in Section 2.3 of the PSAR, responses to requests for additional information, and generally available reference materials as described in the appropriate section of the Standard Review Plan.

2.3.1 Regional Climatology

Located in the broad valley between the Cumberland and the Great Smoky Mountains, the region has a climate strongly influenced by its topography. In winter the Cumberland Mountains retard and weaken cold winter air penetrating from the north. Thus, wintertime high temperatures average near 10°C (50°F); low temperatures will be about 1 to 2°C (mid-30's°F). Temperatures of 0°C (32°F) or lower normally occur on 82 days annually. In summer, the semipermanent high-pressure system of the western Atlantic pushes warm, moist air from the Gulf of Mexico into the region. Again, the mountains moderate temperatures, and summer maximums are near 30°C (86°F). Temperatures in the valley will exceed 32°C (90°F) about 30 days per year, compared with areas west of the mountains where this temperature may be exceeded more than 50 days each year.

The region receives about 1,300 mm (50 in.) of precipitation annually. Precipitation is greatest during winter and early spring and lowest in early autumn. A secondary precipitation maximum, associated with thundershower activity, occurs in July. Thundershowers occur on an average of 53 days a year; approximately 60% of the thunderstorm days occur from May through August. Hail may be associated with thundershowers on 2 to 4 days of the year.

Snowfall accounts for a small part of the total area precipitation, and averages about 280 mm (11 in.) annually. Freezing rain may occur 5 days each year. Between 1953 and 1974, 54 tornadoes occurred within a 10,000-mi^2 area containing the site; this resulted in a mean annual tornado frequency of 2.5 and a recurrence interval for a tornado at the plant site of 1,450 years. The design-basis tornado characteristics selected by the applicants conform to the recommendations of RG 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country. These characteristics are: rotational speed, 290 mph; translational speed, 70 mph; and a total pressure drop of 3 psi occurring at a rate of 2 psi per second. Remnants of hurricanes or tropical storms occasionally affect the area; during the period 1871-1973, nine tropical storms or hurricanes passed through the region. The "fastest mile" of wind recorded in the area has been 33 m per second (73 mph), at Knoxville, Tennessee, in July 1961. An operating basis windspeed (defined as the "fastest mile" windspeed at a height of 30 ft with a return period of 100 years) of 90 mph has been selected by the applicants for the design of the plant.

The emergency cooling tower design is based on the worst 1-day and worst 30-day periods of record in accordance with RG 1.27. The region is expected to experience atmospheric stagnation about 8 days per year. The autumn months had the highest frequency of cases. The maximum snow and ice loading of 40 psf is acceptable for the roofs of safety-related structures. As discussed above, the staff has reviewed available information relative to the regional meteorologial conditions of importance to the safe design and siting of this plant. On the basis of this review, the staff concludes that the applicants have identified and considered appropriate regional meteorological conditions in the design and siting of this plant and, therefore, meet the requirements of 10 CFR 100.10 and PDC 2 (the equivalent of GDC 2 in 10 CFR 50, Appendix A). The design-basis tornado characteristics selected by the applicants conform to the position set forth in RG 1.76 and, therefore, meet the requirement of PDC 5 (the equivalent of GDC 4 of 10 CFR 50, Appendix A) to determine an acceptable design-basis tornado for missile generation.

2.3.2 Local Meteorology

To assess the local meteorological characteristics of the Clinch River site, climatological data from Oak Ridge, Tennessee (10 mi northeast of the site), and Knoxville, Tennessee (25 mi east-northeast), and data collected on site are available.

At Oak Ridge average daily maximum and minimum temperatures range between 31° and 19°C (88° and 66°F) in July, the warmest month, and between 8° and -2°C (47° and 29°F) in January, the coolest month. The extreme maximum temperature recorded was 41°C (105°F); the extreme minimum temperature has been -23°C (-9°F).

Oak Ridge receives about 1,350 mm (53 in.) of rain annually; monthly precipitation exceeds 125 mm (5 in.) between December and March; precipitation is at a minimum during September and October when monthly averages are less than 90 mm (3.5 in.). The maximum 24-hour recorded rainfall has been 190 mm (7.5 in.) in August 1960. Annual snowfall averages 250 mm (10 in.) at Oak Ridge. The maximum 24-hour snowfalls have been 300 mm (12 in.) at Oak Ridge (Mar. 1960) and 460 mm (18 in.) at Knoxville (Nov. 1952).

Heavy fog (visibility 0.25 mi or less) occurs on about 34 days annually at Oak Ridge. Such occurrences may be more frequent at the plant site, which is nearer the river.

Wind data taken from the 10-m level of the onsite permanent meteorological tower for the 1-year period February 17, 1977 through February 16, 1978 indicate prevailing winds from the west-northwest, west, and west-southwest with an annual combined frequency of approximately 29%. The average windspeed at the 10-m level is about 2.5 m per second (5.7 mph) and calm conditions occur less than 0.5% of the time. Neutral (Pasquill type D) and slightly stable (Pasquill type E) conditions occur most frequently, about 36% and 26% of the time, respectively. Moderately stable (Pasquill type F) and extremely stable (Pasquill type G) conditions occur about 15% and 14% of the time, respectively.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant. On the basis of this review, the staff concludes that the applicants have identified and considered appropriate local meteorological conditions in the design and siting of this plant and, therefore, meet the requirements of 10 CFR 100.10 and PDC 2 (GDC 2).

2.3.3 Onsite Meteorological Measurements Program

Since April 1973, a temporary 200-ft instrumented tower had been in operation southwest of the reactor site. In February 1977, two instrumented towers were installed: a 10-m tower south of the site and a permanent 110-m tower southeast of the site.

Simultaneous measurements were taken on the temporary tower and the two newer towers during the period February 16, 1977 to March 2, 1978. In April of 1982 the 10-m and 110-m towers were reinstrumented and was placed back in operation. The data acquisition equipment is located in a trailer at the base of the 110-m tower with data from the 10-m tower being telemetered to this same location. The 10-m tower instrumentation consists of windspeed and wind-direction sensors. The 110-m tower instrumentation consists of windspeed and wind-direction sensors located at the 10-, 60-, and 110-m levels; temperature sensors at the 10-, 60-, and 110-m levels; dew point sensors at the 10-m level; and solar radiation and precipitation sensors at the 1-m level. Data from these systems are recorded by a digital system interfaced with a NOVA 1200 Minicomputer and peripheral equipment. Wind-direction and windspeed values are also recorded by an analog system. A calibration program for the sensors is in effect. The applicant has initiated an adequate data reliability program. In spite of the fact that meteorogical data have been collected on site from 1973 through early 1978, the staff has only utilized the data collected on the present permanent 110-m tower during the period February 17, 1977 through February 16, 1978 for analysis purposes because the earlier meteorological measurements programs either did not conform with RG 1.23 sensitivity-of-data-recovery guidance, had questionable exposure, or required extrapolation of winds from higher levels to the 10-m level. The joint data recovery rate of the 10-m windspeed and wind direction and the temperature difference between 10- and 60-m levels on the permanent tower during the period February 17, 1977 through February 16, 1978 was 97%.

The onsite meteorological measurement system conforms to the guidance of RG 1.23 and has provided adequate data to represent onsite meteorological conditions as required in 10 CFR 100.10. The onsite data provide an acceptable basis for making conservative estimates of atmospheric dispersion conditions used for estimating consequences of design-basis accident and routine releases from the plant.

2.3.4 Short-Term (Accident) Diffusion Estimates

To audit the applicants' estimates, the staff has performed an independent assessment of short-term (less than 30 days), accidental releases from buildings and vents using the direction-dependent atmospheric dispersion model described in RG 1.145, with consideration of increased lateral dispersion during stable conditions accompanied by low windspeeds. Onsite data for 1 year (Feb. 17, 1977 through Feb. 16, 1978) collected on the permanent CRBRP tower were used for this evaluation. Windspeed and wind direction were measured at the 10-m level, and atmospheric stability was defined by the vertical temperature gradient between the 10-m and 60-m levels. A ground-level release with a building wake factor, cA, of 1,208 m was assumed. The relative concentration (χ/Q) for the 0 to 2-hour period was determined to be 1.22×10^{-3} sec/m^3 at a distance of 670 m. The χ/Q values for appropriate time periods at the outer boundary of the low population zone (4023) are:

Time period	χ/Q (sec/m^3)
0-8 hours	1.2×10^{-4}
8-24 hours	8.4×10^{-5}
1-4 days	3.9×10^{-5}
4-30 days	1.4×10^{-5}

The applicants have calculated χ/Q values (generally less than 10% difference) comparable to those calculated by the staff. On the basis of the above evaluation, the staff concludes that the applicants have considered appropriate atmospheric dispersion estimates for assessments of the consequences of radioactive releases for design-basis accidents in accordance with the requirements of 10 CFR 100.11. The atmospheric dispersion estimates provided in this section have been used by the staff in an independent assessment of the consequences of radioactive releases for design-basis accidents.

2.3.5 Long-Term (Routine) Diffusion Estimates

To audit the applicants' estimates, the staff has performed an independent calculation of annual average relative concentration (χ/Q) and relative deposition (D/Q) values using the straight-line Gaussian atmosperic dispersion model described in RG 1.111. Continuous releases only were evaluated and all releases were assumed to be at ground level. The calculations also included an estimate of the maximum increase in calculated relative concentration and deposition resulting from recirculation of air flow not considered in the straight-line trajectory model. The staff and applicants utilized 1 year of onsite data (Feb. 17, 1977 through Feb. 16, 1978) collected on the CRBRP permanent meteorological tower for their long-term, routine release, diffusion estimates.

On the basis of the above evaluation, the staff concludes that the applicants have considered representative atmospheric dispersion estimates for demonstrating compliance with the numerical guides for doses contained in 10 CFR 50, Appendix I. The atmospheric dispersion estimates developed by the staff are included in the assessment of the radiological impact to man resulting from routine releases to the atmosphere contained in the staff's environmental statement.

2.4 Hydrologic Engineering

2.4.1 Introduction

The staff has reviewed the hydrologic engineering aspects of the applicants' design, design criteria, and design bases of safety-related facilities at the Clinch River Breeder Reactor Plant. The acceptance criteria include the applicable PDC, reactor site criteria (GDC in 10 CFR 50), and standards for protection against radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in regulatory guides, ANSI standards, and branch technical positions identified in SRP Sections 2.4-1 through 2.4-14. Conformance to the acceptance criteria provides the bases for concluding that the site and facilities will meet the requirements of 10 CFR 20, 50, and 100 with respect to hydrologic engineering.

2.4.2 Hydrologic Description

The proposed site is located in Roane County, Tennessee, on the north shore of the Clinch River, and about 25 mi west of Knoxville, Tennessee. Proposed plant grade will be about 815 ft mean sea level (MSL).

The main surface hydrologic feature of the site is the Clinch River. The Clinch River has a drainage area of about 16,200 mi^2 and originates in Tazewell County,

County, Virginia, about 330 river miles upstream from the site. The river is regulated by a series of dams both upstream and downstream from the site. Norris Dam, about 62 river miles upstream from the site, impounds up to 2.3×10^6 acre-feet. Melton Hill Dam, which is about 6 river miles upstream from the site, impounds up to 31,500 acre-feet. The Clinch River flows into the Tennessee River, about 16 river miles downstream from the site.

The entire reach of the Clinch River below Melton Hill Dam is in the backwater of Watts Bar Dam about 55 mi downstream on the Tennessee River. Water level at the site is regulated by Watts Bar Dam, Melton Hill Dam, and Fort Loudoun Dam, which is on the Tennessee River upstream of its confluence with the Clinch River. Normal pool elevations range from 735 to 741 ft MSL at the site, depending on the season.

The annual average flow past the site is about 5,380 ft^3 per second since closure of Melton Hill Dam in 1963. The minimum recorded flow past the site was zero, during a period of water treatment in Melton Hill Reservoir to control Eurasian Water Milfoil growth. Future operation of the reservoirs will be coordinated with the operation of the plant to guarantee a minimum flow.

The applicants have identified several municipal and industrial users of surface water that could be affected by the discharge from the plant. The nearest industrial user is about 1.6 mi downstream from the site. The nearest municipal user that could be affected by the plant is on the Little Emory River, a tributary of the Clinch River, where the possibility of occasional reverse flow exists because of operation of the dams.

There is heavy recreational use of the Clinch River during the summer months. The applicants have not identified any irrigation usage of the Clinch River below the site.

Within 20 mi of the site, there are 17 public water supplies using groundwater, and also numerous private wells. Total public and industrial groundwater use within 20 mi of the site is estimated to be about 5×10^6 gpd. The plant will use no groundwater during operation.

The applicants have provided hydrologic descriptions of the site. The staff has reviewed the applicants' information in accordance with procedures in SRP Sections 2.4.1 and 2.4.2, and concludes that it is sufficient to meet the requirements of PDC 2.

2.4.3 Flood Potential

The applicants considered the potential for flooding of the site from several sources; this was independently verified by the staff, consistent with the criteria of RG 1.59, "Design Basis Floods for Nuclear Power Plants." The maximum flood of record, judging from gage records and newspaper accounts, occurred in March 1886, with a reported water level of 764 ft MSL at the site. This flood occurred before construction of the present, extensive TVA dam system. Since completion of the system of dams in March 1973, the maximum water level at the site has been about 750 ft MSL, which is about 65 ft below plant grade. The applicants have estimated that a repetition of the worst flood of record, but with the present system of dams, would yield a water level of about 751 ft MSL, 64 ft below plant grade.

2.4.3.1 Precipitation-Induced River Flooding

The applicants have evaluated the effects of the probable maximum flood (PMF) (defined in RG 1.59) on the Clinch River. The analysis included the overtopping and subsequent failures of Fort Loudoun Dam and Watts Bar Dam on the Tennessee River and Melton Hill Dam on the Clinch River. The failures of Fort Loudoun Dam and Melton Hill Dam increase the estimate of water level at the site; the failure of Watts Bar Dam decreases the estimate, but to modest degrees. Norris Dam would not fail for hydrologic reasons during the PMF. The estimated maximum stillwater level at the site would be 778 ft MSL, 37 ft below plant grade. Wind wave runup from an attendant 40-mph wind would add a maximum of about 4 feet against vertical surfaces. This flood would not be a threat to the site, and is well below the design-basis flood that would be caused by the assumed seismic failure of Norris Dam, as explained in Section 2.4.3.3.

The staff has reviewed the PSAR material presented in accordance with the procedures described in SRP Sections 2.4.2 and 2.4.3 and concludes that the plant meets the requirements of PDC 2 with respect to flooding of streams and rivers caused by heavy precipitation.

2.4.3.2 Intense Local Precipitation

At the staff's request, the applicants have committed to design all storm water discharge facilities, including site drainage, to be capable of withstanding the local probable maximum precipitation (PMP). The design flood will be the 8-hour PMP defined for the Tennessee River Basin.

Roofs will be designed for up to 8 in. of ponding, and will have scuppers to provide for runoff beyond the design levels.

The overall site drainage system will be designed to prevent ponding levels of greater than 6 in. above site grade. All exterior openings in safety-related buildings will be at least 1 ft above site grade.

The staff concurs with the applicants' criteria for the design of the site to withstand the effects of intense local precipitation. Since the final design of features such as roof scuppers, storm sewers, and site grading is not complete at this time, the final staff review must be deferred until the operating license stage.

2.4.3.3 Potential Dam Failures

The applicants investigated several permutations of dam failure concurrent with precipitation-induced floods. Two basic conditions were investigated: (1) a 25-year flood combined with dam failures resulting from the safe shutdown earthquake (SSE) and (2) one-half the PMF coincident with dam failures from the operating basis earthquake (OBE). Reservoirs were all assumed full. The applicants investigated the failures, both individually and in groups, of 11 dams that could influence water levels at the site. The design-basis flood for the proposed site was determined to be caused by the partial seismic failure of

Norris Dam resulting from the OBE, coincident with one-half of the PMF flow (defined in RG 1.59), with the attendant hydrologic failures of the Melton Hill Dam and Watts Bar Dam.

Norris Dam is a large concrete gravity dam, which completely fills a valley on the Clinch River, about 62 mi upstream from the site. The complete disappearance of the dam is not a credible event, but it is conservatively assumed to partially fail by overturning resulting from seismic and hydraulic forces.

Stage-discharge rating curves for Norris Dam in several different failed positions were generated with a scale hydraulic model. Several of the monolithic blocks from which the dam is constructed were assumed to fall over in the downstream direction, but to largely remain where they fall, thereby partially blocking the channel and impeding the river flow. The final state of the debris is speculative, but the design-basis flood was based on reasonable assumptions of the debris configuration.

The mode of dam failure determined by the applicants for the OBE was a 665-ft section of blocks tipped over downstream, with a debris level of 970 ft MSL. Outflow from this assumed dam failure was combined with one-half of the PMF discharge. This would result in the failure of Watts Bar Dam, which would not, however, occur until after the peak water level would be reached at the site, and therefore does not affect the analysis. The maximum stillwater level at the site for this event has been estimated by the applicants to be 804 ft MSL, about 9 ft below plant grade. Maximum wave runup caused by 40-mph winds would add an estimated 5 ft at vertical surfaces to 809 ft MSL, still 6 ft below plant grade. This analysis of flood levels at the site was performed using an unsteady-flow computer model.

The staff has verified the applicants' estimate for flooding at the site using the assumed design-basis failure mode for Norris Dam. Aspects of the simulations were repeated and sensitivity to permutations of the computer model parameters was determined.

The applicants have performed experiments to determine the sensitivity of flood levels at the site for various assumed failure modes of Norris Dam. These sensitivity studies have been presented orally to the staff and also to the Advisory Committee on Reactor Safeguards in connection with the Site Suitability Report. At the request of the staff, the applicants have included the results of these hydraulic model and mathematical model sensitivity studies in an amendment to the PSAR. The applicants' sensitivity studies considered several arbitrary dam failure modes more severe than that chosen for the design basis, including complete disappearance of Norris Dam. These dam failure modes were not determined by any rational mechanism, but were performed strictly as sensitivity experiments. Except for the case of complete disappearance of Norris Dam, which the staff does not consider to be a credible event, the predicted stillwater levels at the site determined from these studies were several feet below plant grade. Waves that could theoretically run up on vertical surfaces above the level of site grade would not endanger the plant because all safety-related buildings will be located on the nuclear island above the stillwater level, and well removed from any effects of waves coming from the river.

On the basis of its review of the PSAR according to SRP Section 2.4.4, the staff concludes that the plant will meet the requirements of PDC 2 with respect to flooding from potential dam failures.

2.4.3.4 Ice Flooding

The staff has concluded that there is no hydrologically related hazard to the plant from the effects of ice in the Clinch River. Significant amounts of ice would not form in the Clinch River because of the temperate climate and frequent changes of water level in the reservoirs. Because the site is more than 70 ft above the normal Clinch River level, an ice jam sufficient to cause plant flooding is not credible. Damage to the river intake or discharge structures is extremely unlikely. In the event of any such damage, however, there is sufficient onsite water storage for plant shutdown, without need of the river structures.

On the basis of its review of the PSAR accidents according to SRP Section 2.4.7, the staff concludes that ice will not cause the design-basis flood and that the plant will meet the guidelines of RG 1.102 and the requirements of PDC 2 with respect to ice flooding.

2.4.5 Cooling Water Supply

2.4.5.1 Description of Normal and Emergency Supply

Normal circulating water will be cooled by mechanical draft cooling towers. Makeup water for the towers will be withdrawn from the Clinch River at an annual average rate of 13.5 ft^3 per second and stored in the cooling tower basins. Of this withdrawal, an average of 5 ft^3 per second will be returned to the river as blowdown, and 8.3 ft^3 per second will be lost as evaporation and drift from the towers.

Water supply from the Clinch River will be withdrawn from two submerged perforated pipe intake structures located about 26 ft from shore. The intake structure has been designed to provide a reliable water supply at the normal minimum anticipated water level of 735 ft MSL, and to withstand a maximum water level of at least 750 ft MSL, the maximum water level seen at the site since the completion of Melton Hill Dam in 1963. Discharge to the river will take place through a simple submerged pipe protruding at a right angle from shore. Both the intake and discharge structures will be protected by concrete aprons.

Although the intake and discharge structures must be considered inoperable during or after severe flooding, neither will be safety related or necessary for the safe shutdown of the plant. Emergency cooling water will be provided by a two-cell seismic Category I mechanical draft cooling tower. Either cell of the tower will be able to safely dissipate the entire heat load of shutdown under the most severe meteorological heat transfer conditions. The emergency tower will have sufficient water in its self-contained seismic Category I basin to operate for a period of at least 30 days under meteorological conditions of maximum evaporation without the need for additional makeup supply.

Final sizing of the emergency cooling tower and basin has not yet been completed, but the applicants will follow the criteria suggested by RG 1.27, "Ultimate Heat Sinks for Nuclear Power Plants." Long-term meteorological data from Oak Ridge Township will be used in the design.

2.4.5.2 Adequacy of Cooling Water Supply

The water level at the site is controlled by operation of the upstream and downstream dams. Sufficient water would be available from Norris reservoir alone to maintain a flow past the site of 710 ft^3 per second for a minimum period of 12 months without any inflow to the reservoir. Plant consumption under normal conditions will be only about 8.3 ft^3 per second. Use of the river intake structure could be lost because of low water level resulting from the unlikely total loss of Watts Bar Dam. It would take several hours for the water level to drop below a usable level at the site, however. There are no safety-related implications of this event or some other event such as the unlikely blockage of the intake by ice, since river water will not be required for plant shutdown. Emergency cooling water will be provided by an onsite seismic Category I cooling tower, with self-contained water storage adequate for plant shutdown.

On the basis of its review, using the procedures described in SRP Sections 2.4.5, 2.4.7, and 2.4.11 and the guidance of RG 1.27, the staff concludes that with respect to low water levels and ice blockages of the water intakes, the plant design is acceptable and will meet the requirements of PDC 2. The staff further concludes that the cooling water supply for the plant will meet the requirements of PDC 2 with respect to hydrologic characteristics. The applicants have not completed final sizing of the emergency cooling tower and water storage basin. The staff will review the emergency cooling and water supply adequacy at the OL stage.

2.4.6 Groundwater

Groundwater occurs at the proposed site primarily in weathered joints and fractures in the subsurface carbonate rock. Borings made at the proposed site indicate that the elevation of the top of continuous rock lies at about 700 ft MSL. The Clinch River bounds the groundwater system and serves as a groundwater sink. Discharge from the aquifer flows directly into the Clinch River and its tributaries.

Piezometric levels at the site measured in January 1974 range from about 770 ft MSL near the nuclear island to 800 ft MSL near the emergency cooling tower. These levels fluctuate from 10 to 25 ft, indicating rapid local recharge from precipitation. Excavation of the site and placement of permeable fill around structures will alter this recharge somewhat. The present groundwater level is above the nuclear plant island foundation level of 720 ft MSL and the emergency cooling tower foundation level of 765 ft MSL. The design-basis level for subsurface hydrostatic loading is 809 ft MSL, which corresponds to the applicants' estimate of the surface water level with runup at the site caused by the design-basis flood in the Clinch River. The design-basis piezometric level for the dynamic analysis of seismic Category I structures is 780 ft MSL, corresponding approximately to the stillwater level plus wave runup caused by the probable maximum flood. The staff agrees that the 809 ft MSL groundwater level would be conservative for the analysis of hydrostatic loadings on safety-related structures.

However, the applicants' choice of the 780 ft MSL piezometric level for dynamic loading required further justification since groundwater levels well above 780 ft MSL have been observed during site monitoring. Moreover, site grading and the placement of permeable fill will alter the groundwater hydrology of the site, but it was unclear whether the net effect of these procedures would be to raise or lower the peak groundwater level for seismic Category I structures. The staff, therefore, asked that the applicants justify why a groundwater level of 780 ft MSL would be conservative for dynamic loading analyses of seismic Category I structures.

The applicants responded, in an amendment to the PSAR, to the staff's requests for additional information on groundwater levels. Currently measured high groundwater levels in the vicinity of the emergency cooling tower are explained by high topographic ridges in the near vicinity, which will be removed when the site is graded. The net effect of the site excavations will be to reduce water levels considerably below present levels, much more characteristic of the rest of the site. The staff, therefore, concludes that the groundwater level of 780 ft MSL would be conservative for the analyses of seismic Category I structures. It will, however, require that the preconstruction estimates of groundwater level be confirmed for the plant at the operating license stage, after all major construction has been completed and the effects of construction dewatering have disappeared.

Groundwater at the site generally moves from topographic highs to topographic lows. Groundwater recharges over a wide surface area, but especially from rock outcroppings and sinkholes. The Clinch River acts as the groundwater sink most of the time. The river channel is in the bedrock, and it is unlikely that any groundwater passing the site could go under the river. Under cases of rapid water-level change, the Clinch River may recharge the water table, temporarily reversing the flow close to shore. This recharge does not affect the long-term trend of groundwater movement, however.

The applicants conducted a survey within a 2-mi radius of the plant site, and found 110 wells and springs. There are 17 public and 7 industrial water supplies that draw on groundwater within a 20-mi radius of the site. Most groundwater use is for single family dwellings and agriculture. Public water supplies within 20 mi of the site withdraw about 4.5×10^6 gpd. Industrial users withdraw about 7×10^5 gpd.

The staff review was based on the guidance of SRP Section 2.4.12. The staff has determined that emergency shutdown of the plant will not depend on groundwater supplies, and that safety-related structures have been designed to conservative groundwater levels for hydrostatic loading analyses and will meet the requirements of PDC 2. The staff further concludes that the site meets the requirements of 10 CFR 100 and Appendix A thereto, 10 CFR 50, and PDC 44 with respect to the effects of groundwater on the site.

2.4.7 Accidental Releases of Liquid Effluents in Ground and Surface Waters

No groundwater users could be adversely affected by an accidental contamination of groundwater at the site. All groundwater users are located upgrade of the site, or on the opposite side of the Clinch River, which is the groundwater sink.

Contamination from an accidental radioactive release to the groundwater would eventually enter the Clinch River.

The staff has evaluated the hydrologic consequences of a postulated radioactive liquid release from the proposed plant. Its analysis conservatively assumed that the entire volume of the intermediate activity level system waste holding tank, 20,000 gal, instantaneously entered the site aquifer, and moved downgrade to the Clinch River. The groundwater travel time would be a minimum of 30 years. This analysis conservatively neglects the sorption of many of the radioactive substances on the soils and rocks of the aquifer and the sediments in the river. The released material would be further diluted by the annual average 4,600-ft^3-per-second flow upon entering the Clinch River. The nearest user is about 9,000 ft downstream of the point at which the groundwater would enter the river.

The staff has estimated that the dilution of the contents of the tank at the nearest downstream point of use would be a minimum of 7×10^6. Even with such extreme conservatism, none of the materials released would exceed the concentrations of 10 CFR 20, Appendix B, Table II.

The staff, therefore, concludes that releases of liquid radioactivity from accidents within the design basis would not pose a threat to public health and safety, and that the plant will meet the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluent. The staff relied on the guidance of SRP Sections 2.4.12 and 2.4.13, RG 1.113, 10 CFR 20, and 10 CFR 100 in performing its analysis.

2.4.8 Technical Specifications and Emergency Operation Requirements

On the basis of its review in accordance with SRP Section 2.4.14, the staff has not identified any hydrologic technical specification or emergency operating plan necessary for the safe operation or shutdown of this plant. In the unlikely event that water from the Clinch River would be unavailable, the ultimate heat sink cooling tower would be used to provide service water for plant shutdown without external water supplies.

2.4.9 Conclusions

The staff concludes that there are no unique hydrologic phenomena related to site flooding, and that the proposed site criteria are consistent with RG 1.59 and the requirements of PDC 2 with respect to flooding.

An adequate water supply can be provided for normal and emergency cooling that will meet the criteria of RG 1.27 and, therefore, will meet the hydrologic criteria of PDC 44.

Surface water and groundwater contamination from unplanned releases of liquid radwaste will result in no adverse effects on users and will, therefore, meet the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluent.

The staff concludes that groundwater levels for subsurface loadings are conservative in accordance with the requirements of PDC 2, but it will require that the preconstruction estimates be confirmed for the plant as built before the operating license is granted.

2.5 Geology and Seismology

2.5.1 Basic Geologic and Seismic Information

The U.S. Atomic Energy Commission (AEC) geosciences staff initially reviewed information pertinent to the Clinch River breeder reactor (CRBR) during the early and middle 1970s. A team consisting of a geologist, seismologist, and geotechnical engineer reviewed site information included in the PSAR, the published literature, and DOE documents regarding the waste disposal injection well program at Oak Ridge, Tennessee. Several reconaissances were made of the site and region around the site. During that review four major issues were identified.

(1) the appropriate SSE for the site, which is located in the Southern Valley and Ridge Tectonic Province

(2) the capability (Appendix A, 10 CFR 100) of faults in the site vicinity

(3) the potential for solution cavities in foundation bedrock

(4) the possibility of earthquakes induced by deep well waste injection on the Oak Ridge reservation 4 mi east of the site.

On the basis of that review, the staff concluded (NRC, Site Suitability Report in the Matter of CRBRP, 1977):

(1) The appropriate SSE design basis should be based on the assumption that the 1897 Giles County, modified Mercalli intensity (MMI) VIII earthquake could occur near the site and the RG 1.60 response spectra anchored at 0.25 g would be an appropriate representation of this earthquake.

(2) Faults at the site and in the vicinity of the site are not capable within the meaning of Appendix A, 10 CFR 100.

(3) There are no significant solution cavities in the foundation rock beneath the plant.

(4) Disposal of radioactive wastes 4 mi east of the site will not increase the earthquake potential at the site.

During its CP review, the NRC staff concentrated on evaluating the considerable amount of new information that has been developed since the mid-1970s, both in the scientific literature and from data obtained from additional investigations at the site. Based on its review of other nuclear sites in the Valley and Ridge Tectonic Province, such as Sequoyah, Watts Bar, Phipps Bend, and Bellefonte, the published scientific literature, and the amended PSAR, the staff concludes that the conclusions reached during the earlier review are still valid.

As advisors to the staff, the U.S. Geological Survey (USGS) has completed its review of the geology and seismology of the CRBR site (Devine, 1983*). The

*See Appendix H for copy of Devine's report.

USGS concludes that the applicants' conclusions with regard to site geology are reasonable in light of current theories of Appalachian tectonics and available data, although the applicants' demonstration of noncapability of faulting was not as definitive as possible. They also concur with the SSE and consider it reasonable, indicating however that a definitive seismological investigation could address the problem of a postulated seismic source zone in eastern Tennessee (see Section 2.5.2.3 of this SER).

The staff concludes that the faults at the site and in the region around the site are not capable. There are, however, additional data which might, if appropriate exposures are available, be utilized to confirm that conclusion. High terraces of probable Pleistocene age are relatively common in the site region. These terraces were used by the Tennessee Valley Authority (TVA) in the Phipps Bend (1975) and Watts Bar (Apr. 1974) geologic investigations to demonstrate, along with other data, that local Valley and Ridge Faults are not capable. It is the staff's opinion that it would be prudent for the CRBR applicants to investigate similar terraces in the vicinity of the site. This should be done by locating terraces in the region of the site where there is a high likelihood that they overlie faults. These terraces should be mapped and the cross-cutting relationships between them and the faults should be determined. Additionally, the applicants should map in cross-section the large terrace in the southeast section of the peninsula on which the site is located. Although no faults are recognized there, it is likely that minor tectonic structures will be found because of the proximity of the Copper Creek Fault. The staff regards this investigation as confirmatory and recommends that it not delay issuance of the construction permit.

During its current review, in addition to reaffirming its former conclusions, the staff identified the following issues:

(1) the significance of new data that have been gathered about sources of eastern seismicity including the 1886 maximum MMI X Charleston earthquake

(2) the significance to the site of the results of work by G. A. Bollinger (1981) in the epicentral area of the 1897 MMI VIII Giles County earthquake.

One of the working hypotheses that has been developed from the research performed in the southeastern United States over the past decade, particularly in the Charleston, South Carolina, region, is that eastern seismicity may be related to a master thrust fault or detachment surface between 4 and 13 km beneath the Piedmont, Blue Ridge, and Valley and Ridge Provinces. If this is true, the low angle thrust faults of the Valley and Ridge Province may be related to it. The staff has formulated an interim position concerning eastern seismicity in general and Charleston seismicity in particular (Devine, 1983). The staff is addressing the uncertainties about eastern U. S. seismicity including a decollement (detachment surface) hypothesis by means of deterministic studies funded by the NRC and probabilistic studies that are being funded by NRC and conducted by Lawrence Livermore National Laboratory.

Because of the speculative nature of all of the eastern seismicity hypotheses, the low probability associated with large earthquakes occurring in the eastern United States, and knowledge of the geology and seismology of the region gained from the literature and review of other sites (Sequoyah, Watts Bar, Bellefonte, Phipps Bend), the staff considers the CRBR seismic design basis to be appropriate. At the conclusion of the shorter term probabilistic research program and during the longer term deterministic studies, the staff will be assessing need for a modified position with respect to specific sites if warranted.

The new information on the Giles County earthquake area is discussed in Section 2.5.2. After reviewing that information the staff concludes that it will have no adverse impact on the seismic design bases of the site. The deterministic and probabilistic studies concerning seismicity of the eastern seaboard that are described above will include consideration of the Giles County earthquake and the possibility of other seismogenic zones nearer to the site.

The staff concludes that the applicants have satisfied the requirements of 10 CFR 100, Appendix A. It also finds that the PSAR conforms to the applicable sections of

(1) SRP (NUREG-0800) Sections 2.5.1, 2.5.2, and 2.5.3
(2) RG 1.70, Rev. 2
(3) RG 1.60, Rev. 1

On the basis of its review of the PSAR and pertinent documents from the published scientific literature, the staff concludes:

(1) The applicants have conducted an adequate investigation of the site and region around the site, and there are no geologic conditions that pose a hazard to the site. However, it is prudent for the applicants to conduct confirmatory investigations as discussed previously.

(2) The maximum earthquake that should be considered at the site is defined by MMI VIII, or magnitude $m_b = 5.8$. The proposed SSE of the RG 1.60 spectrum anchored at 0.25 g is acceptable.

(3) The OBE of the RG 1.60 response spectrum anchored at 0.125 g is adequate.

(4) There are no capable faults at the site or in the site region.

The following sections contain a brief summary of the geological conditions of the CRBR site and the basis for the staff conclusion concerning the geological suitability of the site.

2.5.1.1 Regional Geology

The site is located in the Valley and Ridge Physiographic Province (Fenneman, 1938, and Thornbury, 1965), the southern part of which is equivalent to the Southern Valley and Ridge Tectonic Province (see NUREG-0011). This province consists of ridges and intervening valleys that extend in a northeast-southwest direction for distances exceeding 100 mi. Elevations range from about 500 ft MSL in the valleys to more than 5,000 ft MSL on the higher ridges. The drainage systems follow the valleys and occasionally cut across the ridges through gaps.

The province is underlain by Paleozoic (570 million years before present (mybp) to 240 mybp) sedimentary rocks consisting of limestones, dolomites, shales, siltstones, and sandstones that extend to depths of many tens of thousands of feet. These strata strike northeast and dip to the southeast.

Structurally, the site is located in the Southern Valley and Ridge Tectonic Province. This province is characterized by major thrust faults, many of which are more than 100 mi long, that strike in a northeast-southwest direction and dip at a low angle to the southeast. Large blocks of the Paleozoic sedimentary rocks have been translated along these faults for distances of many miles.

The interpretation of Consortium for Continental Reflection Profiling (COCORP) seismic reflection lines across the southeastern United States indicates the presence of a major detachment surface at depths of 4 to 13 km beneath the Valley and Ridge, Blue Ridge, and Piedmont Provinces (Albaugh et al., 1980; Cook et al., 1979, Harris et al., 1981). If this interpretation is correct, the detachment would probably outcrop somewhere in the Valley and Ridge Province. The major thrust faults that are mapped in that province which were crossed by the COCORP line are shown in the data to be listric to the seismic reflection feature that has been interpreted to be the detachment.

Two of the regional thrust faults are located near the site, the Copper Creek Fault about 3,000 ft southeast of the site, and the White Oak Mountain Fault, 1.7 mi northwest of the site.

As part of the geological investigations of the Clinch River site, samples of fault gouge from the Copper Creek Fault and shear zone at the site were radiometrically dated and shown to be at least 280 million years old. During investigations for the Phipps Bend (TVA, 1975) and Watts Bar (TVA, 1974) sites, undeformed terraces of Pleistocene age (2 mybp to 30,000 ybp (years before present)) were found which overlay faults related to the late Paleozoic Appalachian orogeny, demonstrating their antiquity. This evidence and other evidence described more fully in Section 2.5.3 lead the staff to conclude that the detachment and related thrust faults are not capable.

Based on its review, the staff concludes that these thrust faults in the Valley and Ridge Province, including the Copper Creek and White Oak Mountain Fault, are not capable according to Appendix A, 10 CFR 100.

2.5.1.2 Site Geology

The site lies on a small peninsula formed by a sharp meander in the Clinch River. The peninsula is characterized by the regional northeast-southwest trending topography, which at the site has been modified by erosion into low, rounded ridges and shallow, gentle-sloped swales, or valleys. Elevations in the immediate site area range from about 750 ft MSL, at the river's edge, to more than 1,000 ft MSL atop the local ridges. Ground surface at the site is about 800 ft MSL.

The uppermost rocks beneath the site belong to the Knox Group and the Chickamauga Group. The strata outcrop in northeast-southwest bands, and, because of differences in erosion and weathering characteristics of the rock units, control the ridge and swale topography.

The Knox Group of Late Cambrian-Early Ordovician Age (510 mybp to 490 mybp) is principally a dolomite and outcrops in a zone several hundred feet northwest of the site. It dips southeast and lies at a depth of more than 400 ft beneath the site.

The rocks directly beneath the site belong to the Chickamauga Group of Ordovician age (490 mybp to 430 mybp). The Chickamauga Group at the site has been divided into units A and B. Unit A directly overlies an unconformity on the surface of the Knox dolomite. Unit A Chickamauga is subdivided into a lower siltstone, a middle limestone, and an upper siltstone. Each of these units is several tens of feet thick. The site is founded on the upper siltstone unit. Unit B has not been subdivided like unit A, but it has a lower limestone horizon overlain by undifferentiated Chickamauga. The unit B limestone outcrops in a band southeast of the site and is not involved in site foundations.

Minor geologic structures consisting of small folds, faults, and a shear zone of bedding plane slippage in the lower section of the unit A limestone, have been identified. These features are related to the Late Paleozoic Appalachian orogeny and are therefore not capable. These structures are discussed in more detail in Section 2.5.3.

2.5.1.3 Limestone Solutioning

The site is located in carbonate rock terrain; therefore, the possible presence of solution cavities beneath the plant became a concern early in the review. Throughout the 1970s investigations at the site have been carried out. The staff has followed these studies and has made several visits to the site to examine the karst topography, outcrops of bedrock, and rock core taken at the site.

Before the deposition of the Chickamauga Group, the surface of the Knox dolomite was exposed for a long period of time to weathering and erosion. Karst topography (caverns and sinks) developed. When the deposition of Chickamauga sediments began, the cavities and sinks were filled with material. This material lithified and became as competent as the surrounding rock. These features (paleokarst) are present today at depth in the Knox Group, near its contact with the Chickamauga Group, about 400 ft beneath the site.

Since the strata have been tilted to the southeast, solutioning has been restricted to ground surface and along the bedding planes and joints that generally follow the southeast dipping beds. Solution features at the surface, such as sinks formed by widening along joints, are common in surface outcrops of the Knox dolomite and the limestones. They exist, but are not common, in the siltstones. Weathering and solution-widened joints are much more common in the limestones than in the siltstones, and extend deeper. As jointing becomes less frequent and tighter with depth, there is an approximate lower limit to which significant solutioning has occurred. The applicants have identified that level, referred to as the "depth to continuous rock," and plan to place structural foundations below it at a base level of 712.5 ft MSL. Because of variations in topography and depth to continuous rock, excavation depths will range from 50 to 100 ft below ground surface.

Based on the available data, the staff concludes that there are no significant cavities below the foundations that could cause subsidence or collapse. That conclusion is based on:

(1) the relatively low susceptibility of the unit A upper siltstone (foundation rock) to solutioning

(2) the tightness and low frequency of joints below "top of continuous rock"

(3) the 129 core borings that have been drilled at the site, with close spacing in the power block area. (Core recoveries and rock quality designations (RQD) were very high.)

(4) in-hole geophysics and seismic, and crosshole seismic investigations

(5) surface seismic refraction surveys

(6) a test grouting program which demonstrates by low grout takes that the rock contains few voids

(7) borings drilled to verify the effectiveness of the grouting program

2.5.2 Seismology

2.5.2.1 Maximum Earthquake

Following the procedures set forth in 10 CFR 100, Appendix A, the controlling earthquake or that event which defines the safe shutdown earthquake (SSE) for most eastern U.S. nuclear power plants is the maximum historical earthquake in the tectonic province within which the nuclear power plant site is located. The CRBR site is in the Southern Valley and Ridge Tectonic Province (as discussed in Section 2.5.1). The maximum historical earthquake in this tectonic province and the controlling earthquake for the CRBR site is the 1897 maximum MMI VIII earthquake in Giles County, Virginia. This earthquake occurred before the development of calibrated seismic recording instruments and so there is no way to calculate its magnitude directly. However, a study by Nuttli, Bollinger and Griffiths (1979) has utilized empirically determined relationships between body wave magnitudes of instrumentally recorded earthquakes and (1) their intensity falloff with distance, (2) their total felt area, and (3) the area within the intensity IV isoseismals to determine magnitudes for historical earthquakes where only intensity data were available. Particular attention was given in this study to the 1897 Giles County earthquake. The two most reliable estimates in the authors' opinion (intensity falloff with distance and the intensity IV isoseismal area) each indicated a body wave magnitude of 5.8 for this event.

G. A. Bollinger (1981) presented a paper entitled, "The Giles County, Virginia Seismic Zone--Configuration and Hazard Assessments," at a conference on earthquakes and engineering in the eastern United States. In this paper, Bollinger postulated the existence of a buried fault in Giles County. He delineated this fault plane with the hypocenters of eight recent microearthquakes and four larger pre-1977 felt events. He used the largest extent of the seismic zone defined by these hypocenters, including the maximum extent of the hypocenter location errors in order to calculate the maximum possible fault dimensions.

These calculations resulted in maximum fault plane surface area estimates with a range of from 80 to 800 km^2. Using these maximum dimensions, he then postulated a range of surface wave magnitude of $M_s = 6$ to $M_s = 7$ for the maximum possible earthquake that could be generated by this fault. Bollinger stated that this worst-case postulation is made without regard to the probability of its actual occurrence, without regard to any engineering design considerations, and primarily for application in emergency planning. Using Nuttli's (1981) relation between M_s and m_b for midplate earthquakes results in an m_b range of 5.8 to 6.3 for this largest postulated event.

In view of the fact that the Giles County earthquake is the controlling earthquake for the seismic design of the CRBRP, the staff asked the applicants to provide a discussion on any effect Bollinger's hypothesis might have on the site with respect to the following three items:

(1) the potential of the 1897 earthquake being associated with this specific geologic structure

(2) the potential of an earthquake up to $M_s = 7.0$ being located in Giles County, and any far field ground motion effect (both peak values and response spectrum) at the site from an $M_s = 7.0$ event in Giles County)

(3) the potential of similar seismogenic structures being located near the CRBR site and any effect at the site from earthquakes on these seismogenic structures

Item (1)

In PSAR Amendment 69, the applicants provided a response to the above items in question. In addressing Item (1) above, the applicants state that supporting and contradicting evidence exists for the hypothesis that the 1897 Giles County, Virginia, earthquake occurred in association with the seismogenic zone proposed by Bollinger. In particular they state that:

(a) The proposed seismogenic zone is centered on Pearisburg, Virginia, the locality with the highest reported intensity effects from the 1897 earthquake. Pearisburg was the largest population center of the Giles County, Virginia, area in 1897 and there is a possibility of some bias in the location of this earthquake because population distribution may artificially weight the intensity data.

(b) The lower estimate of the seismogenic zone's size is considered by Bollinger as capable of producing a maximum earthquake of magnitude (M_s) 6.0 which is approximately the same as the 1897 shock.

(c) Not all the instrumentally located earthquakes in the Giles County area show close spatial association with the proposed seismogenic zone. The second largest event reported to have occurred in the Giles County area, the magnitude (m_b) 4.6 earthquake of November 20, 1969, was approximately 20 km northwest of the proposed seismogenic zone.

(d) Focal mechanism solutions for Giles County events have proved to be inconclusive.

Item 2

In addressing Item (2) above the applicants state that the Bollinger study does not attempt to quantitatively estimate ground motion nor does it assess the likelihood of occurrence of the maximum magnitude event and thus it cannot be directly used to establish engineering seismic design criteria. In particular they state that:

(a) The evidence for the fault zone is the distribution of earthquake hypocenters which define a tabular seismogenic zone. It is from this that Bollinger infers the existence of a single fault plane as a means of estimating the magnitude of the maximum possible earthquake associated with the zone. The dimensions of the zone are estimated by moving the computed hypocenters inside their error ellipsoids to achieve maximum and minimum spatial dispersals. Bollinger estimates the maximum magnitude of earthquakes associated with this seismogenic zone by assuming that the range in fault plane area of 80 km^2 to 800 km^2 represents the range of maximum potential rupture area.

(b) Bollinger's estimate of maximum magnitude is based on the additional assumptions that the entire postulated fault plane could rupture during a single earthquake and that the published relationship between fault rupture area and magnitude are valid for the Giles County zone.

(c) There is currently no evidence other than the earthquake locations to demonstrate the existence and orientation of slip on one or more faults.

The applicants also estimated the ground motion effect at the CRBR site from the postulated maximum earthquake, a surface wave magnitude M_s = 7.0, located in Giles County, Virginia, at a distance of about 360 km. Two approaches were used in this ground motion estimate. For one approach, Nuttli's work (1979) was used to estimate the peak ground acceleration at the site. The second approach utilized the hypothetical isoseismal maps produced by Bollinger (1981) which were developed to include the source characteristics of a postulated M_s = 7.0 maximum MMI IX Giles County earthquake and regional attenuation properties of the Appalachians. The results of these methods produces ground motion estimates at the site of 0.01 to 0.036 g peak horizontal acceleration and site intensity estimates in the range V to VI MMI. All of these estimates are less than the SSE.

Item 3

In addressing Item (3) above, the applicants investigated the historical seismicity of the region and reviewed published information concerning basement structure and examined inferred basement lineations for possible correlation with the historic seismicity. The applicants state that the epicenters of well-located events in the area form no discernible spatial pattern. However, since the region has been adequately monitored for only the past 10 to 15 years they could not make unequivocal conclusions as to the existence or nonexistence of spatial trends. A northeast trending magnetic and gravity lineation paralleling the western margin of the Valley and Ridge Province in eastern Tennessee is part of a much larger feature which can be traced from Alabama to New York. The applicants adopt King and Zietz's (1978) position that this linear trend marks a basement discontinuity separating a seismically active crustal block on the southeast from a seismically inactive crustal block on the northwest. The applicants conclude that the lineament is not a seismogenic structure but may represent the western boundary of a seismically active region.

The staff has reviewed the Bollinger report and the applicants' response to staff questions and concludes that the safe shutdown earthquake proposed for the CRBR is adequately conservative to encompass any vibratory ground motion that might result from the type of earthquake postulated by Bollinger to occur on the postulated Giles County trend in the vicinity of Pearisburg, Virginia. This conclusion is based primarily on the following:

(1) If all of Bollinger's assumptions are correct and there is a structure in Giles County that may be able to generate a surface wave magnitude $M_s = 7$ earthquake, then the event would be related to that particular structure and would not be assumed to occur randomly elsewhere in the tectonic province as per Appendix A to 10 CFR 100.

(2) The staff has made its own estimate of ground motion from an $M_s = 7$ earthquake at a distance of 360 km from the site. This results in a vibratory ground motion well below the CRBR SSE. Using Nuttli's (1979) attenuation relation, Tera-Lawrence Livermore National Laboratory's attenuation relation (memorandum from Jackson to Crutchfield, NRC, June 23, 1980), and Nuttli and Hermann's (1981) attenuation relation for an $m_b = 6.3$ ($M_s = 7$) earthquake at a distance of 360 km, peak acceleration and velocity values are obtained that, when used in conjunction with the amplification factors in NUREG/CR-0098, result in a spectrum significantly less than the SSE.

Although Bollinger has postulated a seismogenic structure in Giles County, the evidence presented is tentative and subject to interpretation. This area continues to be investigated by Virginia Polytechnic Institute and State University (VPISU) scientists in an attempt to better identify the presence of a causative structure. There is as yet no direct geological or geophysical evidence to indicate the existence of such a structure. Therefore, the staff considers it currently premature to restrict the recurrence of a Giles County 1897 maximum MMI VIII earthquake to the postulated seismogenic zone and still considers a possible recurrence of this event in the site vicinity to be the controlling earthquake for determining the SSE.

There has also been a postulation that there may be a seismogenic source zone capable of a large earthquake closer to the CRBR site than Giles County. This is discussed in Section 2.5.2.3 and in the referenced USGS seismology and geology review of the Clinch River breeder reactor site (Devine, 1983). The staff concludes that there is insufficient evidence at this time to warrant consideration of such a source in the definition of the SSE at Clinch River.

2.5.2.2 Effects of Human Activity in the Area

A facility for injecting radioactive waste into subsurface strata is located on the Oak Ridge Reservation approximately 4 mi east of the proposed CRBRP site. These injection wells have been used periodically since February 1954 to inject wastes mixed with a cement grout slurry into the Conasauga shale along cracks generated by hydrofracturing. In the early 1970s the staff expressed concern that the injection might induce earthquakes in the area. The applicants undertook a study to investigate this concern.

The applicants report that many small seismic signals resembling earthquakes have been recorded on the seismograph at Oak Ridge. These occurred primarily during normal working hours and analysis of recordings made during injection and noninjection periods indicate that locally generated seismic signals, other than those caused by the machinery and fluids used in the injection process, do not seem to occur any more frequently when injection is in progress. The applicants have conducted analyses of these signals and have concluded that the Oak Ridge National Laboratory injection wells are not inducing earthquakes in the area; the staff concurs in this assessment on the basis of a statistical comparison that indicates that the locally generated seismic signals appear to correlate to the normal work hours and are probably manmade from sources other than injection as they do not appear to correlate to periods of waste injection.

The applicants have committed in response to Question Q323.47 (PSAR Amendment 17, Apr. 1976) to restrict future hydrofracture operations within a defined set of parameters. These parameters are defined by four points on the topographic map of Melton Valley, ORNL-D26364, and based on the Tennessee State System of rectangular coordinates. They are (N557,800, E2,498,500); (N557,800, E2,449,400); (N555,500, E2,497,600); and (N554,900, E2,498,500). The portion of the Conasauga Formation utilized for injection is limited to the approximately 300 ft of red shale occurring between the Rome sandstone and the three limestone beds used as stratigraphic markers. Future operations will be restricted to those locations where this particular stratum occurs in the range of 500 ft to 1,500 ft below the land surface and all operations will be conducted in such a manner that static injection pressure will not exceed 3,000 psig as measured at the well head annulus. Geologic conditions in the area defined for injection wells have been mapped and the location of the major thrust fault (Copper Creek Fault) determined. The injection wells are designed and constructed so that the bottoms of the wells do not extend to the depth of the major fault plane. Therefore, it will not be possible to inject fluid into the fault plane of the Copper Creek Fault.

A new disposal well has been installed and went into use in June 1982. This new facility is located about 800 ft southwest of the well that had been used during the past few years which has now been retired. Tests at the new location have demonstrated that the new disposal well penetrates essentially the same geologic horizon as the old well. The injections in the new well are not expected to have any effect on nearby faults or lateral stress buildup since they are planned to be similar to past injections and these past injections apparently have not produced these effects. The staff therefore concludes that future waste injection will not have an adverse affect on the proposed site.

2.5.2.3 Safe Shutdown Earthquake

In recent nuclear power plant safety analysis report reviews (Sequoyah, Watts Bar, Washington Nuclear Power Unit 3, Perry, and Wolf Creek), the staff has adopted the practice of using site-specific response spectra to assess the adequacy of the proposed SSE. In order to compute site-specific response spectra, it is necessary to characterize the earthquake size, the epicentral distance (distance between the surface location of the earthquake and the site) and the site conditions (soil or rock) being modeled. The 1897 Giles County earthquake is also the controlling earthquake for the seismic design of the Tennessee Valley Authority's Sequoyah, Watts Bar, and Bellefonte nuclear power plants. As part of the justification of the seismic design criteria used for these three nuclear power plants TVA developed a site-specific spectrum. The staff has reviewed and approved this site-specific spectrum as being an adequate representation of the vibratory ground motion from an 1897 Giles County (maximum MMI VIII, body wave magnitude 5.8) type earthquake in the site vicinity (see NUREG-0011 and NUREG-0847). A comparison of the site-specific spectrum developed for an 1897 Giles County type earthquake with the applicants' proposed SSE spectrum, an RG 1.60 response spectrum with a high frequency anchor of 0.25 g, shows that the proposed SSE spectrum exceeds the site-specific spectrum at all frequencies. The staff, therefore, finds the CRBR SSE to be adequate for structures founded on rock.

The United States Geological Survey (USGS) has acted as advisor to the staff in the review of the CRBR PSAR (Devine, 1983). The USGS has also used a probabilistic method in assessing the conservatism of the CRBR SSE.

The staff does not normally use probabilistic methods in the determination of seismic design criteria for nuclear power plants. In the probabilistic method of seismic hazard estimation, zones of potential earthquakes are delineated by seismicity and neotectonics, and predicted rates of earthquake occurrence are estimated by historical seismicity. The ground motion at a site is estimated as a function of magnitude, distance, and site conditions. Knowledge of the input parameters required for a probabilistic seismic hazard analysis in the eastern United States is limited because of short seismic history and unknown neotectonics. Because of this, the input parameters are subjective and uncertain. The results of probabilistic studies are, therefore, currently considered by the staff to be more appropriate for relative comparisons of seismic hazard to gain insight rather than for the absolute determination of hazard.

As part of its review the USGS looked at the exceedance probability of 0.25 g when considered in the light of the assumptions made in producing probabilistic ground motion maps for the eastern United States (Algermissen et al., 1982). Included in the generation of these maps were the assumptions that the seismicity was diffuse and uniform over the Appalachian Province, that the earthquakes were crustal earthquakes that could be modeled as point sources near the surface, and that the maximum magnitude is 7.3. Based on these assumptions the probability of exceeding 0.25-g peak horizontal acceleration is about 2×10^{-4} per year.

In addition to the above study the USGS also performed a probabilistic seismic hazard analysis based on the speculated existence of a local seismic zone.

This zone is based on the hypothesized alignment of nine earthquakes that have been relocated by J. Dewey and D. Gordon (see Devine, 1983). It is theorized that these events may represent a concentration of seismicity in eastern Tennessee and that, although there is insufficient evidence of a specific structure, it is possible that this alignment may represent a basement seismic source zone analogous to that proposed for the Giles County area by Bollinger. In addressing the seismic hazard from this hypothesized seismic source zone a line source 140 km long is assumed and earthquakes were modeled as ruptures. The maximum magnitude was assumed to be $M_s = 7.0$ and the source was assumed to lie 15 km from CRBR. On the basis of these assumptions, the probability of exceeding 0.25 g is about an order of magnitude higher than that calculated assuming no such source exists. Arguments are provided for and against the existence of this hypothetical structure.

The main evidence presented for the existence of the hypothetical structure in the USGS letter (Devine, 1983) is the apparent alignment of the relocated epicenters. Most of the other evidence referenced is equivocal or negative to the existence of the hypothesized structure.

The staff has considered the possibility of a local structure in its review and has examined the seismicity and the literature on this region. The staff sees no compelling argument to associate the nine earthquakes identified by the USGS reviewers with a particular hypothetical structure. Several other alignments of earthquake epicenters in the diffuse pattern of seismicity in this region could also be assumed. Several of the earthquakes assumed to be in the alignment from which the structure is hypothesized have rather poor location statistics. One has an error ellipse with an axis of about 95 km, another event has an error ellipse with an axis of about 60 km, and a third has an error ellipse with an axis of about 50 km. In contrast, the seismic zone hypothesized by Bollinger for the Giles County, Virginia, area has a rather tight alignment of epicenters with good depth estimates which help define a tabular zone and much better location statistics. The Giles County zone is hypothesized on some very intensive studies including the detailed calibration of the seismic velocity model through the use of local explosions.

The USGS reviewers conclude that the selection by the applicants of the Giles County earthquake as the controlling earthquake at the site is reasonable and also concur with the assessments of the maximum intensity and the SSE and the anchoring of the RG 1.60 response spectrum to 0.25-g acceleration. They also note, however, that a notable difference exists in the estimated exceedance probabilities, depending upon whether or not the hypothesized seismic zone exists and is capable of generating large magnitude events.

Probabilistic hazard studies are sensitive to the assumptions made in selecting the input parameters and are illustrated in this case by the difference in earthquake hazard that results from the two different source zonation assumptions used. The USGS letter (Devine, 1983) indicates that the data available at the present time are insufficient to establish the situation one way or the other for this hypothesized seismic zone and therefore, although the CRBRP SSE is reasonable on the basis of present data, a definitive seismological investigation would be required to address the problem with respect to the probabilistic studies of a possible concentrated seismic source zone in eastern Tennessee.

As was stated above, it is the staff's position that the evidence for such a zone is weak, and therefore does not warrant consideration as a capable fault within the meaning of Appendix A to 10 CFR 100. However, additional studies are currently under way through ongoing research efforts in the region. There is a well-distributed network of seismograph stations in the CRBR region and more stations are planned for the near future. These stations are operated by various agencies including TVA, the Tennessee Earthquake Information Center, and Georgia Institute of Technology. Through the use of data from this network, small earthquakes, if they occur in the CRBR area, can be located. Until now an insufficient number of accurately located earthquakes have been recorded to define any extended seismic source zone in the region. The staff recommends that the applicants keep informed on all seismological developments in the site region, such as the occurrence of earthquakes or the results of new research, since this information will have to be provided in the Final Safety Analysis Report.

2.5.2.4 Operating Basis Earthquake

The applicants have proposed an OBE with a peak horizontal acceleration of 0.125 g anchoring an RG 1.60 response spectrum. This OBE is one-half the SSE and the staff finds it adequately conservative within the meaning of Appendix A to 10 CFR 100.

2.5.3 Surface Faulting

Structures of the Southern Valley and Ridge Tectonic Province are described in Section 2.5.1.1. On the basis of available evidence, the staff concludes that the thrust faults that characterize this province are not capable within the meaning of Appendix A to 10 CFR 100. The bases for that conclusion are:

(1) Extensive field research has been conducted in the region with the intent of finding evidence for recent displacement along these faults to explain current seismicity, and none has been found.

(2) Triassic dikes mapped in Virginia penetrate Valley and Ridge Province structures without being offset (Dennison and Johnson, 1971)

(3) In Alabama where Coastal Plain deposits overlie the southern part of the Valley and Ridge Province structure there is no evidence of offset (Butts, 1926).

(4) Where subsidiary faults of the major thrust faults have been mapped in relation to overlying ancient terrace deposits, those terraces have not been offset (i.e., Phipps Bend and Watts Bar site fault investigation; TVA, 1975; TVA, 1974).

(5) Radiometric age dating of gouge taken from the Copper Creek Fault indicates an age of at least 280 mybp.

The closest major fault to the site is the Copper Creek Fault and its trace is located 3,000 ft from the site. At this location, the fault strikes N52°E and dips away from the site to the southeast at an approximate dip of 25°. Displacement is about 7,200 ft with the Rome Formation thrust over Chickmauga Group rocks. This fault has a mapped length of 100 mi, but becomes complex and merges to the north with other faults. The Copper Creek Fault is one of many Late Paleozoic thrusts that developed during the Allegheny orogeny (Pennsylvanian-Permian, 330-240 mybp). These structures are not considered active and are not used in the determination of the SSE. Radiometric dates of 290 ± 10 mybp and 280 ± 10 mybp were obtained for mylonite fault gouge material taken from the fault zone of the Copper Creek thrust. This finding, coupled with lack of evidence of recent offset and an understanding of the tectonic development of the Paleozic thrust faulting in east Tennessee, indicates that this major fault and other small faults in the site area associated with it are tectonically old. Therefore, the staff does not consider them hazardous to the safe operation of a nuclear plant at this location. These faults are not capable faults as defined in "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A, 10 CFR 100.

Considerable new regional geologic and seismic information has been obtained since publication of the site suitability report (SSR), including new data regarding the Giles County and Charleston earthquakes and theories about their source mechanisms. The NRC staff has assessed this new information relative to the proposed CRBRP site, and finds no reason to change its conclusions regarding the suitability of the site. The NRC position on the Charleston earthquake and eastern seismicity is presented in Devine (1983).

Several minor structures have been identified in bedrock at the site. These features include minor folds with amplitudes of several feet, minor faults with inches to several feet of displacement, and a shear zone involving bedding within a zone about 40 ft thick in the lower part of the unit A limestone and the upper few feet of the unit A lower siltstone. This shear zone was identified by slickensides on the bedding planes and minor offsets of fractions of an inch. The slip surfaces have been healed. Because of orientation consistent with regional tectonic deformation and healing, the staff concludes that these features are related to the Appalachian orogeny and are therefore at least 240 million years old.

In their response to NRC Question 230.3R, which requested a map and discussion of the relationship between geologic structures at the site and the high terrace deposits, the applicants indicate that it was not necessary to do this because other lines of evidence have demonstrated the antiquity of faults in the site vicinity. The staff and the USGS agree that the faults are not capable, but in the staff's view, it is prudent to conduct additional investigations of the relationship of tectonic structures if they are present in the site vicinity and "datable" horizons that overlie them. The applicants should map and investigate those locations where there is a good chance to find contacts between high terrace deposits and faults to confirm that the faults are not capable. This can be done in the following manner:

(1) Investigate the high terrace in the southeast portion of the site peninsula to determine whether or not those deposits have been tectonically deformed. Because of the proximity of this terrace to the Copper Creek Fault, it is likely that structures are present there that are genetically related to the Copper Creek Fault; and/or

(2) Locate sites in the subregion around the site where "datable" horizons appear to overlie mapped faults, and investigate those areas to determine whether or not the capping material is offset. Whether the applicants find an appropriately located terrace or not, the study should be documented in a manner similar to that described in Supplement 2 to PSAR Section 2.5 regarding sites where residual soil colluvium were photographed at projected outcrops of the Copper Creek and White Oak Mountain Faults.

Because this work is confirmatory, the staff recommends that it be carried out as a post-CP item.

It is likely that many minor structures, including small faults, will be encountered during excavation at the site. The applicants have committed to geologically map the excavations and promptly notify the staff of any faults discovered there so that field inspections can be made if necessary.

Although the staff expects additional small faults to be found, there is no reason to expect these faults to be younger than Late Paleozoic (more than 240 mybp).

On the basis of available information, the staff concludes that the faults at the site and in the region around the site are not capable as defined in Appendix A, 10 CFR 100. However, it is the staff's opinion that there may be additional evidence in the region in the form of high terrace deposits overlying faults. The staff considers it prudent to examine these cross-cutting relationships to confirm the conclusion that the faults are not capable.

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 General Plant Description

The CRBR site is located in east-central Tennessee approximately 25 mi west of Knoxville and 9 mi southwest of Oak Ridge on undeveloped property which is owned by the U.S. Government and is in the custody of the TVA. The site is bounded on the north by DOE's Oak Ridge Reservation and on the east, south, and west by a meander of the Clinch River between river miles 14.5 and 18.6. (See Figures 2.1 and 2.2.)

The topography of the area is characterized by a series of steep parallel limestone ridges, hills, and knobs which are generally oriented in a northeast to southwest direction. The dominant feature in the site area is Chestnut Ridge, running across the northern part of the site, cresting at an elevation of 1,100 ft above MSL. The main plant grade has been established at 815 ft above MSL, 74 ft above the Clinch River mean water level of 741 ft.

2.5.4.2 Site Investigations

Foundation investigations were started in February 1972. The applicants have reported a total of 129 borings, and 6,360 linear feet of seismic refraction traverses have been accomplished at the site. Additional in situ testing accomplished included seismic up-hole surveys, seismic cross-hole surveys, continuous velocity logging, and Goodman plate jack testing.

The core boring and sampling program was initiated on a 200-ft north-south grid pattern to determine subsurface conditions in the general site area and to define the best location for the nuclear island mat. Additional borings were completed on a 200-ft grid rotated 45° from the initial grid; this provided a 140-ft spaced coring program for the nuclear island area. Holes were later drilled at the center of the 140-ft grid, providing a minimum spacing of 100 ft between borings to complete the coring program for the nuclear island area. In addition, five borings were drilled to pinpoint the best location and mat elevation for the seismic Category I cooling tower. Four additional core borings were drilled to define the location of the foundation for the seismic Category I steam generator maintenance bay structure.

Continuous velocity logging surveys were accomplished in 11 core borings throughout the site, and seismic cross-hole surveys were performed in an array of 15 borings. Seismic up-hole testing was accomplished within the area of the proposed nuclear island mat to provide velocity data for the rock mass in the vertical direction. Goodman plate jack tests to determine the modulus of elasticity of intact rock in the in situ state were accomplished in four holes in the main plant site area at depths ranging between 15 and 150 ft.

Other site investigations reported as accomplished by the applicants relevant to the geotechnical aspect of the site involved a comprehensive office review of available published data including reports, geologic maps, and previous construction data for the area.

On the basis of the information presented in the Preliminary Safety Analysis Report (PSAR), the staff finds that the applicants' site investigation efforts provide adequate coverage of the site area in sufficient detail to provide a high level of confidence that specific subsurface conditions have been adequately defined. The staff's review of data presented reveals no evidence of significant zones of solutioning, caverns, or highly weathered areas in the foundation bedrock which could produce significant subsidence under the loads to be imposed by the proposed structural mats.

2.5.4.3 Foundation Materials

The foundation rock for the site consists of alternating layers and laminations of siltstone, limestone, and shale with some chert. The bedrock is overlain by terrace deposits up to 40 ft thick, weathered rock, and zones of clayey residual soil. Overburden varies in thickness from 8 to 56 ft throughout the site area. The main seismic Category I structures, except for the steam generator maintenance bay and the fuel oil storage tanks, will be founded on a single common structural mat called the nuclear island at el 715 ft located directly on a siltstone stratum termed the Chickamauga unit A upper siltstone. The steam generator maintenance bay will be founded in a limestone formation termed the Chickamauga unit B limestone. Two seismic Category I fuel oil storage tanks will be anchored to a common reinforced concrete mat with base at el 793 ft supported directly by class A structural backfill material overlying the unit A upper siltstone. The seismic Category I cooling tower will be supported by a single mat founded at el 765 ft on the unit A upper siltstone.

Emergency plant service water system supply and return headers and seismic Category I pipes and underground Class 1E electrical ducting will be founded on class A structural backfill (see Section 2.5.4.6.2, "Backfill").

2.5.4.4 Properties of In Situ Materials

Investigative programs were conducted by the applicants to determine static and dynamic engineering properties of the foundation rock at the plant site. The programs consisted of both laboratory and field in situ testing.

2.5.4.4.1 Static Properties

Results of laboratory unconfined compression and Poisson's ratio tests provided the basis for the applicants' selection of the foundation rock static engineering properties. Unit weight and moisture content tests were also performed on representative rock samples. Goodman plate jack tests to determine the modulus of elasticity of rock in the field intact state were also accomplished. Test procedures were in conformance with the following procedures:

(1) unconfined compression--ASTM D2938
(2) Poisson's ratio--ASTM D3148
(3) unit weight--CRBR PSAR, Vol. 3, Sec. 2.5, App. 2B
(4) moisture content--ASTM D2216
(5) Goodman plate jack tests--CRBR PSAR, Vol. 3, Sec. 2.5, App. 2A

The ranges and averages of laboratory test results are presented in Table 2.5-8 of the PSAR. The results listed are representative of the intact rock quality. Because of the geologic discontinuities existing within the areas of the in situ rock which will be stressed by the applied structural loads, reduction of intact rock quality properties is appropriate in selecting a design static rock mass modulus value (Hendron, Ch. 2, pp.21-53, 1975). The applicants' selection of a reduction factor of 0.5 based upon evaluation of rock quality designation (RQD) values determined from in situ rock core boring is considered appropriate and acceptable to the staff.

The resultant properties of the in situ rock mass as determined by the applicants are:

- Unit weight (measured) = 165 pcf
- Poisson's ratio (measured) = 0.3
- Young's modulus (measured and adjusted based on RQD values) = 5×10^5 psi
- Shear modulus (calculated) = 2×10^5 psi

2.5.4.4.2 Dynamic Properties

Seismic up-hole, cross-hole, and continuous velocity logging surveys were performed by the applicants in representative areas of the site rock mass to determine in situ dynamic properties. The results provided information regarding compression and shear wave velocity properties for the in situ rock mass in the vertical and horizontal direction. Shear wave velocities of the foundation rock were measured by each test method. Based upon survey results a range of 5,580 to 6,820 ft per second was selected to be representative of the shear wave velocity for sound foundation rock. The applicants have used an average value of 6,200 ft per second to compute a representative dynamic shear modulus of 1.37×10^6 psi, and a dynamic Young's modulus of 3.55×10^6 psi for the in situ rock.

Because earthquake loading differs in duration and stress level from geophysical test methods, a reduction of moduli calculated from geophysical test results is appropriate in establishing dynamic design rock properties (Hendron, Ch. 2, pp. 21-53, 1975). The applicants have selected a dynamic property reduction factor of approximately 0.42 based upon consideration of rock quality designation values derived from in situ rock core properties and upon comparison of the results of in-place continuous velocity logging measurements with the results of up-hole velocity measurements. The staff considers this approach appropriate and acceptable. The resultant recommended design dynamic moduli for the siltstone foundation rock are:

- Shear modulus (G) = 5.8×10^5 psi ± 25%
- Young's modulus (E) = 1.5×10^6 psi ± 25%

Using data derived from field survey and laboratory tests and similar logic to establish appropriate reduction factors, the applicants have selected the following dynamic moduli values for the unit A upper limestone foundation material:

- Shear modulus (G) = 1.2×10^6 psi ± 25%
- Young's modulus (E) = 3.0×10^6 psi ± 25%

The staff finds these values for siltstone and limestone to be appropriate and the range of values to be sufficiently conservative.

2.5.4.5 Groundwater

Groundwater levels vary throughout the site because of topographic features and the permeability of the underlying rock. The normal ground water level in the vicinity of the main seismic Category I plant structures is approximately 770 ft MSL. Dewatering during excavation will be required to control hydrostatic uplift pressure. Water seepage will be controlled by placing drainage ditches at the toe of the slopes which lead to sump pits placed at appropriate locations along the slopes. Collected water will be pumped to impoundment ponds to allow suspended solids to settle before discharge into the Clinch River. The staff's evaluation of groundwater is presented in Section 2.4 of this SER.

During construction the applicants plan to install slotted drainage pipes in the rock face to control ground water seepage and to provide sufficient pumping capability to pump the anticipated seepage to the established impoundment ponds. Some local areas around the excavation may be grouted if the grouting will benefit the site dewatering program.

2.5.4.6 Excavation and Backfill

2.5.4.6.1 Excavation

Excavation in soil and rock is required to establish the planned foundation grade for the plant structures. After clearing and grubbing the site area of trees, shrubs, and surface vegetation, the applicants plan to strip the topsoil and to stockpile it for use in landscaping at a later date. The applicants have estimated that excavation requirements for the projects will include approximately 1,764,000 yd^3 of overburden, 594,000 yd^3 of weathered rock, and 429,000 yd^3 of sound rock. Excavation in cohesive residual overburden will be cut on slopes of 2 ft horizontal to 1 ft vertical or flatter. Excavation in partially weathered and sound rock will be nominally vertical except where joint patterns and bedding planes control slope stability. Rock bolts will be used where required to maintain stability. Trenches extending to sound bedrock will be excavated for seismic Category I piping.

The applicants have committed in Section 2.5.4 of the PSAR to having a qualified experienced geologist on site during all phases of the excavation for seismic Category I structures to review excavation activities, inspect exposed rock strata, and prepare a detailed geologic map of excavated areas. In addition, the applicants plan to convene a consulting geotechnical review group consisting of rock mechanics and geology specialists to inspect the excavations on a monthly schedule, or more frequently as required by the geologic conditions encountered. The applicants have also committed in Section 2.5.4 of the PSAR to having a qualified engineer monitor and control all rock blasting operations to minimize overbreak rock damage and disturbance at foundation grade.

Upon completion of the excavation area for the main nuclear island mat, the applicants have committed in Section 2.5.4 of the PSAR to accomplishing an investigation program in the western portion of the area where siltstone is underlain by soluble limestone strata of relatively shallow depths. The program will include drilling a series of airtrack holes through the limestone strata, performing geophysical logging, and drilling core boring as required. The results will be evaluated to verify that the homogeneity and bearing characteristic of the rock meet or exceed estimated design values.

After excavation is complete, the applicants plan to cover the various base areas with fill concrete, gunite, or gravel as appropriate. Gunite is to be placed on slopes cut in siltstone to provide protection during exposure. Slopes requiring rock bolting are to be covered with chain link or wire fencing fabric to retain loose rock. Before the mats for seismic Category I structures are poured, a formal approval of the prepared base rock foundation is to be made by the applicants' review group.

The staff's review indicates that the applicants' planned excavation activities and onsite inspection and monitoring program are adequate and acceptable. The staff requests that timely formal notification be given of significant geologic differences encountered during excavation to allow the staff to examine them at the earliest possible time. The staff also requests that a letter copy of the applicants' review group formal acceptance of each completed prepared base for a seismic Category I structure be provided before the structural mats are poured.

2.5.4.6.2 Backfill

The majority of materials removed during onsite excavation will be utilized in nonsafety-related balance-of-plant areas to establish the required final plant grade, to provide foundations for access roads and railroad sections, and to meet other nonsafety-related fill requirements. To meet safety-related requirements, the applicants plan to use two types of controlled backfill materials: (1) lean concrete backfill and (2) class A backfill.

The lean concrete backfill is to be placed directly on competent rock beneath and around the nuclear island common mat and beneath the steam generator maintenance bay and cooling tower. In areas where nominally vertical cuts have been made in the rock, lean concrete will also be placed to extend up to the top of the vertical cut within competent rock adjacent to the structures. The applicants plan to produce the lean concrete on site using locally available sand, aggregate, and water sources and a mix proportioned to obtain a 90-day design strength of 2,000 psi using ASTM C-39 testing procedures.

The class A backfill material is to be well-graded, crushed limestone of the Knox Formation obtained from a quarry or from a crusher-screening operation to be established on site. The applicants propose to control the placement of the class A backfill to provide a minimum Modified Proctor (ASTM D1557) density of 95% or 85% relative density (ASTM D2049), whichever results in the greater dry density. Results of saturated consolidated-undrained triaxial shear tests performed by the applicants on similar crushed limestone materials obtained from a nearby quarry and crusher operation indicate an angle of internal friction (Ø) of 38° is representative of such materials when compacted to 90% Modified Proctor (ASTM D1557-78) density (CRBR PSAR, Vol. 3, Sec. 2.5, App. 2D). The applicants propose to use class A backfill material adjacent to all seismic Category I structures founded on rock from the top of lean concrete fill to the design plant grade elevation of 815 ft. Additionally, two seismic Category I diesel fuel oil storage tanks anchored to a common reinforced concrete mat measuring 72 ft x 46 ft with its base at el 787 ft, and buried piping, electrical ducts, and manholes will be supported on and embedded in class A backfill overlying bedrock. The extent of structural backfill placement is presented in Figures 2.5-38 through 2.5-41 of the PSAR.

2.5.4.7 Foundation Stability

2.5.4.7.1 Static Loading and Settlement

The applicants have estimated that the seismic Category I nuclear island structures located on a 15- to 18-ft-thick common mat founded on competent siltstone will impose a maximum static bearing pressure of 8.9 ksf on the rock. This loading represents the most severe static loading condition the site rock may be expected to experience. The staff has estimated the bearing capacity of the foundation rock for the common mat using the procedures of J. E. Bowles (Ch. 4, p.143, 1977) and has conservatively determined that the estimated bearing capacity of the siltstone provides a factor of safety of greater than 10 for the proposed static loading conditions. The applicants have estimated the average expected settlement of the common mat using elastic theory and have determined that a settlement of less than 0.5 in. would be expected. The applicants have also pointed out that the common mat loading of 8.9 ksf is less than the removed overburden weight and, therefore, expected settlements would be limited to recompression of the foundation rock rebound which will occur during site excavation. The staff has independently verified these findings using the procedures of K. G. Stagg (Ch. 5, pp. 125-156, 1975) and agrees with the applicants that adequate margins of safety exist to ensure the static stability of structures founded on mats in competent rock at the site.

The two seismic Category I fuel oil storage tanks anchored to a common reinforced concrete mat represent the most severe static bearing condition for loads supported on compacted class A backfill materials. The staff has independently estimated the allowable bearing capacity of a typical class A backfill material such as that proposed for use by the applicants when subjected to the expected foundation loading using the procedures of Terzaghi and Peck (pp. 184-202, 1967) and Bowles (Ch. 4, p. 143, 1977). Staff results indicate that the backfill material can be expected to safely support the mat loading with a factor of safety of greater than 40 and an expected settlement of less than 1 in. The staff concludes that the bearing capacity of the fill is acceptable.

2.5.4.7.2 Dynamic Loading

In Section 3.7.1 of the PSAR the applicants have committed to analyze the stability of foundations under earthquake loading. A safe shutdown earthquake (SSE) peak horizontal ground acceleration of 0.25 g at the base of foundations has been selected as appropriate for the site. A peak horizontal ground acceleration of 0.125 g has been selected for the operating basis earthquake (OBE). The design response spectra for horizontal motions for the SSE and the OBE are in accordance with the amplification factors in RG 1.60. The derivation of the design response spectra for this site is discussed in Sections 2.5.2 and 3.7.1 of the PSAR. The staff's evaluation of the earthquake design basis is discussed in Section 2.5.2 of this SER.

2.5.4.7.3 Lateral Loads

For static loading conditions, the applicants proposed to design seismic Category I structure subsurface walls to resist at-rest earth pressures determined using the Rankine theory (Terzaghi and Peck, pp. 184-202, 1967) and full hydrostatic groundwater pressure at all levels below the maximum flood elevation of 809 ft. An at-rest lateral earth pressure coefficient of 0.7 has been selected for design.

For dynamic loading conditions, the applicants have considered the maximum dynamic pressure to be equal to the sum of the initial static pressure attributable to the effect of dry cohesionless backfill plus two additional pressure increments attributable to the dynamic loading of the structural backfill and the groundwater from the base of structures to the normal water table elevation of 780 ft. The applicants propose to calculate the dynamic pressure using the simplified procedure of Seed and Whitman (1970) for approximating the Mononobe-Okabe earth pressure effects combined with groundwater pressures (pore pressures) determined using Westergaard theory (Seed and Whitman, 1970). A horizontal ground acceleration of 0.25 g applied at the base of the structure walls has been selected for estimating inertial forces.

The staff concludes that the applicants' methods of estimating lateral earth pressures on seismic Category I subsurface walls are in accordance with the current state of the art and are sufficiently conservative to allow compliance with the requirements of NRC regulations identified in 10 CFR 50, Appendix A, and 10 CFR 100, Appendix A.

2.5.4.8 Liquefaction Potential

All seismic Category I structures, tanks, piping, and duct banks will be supported on sound competent bedrock or on well-graded class A granular backfill overlying bedrock.

The applicants have committed in Section 2.5.4 of the PSAR to control the placement of the class A backfill to achieve in-place minimum densities of 95% Modified Proctor (ASTM D1557) or 85% relative density (ASTM D2049), whichever results in the greater dry density.

Laboratory testing of representative backfill materials to determine cyclic strength characteristics has not been performed as it is the applicants' position that crushed rock materials, such as the proposed backfill material, when compacted to at least 95% Modified Proctor or 85% relative density would not be susceptible to liquefaction when subjected to the postulated SSE event, and therefore, a dynamic testing program to demonstrate this conclusion is not required.

The staff considers a relative density of 85% or greater to be a significant index of a low potential for liquefaction of the proposed backfill material. The staff also recognizes that the proposed backfill material can be expected to have other favorable properties such as a relatively coarse grain size (gravelly sand), good gradation (SW), and relatively high in-place permeability ($k > 10^{-2}$) which also would tend to increase the resistance of the backfill material to potential liquefaction (Seed, 1979).

Considering all the above factors and commitments, the staff accepts the position of the applicants and concludes with high confidence that well-graded, sound, crushed, granular, backfill material, when compacted to greater than 95% Modified Proctor or 85% relative density (whichever results in the greater dry density), would not be significantly susceptible to liquefaction when exposed to the postulated SSE. Before placement of class A backfill materials, the staff requests that the applicants docket results of gradation testing (ASTM D422) and permeability testing (ASTM D2434) performed on all class A backfill materials proposed for use at the site. The staff also requests that the applicants perform and submit results of Standard Los Angeles Abrasion Tests (ASTM C-131) on any class A backfill materials derived from siltstone to estimate the resistance of the backfill to crushing under the action of compaction operations. All materials having a percentage of wear of greater than 50% as measured by the Los Angeles Abrasion Test shall be subjected by the applicants to in-place compaction testing that will ensure the backfill material is compacted to 95% Modified Proctor or 85% relative density (whichever is greater) based upon the after-compaction gradation of the material.

2.5.4.9 Conclusions

On the basis of the reported results of the applicants' investigations, laboratory and field tests, and analyses presented in the PSAR, the staff concurs with the position of the applicants, with a high level of confidence, that site and plant foundation materials will be adequate to safely support the proposed CRBRP facilities during the planned life of the plant. The staff concludes that the efforts of the applicants relative to defining the nature and engineering properties of the site and plant foundation material meet the requirements of the applicable rules and basic acceptance criteria of the NRC contained in 10 CFR 50 and in 10 CFR 100 and in the regulatory positions contained in regulatory guides pertinent to Section 2.5.4 of the SRP (NUREG-0800) and are, therefore, acceptable.

2.5.5 Stability of Slopes

There are no natural or proposed manmade slopes which could adversely affect the safety of the plant.

2.5.6 Embankments and Dams

There are no existing or proposed earth, rock, or earth and rockfill embankments used for plant flood protection or for impounding cooling water required for the operation of the plant.

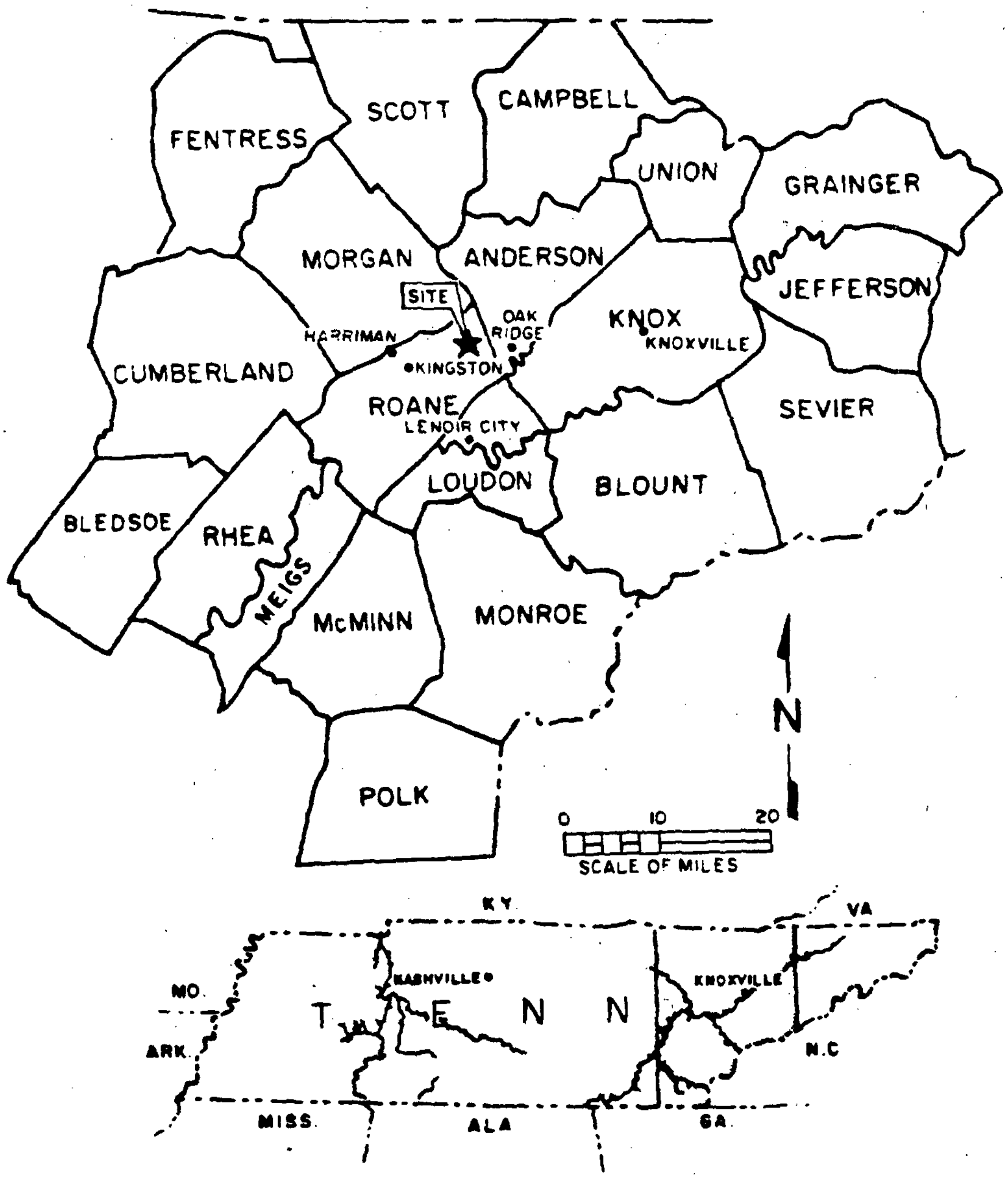

Figure 2.1 Location of Clinch River site in relation to counties and States

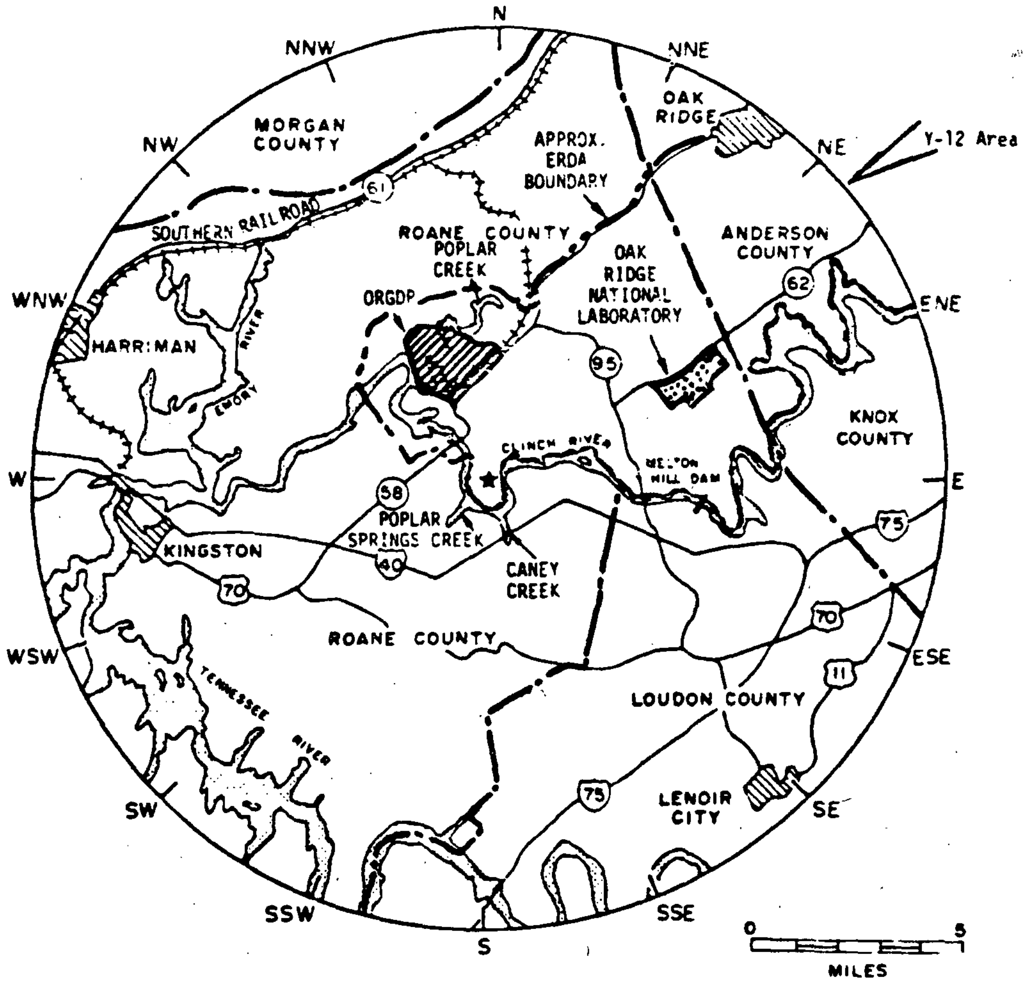

Figure 2.2 Clinch River site local area

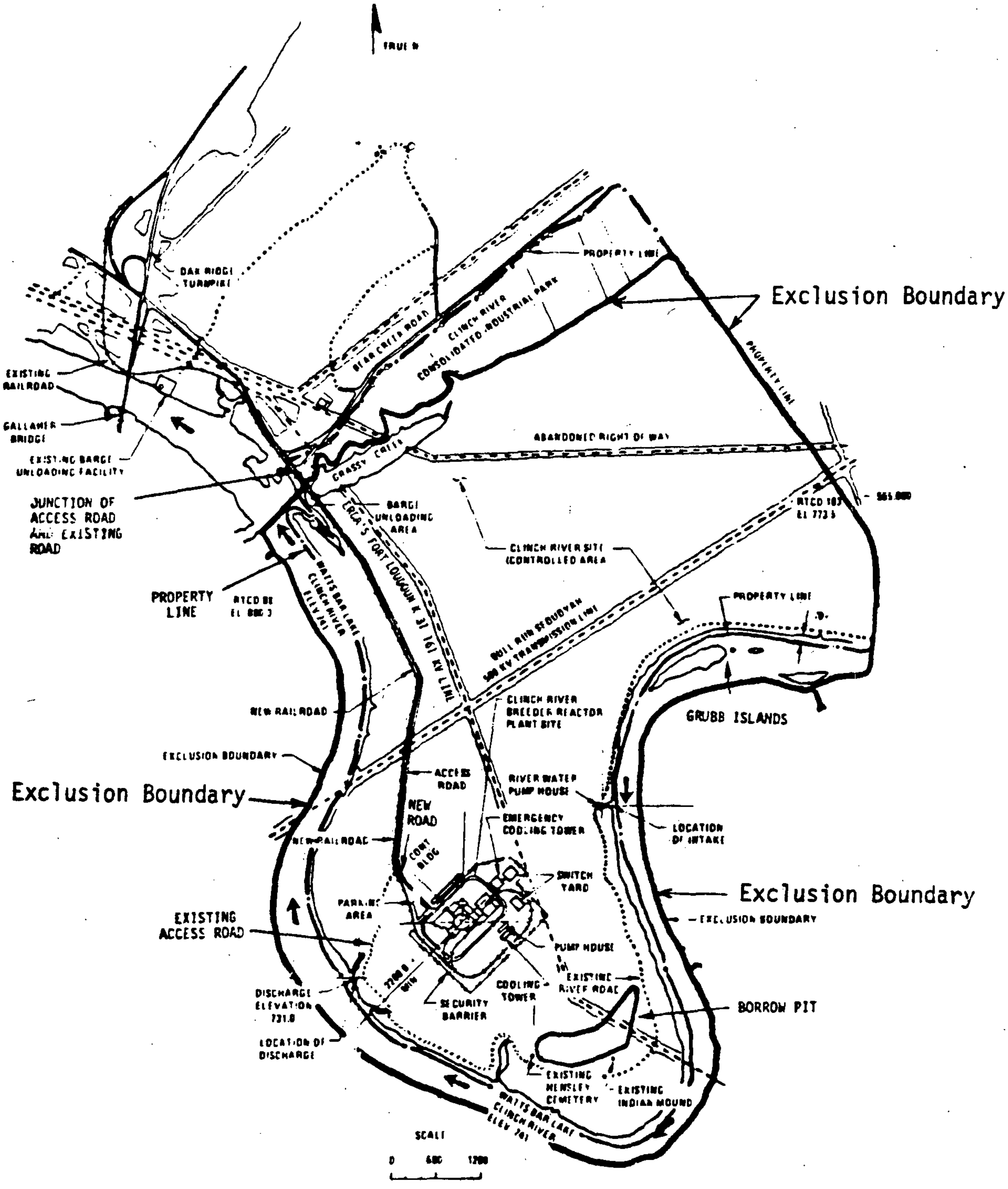

Figure 2.3 Clinch River site exclusion area and property boundaries

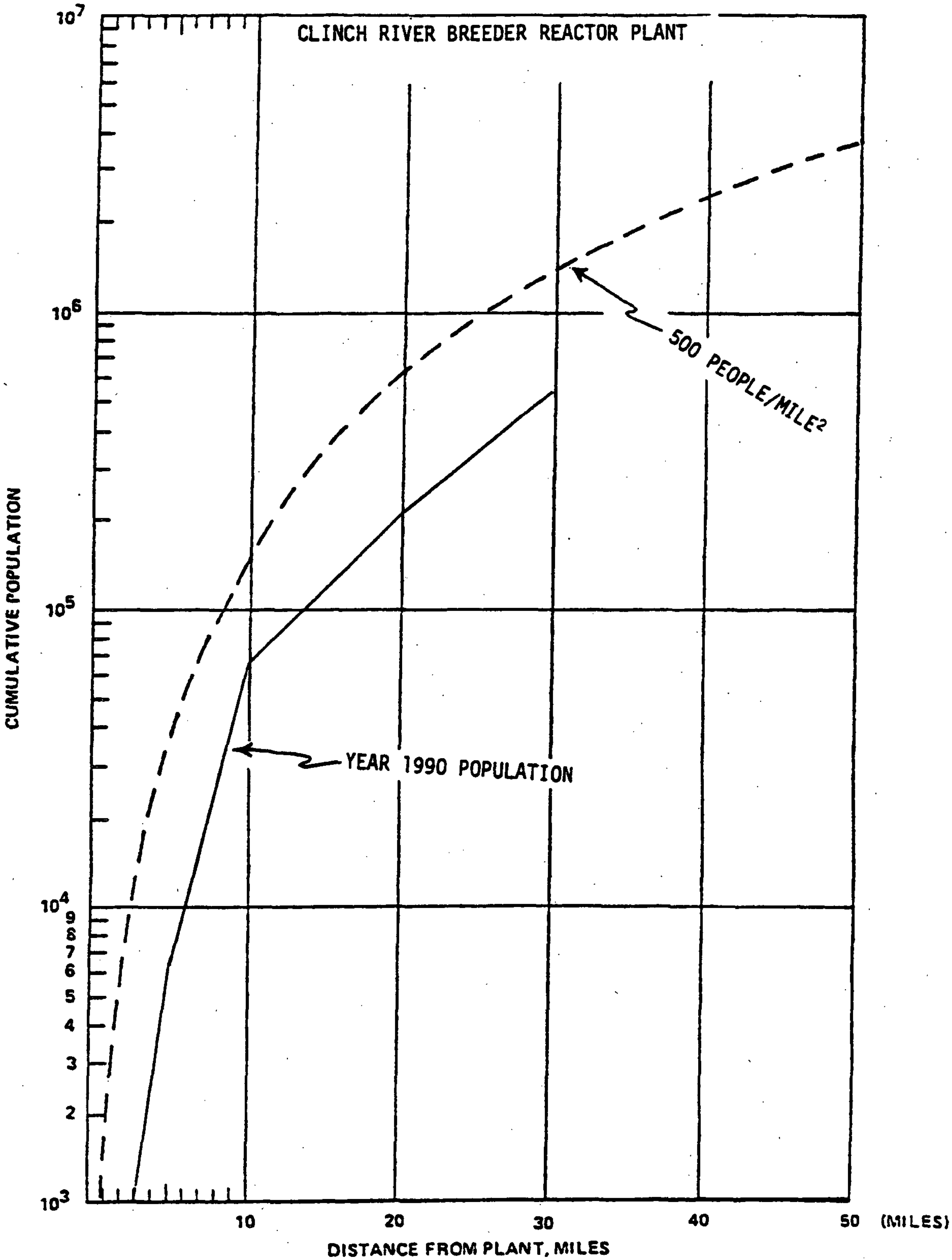

Figure 2.4 Cumulative population distribution (1990)

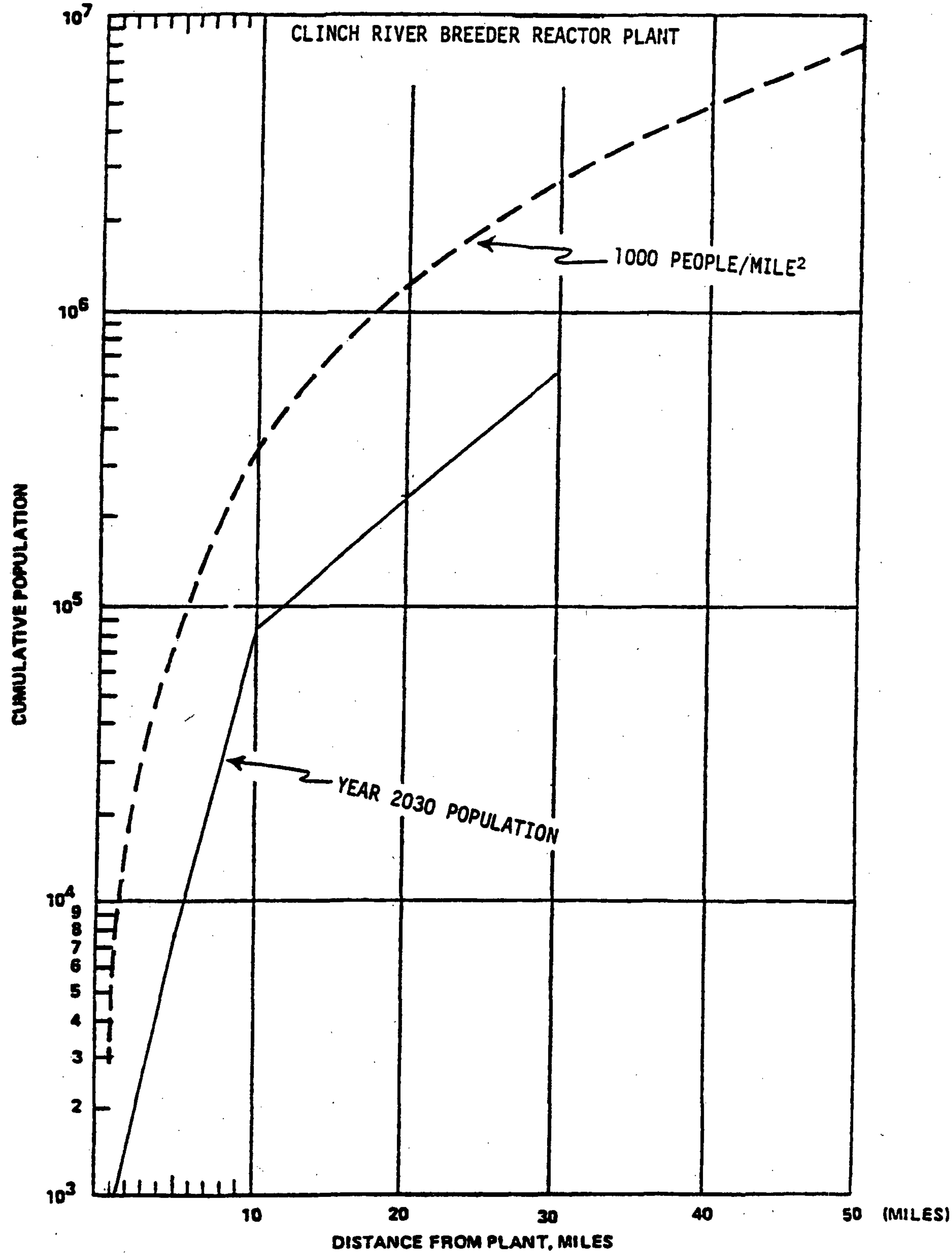

Figure 2.5 Cumulative population distribution (2030)

Table 2.1 1980 census and projected resident cumulative populations

Radius, miles	1980	1990	2030
0-5	4,440	4,680	5,380
0-10	52,040	57,980	67,580
0-20	205,340	202,580	214,280
0-30	516,540	550,180	608,280

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Conformance With General Design Criteria

The purpose of this section is to present the final CRBR principal design criteria, to provide the basis for each criterion, and to explain the differences between the CRBR criteria and the general design criteria of 10 CFR 50, Appendix A.

Paragraph 50.34 of 10 CFR 50 requires that the preliminary safety analysis report (PSAR) for each nuclear power plant lists the principal design criteria for that plant. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety, that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Appendix A to 10 CFR 50 provides a list of general design criteria which establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The general design criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

In July 1974, the staff developed and issued interim principal design criteria for CRBRP. The criteria were based on (1) the general design criteria in Appendix A to 10 CFR 50, (2) general design criteria for liquid metal fast breeder reactor (LMFBR) plants developed and issued for trial use and comment by the American Nuclear Society Standards Committee ANS-54, and (3) draft design safety criteria for CRBRP proposed by the applicants.

On the basis of the comments received on its interim criteria, the staff revised and issued principal design criteria for CRBRP in January 1976. Those criteria were included in Chapter 3.1 of the CRBR PSAR as proposed by the applicants.

As part of the current review of the PSAR, another assessment of the adequacy of the CRBR principal design criteria has been made. The objective of this assessment has been to ensure that the criteria address all aspects of the design important to safety and meet the intent of Paragraph 50.34 of 10 CFR 50.

For CRBR, compliance with the above has been based on ensuring that the criteria

(1) establish requirements for those structures, systems, and components (which are comparable to structures, systems, and components in LWRs) equivalent to or more conservative than the corresponding requirements for LWRs

(2) establish requirements for those structures, systems, and components unique to CRBR that are consistent with their importance to safety and

that reflect an equivalent or more conservative safety approach than that generally applied to LWRs

(3) establish requirements on the CRBR design that will make the likelihood of core disruptive accidents sufficiently low so that they can be excluded from the CRBR design basis

The intent of the criteria is to express the broad requirements that must be met to ensure that the safety of CRBR will be comparable to that of LWRs and that core disruptive accidents will be of sufficiently low likelihood so that they can be excluded from the plant design basis.

The resulting criteria represent the minimum design requirements acceptable to the staff for CRBRP and provide the point of departure for the development of detailed engineering criteria and final design.

The basic method and assumptions used in the development of this final set of principal design criteria were as follows:

(1) The criteria address only those structures, systems, and components associated with accident prevention to ensure the accommodation of events within the design basis and to make the likelihood of core disruptive accidents sufficiently low so that they can be excluded from the CRBR design basis. Because the prevention of accidents, the selection of the design-basis events, and the reduction of the likelihood of accidents beyond the design basis become the key factors that dictate the necessary plant structures, systems, and components, proper selection of the types of design-basis events and identification of the factors that can lead to accidents beyond the design basis are necessary to ensure a complete and adequate set of principal design criteria. Therefore, the approach used in selecting the design-basis events, in identifying the factors that can lead to accidents beyond the design basis, and in treating accidents beyond the design basis is considered important and is summarized below:

    (a) A set of design-basis events was developed by the applicants and reviewed by the staff. These events define conditions under which the plant design is assessed and for which protective systems and/or features must be provided to ensure accommodation of the event. The design-basis events include those that are expected to occur one or more times during the life of the plant (called normal operation, which includes anticipated operational occurrences) and those that are not expected to occur during the life of the plant but that are chosen as upper-bound events for design purposes to envelop all events considered credible (called postulated accidents). The selection of the design-basis events and their relative likelihood of occurrence involved such considerations as the CRBR design, experience at other nuclear power plants, likelihood of occurrence, plant lifetime, and plant operating modes. The acceptability of these design-basis events is based on engineering judgment. The design-basis events for CRBR are described in Section 15 of this report.

    Specific limits or acceptance criteria are established for determining the acceptability of the plant response to the design-basis events

Different limits may be applied for different categories of design-basis events. These limits are discussed elsewhere in this report.

(b) The factors that can lead to core disruptive accidents in CRBR and those features necessary to make the likelihood of core disruptive accidents sufficiently low so that they can be excluded from the CRBR design basis were addressed by the staff in a letter R. P. Denise (NRC) to L. W. Caffey (CRBR Project Office-DOE), dated May 6, 1976. The principles discussed in that letter are embodied in the CRBR principal design criteria.

(c) The proposed plant was also analyzed to determine its ability to withstand accidents beyond the design basis. The types of accidents analyzed, the acceptance criteria, the margin (and the addition of any features to the plant to provide additional margin) for accommodation are based on specific analysis of core disruptive accidents and engineering judgment. Design criteria for these additional features are addressed separately in Appendix A to this SER.

(2) The criteria were based on the general design criteria for LWRs contained in Appendix A to 10 CFR 50. In the development of the CRBR criteria, the staff considered the guidance in Appendix A to 10 CFR 50 as follows:

(a) Where there was no substantial difference between CRBRP and LWRs, the staff considered the LWR criteria applicable and adopted the appropriate criteria.

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

(c) On the basis of its review, the staff identified and developed additional criteria for CRBRP where there were significant differences between LWR plants and the CRBRP.

The criteria in Appendix A to 10 CFR 50 were used to the maximum extent possible. Wording changes were made only to adapt the criteria to CRBR terminology, for completeness, or to add additional requirements or conservatisms deemed appropriate because of the inherent differences between LWRs and CRBR or the more limited operating experience with LMFBRs versus LWRs. Adhering as closely as possible to the wording and requirements of the LWR criteria is considered appropriate because a level of safety equivalent to that of an LWR is the goal and the LWR criteria are implemented by existing NRC guides and technical positions that could be impacted by changes in the criteria.

In addition, because the resolution of certain generic issues, such as station blackout, may affect the design criteria for all future plants, the staff did not try to prejudge the resolution of these issues by modifying the principal design criteria for CRBR to address these issues. Instead, the staff addresses them on a case-by-case basis in Appendix B of this report.

(3) Draft ANS Std. 54.1 "General Safety Design Criteria for an LMFBR Nuclear Power Plant," July 1981, was considered along with the principal design criteria for the fast flux test facility (FFTF) and the Southwest Experimental Fast Oxide Reactor (SEFOR). These criteria have been examined for their applicability to CRBR and, where appropriate, have been adopted.

The basis for a decision regarding the adequacy and completeness of the CRBR criteria involves a judgment of whether the design-basis-event spectrum identifies the types of design-basis events applicable to CRBR and whether the criteria adequately address (1) the safety function of those structures, systems, and components associated with accident prevention and accommodation of the design-basis events and (2) those factors that will reduce the likelihood of core disruptive accidents sufficiently so that they can be excluded from the design basis.

The staff has reviewed the design-basis-event spectrum and considers it complete in addressing the various types of events that should be included in the design basis for CRBR. In reaching this conclusion the staff considered such factors as experience at other nuclear power plants and the planned operating modes of CRBR, discussed in more detail in Section 15 of this report. The design criteria are intended to address the design features necessary in the plant to accomplish the following basic safety functions:

(1) prevention of accidents
(2) shutdown of the reactor
(3) removal of decay heat
(4) containment of radioactive material

Other considerations affecting plant design such as sabotage and radiation protection of plant personnel are addressed in the Code of Federal Regulations and design criteria supplementing these requirements were not considered necessary. In determining the completeness of the criteria, the staff considered (1) the general design criteria of 10 CFR 50, Appendix A, as guidance; (2) the unique characteristics of CRBR; (3) the factors discussed in the May 6, 1976 letter from R. P. Denise to L. W. Caffey; (4) draft ANS Std. 5.4.1; and (5) the design criteria of FFTF and SEFOR in developing the CRBR criteria. In general, criteria equivalent to LWR criteria were developed except in the areas of reactor shutdown systems and decay heat removal where requirements more conservative than the corresponding LWR requirements were imposed on CRBR to provide additional assurance that these functions are performed. These areas were chosen for additional conservatism because of the unique characteristics of CRBR (positive sodium void coefficient and potential for recriticality of molten fuel) which places added emphasis on performing these functions.

The staff believes that the principal design criteria given in this section establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety and will result in a CRBR design that presents a risk to public health and safety no greater than that from an LWR. If, in the future, new information bearing on these criteria comes to light which suggests that modifications are necessary, it will be handled on a case-by-case basis. This is consistent with the policy in 10 CFR 50, which acknowledges the possibility of addition to the general design criteria.

These criteria have been transmitted to the applicants (letter, P. S. Check to J. R. Longenecker, Dec. 27, 1983) for implementation. Accordingly, in Section 3.1 of the PSAR, the applicants have provided for each criterion a brief summary of how the CRBR design complies with the criterion. In many instances reference is made to other PSAR sections for details. The staff's review of the applicants' compliance is addressed in other sections of this SER as shown in Table 3.1.

Listed below are the CRBR principal design criteria. Provided with each criterion is an identification of changes from the 10 CFR 50, Appendix A criteria, and a justification for its inclusion or a justification of omission for each criterion from 10 CFR 50, Appendix A, not included in the CRBR set.

3.1.1 CRBR Principal Design Criteria

3.1.1.1 Definitions

Anticipated Operational Occurrences. Anticipated operational occurrences are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include, but are not limited to, an inadvertent control rod withdrawal, tripping of sodium circulating pumps, loss of all offsite power, and tripping of the turbine generator set.

Fuel Design Limits. Fuel design limits are those limits such as temperature, burnup, fluence, and cladding strain, that are specified by the designer for normal operation and anticipated operational occurrences beyond which fuel rod failure may occur.

Heat Transport System. The heat transport system (HTS) is the aggregate of systems and/or components containing the heat transport fluids and used for extracting heat from the reactor and transporting it to the equipment used for electrical power conversion during normal operation or, after plant shutdown, to an ultimate heat sink. As such it includes the reactor residual heat extraction system. It does not include systems whose prime function is the cooling of structures or equipment.

Intermediate Coolant Boundary. The intermediate coolant boundary consists of those components such as heat exchangers, piping, pumps, tanks, and valves, that are part of the intermediate coolant system or are connected to the intermediate coolant system up to and including any and all of the following:

(1) the passive barrier between the intermediate coolant and the working fluid of other portions of the heat transport system

(2) the first valve normally closed or automatically isolable during normal reactor operation in piping that does not penetrate reactor containment

(3) the outermost containment isolation valve in piping that penetrates reactor containment

Intermediate Coolant System. The intermediate coolant system consists of those components, such as intermediate pumps, steam generator, expansion tanks, and connecting piping, that contain intermediate coolant and are necessary to transport core heat from the primary coolant system to the steam system.

Normal Operation. Normal operation consists of steady-state operation and those departures from steady-state operation which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. It includes conditions such as startup, normal shutdown, standby, load following, anticipated operational occurrences, operation with specific equipment out of service as permitted by Technical Specifications, and routine inspection, testing, and maintenance of components and systems during any of these conditions, if consistent with the Technical Specifications.

Nuclear Power Unit. A nuclear power unit consists of a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Postulated Accidents. Postulated accidents are those events that, although not expected to occur, are selected, in addition to normal and anticipated operational occurrences, for establishing design bases of systems, components, and structures. They represent bounding events that envelop variations in the types of accidents considered and are the upper-bound design-basis events. Postulated accidents together with normal operation and anticipated operational occurrences represent the total spectrum of design-basis events.

Reactor Coolant Boundary. The reactor coolant boundary consists of those components, such as the vessel, heat exchangers, piping, pumps, tanks, and valves, that are part of the reactor coolant system or are connected to the reactor coolant system up to and including any and all of the following:

(1) the second of two valves normally closed or automatically isolable during normal reactor operation

(2) the passive barrier between the reactor coolant and the working fluid of other portions of the heat transport system

Reactor Coolant System. The reactor coolant system consists of those components, such as the reactor vessel, primary pumps, intermediate heat exchanger, valves, and connecting piping, that contain primary radioactive coolant and are necessary to transport reactor core heat to the intermediate coolant system.

Reactor Residual Heat Extraction System. The reactor residual heat extraction system is that portion of the heat transport system which, after plant shutdown, is capable of extracting heat from the reactor coolant and transporting this heat to an ultimate heat sink.

Single Failure. A single failure is an occurrence that results in loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.*

3.1.1.2 CRBR Principal Design Criteria

Criterion 1--Quality Standard and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Justification--This criterion is identical to General Design Criterion (GDC) 1 in 10 CFR 50, Appendix A. The intent of this criterion is to require that work important to safety be performed in a fashion that ensures the end product will meet all the design and construction standards that apply. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 2--Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect

(1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety functions to be performed

Justification--This criterion is identifical to GDC 2 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the plant be designed to withstand natural phenomena that could affect the ability of the plant's safety

*Single failure of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure is under development.

systems to perform their functions. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 3--Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components.

Justification--This criterion is identifical to GDC 3 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the plant be designed and constructed to minimize the effect of fires on plant systems and hardware important to safety. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 4--Protection Against Sodium and NaK Reactions

Systems, components, and structures containing sodium or NaK shall be designed and located to limit the consequences of chemical reactions resulting from a sodium or NaK spill. Special features such as inert atmosphere vaults shall be provided as appropriate for the reactor coolant system. Fire-control systems and means to detect sodium, NaK, or their reaction products shall be provided to limit and control the extent of such reactions to ensure that the functions of components important to safety are maintained. Means shall be provided to limit the release of reaction products to the environment, as necessary, to protect plant personnel and to avoid undue risk to the public health and safety. Material that might come in contact with sodium or NaK shall be chosen to minimize the adverse effects of possible chemical reactions or microstructural changes. In areas where sodium or NaK chemical reactions are possible, structures, components, and systems important to safety, including electrical wiring and components, shall be designed and located so that the potential for damage by sodium chemical reactions is minimized. Means shall be provided as appropriate to minimize possible contacts between sodium/NaK and water. A single failure of a passive boundary shall not permit the contact of primary coolant with water/steam. The effects of possible interactions between sodium/NaK and concrete shall be considered in the design.

The sodium-steam generator system shall be designed to detect sodium-water reactions and limit the effects of the energy and reaction products released by such reactions so as to prevent loss of safety functions of the heat transport system.

Justification--Because of the use of sodium and NaK as coolants, this criterion is unique to CRBR. The intent of this criterion is to require that the plant be designed and constructed with special consideration given to the effects of sodium and NaK, including the detection, consequences, and mitigation of sodium and NaK reactions and spills.

Because of the high chemical activity of sodium and NaK (react vigorously with water and oxygen), leaks or spills of this material can lead to chemical reactions, fires, and combustion products not possible in LWRs. This high chemical activity requires that special measures be taken to prevent contact of the liquid metal with water, concrete, and oxygen and to extinguish any liquid-metal fires that do occur. In addition, means to detect liquid-metal spills and to protect other plant equipment and personnel from the corrosive and potentially radioactive combustion products of liquid-metal fires are required. Thus, a new criterion addressing protection against liquid-metal reactions is considered appropriate for CRBR.

Criterion 5--Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including anticipated operational occurrences, maintenance, and testing, as well as with postulated accidents, including the effects of sodium and NaK and their aerosols and combustion products. These structures, systems, and components shall be appropriately protected against dynamic effects, such as the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear-power unit.

Justification--This criterion is identical to GDC 4 in 10 CFR 50, Appendix A, except that (1) the words "including loss-of-coolant accidents" have been removed from the end of the first sentence, since design features have been added to the plant to reduce the likelihood of total loss-of-coolant accidents (see Criterion 27), (2) the words "including anticipated operational occurrences" have been added to the first sentence to ensure the criterion applies to all design-basis events, and (3) the words "including the effects of Na and NaK and their aerosols and combustion products" have been added to the first sentence to emphasize this unique aspect of CRBR. The intent of this criterion is to require that the plant be designed and constructed to withstand the effects of normal and abnormal operation without causing a loss of other plant systems or hardware important to safety. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 6--Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Justification--This criterion is identifical to GDC 5 in 10 CFR 50, Appendix A. The intent of this criterion is to ensure that components and systems important to safety and shared by facilities will not preclude safe shutdown in one facility in the event of an accident in another facility. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 7--Sodium Heating Systems

Heating systems shall be provided as necessary for systems and components important to safety that contain, or may be required to contain, sodium or sodium aerosol. The heating systems and their controls shall be appropriately designed with suitable redundancy to ensure that the temperature distribution and rate of change of temperature in sodium systems and components are maintained within design limits assuming a single failure. The heating system shall be designed so that its failure will not impair the safety function of associated systems and components.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that systems important to safety that contain sodium or sodium aerosols and that require a controlled temperature for the system to perform its safety function be provided with a heating system capable of ensuring that desired temperatures are maintained and designed to preclude overheating the components to which they are attached. Because sodium freezes at 208°F, external heat is required to be supplied to the sodium systems under certain plant conditions to keep the sodium molten and to keep sodium aerosol from condensing and plugging flow paths exposed to sodium vapor. Certain portions of the sodium and sodium cover gas systems are considered important to safety and require external heat to enable them to perform their functions (e.g., pressure relief lines and standby decay heat removal systems). Accordingly, a heating system is required that will maintain the desired temperature, provide sufficient instrumentation to verify these temperatures, and not be an initiator of a failure of the sodium system. Thus, a new criterion addressing requirements on sodium heating systems is considered appropriate for CRBR.

Criterion 8--Reactor Design

The reactor and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Justification--This criterion is identical to GDC 10 in 10 CFR 50, Appendix A, except that the word "core" was removed in the first line so as not to imply this criterion should be limited to the core. The intent of this criterion is to require that fuel design limits not be exceeded during normal and anticipated operational occurrences. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 9--Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Justification--This criterion is identical to GDC 11 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the nuclear characteristics of the core provide a prompt negative nuclear feedback in response to positive reactivity insertions while the plant is operating in the power range. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 10--Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Justification--This criterion is identical to GDC 12 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the plant be designed to prevent power oscillations that can result in exceeding fuel design limits. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 11--Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, including anticipated operational occurrences, and for postulated accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Justification--This criterion is identical to GDC 13 in 10 CFR 50, except that the words "for anticipated operational occurrences" have been changed to "including anticipated operational occurrences" to be consistent with the definitions, the word "postulated" has been added in front of the word "accident" to be consistent with the definitions, and the word "pressure" has been removed from the phrase "reactor coolant pressure boundary" to conform to CRBR terminology. The intent of this criterion is to require sufficient instrumentation and control to monitor and maintain system variables within their prescribed values and also to monitor key parameters throughout normal and postulated accident ranges. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 12--Reactor Coolant Boundary

The reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Justification--This criterion is identical to GDC 14 in 10 CFR 50, Appendix A, except that the word "pressure" has been removed from the phrase "reactor coolant pressure boundary" to conform to CRBR terminology. The intent of this criterion is to require high integrity of the reactor coolant boundary and low probability of gross rupture, to maintain adequate core cooling, and to contain the radioactive coolant. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 13--Reactor Coolant System Design

The reactor coolant system and associated control, protection, auxiliary, and sodium heating systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Justification--This criterion is identical to GDC 15 in 10 CFR 50, Appendix A, except that the words "and sodium heating systems" were added to address the CRBR design and the word "pressure" has been removed from the phrase "reactor coolant pressure boundary" to conform to CRBR terminology. The intent of this criterion is to require that the reactor coolant system and its associated systems be designed to preclude subjecting the reactor coolant boundary to conditions in excess of its normal design conditions. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 14--Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Justification--This criterion is identical to GDC 16 in 10 CFR 50, Appendix A. The intent of this criterion is to require that a reactor containment system act as a final barrier against the uncontrolled release of radioactive material to the environment. The containment and associated systems include the containment building, its penetrations, and any systems that act as extensions of containment. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 15--Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded as a result of normal operation, including anticipated operational occurrences, and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded. One of the these circuits shall be designed to be available within a few seconds following any postulated accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Justification--This criterion is identical to GDC 17 in 10 CFR 50, Appendix A, except that the word "pressure" has been removed from the phrase "reactor coolant pressure boundary" to conform to CRBR terminology and the reference to "loss-of-coolant accident" has been removed, since Criterion 27 requires features to reduce the likelihood of a total loss of coolant.

In addition, the words "normal operation, including" have been added to the second sentence to include all design-basis events. The intent of this criterion is to require a highly reliable onsite and offsite electrical power system to ensure power to those systems and components important to safety. The reliability of the electrical power system is intended to be sufficiently high to support the requirements of the systems it serves. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 16--Inspection and Testing of Electric Power System

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Justification--This criterion is identical to GDC 18 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the electric power systems for the plant be designed to allow for periodic testing and inspection to verify their operability. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 17--Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under postulated accident conditions (including those conditions addressed in Criterion 4). Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and with a design capability for subsequent control of the reactor at any coolant temperature lower than that during the hot shutdown condition.

Justification--This criterion is identical to GDC 19 in 10 CFR 50, Appendix A, except (1) the words "including loss-of-coolant accident" have been removed from the first sentence and replaced with words addressing sodium and NaK reactions since this is the main concern in a coolant leakage situation in CRBR, (2) the word "postulated" has been added in the first sentence to be consistent with the definitions, and (3) the last sentence has been changed to remove reference to cold shutdown and replace it with conditions applicable to CRBR.

The intent of this criterion is to require that the control room be designed to permit access and occupancy under all normal and postulated accident conditions and that in the event the control room is uninhabitable or that its reactor shutdown or decay heat removal functions cannot otherwise be performed, alternate shutdown locations, independent of the control room instrumentation and controls, be provided to perform those functions. As such, the intent of this criterion is not unique to LWRs and with the above modifications is considered applicable to CRBR.

Criterion 18--Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense postulated accident conditions and to initiate the operation of systems and components important to safety.

Justification - This criterion is identical to GDC 20 in 10 CFR 50, Appendix A, except that the word "postulated" has been added to Item (2) to be consistent with the definitions. The intent of this criterion is to require that the plant protection system sense offnormal conditions and automatically actuate appropriate protective systems important to safety. As such this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 19--Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation including a capability to test channels independently to determine failure and losses of redundancy that may have occurred.

Justification--This criterion is identical to GDC 21 in 10 CFR 50, Appendix A. The intent of this criterion is to require a highly reliable plant protection system. As such this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 20--Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena and of normal operation, including anticipated operational occurrences, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Justification--This criterion is identical to GDC 22 in 10 CFR 50, Appendix A, except that the words "including anticipated operational occurrences," have been added to the first sentence for completeness. The intent of this criterion is to require that the plant protection system be designed to have diverse independent channels thus minimizing the potential for common mode failure. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 21--Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, sodium, sodium reaction products, and radiation) are experienced.

Justification--This criterion is identical to GDC 23 in 10 CFR 50, Appendix A, except the words "sodium, sodium reaction products" have been added to cover additional adverse environments possible in CRBR. The intent of this criterion is to require that the plant protection system be designed in a fail-safe fashion. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 22--Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Justification--This criterion is identical to GDC 24 in 10 CFR 50, Appendix A. The intent of this criterion is to ensure that where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels will be capable of providing the protective action even when degraded by a second random failure. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 23--Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods.

Justification --This criterion is identical to GDC 25 in 10 CFR 50, Appendix A, except that the words "(not ejection or dropout)" were removed since they do not apply to CRBR. The intent of this criterion is to require that the plant protection system be designed to terminate any anticipated operational occurrences involving the reactivity control system initiated by a single malfunction without exceeding acceptable fuel design limits. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 24--Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One system shall be capable of independently and reliably sensing and responding to offnormal conditions to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as a stuck rod, specified acceptable fuel design limits are not exceeded. The other system shall be capable of independently and reliably sensing and responding to offnormal conditions to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as a stuck rod, the capability to cool the core is maintained. Each system shall have sufficient worth, assuming failure of any single active component, to shut down the reactor from any operating condition to zero power and maintain subcriticality at the hot shutdown temperature of the coolant, with allowance for the maximum reactivity associated with any anticipated operational occurrence or postulated accident. One of the systems shall be capable of holding the reactor core subcritical for any coolant temperature lower than the hot shutdown temperature.

Justification--This criterion is similar to GDC 26 in 10 CFR 50, Appendix A, with major changes as noted below. The words "(xenon burnout)" were removed since this is not a major concern in LMFBRs and the words "under cold conditions" at the end of the last sentence were replaced with the words "for any coolant temperature lower than the hot shutdown temperature" to conform to CRBR terminology to cover all expected conditions below the hot shutdown condition. In addition, the first paragraph was revised to require that each reactivity control system be able to independently sense and terminate anticipated operational occurrences assuming a single failure. Also, the next to the last sentence was added to specify requirements on shutdown margin thus ensuring that the reactivity control systems would be able to insert sufficient worth to shut down the reactor and overcome any positive reactivity inserted by the design-basis event. The intent of this criterion is to require two independent reactivity control systems of different design, each capable of responding to offnormal events. One system is to maintain the fuel within acceptable design limits; the other system is to maintain core coolability (assuming the first system will not respond). For each system this criterion pertains to both the sensing portion of the system (including its associated electronics and logic) and the mechanical portion of the system. This is a more conservative criterion than that applied to LWRs. It is the staff's position that because of the inherent differences in nuclear characteristics between LWRs and CRBR (i.e., positive sodium void coefficient and potential for recriticality), two independent, diverse, redundant shutdown systems should be provided for CRBR to reduce the likelihood of failure to scram. Requiring the second system to only maintain core coolability is considered acceptable since it would take a complete failure of the first system to get to this condition, an event which is beyond the design basis. As such, this criterion, as modified, is considered appropriate for CRBR.

Criterion 25--Reactivity Control Systems Capability

The reactivity control systems shall be designed to have an independent capability of reliably sensing and responding to offnormal conditions to assure that under postulated accident conditions and with appropriate margin for malfunctions such as a stuck rod, the capability to cool the core is maintained.

Justification--This criterion is similar to GDC 27 in 10 CFR 50, Appendix A, except for the following major changes: (1) the words "in conjunction with poison addition by the emergency core cooling system" have been removed since CRBR will not have an emergency core cooling system and (2) the requirement for a combined capability has been modified to require that each system independently be capable of sensing and shutting down the reactor in response to postulated accident conditions (assuming a single failure) for the same reason as that stated in the justification for Criterion 24. As such, this criterion as modified is considered appropriate for CRBR.

Criterion 26--Heat Transport System Design

The heat transport system shall be designed to reliably remove heat from the reactor and transport the heat to the turbine generator or ultimate heat sinks under all plant conditions of normal operation, including anticipated operational occurrences, and postulated accidents. Consideration shall be given to provision of independence and diversity to provide adequate protection against common mode failures. The system safety functions shall be to

(1) provide sufficient cooling to prevent exceeding specified acceptable fuel design limits during normal operation and following anticipated operational occurrences,

(2) maintain integrity of the reactor coolant boundary that is sufficient to provide adequate core cooling following postulated accidents

Following the loss of a flow path, the heat transport system shall include at least two independent flow paths, each capable of performing the safety functions following shutdown.*

*This requirement is not intended to preclude two-loop operation, provided the system safety functions can be appropriately met.

The system shall include suitable interconnections, leak detection, and isolation and containment capability to ensure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the safety function can be accomplished assuming a single failure.

Justification--This criterion is unique to CRBR. The intent of this criterion is to ensure that the heat transport system (HTS) will be designed for normal operation, including anticipated operational occurrences, and will be able to withstand postulated accidents and that at least two flow paths will be available for decay heat removal. In CRBR the HTS will remove reactor heat during all normal and offnormal conditions. No system equivalent to an emergency core cooling system will be provided. Accordingly, it is considered important that the HTS be able to accommodate all design-basis events assuming a single failure and a principal design criterion addressing this point is considered appropriate.

Criterion 27--Assurance of Adequate Reactor Coolant Inventory

The reactor coolant boundary and associated components, control, and protection systems shall be designed to limit loss of reactor coolant so that an inventory adequate to perform the safety functions of the heat transport system is maintained under normal operation, including anticipated operational occurrences, and postulated accident conditions.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the HTS design provide for retention of sufficient sodium inventory (in the event of a leak) to ensure adequate decay heat removal capability. Since leaks in the coolant system do not lead to the coolant flashing to vapor, a coolant makeup system or an emergency core cooling system similar to that in LWRs is not required for CRBR provided that it can be shown that leaks in the coolant system will not lead to uncovering the core or loss of core cooling. To show the above it is necessary to design the plant for sufficient retention of coolant inventory in the event of a leak to maintain a decay heat removal path by ensuring that the physical design of the system will include features (such as elevated piping, guard vessels, and expansion tanks) that will preclude loss-of-coolant inventory that could lead to inadequate core cooling. This criterion requires that such features be a part of the CRBR design and is considered appropriate for a principal design criterion.

Criterion 28--Quality of Reactor Coolant Boundary

Components which are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and to the extent practical, identifying the location of the source of reactor coolant leakage.

Justification--This criterion is identical to GDC 30 in 10 CFR 50, Appendix A. The intent of this criterion is to require the use of quality standards in the design, fabrication, and testing of primary reactor coolant boundary components and to provide the capability to detect leaks in the system. For CRBR this criterion has an additional intent, namely that of minimizing the likelihood of leaks greater than those assumed in the design basis. As such, this criterion is not considered unique to LWRs and is considered applicable to CRBR.

Criterion 29--Fracture Prevention of Reactor Coolant Boundary

The reactor coolant boundary shall be designed with sufficient margin to assure that when stressed under normal operation, including anticipated operational occurrences, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under normal operation, including anticipated operational occurrences, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry and irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Justification--This criterion is identical to GDC 31 in 10 CFR 50, Appendix A, except that the words "including anticipated operational occurrences" were added in the first and second sentences to include all design-basis events and the words "service degradation of material properties, creep, fatigue, stress rupture" and "effects of coolant chemistry" were added in the second sentence to address the unique concerns of CRBR because of the high design and operating temperatures of the system and the proposed use of sodium as a coolant. The intent of this criterion is to require that the primary reactor coolant boundary components be designed to avoid brittle and rapidly propagating fracture modes thus minimizing the likelihood of leaks greater than those assumed in the design basis. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 30--Inspection of Reactor Coolant Boundary

Components which are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection and testing of areas and features important to safety to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program.

Justification--This criterion is identical to GDC 32 in 10 CFR 50, Appendix A, except that the words "pressure" and "for the reactor pressure vessel" were removed to use terminology applicable to CRBR and because the inspection requirements are intended to apply to the entire primary boundary not just the reactor vessel. The intent of this criterion is to require that the design allow for periodic inspection and an appropriate material surveillance program for the primary coolant boundary. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 31--Intermediate Coolant System

The intermediate coolant system shall be designed to transport heat reliably from the reactor coolant system to the steam/feedwater systems as required for the reactor coolant system to meet its safety functions under all plant conditions of normal operation, including anticipated operational occurrences, and postulated accident conditions. The intermediate coolant system shall contain coolant that is not chemically reactive with the reactor coolant.

A pressure differential shall be maintained across a passive boundary between the reactor coolant system and the intermediate coolant system so that any leakage would tend to flow from the intermediate coolant system to the reactor coolant system unless it can be shown that other provisions are acceptable on some defined basis.

Justification--This criterion is unique to CRBR because LWRs do not have intermediate loops. The intent of this criterion is to require an intermediate heat transfer loop between the steam system and the reactor coolant system, to require that the intermediate system have greater pressure than the primary loop (thus preventing the leakage of radioactive sodium into the nonradioactive intermediate system), and to require a coolant that will not react chemically with the primary sodium.

In CRBR, heat from the reactor coolant system will be transferred to an intermediate sodium loop and then to the steam system. The main purpose in having an intermediate loop is to separate the radioactive primary sodium from the steam thus avoiding the possibility of a sodium-water reaction involving radioactive sodium. To help accomplish this function, the intermediate system pressure is maintained at a higher level than the pressure in the reactor coolant system. Given such a loop, it is also necessary that it reliably transfer reactor heat to the steam system to meet its safety function for decay heat removal. Additionally, because there are no containment isolation valves in the intermediate system, it is considered a closed system and acts as an extension of containment. Given these important safety functions, the application of a criterion on the intermediate system is considered appropriate for CRBR.

Criterion 32--Fracture Prevention of Intermediate Coolant Boundary

The intermediate coolant boundary shall be designed with sufficient margin to ensure that when stressed under normal operation (including anticipated operational occurrences), maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under normal operation (including anticipated operational occurrences), maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry and irradiation on material properties, and (3) residual, steady-state, and transient stresses, and (4) size of flaws.

Justification--This criterion is unique to CRBR because LWRs do not have intermediate loops. It is identical to CRBR Criterion 29. The intent of this criterion is to require that the intermediate coolant boundary components be designed to avoid brittle and rapidly propagating fracture modes. Since the intermediate system provides important safety functions (i.e., decay heat removal path and isolation of the radioactive reactor coolant from the steam system), a criterion for fracture prevention similar to that for the reactor coolant system is considered appropriate for CRBR.

Criterion 33--Inspection and Surveillance of Intermediate Coolant Boundary

Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection of areas and features important to safety to assess their structural and leaktight integrity and (2) an appropriate material surveillance program for the intermediate coolant boundary. Means shall be provided for detecting intermediate coolant leakage.

Justification--This criterion is unique to CRBR because LWRs do not have intermediate loops. It is similar to CRBR Criterion 30. The intent of this criterion is to require that the design allow for periodic inspection, an appropriate material surveillance program, and leak detection capability. Because the intermediate system provides important safety functions, a criterion for inspection, surveillance, and leak detection similar to that for the primary system is considered appropriate for CRBR.

Criterion 34--Reactor and Intermediate Coolant and Cover Gas Purity Control

Systems shall be provided to monitor and maintain reactor, intermediate coolant, and cover gas purity within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, (3) radionuclide concentrations, and (4) detection of sodium-water reactions.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the plant have systems to monitor and maintain the purity of the reactor, the intermediate sodium, and the cover gas for the reactor coolant and intermediate system. The purity of the sodium coolant directly affects the corrosion rate of the components exposed to the sodium. Since the design of the plant takes into consideration an allowance for corrosion, the purity of the sodium must be monitored and kept within a range consistent with the corrosion allowance. In addition, sodium with a high level of impurities can tend to plug flow passages in cooler areas of the system and thus it is important to maintain the sodium plugging temperature below the minimum sodium temperature in the system. The purity of the sodium cover gas affects the purity of the sodium, the formation of sodium compounds in the cover gas space (with a resulting potential for blockage of gas flow paths), the ability to detect fission gas leakage from fuel or blanket pins, and the ability to detect sodium-water reactions in the steam generators. Maintaining limits on cover gas purity is thus required to ensure that certain plant safety systems will operate as designed. On the basis of the above considerations, this criterion was found appropriate for CRBR.

Criterion 35--Reactor Residual Heat Extraction System.

A reactor residual heat extraction system shall be provided to reliably transfer residual heat from the reactor coolant system to ultimate heat sinks under all plant shutdown conditions following normal operation, including anticipated operational occurrences, and postulated accident conditions. A passive boundary shall normally separate reactor coolant from the working fluids of the reactor residual heat extraction system. Any fluid in the residual heat extraction system that is separated from the reactor coolant by a single passive barrier shall not be chemically reactive with the reactor coolant.

Suitable redundancy, independence and diversity in systems, components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure, with at least two flow paths remaining available for residual heat removal.*

Justification--This criterion is comparable to GDC 34 in 10 CFR 50, Appendix A, except that (1) the word "reliably" was added to the first sentence to specify reliability is a design goal, (2) the third sentence was added to address coolant compatability, (3) the words "independence and diversity in systems" were added to the second paragraph to require that independence and diversity be provided as necessary in decay heat removal paths for reliable decay heat removal, and (4) the words "with at least two flow paths remaining available for residual heat removal" were added to the second paragraph to specify the minimum redundancy. The intent of this criterion is to require reliable means of removing reactor decay heat assuming loss of offsite or onsite power and a single failure that could remove one or more of the four available flow paths from service. This criterion requires that such systems be part of the CRBR design.

The requirement for independence, diversity, and at least two flow paths was added to help minimize the probability of a loss-of-cooling event that could lead to core disruption. It is the staff's position that at least two heat removal paths remain available following a design-basis event and a single failure to provide additional margin for accommodation of failures in the decay heat removal paths. This is a more conservative position than that for LWRs and is considered appropriate because in CRBR a loss of core cooling could lead to a core configuration more reactive than a similar event in a LWR. The statement on coolant compatibility was added to require that the potential for chemical reactions with the radioactive primary coolant be minimized. This is consistent with a similar requirement on the intermediate HTS (Criterion 31).

Criterion 36--Inspection of Reactor Residual Heat Extraction System

The reactor residual heat extraction system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure integrity and capability of the system.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the design of the systems for reactor decay heat removal allow provision for periodic inspection of piping and components to ensure that the system integrity is intact. Having the capability to adequately remove reactor decay heat is important to safety, both in LWRs and in sodium-cooled plants; however, because of the added concern of geometry changes and recriticality on loss of core cooling in CRBR, periodic inspection of these systems for integrity and capability is considered appropriate.

*This requirement is not intended to preclude two-loop operation provided the system safety functions can be appropriately met.

Criterion 37--Testing of Reactor Residual Heat Extraction System

The reactor residual heat extraction system shall be designed to permit appropriate periodic pressure and functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the complete system, and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and following postulated accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the design of the paths for reactor decay heat removal allow provision for periodic testing to ensure that they perform as designed. Having the capability to adequately remove reactor decay heat is important to safety, and in view of the concerns expressed in the discussion of Criteria 35 and 36, requiring periodic testing of these paths is considered appropriate for CRBR.

Criterion 38--Additional Cooling Systems

In addition to the heat rejection capability provided by the reactor residual heat extraction system, systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as necessary. The system safety function shall be to transfer the combined heat load of these structures, systems, and components as required for safety under normal operation, including anticipated operational occurrences, and postulated accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Justification--This criterion is identical to GDC 44 in 10 CFR 50, Appendix A, except that (1) its applicability and title have been extended to cover all additional cooling systems, not just cooling water, (2) the first sentence has been modified to specifically exclude the reactor residual heat extraction system (since it is addressed in Criteria 35, 36, and 37) and to add the words "as necessary" at the end of the sentence, and (3) the second sentence has been modified to add the words "including anticipated operational occurrences and postulated" to make it clear that all design-basis events are included. The intent of this criterion is to require cooling to other components and systems important to safety (which require a controlled temperature to perform their safety function) assuming loss of offsite or onsite power and a single failure. The reliability of the additional cooling systems is intended to be sufficient to support the requirements of the systems it serves. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 39--Inspection of Additional Cooling Systems

The additional cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.

Justification - This criterion is identical to GDC 45 in 10 CFR 50, Appendix A, except that the applicability has been extended to cover all additional cooling systems, not just cooling water. The intent of this criterion is to require that the design of cooling systems that provide cooling to components and systems important to safety allow provisions for periodic inspection of important components. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 40--Testing of Additional Cooling Systems

The additional cooling systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of their components, (2) the operability and the performance of the active components of the systems, and (3) the operability of the complete systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Justification --This criterion is identical to GDC 46 of 10 CFR 50, Appendix A, except that the applicability has been extended to cover all additional cooling systems, not just cooling water, and the words "for reactor shutdown and for loss-of-coolant accidents" have been removed so as not to limit the testing to those conditions. The intent of this criterion is to require that the design of the cooling systems that provide cooling to components and systems important to safety allow provisions for periodic testing to ensure that the system still performs as designed. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 41--Containment Design Basis

The reactor containment structure, including access openings and penetrations, and if necessary, in conjunction with additional postaccident heat removal systems including ex-vessel systems, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, including anticipated operational occurrences, and any of the postulated accidents. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as decay heat in released fission products, potential spray or aerosol formation, and potential exothermic chemical reactions; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservatism of the calculational model and input parameters.

Justification--This criterion is identical to GDC 50 in 10 CFR 50, Appendix A, except that the first sentence has been changed to add the word "including" between "operation" and "anticipated" to be consistent with the definition of anticipated operational occurrences and so that it does not refer to the containment heat removal system but rather to the more general postaccident heat removal systems, and the second sentence has been changed so that it does not refer to metal-water reactions and emergency core cooling but rather to conditions applicable to CRBR such as decay heat, spray and aerosol formation, and potential exothermic chemical reactions. The intent of this criterion is to require that the containment building and any associated penetrations or extensions be designed to accommodate with margin the conditions resulting from all postulated accidents. As such this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 42--Fracture Prevention of Reactor Containment Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under normal operation, including anticipated operational occurrences, maintenance, testing, and postulated accident conditions (1) its metallic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during normal operation, including anticipated operational occurrences, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Justification--This criterion is identical to GDC 51 to 10 CFR 50, Appendix A, except that in the first Item the word "ferritic" has been changed to "metallic" to be less specific regarding the containment material and the words "including anticipated operational occurrences" have been added in the first and second sentences to include all design-basis events. The intent of this criterion is to require that the containment building design provide sufficient margin to avoid brittle failure under all postulated loading conditions. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 43 -- Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Justification--This criterion is identical to GDC 52 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the containment building and equipment inside it be designed to allow periodic leak testing to verify their integrity. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 44--Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Justification--This criterion is identical to GDC 53 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the containment building be designed to permit periodic inspection and leak testing of individual penetrations that rely on resilient seals or bellows seals. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 45--Piping Systems Penetrating Containment

Piping systems penetrating reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Justification--This criterion is identical to GDC 54 to 10 CFR 50, Appendix A. The intent of this criterion is to require that piping penetrations through the containment building be designed to be isolable, if necessary for safety, and that the isolation valves be capable of periodic leak testing. As such this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 46--Reactor Coolant Boundary Penetrating Containment

Each line that is part of or directly connected to the reactor coolant boundary and that penetrates reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Justification--This criterion is identical to GDC 55 in 10 CFR 50, Appendix A, except that the words "or directly connected to" have been added in the first sentence to also include supporting systems (such as drain lines and purification system lines) that are connected to and contain primary reactor coolant. The intent of this criterion is to provide guidance on acceptable configurations for isolation valves for piping systems containing primary reactor coolant that penetrate the containment building. As such, it is not unique to LWRs and is considered applicable to CRBR.

Criterion 47--Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Justification--This criterion is identical to GDC 56 in 10 CFR 50, Appendix A. The intent of this criterion is to provide guidance on acceptable configurations for isolation valves for piping systems that penetrate the containment building and that are open to the containment atmosphere. As such, it is not unique to LWRs and is considered applicable to CRBR.

Criterion 48--Closed System Penetrating Containment

Each line that penetrates primary reactor containment and is neither part of nor directly connected to the reactor coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve, unless it can be demonstrated that containment isolation provisions for a specific class of lines are acceptable on some other defined basis. The isolation valve, if required, shall be either automatic or locked closed or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Justification--This criterion is identical to GDC 57 in 10 CFR 50, Appendix A, except that (1) the words "nor directly connected to" have been added to the first sentence to exclude other systems that penetrate containment and that contain primary coolant since they are addressed in Criterion 46 and (2) a provision to allow an alternate scheme for isolation has been added to allow more flexibility in meeting this requirement. The intent of this criterion is to provide guidance on acceptable containment isolation valve configurations for piping systems not open to containment atmosphere and not containing primary reactor coolant. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 49--Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, sodium aerosols or combustion products, and other substances that may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. The effects of sodium leakage and its potential reaction with oxygen as well as its potential for hydrogen generation when in contact with concrete should be considered.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Justification--This criterion is identical to GDC 41 in 10 CFR 50, Appendix A, except that the words "sodium aerosols or combustion products" have been added in the first sentence and the second sentence has been added to address the effects of sodium, which are unique to CRBR. The intent of this criterion is to require systems to control the composition of the containment atmosphere thus minimizing the potential for violating containment integrity and reducing the concentration of fission products in the atmosphere. This in turn leads to lower releases to the outside atmosphere. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 50--Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

Justification--This criterion is identical to GDC 42 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the design of containment atmosphere cleanup systems allow provision for periodic inspection of important components. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 51--Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Justification--This criterion is identical to GDC 43 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the design of the containment atmosphere cleanup systems allow provision for periodic testing to ensure that the systems perform as designed. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 52--Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Justification--This criterion is identical to GDC 60 in 10 CFR 50, Appendix A. The intent of this criterion is to require that the plant have provisions for the controlled release of gaseous, liquid, and solid radioactive waste. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 53--Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal operation, including anticipated operational occurrences, and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. The fuel handling and its interfacing systems shall be designed to minimize the potential for fuel management errors that could result in fuel rod failure.

Justification--This criterion is identical to GDC 61 to 10 CFR 50, Appendix A, except that the words "operation, including anticipated operational occurrences" have been added in the first sentence to make it clear that all design-basis events are included and the last sentence was added to require features to minimize fuel management errors so that such errors can be eliminated from the CRBR design-basis-event spectrum. The intent of this criterion is to provide design guidelines for the fuel storage, fuel handling, radioactive waste, and other systems containing radioactivity and to require design features to minimize the potential for fuel management errors that could lead to fuel rod failure. As such, it is not unique to LWRs and is considered applicable to CRBR.

Criterion 54--Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Justification--This criterion is identical to GDC 62 in 10 CFR 50, Appendix A. The intent of this criterion is to require positive means of preventing criticality in the fuel storage, fuel handling and refueling process. As such, it is not unique to LWRs and is considered applicable to CRBR.

Criterion 55--Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactve waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Justification--This criterion is identical to GDC 63 in 10 CFR 50, Appendix A. The intent of this criterion is to require monitoring of fuel and waste storage systems to ensure adequate heat removal and excessive radiation levels that are not excessive. As such, it is not unique to LWRs and is considered applicable to CRBR.

Criterion 56--Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Justification--This criterion is identical to GDC 64 in 10 CFR 50, Appendix A, except that the words "spaces containing components for recirculation of loss-of-coolant accident fluid" have been removed because CRBR does not require such spaces and the word "including" has been added between "operation" and "anticipated" to be consistent with the definition of anticipated operational occurrences. The intent of this criterion is to require monitoring capability to detect and measure the radioactivity discharged to the plant environment and from the plant's effluent paths to the outside atmosphere. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 57--Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of events such as rod runout, steamline rupture, changes in reactor coolant temperature and pressure, and cold sodium addition.

Justification--This criterion is identical to GDC 28 in 10 CFR 50, Appendix A, except that the words "rod dropout" have been changed to "rod runout" and the words "cold water addition" have been changed to "cold sodium addition" to be consistent with CRBR terminology. Also, the word "pressure" has been dropped from the terms "pressure boundary" and "pressure vessel" and the words "rod ejection (unless prevented by positive means)" have been dropped from the last sentence since they do not apply for CRBR because of the low system operating pressure. The intent of this criterion is to require that the plant systems that can add reactivity to the core be designed to limit reactivity insertions to values that are consistent with the capability of the protection systems and will not result in loss of coolant boundary or affect the ability to cool the core. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 58--Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Justification--This criterion is identical to GDC 29 in 10 CFR 50, Appendix A. The intent of this criterion is to require highly reliable protection and reactivity control systems. As such, this criterion is not unique to LWRs and is considered applicable to CRBR.

Criterion 59--Fuel Rod Failure Propagation. Features shall be provided to limit propagation of stochastic fuel rod failures. These features may be inherent in the design of the fuel and blanket assemblies to eliminate or mitigate propagation or may include monitoring systems to detect pin failures in time to permit appropriate measures to be taken. The features provided shall be sufficient to limit propagation of each failure to the assembly in which it is located.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the design be capable of preventing fuel failure propagation, which could lead to a disruption of a significant fraction of the core. Because of the design differences between CRBR and LWR fuel and the limited experience with LMFBR fuel failures, in comparison with experience with LWR failures, especially the behavior of breached fuel pins that continue to be irradiated, it is considered appropriate to require features to limit fuel failure propagation.

Criterion 60--Flow Blockage

The reactor internals and core assemblies shall be designed to minimize the potential for flow blockage or flow restriction to one or more core assemblies by loose parts or by core assembly loading errors sufficient to cause fuel rod failure.

Justification--This criterion is unique to CRBR. The intent of this criterion is to require that the reactor and core assembly design incorporate features to minimize the potential for flow blockage while the assemblies are in the reactor core so that flow blockage can be eliminated as a design-basis event. Because the core assemblies in CRBR are ducted assemblies, blockages or restrictions at the inlet of an assembly affect flow through the entire assembly and could cause fuel failure such as the one that occurred in the Fermi-1 reactor. This potential can be greatly reduced if the core support structure and core assembly inlet regions are appropriately designed. To eliminate flow blockage events from the CRBR design basis, a criterion requiring that such design features be provided is considered necessary for CRBR.

3.1.2 10 CFR 50, Appendix A, Criteria Not Included

The following criteria from 10 CFR 50, Appendix A, were not applied to CRBR. Justification as to why they were not applied follows each criterion.

GDC 33--Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Justification for Exclusion--CRBR operating conditions below the sodium saturation temperature and design provisions to ensure adequate sodium inventories are required in Principal Design Criterion 27 to ensure that leaks do not uncover the core or stop sodium circulation and core cooling. In addition, to stop core cooling would require the loss of the entire HTS which is not considered a credible event and is not in the plant design basis. In view of the above consideration, there is no need for a separate reactor coolant makeup, system, and this criterion was not applied to CRBR.

GDC 35--Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Justification for Exclusion--In CRBR, emergency core cooling will be provided through the reactor residual heat removal system (see Criterion 35). Since under all postulated accident conditions the sodium temperature remains below the sodium saturation temperature and design criteria have been added to ensure an adequate coolant inventory and decay heat removal paths, there is no need for a separate emergency core cooling system and this criterion was not applied to CRBR.

GDC 36--Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Justification for Exclusion--Without an emergency core cooling system, this criterion is not applicable.

GDC 37--Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Justification--Without an emergency core cooling system, this criterion is not applicable.

GDC 38--Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Justification for Exclusion--No postulated design-basis events for CRBR will cause the containment temperature to exceed its design temperature or pressure. However, CRBR may have a containment heat removal system and a controlled venting system to provide margin for events that are beyond the design basis. In view of the fact that a containment heat removal system was not required for any design-basis events, this criterion was not applied to CRBR.

GDC 39--Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping, to assure the integrity and capability of the system.

Justification for Exclusion--Without a design requirement for a containment heat removal system, this criterion is not applicable.

GDC 40--Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Justification for Exclusion--Without a design requirement for a containment heat removal system, this criterion is not applicable.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

The staff reviewed structures, systems, and components important to safety according to the guidance and recommendations provided in SRP Section 3.2.1 (NUREG-0800).

3.2.1.1 Area of Review

CRBR Principal Design Criterion (PDC) 2, in part, requires that structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The earthquake for which these plant features will be designed is defined as the safe shutdown earthquake (SSE) in 10 CFR 100, Appendix A. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those plant features that will be designed to remain functional if an SSE occurs are designated seismic Category I in RG 1.29. In the PSAR Tables 3.2-1, "Seismic Category I Structures," and 3.2-2, "Preliminary List of Seismic Category I Mechanical System Components and Assigned Safety Classes," Section 1.2, "General Plant Description," and portions of other sections as required to supplement Section 1.2 were reviewed for compliance with these criteria.

The staff reviewed the seismic classification of those structures, systems, and components (including their foundations and supports) which are important to safety and will be designed to withstand, without loss of function, the effects of an SSE and are specified as seismic Category I by the applicants in their Safety Analysis Report (SAR). This review is performed for both construction permit (CP) and operating license (OL) applications. The staff's review of seismic Category I items includes the following plant features: structures, dams, ponds, cooling towers, reactor internals, fluid systems important to safety that are identified in RG 1.26 ventilation systems, standby diesel generator auxiliary systems, fuel-handling systems, and cranes.

The applicants' proposed seismic classification is in part presented in Table 3.2-1 (PSAR) which identifies those structures, systems, and components that are designated seismic Category I. The table identifies all activities affecting the safety-related functions of these seismic Category I plant features which also meet the pertinent quality assurance requirements of 10 CFR 50, Appendix B. Details of the seismic classification of these plant features are shown on plot plans, general arrangement drawings, and on piping and instrumentation diagrams.

Portions of structures and fluid systems that are seismic Category I are clearly identified. For fluid systems important to safety, the classification tables in the SAR identify system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, have suitable footnotes defining interfaces, and are in sufficient detail so that there is a clear understanding of the extent of those portions of the system that are classified as seismic Category I.

3.2.1.2 Acceptance Criteria and Basis

Acceptance criteria are based on meeting the relevant requirements of the following regulations.

(1) CRBR PDC 2, as it relates to the requirements that systems and components important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions.

(2) 10 CFR 100, Appendix A, as it relates to certain systems and components being designed to withstand the SSE and remain functional. These plant features are those necessary to ensure:

    (a) the integrity of the reactor coolant boundary

    (b) the capability to shut down the reactor and maintain it in a safe shutdown condition

(c) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines exposures of 10 CFR 100

(3) CRBR PDC 4 as it relates to the systems and components necessary for protection against sodium and NaK reactions. These systems and components are important to safety and therefore must meet the relevant requirements of Item 1 above.

(4) CRBR PDC 5 as it relates to protection against operational occurrences and postulated accidents for systems and components important to safety.

To meet these requirements, RG 1.29 was used. This regulatory guide describes an acceptable method of identifying the plant features which should be classified seismic Category I. Although it was written for a light-water reactor design, RG 1.29 is applicable directly to a sodium design because its requirements are specified in terms of the functions which must be performed to safely shut down the reactor and maintain it in a safe shutdown condition. Two changes were required: (1) since the primary coolant system operates at pressures and temperatures well below the sodium flash point, no emergency core cooling system is required, and (2) sodium is violently reactive with both water and air. Therefore, the functions of preventing and mitigating such reactions must be included with those of RG 1.29.

3.2.1.3 Review Evaluation

The review demonstrated that all systems and components necessary to meet the requirements of the amended RG 1.29 are identified or committed to be identified in Tables 3.2-1 and 3.2-2 of the PSAR.

3.2.1.4 Evaluation Summary

Systems and components (excluding electrical features) that are important to safety and that are required to withstand the effects of an SSE and remain functional have been classified as seismic Category I items and have been identified in an acceptable manner in Tables 3.2-1 and 3.2-2 (PSAR), and on system piping and instrumentation diagrams in the SAR. Other systems and components that may be required for operation of the facility (excluding electrical features) need not be designed to seismic Category I include those portions of Category I systems such as vent lines, drain lines, fill lines, and test lines on the downstream side of isolation valves and those portions of the system which are not required to perform a safety function.

The staff concludes that the systems and components important to safety within the scope of the staff's review have been properly classified as seismic Category I items and meet the requirements of CRBR PDC 2, "Design Bases for Protection Against Natural Phenomena," 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," and CRBR PDC 4, "Protection Against Sodium and NaK Reactions." The staff concludes that the applicants have met the requirements of CRBR PDC 2, 4, and 5 by properly classifying their systems and components important to safety as seismic Category I items in accordance with the positions of RG 1.29 as modified for a sodium facility. The staff further concludes that the identified systems and components are the plant features necessary to ensure (1) the integrity of the reactor coolant boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent and mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

3.2.2 System Quality Group Classification

The staff reviewed the components detailed below in Section 3.2.2.1 according to the guidance and recommendations provided in SRP Section 3.2.2 (NUREG-0800).

3.2.2.1 Area of Review

The review covered (1) the applicants' classification system for safety-related components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems, important to safety, and (2) the assignment by the applicants of quality groups to those portions of systems required to perform safety functions. In addition, the review covered the tables and suitable piping and instrumentation diagrams upon which the applicants presented the fluid systems important to safety, the associated quality group classifications, quality assurance requirements, and class designation in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1 (hereinafter referred to as "the Code"). PSAR Section 3.2.2 contains material pertinent to this review.

The review covered the construction of systems and components designated as Quality Groups A, B, C, in accordance with the appropriate regulatory guides, industry codes, and standards.

3.2.2.2 Acceptance Criteria and Basis

Acceptance criteria are based on meeting the relevant requirements of the following regulations.

(1) CRBR PDC 1 and 10 CFR 50.55a as they relate to the requirement that systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

(2) CRBR PDC 4 states that systems, components, and structures containing sodium shall be designed to limit the consequences of sodium or NaK chemical reactions resulting from a sodium or NaK spill.

(3) American National Standard (ANS) 54.6, "LMFBR Safety Classification and Related Requirements (Draft)," provides specific guidance on the safety classifications and the corresponding quality and seismic requirements of structures, systems, and components of commercial LMFBR nuclear power plants. The guidelines are analogous to those provided for light-water-cooled reactors in RGs 1.26 and 1.29, and ANSI Stds. N18.2 (PWR) and N212 (BWR). As a note of caution, at the time of this review ANS 54.6 is in a preliminary draft stage and has not been approved by ANSI. Since its release in October 1979 was for trial use and comment, this standard may be subject to revision or withdrawal before final issuance.

To meet the requirements of CRBR PDC 1 and 10 CFR 50.55a, RG 1.26 and ANS 54.6 (Draft) were used. These criteria provide an acceptable method for determining quality standards for Quality Groups A, B, C, and D sodium-, water-, and steam-containing components important to safety of CRBRP.

3.2.2.3 Review Evaluation

The applicants have provided details of the safety-related components and systems and corresponding quality standards in PSAR Section 3.2.2. The applicants have adhered to the requirements of 10 CFR 50.55a and RG 1.26 in defining the correspondence of quality groups and ASME Code classifications. In recognition of the differences in plant design of certain systems between LWRs and LMFBRs, the intent of RG 1.26 have been interpreted conservatively. In PSAR Sections 3.2.2.1 through 3.2.2.3 the applicants have provided details of the criteria used to classify systems and components important to safety. The designation of safety classes conforms to the requirements of RG 1.26 to the extent to which this document is applicable to a liquid metal cooled reactor. In addition, the CRBR safety class criteria were found to be in close agreement with the relevant sections of ANS 54.6 (Draft), "LMFBR Safety Classification and Related Requirements." Consequently, it was determined that the guidelines of LWR SRP Section 3.2.2 and its references, as modified by the relevant sections of CRBRP PDC 4 and ANS 54.6 (Draft), are applicable to the CRBRP.

The fluid system classification boundaries cannot be indicated until the piping and instrumentation drawings have been completed. However, the applicants have committed to using piping valves or vessel nozzles as the safety class boundaries. Table 3.2-2 (PSAR) identifies, in an acceptable manner, changes in quality group classification within piping systems.

3.2.2.4 Evaluation Summary

On the basis of its review of CRBR PSAR Section 3.2.2, the staff's evaluation supports the following conclusions.

Safety-related components of fluid systems important to safety such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves have been classified Quality Groups A, B, C, or D and have been identified in an acceptable manner in Table 3.2-2 (PSAR) and on system piping and instrumentation diagrams in the PSAR. These components have been constructed to quality standards commensurate with the importance of Quality Groups A and B (ASME Section III, Classes 1 and 2) reactor coolant boundary components is discussed in SRP Section 5.2.1. Other Quality Group B components of systems identified in Positions C.1.a through C.1.e of RG 1.26 are constructed to ASME Section III, Class 2. Components in systems identified in Positions C.2.a through C.2.d of RG 1.26 are constructed to Quality Group C standards, ASME Section III, Class 3. Components in systems identified in Position C.3 of RG 1.26 are constructed to Quality Group D standards such as ASME Section VIII and ANSI Std. B31.1.

The staff concludes that the safety-related components of fluid systems important to safety have been properly classified as Quality Groups A, B, C, or D items and meet the requirements of CRBR PDC 1, 4, and 5. The staff arrived at this conclusion because the applicants met the requirements of CRBR PDC 1, 4, and 5 by properly classifying these safety-related components important to safety as Quality Groups A, B, C, or D in accordance with the positions of RG 1.26 and ANS 54.6 (Draft) and by the staff's conclusion that the identified safety-related components are those necessary (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant boundary, (2) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) to contain radioactive materials.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

All seismic Category I structures exposed to wind forces will be designed to withstand the effects of the design wind. The design wind specified has a velocity of 90 mph based on a recurrence interval of 100 years.

The procedures that will be used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors will be in accordance with ANSI A58.1-1972. This document is acceptable to the staff.

The staff concludes that the plant design will be acceptable and will meet the requirements of CRBR PDC 2 and the recommendations of SRP Section 3.3.1. This conclusion is based on the following.

The applicants will meet the requirements of CRBR PDC 2 with respect to the capability of the structures to withstand design wind loading by:

(1) appropriate consideration for the most severe wind as discussed in Section 2.3.2 of this SER.

(2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena as indicated in Section 3.8 of this SER

(3) consideration of the importance of the safety function to be performed

The applicants will meet these requirements by using ANSI A58.1-1972, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and to select pressure coefficients corresponding to the structural geometry and physical configuration.

The applicants will design the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings determined to be appropriate for the site so that Item (1) listed above will be met. In addition, the design of seismic Category I structures, as required by Item (2) listed above, will consider in an acceptable manner load combinations involving the most severe wind load and other loads resulting from normal plant operation and/or accident conditions as delineated in Section 3.8 of this SER.

The procedures to be used for determining the loadings on structures induced by the design wind specified for the plant are acceptable, since these procedures have been used in the design of conventional structures and proven to provide a conservative basis, which, together with other engineering design consideration, ensures that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of design-basis winds, the structural integrity of the plant structures to be designed for the design wind will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item (3) listed above.

3.3.2 Tornado Loadings

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant will be designed to resist a tornado of 360-mph tangential wind velocity and 5- to 70-mph translational wind velocity. The simultaneous atmospheric pressure drop is assumed to be 3 psi in 2.0 sec. Tornado missiles are also considered in the design as discussed in Section 3.5 of this SER.

The procedures that will be used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings discussed in Section 3.3.1 of this report. The tornado missile effects will be determined using procedures discussed in Section 3.5 of this SER. The total effect of the design tornado on seismic Category I structures will be determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop, and tornado-associated missiles. Structures will be arranged on the plant site and protected in such a manner that collapse of structures not designed for the tornado will not affect safety-related structures.

The staff concludes that the plant design will be acceptable and will meet the requirements of CRBR PDC 2 and the recommendations of SRP Section 3.3.2. This conclusion is based on the following.

The applicants will meet the requirements of CRBR PDC 2 with respect to the structural capability to withstand design tornado wind loading and tornado missiles by:

(1) appropriate consideration of the most severe tornado as discussed in Section 2.3.2 of this SER

(2) appropriate combinations of the effects of this severe natural phenomenon with those resulting from normal plant operation and/or accident conditions

(3) consideration of the importance of the safety function to be performed

The applicants will meet these requirements by using ANSI A58.1 and American Society of Civil Engineers Paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity generated by the tornado into an effective pressure on structures and to select pressure coefficients corresponding to the structural geometry and physical configuration.

The applicants will design the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings determined to be appropriate for the site so that the requirements of Item (1) listed above will be met. In addition, the design of seismic Category I structures, as required by Item (2) listed above, will include, in an acceptable manner, load combinations involving the most severe tornado load and the loads resulting from normal plant operation and/or accident conditions. The procedures to determine the loadings on structures induced by the design-basis tornado specified for the plant are acceptable, since these procedures have been used in the design of conventional structures and proven to provide a conservative basis, which, together with other engineering design considerations, will ensure that the structures will withstand such severe environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design-basis tornado, the structural integrity of the plant structures that have to be designed for the tornadoes will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item (3) listed above.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

To ensure conformance with the requirements of GDC 2 with respect to protection against flooding, the staff reviewed the overall plant design and protection against the effects of flooding for those structures, systems, and components whose failure could prevent shutdown or result in an uncontrolled release of significant radioactivity.

The facility's flood protection design criteria and bases and preliminary design were reviewed in accordance with SRP Section 3.4.1. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedure" portion of the SRP section. The review procedures provide guidelines to verify that the flood protection design criteria and the bases and the facility's flood protection preliminary design meet LWR PDC 2, related to the system's capability to withstand the effects of flooding, is identical to GDC 2. Acceptance is therefore based on meeting the guidance of RG 1.102.

The applicants will provide protection against the inundation and the static and dynamic effects of postulated flooding of safety-related structures, systems, and components by the "dry site" method as defined in RG 1.102.

At Clinch River, the probable maximum flood (PMF) level is expected to be 777.50 ft MSL and the design-basis flood (DBF) including wind-wave runup is expected to be 809.2 ft MSL (see Section 2.4 of this SER for further discussion). All safety-related facilities, systems, and components housed in seismic Category I structures with exterior access or entrances will be either at or above the plant grade elevation of 8150.0 ft MSL. Water stops will be provided in all construction joints of foundations and walls below grade level, and waterproofing will be provided on the outside faces of all exterior walls below grade and on the underside of all foundation mats. There will be no exterior penetrations into the reactor containment structure below grade level; all penetrations into containment below the maximum flood or groundwater level will be routed through the interior of other seismic Category I structures which are also designed as watertight reinforced concrete structures. Exterior penetrations into other seismic Category I structures below grade level will be designed to be watertight. As additional precautionary measures, watertight doors and equipment hatches will be provided in select portions of the plant, and a flood warning system will be developed for the facility.

Within structures, the protection of safety-related systems and components against flooding caused by the failure of fluid systems is described under Section 9.15 of this SER.

On the basis of its review, the staff concludes that the design criteria and bases of the facility for flood protection (and preliminary design of the facility's flood protection) satisfy the requirements of PDC 2 with respect to protection against natural phenomena, conform to the guidelines of RG 1.102 and are, therefore, acceptable.

3.4.2 Water Level (Flood) Design Procedures

The design flood level resulting from the most unfavorable condition or combination of conditions that produces the maximum water level at the site is discussed in Section 2.4. The hydrostatic effect of the flood will be considered in the design of all seismic Category I structures exposed to the water head.

The procedures used to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable, since these procedures provide a conservative basis for engineering design to ensure that the structures will withstand such environmental forces.

The staff concludes that the plant design will be acceptable and will meet the requirements of CRBR PDC 2 and the recommendations of SRP Section 3.4.2. This conclusion is based on the following.

The applicants will meet the requirements of CRBR PDC 2 with respect to the structural capability to withstand the effects of the flood or highest groundwater level by:

(1) appropriate consideration of the most severe flood as discussed in Section 2.4.3 of this SER

(2) appropriate combinations of the effects of normal plant operation and/or accident conditions with the effects of the natural phenomena

(3) consideration of the importance of the safety function to be performed

The applicants will design the plant structures with sufficient margin to prevent structural damage during the most severe flood or groundwater levels and the associated dynamic effects determined to be appropriate for the site so that the requirements of Item (1) listed above will be met. In addition, the design of seismic Category I structures, as required by Item (2) listed above, will include, in an acceptable manner, load combinations that occur as a result of the most severe flood or groundwater-related load and the loads resulting from normal and accident conditions.

The procedures to be used for determining the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable, since these procedures have been used in the design of conventional structures and proven to provide a conservative basis, which, together with other engineering design considerations, will ensure that the structures will withstand such environmental forces.

The use of these procedures will provide reasonable assurance that, in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required, thus satisfying the requirement of Item (3) listed above.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

See Section 3.5.1.2.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Protection of structures, systems, and components important to safety against postulated internally generated missiles associated with plant operations, such as missiles generated by rotating or pressurized equipment, is required by GDC 4 (PDC 5).

The facility's internally generated missiles design criteria and bases and preliminary design of the facility's missile protection were reviewed in accordance with SRP Sections 3.5.1.1 and 3.5.1.2 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP sections was performed according to the guidelines provided in the "Review Procedure" portion of the SRP sections. The review procedures provide guidelines to verify that the design criteria and bases and preliminary design of the facility's missile protection meet applicable portions of the following LWR general design criterion of 10 CFR 50:

> GDC 4, with respect to protecting structures, systems and components important to safety from the effects of missiles, is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition requiring protection.

SRP Sections 3.5.1.1 and 3.5.1.2 are applicable to the review of the CRBR protection against the effects of internally generated missiles. Conformance with acceptance criteria of SRP Sections 3.5.1.1 and 3.5.1.2 provides the bases for concluding that the facility design criteria and bases and the preliminary design of the facility missile protection are acceptable.

The applicants have stated that protection against postulated missiles will be provided by any one or a combination of compartmentalization, physical barriers, separation, orientation, restraints, and equipment design in order to:

(1) maintain containment integrity

(2) bring the reactor to a safe shutdown under all plant conditions
(3) prevent sodium/water (steam) reaction

Redundant components of each safety-related system are arranged so that a potential missile could not damage or incapacitate both trains of the system. Stored strain energy, contained fluid energy, and rotational energy will be considered in the applicants' evaluation of potential missile sources.

Rotating components such as the pumps and drive mechanisms for the primary heat transport system (PHTS), intermediate heat transport system (IHTS), steam generator system (SGS), and steam generator auxiliary heat removal system (SGAHRS) have been, or will be evaluated as sources of potential missiles. The applicants have stated that a postulated impeller failure of a recirculation pump in the SGS would not result in a missile with sufficient energy to penetrate the pump casing. The auxiliary feedwater pumps associated with the SGAHRS are not considered as credible missile sources.

Pressurized component missiles result from the sudden release of stored strain energy (in the case of nuts, bolts, and studs) or of confined fluid energy (in the case of equipment or vessels that contain highly pressurized fluids). Components such as valve bonnets, relief valve parts, hardware retaining bolts or instrument wells will be evaluated to determine their potential for generating missiles.

On the basis of its review, the staff concludes that through the use of compartmentalization, barriers, separation, orientation, restraints and equipment design, the facility's design criteria and bases and the preliminary design of the facility to maintain the capability for a safe plant shutdown in the event of internally generated missiles are in conformance with the requirements of PDC 5 (GDC 4) with respect to missile protection, and are, therefore, acceptable.

3.5.1.3 Turbine Materials

During the past several years the results of turbine inspections at operating nuclear (LWR) facilities have shown that cracking to various degrees has occurred at the inner radius of turbine discs, particularly those of Westinghouse design. Within this period, a Westinghouse turbine disc failure has occurred at one facility--Yankee Atomic Electric Company. Furthermore, recent inspections of General Electric turbines also have resulted in the identification of disc keyway cracks. The staff has been following this development closely and has set turbine missile generation probability guidelines for establishing the frequency of turbine disc inspections and for maintenance and testing of turbine control and overspeed protection systems to preclude missile-producing failures. The domestic turbine manufacturers are in the process of establishing models and methods for calculating, turbine missile generation probabilities for their respective turbine systems.

Although large steam turbines and their auxiliaries are not safety systems as defined by NRC regulations, failures that occur in these turbines can produce large, high-energy missiles. If a missile were to strike safety-related structures, systems, or components, it could render them unavailable to perform their safety functions. Although this is unlikely because of intervening structures, some plant damage would be inevitable. CRBR PDC 5, "Environmental and Missile Design Bases," which is similar to GDC 4 requires, in part, that structures, systems, and components important to safety be appropriately protected against the effects of missiles that might result from equipment failures. RG 1.115 and SRP Sections 3.5.1.3, 10.2, and 10.2.3 contain present NRC guidelines for evaluating the turbine, analyzing the plant layout, and minimizing the risk to safety-related structures, systems, and components resulting from potential turbine missiles at LWR facilities. However, staff review procedures deviate somewhat from those described in these guidelines so as to reflect the staff's primary objective, which is to preclude missile-producing turbine failure.

In view of current experience and NRC safety objectives, the staff proposes to emphasize the turbine missile generation probability (i.e., turbine generator system (TGS) integrity) in its reviews of the turbine missile issue and eliminate the need for elaborate and ambiguous analyses of strike and damage probabilities given an assumed turbine failure rate. Although straightforward in principle, the latter calculations have to be based on detailed facility information and assumptions regarding missile shape and size, missile energies, barrier penetration potential, and ultimately the likelihood of striking and damaging a facility safety system. Generally, there are significant differences between licensees' or applicants' submittals and the final evaluation by the staff. Nevertheless, the staff concludes, on the basis of its review experience, that the probability of a turbine missile striking and damaging a safety system can reasonably be expected to fall in a relatively narrow range depending on turbine orientation. More refined analyses or additional calculations for other facilities are unlikely to change this conclusion. Therefore, expensive and time-consuming strike probability analyses on the part of applicants or licensees and/or the NRC staff are judged to be unwarranted.

The new review procedure requires that all nuclear steam turbine manufacturers develop volumetric (ultrasonic) examination techniques suitable for inservice inspection of turbine discs and shaft (without removing discs from the shaft), and prepare reports for NRC review that describe their methods for determining turbine missile generation probabilities. The design-speed missile generation probability (P_{11}) is to be related to disc design parameters, material properties, and the inservice volumetric (ultrasonic) disc inspection interval. The destructive overspeed missile generation probability (P_{12}) is to be related to the turbine governor and overspeed protection system's speed sensing and tripping characteristics, the design and arrangement of main steam control and stop valves and reheat steam intercept and stop valves, and the inservice testing and inspection intervals for system components and valves. Perhaps the most significant outcome of this requirement and subsequent analyses is the specification of the total turbine missile generation probability P_1 (i.e., $P_{11} + P_{12}$) as a time-dependent quantity relatively sensitive to turbine maintenance schedules. Furthermore, if, during the history of plant operation, disc cracks are discovered on inspection or valve failures occur on testing, the variation of P_1 with time (inspection or test interval length) changes, requiring appropriate alterations in licensee operating procedures in accordance with the new variation of P_1 with time.

With regard to Clinch River, the staff considers the proposed orientation of the CRBR turbine generator favorable, even though two diesel oil storage tanks will be within the low trajectory turbine missile strike zone (see RG 1.115), since these tanks will be 5 ft below grade and about 20 ft apart. The staff believes that as a result of this design choice, the applicants have reduced, by about an order of magnitude, the probability of unacceptable damage to safety-related systems (compared with a design with unfavorable turbine orientation). The staff does not conclude, however, that a favorable orientation alone is sufficient to ensure protection of safety-related systems. Rather, the staff concludes that such protection is ensured by the measures to be taken by the applicants to obtain a level of TGS reliability that will keep the turbine missile generation probability small. The PSAR contains no inservice inspection program (describing procedures and schedules) for the turbine rotors, and the turbine control and overspeed protection systems. Justification of a proposed turbine maintenance program is critical to determining the acceptability of the turbine missile risks for CRBR. The staff's position on inservice inspection of the turbine discs is addressed in Section 5.12.3.9 of this SER.

In summary, the staff concludes that appropriate inservice inspection programs demonstrating adequate turbine maintenance must be developed to ensure an acceptable turbine missile risk for CRBR. This subject should be addressed in the FSAR.

3.5.1.4 Missiles Generated by Natural Phenomena

GDC 2 (PDC 2) requires that structures, systems, and components essential to safety be designed to withstand the effects of natural phenomena; GDC 4 (PDC 5) requires that these same plant features be protected against missiles. The missiles of concern are those generated by natural phenomena such as tornadoes.

The facility's design criteria and bases and preliminary design for protection against tornado missiles were reviewed in accordance with SRP Section 3.5.1.4. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedure" portion of the SRP section. The review procedures provide guidelines to verify that the design criteria and bases and preliminary design of the facility's design for external missile protection meet applicable portions of the following LWR General Design Criteria of 10 CFR 50:

(1) GDC 2, related to the system being capable of withstanding the effects of tornadoes, is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.76.

(2) GDC 4, related to protecting structures, systems, and components important to safety from the effects of missiles, is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition requiring protection.

The applicants' analysis for tornado missiles will be based upon missile spectrum A of SRP Section 3.5.1.4, and will be applicable to all seismic Category I structures. The spectrum includes the weight, velocity, kinetic energy, impact area, and height for missiles in a tornado zone 1 site, as identified in RG 1.76. Discussion of the protection afforded safety-related equipment from the identified tornado missiles is provided in Section 3.5.2 of this SER. Discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER.

On the basis of its review, the staff concludes that the missile spectrum to be considered in the design of the facility was appropriately selected and conforms to the requirements of PDC 2 and 5 (GDC 2 and 4) with respect to protection against natural phenomena and missiles and the guidelines of RG 1.76 with respect to identification of missiles generated by natural phenomena, and is, therefore, acceptable.

3.5.2 Structures, Systems, and Components To Be Protected from Externally Generated Missiles

GDC 4 (PDC 5) states that all structures, systems, and components essential to the safety of the plant shall be protected from the effects of missiles. The applicants have identified the safety-related structures and systems requiring protection from externally generated missiles (see Section 3.5 of the PSAR). These safety-related structures are designed to withstand the effects of postulated tornado-generated missiles including vertical missiles, without damage to safety-related equipment. The tornado spectrum is discussed in Section 3.5.1.4 of this SER.

The applicants' design criteria and bases regarding structures, systems, and components requiring protection from external missiles were reviewed in accordance with SRP Section 3.5.2 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedure" portion of the SRP section. The review procedures provide guidelines to verify that the applicants' design criteria and bases meet applicable portions of the following LWR General Design Criteria of 10 CFR 50:

(1) GDC 2, related to the system being capable of withstanding the effects of tornadoes, is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RGs 1.27 and 1.117, and appropriate portions of RG 1.13.

(2) GDC 4, with respect to protecting structures, systems, and components important to safety from the effects of missiles, is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition requiring protection.

Except for the differences noted above, SRP Section 3.5.2 is applicable to the review of the applicants' design criteria for tornado missile protection. Conformance with the acceptance criteria of SRP Section 3.5.2, except as noted, provides the bases for concluding that the design criteria and bases are acceptable.

The applicants have identified the plant's outdoor features, including air intakes and exhaust, which may be required to perform a safety function coincident with or following the occurrence of a tornado. All safety-related systems and components and stored fuel are located within tornado-missile-protected structures or are provided with tornado-missile barriers. Components such as the diesel fuel oil storage tanks will be buried underground at a sufficient depth below the plant grade to preclude damage from tornado-generated missiles. The ultimate heat sink for the removal of heat for essential shutdown equipment is comprised of the emergency cooling towers and emergency cooling tower basin. The emergency cooling tower basin structure which serves as a reservoir and intake structure for the emergency plant service water system is designed to withstand the effects of tornadoes. Essential piping from the outdoor intake structure is protected from missiles throughout its length by missile protection slabs or seismic Category I pipe tunnels. The emergency cooling towers and emergency cooling tower basin are further discussed in Section 9.9.4 of this SER.

On the basis of its review, the staff concludes that the design criteria and bases for the facility regarding protection from the effects of natural phenomena such as tornadoes, conform to the requirements of PDC 2 and 5 (GDC 2 and 4) with respect to missile and environmental effects, and meet the guidelines of RGs 1.27 and 1.117, and the intent of RG 1.13 concerning the protection of safety-related plant features from tornado missiles, and are, therefore, acceptable.

3.5.3 Barrier Design Procedures

The plant's seismic Category I structures, systems, and components are to be shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures will include tornado-generated missiles, rotating component failure missiles, and site proximity missiles.

The procedures to be used in the design of the structures, shields, and barriers to resist the effect of the missiles were reviewed and found to be adequate. The analysis of structures, shields, and barriers to determine the effects of missile impact will be accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact will be investigated. This will be accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles will be prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is to be determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, will be combined with other applicable loads as discussed Section 3.8 of this SER.

The staff concludes that the barrier design will be acceptable and will meet the recommendations of SRP Section 3.5.3 and the requirements of CRBR PDC 2 and 3 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to equipment that must withstand the effects of natural phenomena (tornado missiles) and environmental effects including the effects, of missiles, pipe whipping, and discharging fluids. This conclusion is based on the following.

The procedures to be used for determining the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant are acceptable, since these procedures provide a conservative basis for engineering design to ensure that the structures or barriers will be adequately resistant to withstand the effects of such forces.

The use of these procedures will provide reasonable assurance that if design-basis missiles should strike seismic Category I structures or other missiles shields and barriers, the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures will, therefore, be adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of CRBR PDC 2 and 5.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

Piping breaks are analyzed in Section 3.9 of this SER.

3.7 Seismic Design

The conclusions in the following sections reflect the results of the staff's review and evaluation of the applicants' PSAR and also of the staff's structural audit at Burns and Roe, Inc., the architect-engineer for CRBRP. The staff's audit findings and concerns and their resolutions are contained in the Appendix 3.8-A of this SER.

3.7.1 Seismic Input

In the design of CRBR seismic Category I structures, systems, and components, an operating basis earthquake (OBE) of 0.125 g and a safe shutdown earthquake (SSE) of 0.25 g are specified. The input seismic design response spectra (OBE and SSE) are defined at foundation level of the nuclear power plant structures. These design response spectra are identical to the design response spectra defined in RG 1.60, Revision 1, scaled to 0.25 g maximum horizontal ground acceleration. Except for soil damping, values used in the design are in accordance with RG 1.61. For soil, damping values are determined on the basis of the soil shear strains as indicated in Table 3.7-2A of the PSAR.

The synthetic time history used for the seismic design of seismic Category I plant structures, systems, and components will be adjusted in amplitude and frequency content to obtain response spectra that envelop the response spectra specified for the site.

The seismic inputs to seismic Category I structures, systems, and components are adequately defined. The applicants will demonstrate that the requirements of RGs 1.60 and 1.61 are met. The discussion of the development of the free-field ground motion response spectrum and its adequacy is addressed in Section 2.5 of this SER.

The staff concludes that the seismic design parameters to be used in the design of the plant structure will be acceptable and will meet the recommendations of SRP Section 3.7.1 and the requirements of CRBR PDC 2 and Appendix A to 10 CFR 100. This conclusion is based on the following.

The applicants will meet the relevant requirements of CRBR PDC 2 and Appendix A to 10 CFR 100 by taking into consideration magnitudes of earthquakes appropriate for the site as discussed in Section 2.5.2 of this SER.

The seismic design response spectra (OBE and SSE) to be applied in the design of seismic Category I structures, systems, and components will comply with the requirements of RG 1.60. Except for soil, the specific percentage of critical damping values to be used in the seismic analysis of seismic Category I structures, systems, and components is in conformance with RG 1.61. The synthetic time history to be used for the seismic design of seismic Category I plant structures, systems, and components will be adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site. Conformance with the requirements of RGs 1.60 and 1.61 ensures that the seismic inputs to seismic Category I structures, systems, and components will be adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loading.

3.7.2 Seismic System Analysis

3.7.3 Seismic Subsystem Analysis

The scope of review of the seismic system and subsystems analysis for the plant includes the seismic analysis methods for all seismic Category I structures, systems, and components. It includes review of procedures for modeling, soil-structure interaction, development of floor response spectra, torsional effects, evaluation of Category I structure overturning, and determination of composite damping. The review includes the design criteria and procedures for evaluating the interaction of nonseismic Category I structures and piping with seismic Category I structures and the piping and the effects of parameter variations on floor response spectra. The review also includes criteria for seismic analysis procedures for seismic Category I buried piping outside the containment.

The system and subsystem analysis will be performed by the applicants on an elastic basis. Modal response spectrum multidegree of freedom and time history methods will form the bases of the analysis of all major seismic Category I structures, systems, and components. When the modal response spectrum method is used, governing response parameters will be combined by the square root of-the-sum-of-the-squares rule. However, modes with closely spaced frequencies will be combined in accordance with requirements of RG 1.92.

Torsional effects for nonsymmetrical structures are considered by computing the torsional as well as the mass moment of inertia and the eccentricity between centers of rotation and mass. Torsional effects in symmetrical structures are accounted for by assuming a 5% eccentricity for the diameter or horizontal dimension perpendicular to the earthquake direction. Damping will be accounted for by modal damping ratios specified as a percentage of critical in accordance with RG 1.61.

Floor spectra inputs used for design and test verifications of structures, systems, and components will be generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be used for all structures, systems, and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning will be considered.

For buried seismic Category I structures or those founded on soil, the applicants state that the finite element computer program FLUSH will be used to establish the response spectra that will be applied to the foundation of these structures. The free-field motion is to be applied at the top of an assumed rock outcrop at the finished grade elevation, and the response spectra calculated at the "free field" foundation level of these soil-supported or buried structures will envelope the design response spectra.

The other major seismic Category I structures are supported on slanted layered rock and are interconnected structures with a common foundation base mat (nuclear island). One acceptable procedure is to consider a rock-supported structure using a fixed-base model and apply the design input motions directly to the model at the foundation level. Instead, the applicants have chosen to create a rock-structure interaction model for this nuclear island. A static analysis with a finite element model has been used to derive the spring constants at the rock-structure interface and the half-space theory has been used to determine the equivalent damping values. These two approaches are inconsistent (for details, see Appendix 3.8-A of this SER). To resolve the staff's reservations on the adequacy of the assumptions on which the rock-structure interaction model is based, the applicants have made an analysis of the nuclear island by considering it to be fixed at the base. A comparison of the results of the two analyses indicates that the response spectra generated with the rock-structure interaction model generally envelope those generated with the fixed-base model. As a result of such finding and on the basis of the judgment that there should be little effect from rock-structure interaction, the staff accepts the applicants' adoption of the rock-structure interaction model.

The staff concludes that the plant design will be acceptable and will meet the recommendations of SRP Sections 3.7.2 and 3.7.8 and the requirements of CRBR PDC 2 and Appendix A to 10 CFR 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that their design reflects

(1) consideration of the magnitude of the most severe design earthquake appropriate for the site as discussed on Section 2.5.2 of this SER with an appropriate margin (CRBR PDC 2) (consideration of two levels of earthquakes, OBE and SSE (Appendix A, 10 CFR 100))

(2) appropriate combinations of the effects of normal and accident conditions with the effect of the natural phenomena

(3) appropriate consideration of the safety function to be performed (CRBR PDC 2) (the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except that an equivalent static loads method may be used if it can be demonstrated to be adequate (Appendix A, 10 CFR 100))

The applicants will meet the requirements of Item (1) listed above by using the acceptable seismic design parameters in accordance with SRP Section 3.7.1. The combination of seismic loads with those resulting from normal and accident conditions in the design of seismic Category I structures as specified in SRP Sections 3.8.1 through 3.8.5 is in conformance with Item (2) listed above.

With the use of the soil-structure interaction analysis verified to be adequate, the staff concludes that the use of the seismic analysis procedures and criteria delineated above by the applicants will provide an acceptable basis for the seismic design, which is in conformance with the requirements of Item (3) listed above.

3.7.4 Seismic Instrumentation Program

The type, number, location, and utilization of strong-motion accelerographs to record seismic events and to provide data on the frequency, amplitude, and phase relationship of the seismic response of the seismic Category I structures comply with RG 1.12. Supporting instrumentation will be installed on seismic Category I structures, systems, and components to provide data for the verification of the seismic responses determined analytically for such seismic Category I items.

The staff concludes that the seismic instrumentation system provided for the plant will be acceptable and will meet the recommendations of SRP Section 3.7.4 and the requirements of CRBR PDC 2; 10 CFR 100, Appendix A, and 10 CFR 50.55a. This conclusion is based on the following.

The applicants will meet the recommendations of SRP Section 3.7.4 and the requirements of 10 CFR 100, Appendix A, by providing the instrumentation that is capable of measuring the effects of an earthquake as required by CRBR PDC 2. The applicants will meet the requirements of 10 CFR 50.55a by providing the inservice inspection program that will verify operability by the performance of channel checks, calibrations, and functional test at acceptable intervals. In addition, the installation of the specified seismic instrumentation in the reactor containment structure and other seismic Category I structures, systems, and components will constitute an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. Notification of seismic activity to the control room operator is to be by both audio and visual means. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis in evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with RG 1.12.

3.8 Design of Seismic Category I Structures

The conclusions in the following sections reflect the result of the staff's review and evaluation of the applicants' PSAR and also of the staff's structural audit at Burns and Roe, Inc., the architect-engineer of CRBRP. The staff's audit findings and concerns and their resolution are contained in Appendix 3.8A of this SER.

3.8.1 Concrete Containment

CRBR will not have a concrete containment. The steel containment structure will be surrounded by a low leakage concrete confinement structure with an annulus space separating the two structures.

3.8.2 Steel Containment

The reactor and primary heat transport system (PHTS) will be completely enclosed in a steel containment composed of a vertical cylindrical shell with an ellipsoidal-spherical dome on the top and fixed on a concrete mat covered with a flat steel plate at the bottom. The cylinder part of the vessel will have an inside diameter of 186 ft and a height of approximately 169 ft from the top of the base mat steel liner. The dome will have a height of about 75 ft. The geometry and shell thicknesses at the cylinder dome intersection have been chosen so that the containment shell will experience no compressive stresses from internal pressures. This will preclude any potential buckling problem at the junction of the dome and the cylinder. Above the operating floor the vessel will be free standing and stiffened with circumferential beams. Below the operating floor the steel shell will be sandwiched between two concentric reinforced concrete walls. These concrete walls will prevent the lower portion of the steel vessel from buckling. The steel-lined concrete mat will be designed according to Division 2 of the ASME Code, Section III.

The steel containment will be designed, fabricated, constructed, and tested as a Class MC vessel in accordance with Subsection NE of the ASME Code, Section III, Division 1, 1974 Edition, and Subsection CC of the ASME Code, 1975 Edition. In the staff's opinion the 1980 Edition of the ASME Code should be used. The staff has also asked the applicants to evaluate the effect on the design of the containment if Code Case N-284 (1980) is used to evaluate the buckling capability.

In an attempt to resolve this issue, the applicants have compared the PSAR commitments to the 1980 Code and studied the significant code changes. On the basis of the results of the comparison made by the applicants, it has been determined that the containment designed on the basis of the 1974 Edition can meet the requirements of the 1980 Edition, and Code Case N-284.

The staff concludes that the design of the steel containment will be acceptable and will meet the recommendations of SRP Section 3.8.2 and the relevant requirements of 10 CFR 50.55a and CRBR PDC 1, 2, 3, 5, 14, and 41. This conclusion is based on the following.

(1) The applicants will meet the recommendations of SRP Section 3.8.2 and the requirements of 10 CFR 50.55a and CRBR PDC 1 with respect to ensuring that the steel containment will be designed, fabricated, erected, constructed, tested, and inspected to the quality standards indicated below.

(2) The applicants will meet the requirements of CRBR PDC 2 by designing the steel containment to withstand the most severe earthquake that will be established for the site, with sufficient margin and the combination of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicants will meet the requirements of CRBR PDC 5 by ensuring that the design of steel containment will be such that it will be capable of withstanding the dynamic and thermal effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicants will meet the requirements of CRBR PDC 14 by having the steel containment so designed that it essentially will be a leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

(5) The applicants will meet the requirements of CRBR PDC 41 by designing the steel containment to accommodate, with sufficient margin, the design leakage rate and the calculated pressure and temperature conditions resulting from accident conditions, and by ensuring that the design conditions will not be exceeded during the full course of the accident condition. In meeting these design requirements, the applicants will use the recommendations of regulatory guides and industry standards indicated below. The applicants also will perform appropriate analysis that will demonstrate that the ultimate capacity of the containment will not be exceeded and will establish a reasonable margin of safety for the design.

The criteria to be used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed on the structure during its service lifetime will be in conformance with established criteria, codes, standards, and guides designated in the RG 1.57 and the industry standard, ASME Code, Section III, Division 1, Subsection NE, and Division 2.

The use of these criteria as defined by applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements will provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function. A seismic Category I concrete shield building will protect the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building.

3.8.3 Concrete and Structural Steel Internal Structures

Except for the polar crane support system, major internal structures will be located below the operating floor level. The cylindrical concrete wall will carry horizontal shears to the foundation mat, and the inside layer of the sandwiched cylinder concrete wall will form the pressure boundary in local cell areas. Internal structures will consist primarily of cells constructed of reinforced concrete. Cells that will contain radioactive liquid-metal systems or piping and have the potential for a leak of Na or NaK are to be provided with steel cell liners as a leaktight barrier to maintain an inert atmosphere in the cells under normal operating conditions. They are also an engineered safety feature (ESF) in that they act as a barrier to prevent sodium-concrete reactions. As a leaktight membrane, the cell liners are not considered to contribute any structural strength to the overall strength of the cells. However, the cell liners themselves should be strong enough so that their leaktightness integrity can be maintained. Cell liners will consist of steel plate that will be supported by structural concrete with a grid of Nelson studs or, in the case of floor liners, supported by rolled steel sections embedded in the structural concrete. An air gap between the plate and structural concrete will allow escape of water vapor and noncondensible gases generated as concrete is heated during a spill. Major codes being used in the design of the cell-liner system are the ASME Code, 1974 Edition, and the American Institute of Steel Construction (AISC) specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition.

The cell-liner system is being designed to resist load combinations that include the appropriate normal loads, severe environmental loads, extreme environmental loads, and abnormal loads, including those generated during both a small liquid-metal spill (SLMS) and a postulated large liquid-metal spill (PLLMS). Design and analysis procedures used for the cell-liner system include plate and beam theory where linear analysis is appropriate and elastic-plastic finite-element analysis where stresses exceed yield. Analyses of the linear system, including areas of penetrations, consider appropriate regions of the system and consistent boundary conditions for predicting liner-buckling patterns and stresses and strains at high temperatures.

Design criteria for the liner and anchors include stress and strain limits and limits on mechanical loads and displacements. Except for the limits proposed for the PLLMS load condition, the allowable limits are in conformance with established criteria, ccdes, standards, and specifications acceptable to the staff. Limits proposed for the PLLMS are addressed further in Appendix 3.8A of this SER.

All interior structures will be designed as seismic Category I structures. The major code being used in the design of concrete internal structures is American Concrete Institute (ACI) Std. 349-76, "Code Requirements for Nuclear Safety-Related Concrete Structures." For steel internal structures, the AISC specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," is being used.

The containment internal structures are being designed and proportioned to resist load combinations that include normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. The materials of construction and their fabrication, construction, and installation will be in accordance with the ACI 349-76 and RG 1.142 for concrete structures and the AISC specification for steel structures.

The staff concludes that the design of the containment internal structures will be acceptable and will meet the recommendations of SRP Section 3.8.3 and the relevant requirements of 10 CFR 50.55a and CRBR PDC 1, 2, 5, 6, and 41. This conclusion is based on the following.

(1) The applicants will meet the requirements of 10 CFR 50.55a and CRBR PDC 1 with respect to ensuring that the containment internal structures will be designed, fabricated, erected, constructed, tested, and inspected to

quality standards commensurate with the safety functions to be performed by meeting the guidelines of the regulatory guides and industry standards indicated below.

(2) The applicants will meet the requirements of CRBR PDC 2 by designing the containment internal structures to withstand the most severe earthquake that will be established for the site, with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicants will meet the requirements of CRBR PDC 4 with respect to ensuring that systems, components, and structures containing sodium will be designed to limit the consequences of sodium chemical reactions resulting from a sodium spill.

(4) The applicants will meet the requirements of CRBR PDC 5 by ensuring that the design of the internal structures will be such that they will be capable of withstanding the dynamic and thermal effects associated with missiles, pipe whipping, and discharging fluids.

(5) The applicants will meet the requirements of CRBR PDC 6 by demonstrating that structures, systems, and components will not be shared between units or, if shared, by demonstrating that sharing will not impair their ability to perform their intended safety functions.

(6) The applicants will meet the requirements of CRBR PDC 41 by designing the containment internal structures to accommodate, with sufficient margin, the calculated pressure and temperature conditions resulting from accident conditions and by ensuring that the design conditions will not be exceeded during the full course of the accident condition. In meeting these design requirements, the applicants have used the recommendations of regulatory guides and industry standards indicated below.

The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed during the service lifetime are in conformance with established criteria and with the codes, standards, and specifications acceptable to the staff. They include the positions of RGs 1.10, 1.15, 1.55 (these guides have been combined into RG 1.136), and 1.142 and industry standards ACI Std. 349 and ACI Std. 531; ASME Code, Section III, Subsections NE and NF; and the AISC specifications for the design, fabrication, and erection of structural steel for buildings.

In addition, ANSI Std. N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," as endorsed by RG 1.94, is being used.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combination; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements will provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

The concrete and steel seismic Category I structures will be designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, OBE, and SSE; loads generated by postulated ruptures of high-energy pipes (such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes). The functional requirements of accommodation systems for beyond-design-basis accidents are considered in Appendix A.3.10 of this SER.

The design and analysis procedures that will be used for these seismic Category I structures are comparable to those approved on previously licensed LWR applications and, in general, are in accordance with procedures delineated in ACI 349-76 for concrete structures and the AISC specification for steel structures.

The various seismic Category I structures will be designed and proportioned to remain within the limits established by the staff under the various load combinations. These limits for concrete and steel structures are, in general, based on ACI 349-76 and the AISC specification, respectively, and on Appendix A to SRP Section 3.8.4 for masonry wall. They will be modified as appropriate for load combinations that are considered extreme.

The materials of construction and their fabrication, construction, and installation are in accordance with ACI 349-76 for concrete structures and the AISC specification for steel structures.

3.8.4 Other Seismic Category I Structures

Seismic Category I structures other than the steel containment will include the reactor service area of the reactor service building, control building, steam generator building, diesel generator building, emergency cooling tower structure, diesel fuel storage tank foundation, electric manholes, and confinement building. All except the emergency cooling tower structure, diesel fuel storage tank foundation, and electric manholes will have foundations that will be part of a common nuclear island mat.

The confinement building will be a reinforced concrete cylindrical enclosure with a spherical dome. It will be located external to and concentric with the steel containment. Some interior walls of the control building and steam generator building will be made of reinforced concrete block. Radiation shield walls in the reactor service building will be constructed of reinforced concrete. All other portions of the structures will be structural steel or reinforced concrete. The structural components will consist of slabs, walls, beams, and columns. The major code that will be used in the design of concrete seismic Category I structures is ACI 349-76, "Code Requirements for Nuclear Safety-Related Concrete Structures." For steel seismic Category I structures, the AISC specification will be used. The design of the reinforced masonry walls is in accordance with the design criteria as delineated in Appendix 3.8-D of the PSAR. These criteria are in general in conformance with Appendix A to SRP Section 3.8.4.

In the reactor service building and the steam generator building, there will be air-filled, unlined cells containing nonradioactive liquid-metal systems and piping. To prevent chemical reactions between Na or NaK and concrete, catch pan systems will be provided in these cells. The system will include catch pans, fire suppression decks covering the catch pan open area, insulation between the catch pan and structural components, and interconnections between adjacent catch pan cells. The catch pan system with its support framing will be designed to contain a large Na or NaK spill and maintain structural integrity at accident temperatures and pressures resulting from a sodium fire. System structures and supports are being designed as seismic Category I components. Major codes being used in the design of these structures are the ASME Code, 1977 Edition, and the AISC specification, "Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition.

The structures are being designed to resist load combinations that include the appropriate normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. Design and analysis procedures being used for the catch pan system include the plate or beam theory, finite-element analysis, and/or scale model tests. Elastic-plastic analysis will be used where yielding is allowed under accident conditions.

Strain allowable limits used in the design of steel plate have been reviewed and found to be similar to those proposed by the applicants for cell liners. The evaluation of those criteria in Sections 3.8.2 and 3.8.3 also apply here. Design allowable limits for attachments will be in conformance with those given in the AISC specification for resisting mechanical loads in construction, test, and normal categories.

The criteria that are being used in the analysis, design, and construction of all of the plant's seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime will be in conformance with established criteria, codes, standards, and specifications acceptable to the staff.

The staff concludes that the design of safety-related structures other than containment, which is evaluated in Section 3.8.2 of this SER, will be acceptable and will meet the recommendations of SRP Section 3.8.4 and the relevant requirements of 10 CFR 50.55a and CRBR PDC 1, 2, 5, and 6. This conclusion is based on the following.

(1) The applicants will meet the requirements of 10 CFR 50.55a and CRBR PDC 1 with respect to ensuring that the safety-related structures other than containment will be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) The applicants will meet the requirements of CRBR PDC 2 by designing the safety-related structures other than containment to withstand the most severe earthquake that will be established for the site with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicants will meet the requirements of CRBR PDC 4 with respect to ensuring that systems, components, and structures containing sodium will be designed to limit the consequences of sodium chemical reactions resulting from a sodium spill.

(4) The applicants will meet the requirements of CRBR PDC 5 by ensuring that the safety-related structures will be capable of withstanding the dynamic and thermal effects associated with missiles, pipe whipping, and discharging fluids.

(5) The applicants will meet the requirements of CRBR PDC 6 by demonstrating that structures, systems, and components will not be shared between units or, if shared, by demonstrating that sharing will not impair their ability to perform their intended safety functions.

The criteria to be used in the analysis, design, and construction of all of the plant's seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime will be in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include the positions of RGs 1.10, 1.15, and 1.142 and industry standards ACI-349, ACI-531, and the AISC specification, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and inservice surveillance requirements will provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, as well as various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairing the structural integrity or the performance of required safety functions.

3.8.5 Foundations

All seismic Category I structures except for the emergency cooling tower structures, diesel fuel storage tank foundation, and electric manholes will have a common nuclear island mat constructed of reinforced concrete. Other seismic Category I structures will be built on a concrete foundation, competent rock, or compacted structural backfill. Appropriate codes have been specified--the portion of the mat under the reactor containment building will meet the requirements of Subsection CC of the ASME Code, Section III, Division 2, and the remainder will meet ACI 349-76. These concrete foundations will be designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, OBE, and SSE; and loads generated by postulated ruptures of high-energy pipes.

The design and analysis procedures that will be used for these seismic Category I foundations are the same as those approved on previously licensed LWR applications and, in general, are in accordance with procedures delineated in ACI 349-76 and ASME Code, Section III, Division 2, for the containment portion of the common mat. The various seismic Category I foundations are being designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 349-76, modified as appropriate for load combinations that are considered extreme. The materials of construction and their fabrication, construction, and installation will be in accordance with ACI 349-76 and, where applicable, with ASME Code, Section III, Division 2.

The criteria that will be used in the analysis, design, and construction of all of the plant's seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed on each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff.

The staff concludes that the design of the seismic Category I foundations will be acceptable and will meet the recommendations of SRP Section 3.8.5 and the relevant requirements of 10 CFR 50.55a and CRBR PDC 1, 2, 5, and 6. This conclusion is based on the following.

(1) The applicants will meet the requirements of 10 CFR 50.55a and CRBR PDC with respect to ensuring that the seismic Category I foundations will be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) The applicants will meet the requirements of CRBR PDC 2 by designing the seismic Category I foundations to withstand the most severe earthquake that will be established for the site, with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicants will meet the requirements of CRBR PDC 5 by ensuring that the seismic Category I foundations will be capable of withstanding the dynamic and thermal effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicants will meet the requirements of CRBR PDC 6 by demonstrating that structures, systems, and components will not be shared between units or, if shared, by demonstrating that sharing will not impair their ability to perform their intended safety functions.

The criteria to be used in the analysis, design, and construction of all of the plant's seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed on each foundation during its service lifetime will be in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include the positions of RG 1.142 and industry standards, ASME Code, Section III, Division 2, ACI 349-76, and the AISC specification, "Specification for Design, Fabrication, and Erection of Structural Steel for Building.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements will provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability for the performance of required safety functions.

APPENDIX 3.8A: SUPPLEMENT TO SECTION 3.8 OF THE SAFETY EVALUATION REPORT--STRUCTURAL AUDIT REPORT

The purpose of this appendix is to document resolution of the findings and concerns identified during a structural audit performed at Burns and Roe, the CRBRP architect-engineer, in Oradell, New Jersey, on June 22-25, 1982. The staff was accompanied by its consultants from Los Alamos National Laboratory. The design of cell liners, the steel containment, and the general seismic design were audited for design criteria and the execution of the design to incorporate these criteria.

Table 3.8A.1 lists the findings identified during the structural audit. All items except for first two dealing with cell liners have been resolved through appropriate actions by the applicants. The applicants are developing new finite element models that should be appropriate for analysis of the cell liners. The staff expects that the models will be adequate for resolving these two items.

In addition to the audit findings, three ongoing concerns were reported. These are discussed in the sections that follow.

3.8A.1 Cell Liner Failure Criteria

The first concern is that the cell liner failure criteria are not appropriate. Failure criteria for the liners are not specifically addressed by current codes or the NRC Standard Review Plan (SRP) for LWRs (NUREG-0800). The most appropriate criteria given in current codes are those given in Division 2 of Section III of the ASME Boiler and Pressure Vessel Code for steel liners in concrete containments. The cell liners are similar in design to the liners in concrete containments and will experience similar loads except that for CRBR the cell liners could be exposed to much higher temperatures. This is especially true for faulted conditions where hot liquid sodium spills onto the liner surface. For small spills, strains are not expected to exceed those limits prescribed by the ASME Code for containment liners under faulted conditions. However, for large spills (Load Combination (D) in the PSAR), the liner temperatures will cause strains that exceed those limits. This means that additional criteria not prescribed in current codes are needed for load combinations involving large spills. These are discussed further below. The applicants have used stress and strain limits consistent with the ASME Code for all other load combinations.

Stress limits are provided for primary membrane and membrane plus bending stresses caused by mechanical loads. These limits are in accordance with Appendix F of Section III of the ASME Code. Appendix F specifically provides rules for evaluation of service loadings with Level D service limits.

The applicants have extended criteria given in Appendix F to apply to all load levels. This is not normally acceptable, but, because the liners are not counted on for carrying mechanical loads, the staff feels that the criteria are acceptable in this case.

The acceptance criteria needed for evaluating the cell liner system's ability to accommodate load combinations involving large spills (Level D, Faulted Conditions) are needed to ensure that the liner does not fracture, allowing sodium to reach the concrete behind the liner. Potential criteria could be any of the following.

(1) The criteria specified in Paragraph CC-3720 of Division 2 of Section III of the ASME Boiler and Pressure Vessel Code.

(2) Criteria given in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code.

(3) Once the local strain criterion is accepted for the primary heat transport system (PHTS) boundary for structural margins beyond the design basis (SMBDB) scenario it should be acceptable for cell liners.

(4) The criteria specified by the applicants will be acceptable if an appropriate test and analysis program is performed to show that it is conservative.

The first choice given above is not acceptable because those criteria given in Division 2 of the ASME Code, apply to liners that are not expected to get nearly as hot as the CRBR cell liners. It would be impossible for the cell liners to meet these criteria. Appendix F to Section III of the ASME Code applies to Level D loads, but it is meant to limit primary stresses that could lead to plastic instability. Because, for the cell liners, the staff is concerned primarily with secondary loads leading to local ductile fracture, the Appendix F criteria are not appropriate.

The criterion proposed by the applicants to protect the PHTS boundary from undergoing ductile fracture during an energetic core disruptive accident (CDA) may also be appropriate for the cell liners. The applicants have shown that the criterion is adequate for certain stress fields, but need to do further work to show it to be appropriate for stress fields with high shear (see Appendix A discussing SMBDB). The NRC staff has recommended that the applicants evaluate this criterion for cell liners. If it were to be applied to cell liners for Level D conditions, the factor of safety should be higher than that for beyond-design-basis-accident conditions.

The fourth choice for a failure criterion is currently being used by the applicants, and involves limiting the effective strain (Vom Mises strain) to 0.50 of the ultimate strain (strain when necking begins in a uniaxial tensile specimen) for membrane strains and 0.67 of the ultimate strain when both bending and membrane strains are present. The staff's current position (March 1983) is that this criterion will be acceptable, if a thorough analysis and test program proves it to be consistently conservative. The juncture of the cell liner with the anchor studs is the critical region and, as such, must be carefully analyzed and tested with the most severe realistic conditions that would be expected for a large sodium spill. The large spill test, LT-1, performed at Hanford Engineering Development Laboratory (HEDL) did not submit the liner to expected stress levels because of early failure of the anchor stud/liner interface. Results of this test cannot, therefore, be used as proof that the liner design is sufficiently conservative.

The applicants have submitted a plan to the NRC staff where a thorough analysis and test program of the cell liner system will be performed. Results of analysis will provide input to the test program. The tests and analyses will either substantiate the current criterion or lead to a more appropriate one.

As of March 1, 1983, the applicants have finished analyses of the liner in several critical areas including highly stressed regions near rectangular embedded plates and reinforced circular penetrations. These analyses are being used to determine the most highly stressed region in the cell liner system. When this region, which is expected to be at a stud/liner interface near a penetration, is determined, a detailed three-dimensional analysis will be performed. This analysis will have to accurately predict the three-dimensional stress field at the particular stud/liner interface where failure is determined to be most probable. Where the liner buckles inward and bears against the concrete, the analytical model will have to include appropriate boundary conditions that allow the liner to slide relative to the concrete.

After completion of the analytical work, the NRC will require a test of the cell liner system that will provide realistic loading to a physical model representative of the region where failure is expected to occur first. The model should be loaded to the Level D loads and then beyond to determine its margin to failure.

Results of analyses to date do not give a clear indication whether either the current failure criteria or the liner design itself will be appropriate for Level D loads. However, with the analytical/test research and development program that the applicant has presented, the staff believes that any weaknesses will be identified. If the program shows that the criteria are not consistently conservative, they can be easily changed to make them appropriate. If the program shows that the design is not appropriate, several viable fallback options exist for changing the design. These include such alternatives as:

(1) making the liner studs more compliant
(2) designing studs that would fail before the liner fractured
(3) reinforcing the area at the stud-liner interface.

On the basis of the proposed research and development program and the available fallback positions, the staff believes there is some certainty that the cell liners can be designed to provide a sufficiently safe barrier between sodium and concrete.

Similar to the cell liner itself, the cell liner stud anchor failure criteria are generally in line with Division 2 of Section III of the ASME Code, except for strain limits for the anchors subjected to Level D, large spill conditions. For this case, the same criteria are used as for the liner itself. The prescribed strain limits are acceptable here, because the consequences of the stud failure are not nearly so great as failure of the liner itself. In fact, stud failure would generally relieve stresses in the liner and make its failure less likely.

3.8A.2 Appropriate Codes

Early in the licensing process the NRC staff decided that, where possible, the CRBRP review should follow the current NRC Standard Review Plan (SRP) for LWRs (NUREG-0800). The SRP requires that the 1980 Edition on the ASME Boiler and Pressure Vessel Code be applied to various aspects of the structural design. In particular, the containment and its penetrations should be designed in accordance with Section III of the current (1980) Code. The applicants used the 1974 version of the ASME Boiler and Pressure Vessel Code because the bulk of plant design was started before the 1980 version of the Code was published. The staff's position is that, in accordance with the SRP, the 1980 Code should be used, unless the 1974 version can be shown to give an equivalently safe design.

To resolve this concern, the applicants performed a thorough review involving a qualitative comparison of the two other versions ASME Code (Question 220.25). As a part of this review, the buckling criteria used for containment design that were based on the 1974 Code and additional criteria given in the PSAR were compared with the most recent Code Case N-284. Based on this review, where there was a question whether the 1974 version of the Code was less conservative than the 1980 version, quantitative comparisons were performed. This involved comparison of design requirements for several of the most highly stressed (as determined from the containment design report) containment penetrations and determination of whether margins of safety for containment buckling met Code Case N-284 requirements.

Evaluation of three different penetrations in different areas of the containment showed that the design based on the 1974 version of the Code, along with standard design practices used by the vendor, is conservative. Results of the buckling analysis show that the design margins of the containment vessel, as designed based on criteria given in the PSAR, exceed the requirement of Code Case N-284.

3.8A.3 Seismic Model

The applicants' seismic model of the nuclear island presented in the PSAR is a lumped-mass model with springs to simulate compliance of the slanted rock layers underlying the CRBRP site. A detailed finite-element model that included the rock layers with actual interface angles was used to define the springs. Only one spring in each direction was used to connect the single mass lump representing the plant foundation mat to ground. Half-space theory was used to determine damping constants for the model.

The staff considers the above approach to be nonstandard, especially for a plant constructed on rock. The use of half-space theory to determine damping constants along with using the finite element method to determine stiffness is inconsistent. However, the applicants showed that stiffness constants derived from half-space theory were very close to those derived with the finite element model, which are assumed to be more accurate. Because the analysis method is nonstandard, the applicants performed additional analyses to substantiate the model presented in the PSAR.

To establish the fact that the rock strata underlying the plant are compliant enough to be considered in the analysis, the applicants modeled the plant and underlying rock with the FLUSH computer code. Results of this analysis showed that there is some soil (rock) structure interaction and that it was treated conservatively in the original lumped mass model. Results of this analysis were also used to show that the foundation mat is sufficiently stiff to include as one lump in the lumped mass model.

Further verification of the original model was provided by eliminating the soil springs and performing a fixed-base analysis. As expected, results showed that, even though present, there is very little soil structure interaction. Response spectra generated with the analysis were enveloped by the design response spectra generated with the original analyses except at frequencies representing fundamental frequencies of the plant in the soil springs. At these frequencies the design spectra were exceeded only slightly.

Based on the above confirmatory analyses, the staff believes that conservative results can be obtained with the original lumped mass model.

Table 3.8A.1 Audit findings and resolution

Findings	Resolutions
<u>Cell liner design</u>	
Improper boundary conditions on finite element models	Regions that were improperly modeled are being remodeled with appropriate boundary conditions.
Finite element meshes too coarse at discontinui ties to accurately predict high stress gradients	Discontinuities, in particular anchor stud/cell liner interfaces, are being remodeled with locally refined meshes
Triplanar corners not analyzed for shallow pool spills where the floor would be at much higher temperature than walls	Applicants have shown that the maximum thermal gradient at this location for a shallow pool is not severe enough to cause failure of the liner
<u>Containment design</u>	
Ultimate capacity prediction for containment not complete	Further analyses were performed to complete the ultimate capacity prediction. For DBA conditions, the capacity is governed by buckling of the equipment hatch
Shear capacity concrete wall sandwiching the steel containment below the operating floor not evaluated	Design was evaluated by applicants and found to be adequate
<u>Seismic analysis</u>	
Evaluation of equipment and personnel airlock for seismic loads oversimplified in that skew penetra tion was treated as being radial	This structure was reanalyzed and found to be accept- able as designed. Stresses governing design did not increase
Fundamental dome breathing mode simulated by lumped mass model needs to be verified	A more refined analysis was performed with a different computer code. The model was found to provide a suffi- cient simulation of the dome breathing mode
Torsional moments in the steam generator building and reactor service building seem large compared with other internal loads	Analytical models were reviewed and found to be cor- rect. Analyses were rerun for North-South direction without torsional mass. Much lower torsional loads showed that relatively large torsional masses are the primary cause of large loads. Torsional moments are

Table 3.8A.1 (continued)

Findings	Resolutions
Seismic analysis (cont.)	
	expected to be higher than those in LWRS because of differences in plant design
Interface between seismic analysis and structural design groups needs to be reviewed	Applicant verified interface between two groups is controlled by an adequate QA program and other standard project procedures
In some cases the applicants apparently did not correctly use the SRSS rule	Complete structure design was rexamined by the applicants to ensure that the SRSS rule was consistently applied
SRSS rule improperly used for analysis of confinement building	Analysis was redone with appropriate use of rule. Results did not indicate any need for change in design
General	
Design Book C-27.RC-6.105 has an incorrect formula for shear capacity of concrete in compression	This formula was apparently miscopied and was not actually used in any calculations. The applicants reviewed all calculations using this formula and found them correct
Design procedure for structures or structural elements in contact with compressible materials needs to be addressed in PSAR	General design procedures were provided and found to be acceptable
All significant computer codes need to be addressed in PSAR	Appendix A of PSAR has been updated to provide required computer code documentation

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The staff reviewed special topics for mechanical components as detailed in Section 3.9.1.1 according to the guidance and recommendations provided in SRP Section 3.9.1 (NUREG-0800).

3.9.1.1 Area of Review

The review covered information in the SAR concerning methods of analysis for seismic Category I components and supports, including both those designated as ASME Code Section III, Classes 1, 2, 3, or CS (core support) and those not covered by the Code. Certain aspects of dynamic system analysis methods are discussed in SRP Section 3.9.2 as well as in SRP Section 3.9.1. Information was also reviewed concerning design transients for Code Class 1 and CS components and supports. PSAR Sections 3.9, 4.2.2, Appendix B, and portions of Sections 5.2 through 5.6 contain material pertinent to this review. The following specific subjects were reviewed under SRP Section 3.9.1.

(1) transients which are used in the design and fatigue analyses of all Code Class 1 and CS components, and supports and reactor internals

(2) the method used to determine the seismic cyclic loading used for fatigue analysis of appropriate components and supports

(3) description and verification of all computer programs which will be used in analyses of seismic Category I Code and non-Code items listed in this SRP section

(4) description of any experimental stress analysis programs which will be used in lieu of theoretical stress analyses

(5) description of the analysis methods which will be used if the applicants elect to use elastic-plastic stress analysis methods in the design of any of the above-noted components

3.9.1.2 Acceptance Criteria and Basis

The acceptance criteria are based on meeting the relevant requirements of CRBR PDC 1, 2, 5, 12, and 13; 10 CFR 50, Appendix B; 10 CFR 50.55a; and 10 CFR 100, Appendix A. The relevant requirements of these regulations are indicated below.

(1) CRBR PDC 1 as it relates to components important to safety being designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety function to be performed

(2) CRBR PDC 2 as it relates to safety-related mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety function

(3) CRBR PDC 5 as it relates to protection against operational occurrences and postulated accidents for systems and components important to safety

(4) CRBR PDC 12 as it relates to the reactor coolant boundary being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

(5) CRBR PDC 13 as it relates to the mechanical components of the reactor coolant system being designed with sufficient margin to ensure that the design conditions of the reactor coolant boundary are not exceeded during any condition or normal operation, including anticipated operational occurrences

(6) 10 CFR 50, Appendix B, as it relates to design quality control

(7) 10 CFR 100, Appendix A, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics

Specific criteria necessary to meet the relevant requirements of the regulations listed above are as follows.

(1) To meet the requirements of CRBR PDC 1, 2, 5, 12, and 13, and 10 CFR 100, Appendix A, the applicants shall provide a complete list of transients to be used in the design and fatigue analysis of all ASME Code Class 1 and core support components, supports and reactor internals within the reactor coolant boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination shall be included. All transients such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control systems or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix A of 10 CFR 100, and design-basis events, that are contained in the Code-required "Design Specifications" for the components of the reactor coolant boundary shall be specified, including reactor internals and core support structures.

    The section of the applicants' PSAR which pertains to transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions resulting from those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in RG 1.68 and SRP Section 3.9.3. Transients and resulting loads and load combinations with appropriate specified design and service limits must provide a complete basis for design of the reactor coolant boundary for all conditions and events expected over the service lifetime of the plant.

(2) To meet the requirements of 10 CFR 50, Appendix B, and CRBR PDC 1, 2, 12, and 13, a list of computer programs that will be used (preferably programs which are recognized and widely known) in dynamic and static analyses to

determine the structural and functional integrity of seismic Category I Code and non-Code items, and the analyses to determine stresses shall be provided. For each program the following information shall be provided to demonstrate its applicability and validity:

(a) the author, source, dated version, and facility

(b) a description, and the extent and limitation of its application

(c) the computer program solutions to a series of test problems which shall be demonstrated to be substantially similar to solutions obtained from any one of sources i through iv, and source v:

(i) hand calculations

(ii) analytical results published in the literature

(iii) acceptable experimental tests

(iv) by a similar program acceptable to the staff

(v) the benchmark problems prescribed in NUREG/CR-1677

A summary comparison of the solution obtained by using sources i through iv shall be provided, in either graphical or numerical form. For source v, the complete computer printout of the input and the solution shall be submitted for every benchmark problem. These solutions may be referenced, and need not be resubmitted, in subsequent license application provided the information submitted under 2a and 2b remains unchanged.

Satisfactory agreement of computer and test solutions, usually within a ±5% error band, provides verification of the quality and adequacy of the computer programs to perform the functions for which they were designed.

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicants with a request that, unless conformance with the staff's acceptance criteria is agreed upon, additional technical justification be submitted.

(3) To meet the requirements of CRBR PDC 1, 2, 12, and 13 when experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I Code or non-Code items, the section of the SAR discussing the experimental stress analysis methods will be acceptable if the information provided meets the provisions of Appendix II of the ASME Code, Section III, Division 1, and as in the case of analytical methods, if the information provided is sufficiently detailed to show the validity of the design to meet the provisions of the Code-required "Design Specifications."

(4) To meet the requirements of CRBR PDC 1, 2, 12, and 13 when Service Level D limits are specified by the applicant for Code Class 1 and core support

components, and for supports, reactor internals, and other non-Code items, the methods of analysis used to calculate the stresses and deformations shall conform to the methods outlined in Appendix F of the ASME Code, Section III, Division 1, subject to the conditions discussed below.

If the applicants employ an elastic or an elastic-plastic method of analysis to evaluate the design of safety-related Code or non-Code items for which Service Level D limits have been specified (NB-3225 and Appendix F of the ASME Code, Section III, Division 1), the review covers the following points:

(a) The applicants must demonstrate that the stress-strain relationship for component materials that will be used in the analysis is valid. The ultimate strength values at service temperature must be justified.

(b) The analytical procedures to be used in the analysis are reviewed to determine the validity of the analysis. If a computer program is used, the applicable requirements of specific criterion 2 above shall be met.

(c) If elastic system analysis is used, its application may require detailed review and justification if applied to the analysis of systems which contain active components with close tolerances, or systems in which the sequence of load application could significantly affect the actual stress distribution.

(d) If elastic, elastic-plastic, or limit analysis methods are used for components in conjunction with elastic or elastic-plastic system analyses, the basis upon which these procedures are used is reviewed. The applicants shall provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the methods used for the system analysis are based.

The applicability of high temperature ($T > 800°F$ for austenitic materials, $T > 700°F$ for ferritic materials) acceptance criteria is presented in Section 3.9.9.

3.9.1.3 Review Evaluation

The guidelines of SRP Section 3.9.1 ensure that the applicants have provided sufficient information in the SAR to demonstrate that the design transients and resulting loads, corresponding design and service criteria, and computer programs used for analyses are acceptable. These guidelines address major concerns, which are applicable to all reactor plants. Consequently, it was determined that the LWR SRP Section 3.9.1 and its references are almost entirely applicable to the CRBR.

The review consisted of ensuring that the methods of analysis for seismic Category I ASME Code and non-Code items together with the design transients and use of computer programs for analysis will provide a complete basis for design of the reactor coolant boundary over the full range of system conditions expected in service.

The applicants have provided details of the general plant transient data for CRBR in Appendix B of the PSAR. This appendix is a compilation of the events which comprise the CRBR design duty cycle, as well as the method of selection of the umbrella transients. The data have been reviewed and have been found to be satisfactory, providing a complete list of the transients, corresponding frequency of temperature and pressure conditions, and conservative formulation of umbrella transients.

A list was compiled of all computer codes found in Appendix A of the PSAR which can be used for static and dynamic analysis. All of the codes on the list were fully reviewed and found acceptable.

The applicants have committed to not using any experimental stress analysis program in lieu of theoretical methods of stress analysis.

The applicants have committed to a program of simplified and rigorous inelastic analysis, if acceptance limits cannot be satisfied by the more conservative elastic method. Although details of the inelastic methods are not presented in the PSAR, the applicants have stated that these methods will conform to the requirements of DOE's Std. RDT F9-4T and to the guidelines of DOE's RDT F9-5T. Paragraph 4 of RDT F9-5T provides a description of methods for time-independent elastic-plastic analysis and time-dependent creep analysis. For components and systems which undergo high-temperature service, the rules of Code Cases 1592 through 1596 are acceptable and will be applied as appropriate. The evaluation of high temperature ($T > 800^{\circ}F$) acceptance criteria is presented in Section 3.9.9 of this SER.

The dynamic analyses of systems, components, and equipment subjected to vibratory loadings, including those that result from fluid flow and postulated seismic events are reviewed in SRP Section 3.9.2. The acceptance criteria for each transient loading condition or combination thereof are reviewed in SRP Section 3.9.3 for ASME Code Classes 1, 2, and 3, or CS components and supports.

3.9.1.4 Evaluation Summary

The staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components are acceptable and meet the relevant requirements of CRBR PDC 1, 2, 12, and 13; 10 CFR 50, Appendix B; 10 CFR 50.55a; and 10 CFR 100, Appendix A. This conclusion is based on the following:

(1) The applicants have met the relevant requirements of CRBR PDC 12 and 13 by demonstrating that the design transients and resulting loads and load combinations with appropriate specified design and service limits which the applicants have used for designing Code Class 1 and CS components and supports, and reactor internals provide a complete basis for design of the reactor coolant boundary for all conditions and events expected over the service lifetime of the plant.

(2) The applicants have met the relevant requirements of CRBR PDC 2 and 10 CFR, 100, Appendix A, by including seismic events in design transients which serve as design basis to withstand the effects of natural phenomena.

(3) The applicants have met the relevant requirements of 10 CFR 50, Appendix B, and CRBR PDC 1 by having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I, Code Classes 1, 2, and 3, and CS structures, and non-Code structures within the present state-of-the-art limits, and by having design control measures which are acceptable to ensure the quality of the computer programs.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff reviewed dynamic testing and analysis of systems, components, and equipment as detailed in Section 3.9.2.1 according to the guidance and recommendations provided in SRP Section 3.9.2.

3.9.2.1 Area of Review

The review covered the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those resulting from fluid flow and postulated seismic events to ensure conformance with CRBR PDC 1, 2, 5, 12, and 13. Sections 3.7, 3.9.1, and 4.2.2 of the PSAR were reviewed for conformance with these requirements. The review covered the following specific areas:

(1) Piping vibration, thermal expansion, and dynamic effect testing should be conducted during startup testing. The systems to be monitored should include (a) all ASME Section III, Classes 1, 2, and 3 systems, (b) high-energy piping systems inside seismic Category I structures, (c) high-energy portions of systems whose failure could reduce to an unacceptable safety level the functioning of any seismic Category I plant feature, and (d) seismic Category I portions of moderate-energy piping systems located outside containment. The supports and restraints necessary for operation during the life of the plant are considered to be parts of the piping system. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions anticipated during service and to confirm that normal thermal motion will not be restrained. The test program description should include a list of different flow modes, a list of selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibration or indications of normal thermal motion restrain occurs.

(2) The following areas which are related to the seismic system analysis described in the applicants' Safety Analysis Report (SAR) were reviewed.

    (a) Seismic Analysis Method

    For all seismic Category I systems, components, equipment, and their supports (including supports for conduit and cable trays, and ventilation ducts), the applicable seismic analysis methods (response spectra, time history, equivalent static load) were reviewed. The manner in which the dynamic system analysis method will be performed was reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees or freedom was reviewed. The manner in which consideration will be given in the seismic dynamic analysis to maximum relative displacements between supports was reviewed. In addition, other significant effects that will be accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response were reviewed.

(b) Basis for Selection of Frequencies

As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure were reviewed.

(c) Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of systems and components were reviewed.

(d) Combination of Modal Responses

When a response spectrum approach is used for calculating the seismic response of systems or components, the phase relationship between various modes is lost. Only the maximum responses for each mode can be determined. The maximum responses for modes do not in general occur at the same time and these responses have to be combined according to some procedure selected to approximate or bound the response of the system. When a response spectra method was used, the description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) was reviewed, including that for modes with closely spaced frequencies.

(e) Analytical Procedures for Piping Systems

The analytical procedures applicable to seismic analysis of piping systems, including methods used to consider differential piping support movements at different support points located within a structure and between structures, were reviewed.

(f) Multiply Supported Equipment and Components with Distinct Inputs

The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs were reviewed.

(g) Use of Constant Vertical Static Factors

Where applicable, justification for the use of constant static factors as vertical response loads for designing seismic Category I systems, components, equipment and their supports in lieu of the use of a vertical seismic system dynamic analysis was reviewed.

(h) Torsional Effects of Eccentric Masses

The criteria and procedures that are used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic system analyses were reviewed.

(i) Seismic Category I Buried Piping Systems

For seismic Category I buried piping, the seismic criteria and methods which consider the effect of fill settlement including pipe profile and pipe stresses, the movements at support points, penetrations, and anchors were reviewed.

(j) Interaction of Other Piping With Category I Piping

The seismic analysis procedures to account for the potential failure of nonseismic Category I piping systems in the seismic design of seismic Category I piping were reviewed.

(k) Criteria Used for Damping

The criteria to account for damping in systems, components, equipment, and their supports were reviewed.

(3) Dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions should be analyzed. The purpose of this analysis is to predict the vibration behavior of the components, so that the input forcing functions and the level of response can be estimated. Before conducting the analysis, the specific locations for calculated responses, the considerations in defining the mathematical models, the interpretation of analytical results, the acceptance criteria, and the methods of verifying predictions by means of tests should be determined.

(4) Flow-induced vibration testing of reactor internals should be conducted during the preoperational and startup test program. The puprose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude or structural damage. The test program description included a list of flow modes, a list of sensor types and locations, a description of test procedures and methods to be used to process and interpret the measured data, a description of the visual inspections to be made, and a comparison of the test results with the analytical predictions.

(5) Dynamic system analyses should be performed to confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from normal and accident conditions in combination with the SSE. The review covered the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

(6) A discussion should be provided which will describe the methods to be used to correlate results from the reactor internals vibration test with the analytical results from dynamic analyses of the reactor internals under steady-state and operational flow transient conditions.

In addition, test results from previous plants of similar characteristics may be used to verify the mathematical models used for the loading condition of postulated normal and accident conditions in combination with the SSE by comparing such dynamic characteristics as the natural frequencies. The review covered the methods to be used for comparison of test and analytical results and for verification of the analytical models.

3.9.2.2 Acceptance Criteria and Basis

The acceptance criteria are based on meeting the relevant requirements set forth in CRBR PDC 1, 2, 5, 12, and 13. The relevant requirements are as follows:

(1) CRBR PDC 1 as it relates to the testing and analysis of systems, components, and equipment with appropriate safety functions being performed to appropriate quality standards.

(2) CRBR PDC 2 as it relates to systems, components, and equipment important to safety being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena (SSE).

(3) CRBR PDC 5 as it relates to systems and components important to safety being appropriately protected against the dynamic effects of discharging fluids.

(4) CRBR PDC 12 as it relates to systems and components of the reactor coolant boundary being designed so as to have an extremely low probability of rapidly propagating failure or of gross rupture.

(5) CRBR PDC 13 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the reactor coolant boundary will not be breeched during normal operating conditions including anticipated operational occurrences.

For this area of review, the criteria for the CRBR (PDC) are essentially the same as those of LWRs (GDC). Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

(1) Relevant requirements of CRBR PDC 12 and 13 will be met if vibration, thermal expansion, and dynamic effects testing will be conducted during startup functional testing for piping, and its supports and restraints. The purpose of these tests is to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the Code and to confirm that no unacceptable restraint of normal thermal motion occurs.

An acceptable test program to confirm the adequacy of the designs should consist of the following:

(a) A list of systems that will be monitored.

(b) A listing of the different flow modes of operation and transients such as pump trips, valve closures, and so forth, to which the components will be subjected during the tests. For example, the transients associated with the reactor coolant system heatup tests should include, but not necessarily be limited to: (i) reactor coolant pump start, (ii) reactor coolant pump trip.

(c) A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during tests. For each of these selected locations, the deflection (peak-to-peak) or other appropriate criteria, to be used to show that the stress and fatigue limits are within the design levels, should be provided.

(d) A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot positions.

(e) A description of the thermal motion monitoring program, that is, verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured.

(f) If vibration should be noted beyond the acceptance levels set by the criteria of (c), above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints will be determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in (d), above, a description should be provided of the corrective action to be taken to ensure that the snubber is operable.

(2) To meet the relevant requirements of CRBR PDC 2, the acceptance criteria for the area of review described earlier in Section 3.9.2.1 are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all seismic Category I systems, components, equipment, and their supports to function as needed during and after an earthquake.

(a) Seismic Analysis Methods

The seismic analysis of all seismic Cateogry I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts) should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.

- Dynamic Analysis Method

  A dynamic analysis (e.g., response spectra method, time history method) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items:

  - Use of either the time history method or the response spectra method.

  - Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all seismic Category I and applicable non-seismic Category I systems and plant equipment. The number is considered adequate when additional degrees of freedom do not result in more than a 10% <u>increase</u> in responses. Alternately, the number of degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 Hz.

  - Investigation of a sufficient number of modes to ensure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10% <u>increase</u> in responses.

  - Consideration of maximum relative displacements among supports of seismic Category I systems and components.

  - Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.

- Equivalent Static Load Method

  An equivalent static load method is acceptable if:

  - Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar systems may be submitted in support of the use of the simplified method.

  - The design and associated simplified analysis account for the relative motion between all points of support.

  - To obtain an equivalent static load of equipment or components which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectra. A factor of less than 1.5 may be used if adequate justification is provided.

  In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable.

For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.

(b) Determination of Number of Earthquake Cycles

During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) should be assumed. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed.

(c) Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than half or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range will be acceptable if the equipment will be adequately designed for the applicable loads.

(d) Three Components of Earthquake Motion

Depending upon the basic methods used in the seismic analysis, that is, response spectra or time history method, the following two approaches are considered acceptable for the combination of three-dimensional earthquake effects.

- Response Spectra Method

  When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model.

- Time History Analysis Method

  When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. The first method is to obtain maximum responses that could result from each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in the response spectra method except that the maximum responses are calculated using the time history method instead of the response spectra method. The second method is to obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable, the earthquake motions specified in the three different directions should be statistically independent.

(e) Combination of Modal Responses

When the response spectra method of analysis is used to determine the dynamic response of damped linear systems, the most probable response is obtained as the square root of the sum of the squares of the response from individual modes. Thus, the most probable system response, R, is given by

$$R = \left[\sum_{k=1}^{N} R_k^2 \right]^{1/2}$$

where R_k is the response for the k^{th} mode and N is the number of significant modes considered in the modal response combination.

When modes with closely spaced modal frequencies exist, an acceptable method for obtaining the system response is to take the absolute sum of the responses of the closely spaced modes and combine this sum with other remaining model responses using the square root of the sum of the squares rule. Two modes having frequencies within 10% of each other are considered as modes with closely spaced frequencies.

(f) Analytical Procedures for Piping Systems

The seismic analysis of Category I piping may used either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are as given in Section 3.9.2.2(2)(a) above.

(g) Multiply Supported Equipment and Components with Distinct Inputs

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, by using the floor response spectra. For the latter option, the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration (in g) at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacement can then be imposed on the supported item in the most unfavorable combination. The responses that result from the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra involves determination of dynamic responses resulting from the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors and applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination using static analysis procedures.

In lieu of the response spectra approach, time histories of support motions may be used as excitations to the systems. Because of the increased analytical effort compared with the response spectra techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared with the response spectra envelope method for multiply supported systems.

(h) Use of Constant Vertical Static Factors

The use of constant vertical load factors as vertical response loads for the seismic design of all seismic Category I systems, components, equipment, and their supports in lieu of the use of a vertical seismic system dynamic analysis is acceptable only if it can be justified that the structure is rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction is more than 33 Hz.

(i) Torsional Effects of Eccentric Masses

For seismic Category I systems, if the torsional effect of an eccentric mass such as a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined on a case-by-case basis.

(j) Seismic Category I Buried Piping Systems

For seismic Category I buried piping systems, the following items should be considered in the analysis:

- The inertial effects of an earthquake upon buried piping systems should be adequately accounted for in the analysis.

Use of the procedures described in Newmark, Blume, and Kapur (1973) and Newmark (1972) are acceptable.

- The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, and so forth should be adequately considered. Use of the procedures described in Hetenyi (1946) is acceptable.

- When applicable, the effects of local soil settlements, soil arching, and so forth, should also be considered in the analysis.

(k) Interaction of Other Piping With Seismic Category I Piping

To be acceptable, each nonseismic Category I piping system should be designed to be isolated from any seismic Category I piping system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I piping system. If it is not feasible or practical to isolate the seismic Category I piping system, adjacent nonseismic Category I piping should be analyzed according to the same seismic criteria as applicable to the seismic Category I piping system. For nonseismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the nonseismic Category I piping should be simulated in the modeling of the seismic Category I piping. The attached nonseismic Category I piping, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the seismic Category I piping.

(l) Criteria Used for Damping

RG 1.61 provides acceptable values which may be used. The use of alternate damping values requires justification.

(3) Relevant requirements of CRBR PDC 1 and 5 are met as given below. The following guidelines, in addition to RG 1.20, apply to the analytical solutions to predict vibrations of reactor internals.

(a) The results of vibration calculations should consist of the following:

- Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, and the response causing the maximum stress amplitude should be calculated.

- The dynamic properties of internal structures, including the natural frequencies, the dominant mode shapes, and the damping factors should be characterized. If analyses are performed on a component structural element basis, the existence of dynamic coupling among component structural elements should be investigated.

- The response characteristics, such as the dependence on fluid induced excitation forces, the flow path configuration, coolant recirculation pump frequencies, and the natural frequencies of the internal structures, should be identified.

- Acceptance criteria for allowable responses should be established, as should criteria for the location of vibration sensors. Such criteria should be related to the Code-allowable stresses, strains, and limits of deflection that are established to preclude loss of function with respect to the reactor core structures and fuel assemblies.

(b) The forcing functions should account for the effects of transient flow conditions and the frequency content. Acceptable methods for formulating forcing functions for vibration prediction include the following:

- Analytical method: based on standard fluid-dynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.

- Test-analysis combination method: based on data obtained from plant tests or scaled model tests (e.g., velocity or pressure distribution data), forcing functions should be formulated which will include the effects of complex flow path configurations and wide variations of pressure distributions.

- Response-deduction method: based on a derivation of response characteristics from plant or scaled model test data, forcing functions should be formulated. However, since such functions may not be unique, the computational procedures and the basis for the selection of the representative forcing functions should be described.

(c) Acceptable methods of obtaining dynamic responses for vibration predictions are as follows:

- Force-response computations are acceptable if the characteristics of the forcing functions are predetermined on a conservative basis and the mathematical model of the reactor internals is appropriately representative of the design.

- If the forcing functions are not predetermined, either a special analysis of the response signals measured from reactor internals of similar design may be performed to predict amplitude and modal contributions, or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used.

(d) Vibration predictions should be verified by test results. If the test results differ substantical from the predicted response behavior, the vibration analysis should be appropriately modified to improve the agreement with test results and to validate the analytical method as appropriate for predicting response of the prototype unit, as well as of other units where confirmatory tests are to be conducted.

(4) Relevant requirements of CRBR PDC 1 and 5 are met as given below. The preoperational vibration test program for the internals should conform to the requirements specified in RG 1.20, including vibration prediction, vibration monitoring, data reduction, and surface inspection. The test program to demonstrate design accuracy of the reactor internals should include, but not necessarily be limited to, the following:

(a) The vibration testing should be conducted with the fuel elements in the core or with dummy elements which provide equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if it can be demonstrated that testing in this mode is conservative.

(b) A brief description of the vibration monitoring instrumentation should be provided, including instrument types and diagrams of locations, which should include the locations having the most severe vibratory motions or having the most effect on safety functions.

(c) The planned duration of the test for the normal operation modes to ensure that all critical components are subjected to at least 10^6 cycles of vibration should be provided. The allowable stress shall be taken as 50% of the fatigue design allowable stress shown at 10^6 cycles. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more will be acceptable.

(d) Testing should include all of the different flow modes of normal operation and upset transients. The proposed set of flow modes are acceptable if they provide a conservative basis for determining the dynamic response of the reactor internals.

(e) The methods and procedures to be used to process the test data to obtain a meaningful interpretation of the core structure vibration behavior should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and the possible effects on safety functions.

(f) Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.

(5) Relevant requirements of CRBR PDC 2 and 5 are met as given below. Dynamic systems analyses should be performed to confirm the structural

design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe normal and accident events in combination with the SSE.

Mathematical models used for dynamic system analysis for normal and accident events in combination with the SSE effects should include the following:

(a) Modeling should include reactor internals and dynamically related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the basis of modeling should be developed and described.

(b) Mathematical models should be representative of system structural characteristics, such as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

(c) Any system structural partitioning and directional decoupling employed in the dynamic system modeling should be justified.

(d) The effects of flow upon the mass and flexibility properties of the system should be discussed.

Typical diagrams and the basis for postulating the normal and accident events induced forcing functions should be provided, including a description of the governing equations and the assumptions used for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semiempirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions.

The methods and procedures used for dynamic system analyses should be described, including the governing equations of motion and the computational scheme used to derive results. Time domain forced response computation is acceptable for both normal and accident events and SSE analyses. The response spectra modal analysis method may be used for SSE analysis.

The stability of elements in compression should be investigated.

Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports.

(6) Relevant requirements of CRBR PDC 1 are met as given below. Regarding the correlation to be made of tests and analyses of reactor internals, a discussion covering the following items to ensure the adequacy and sufficiency of the tests and analysis results should be provided:

(a) Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for possible verification of the mathematical model used in the analysis.

(b) Comparison of the analytically obtained mode shapes with the shape of measured motion for possible identification of the modal combination or verification of a specific mode.

(c) Comparison of the response amplitude time variation and the frequency content obtained from test and analysis for possible verification of the postulated forcing function.

(d) Comparison of the maximum responses obtained from test and analysis for possible verification of stress levels.

(e) Comparison of the mathematical models used for dynamic system analysis under operational flow transients and under combined loadings, to note similarities.

3.9.2.3 Review Evaluation

The applicants' submittal of an analysis and testing program on safety-related piping systems, mechanical components, and reactor internal structures has been reviewed. The PSAR contains a commitment to conduct a piping steady-state vibration, thermal expansion, and operational transient test program. Experience obtained from the FFTF program has been included in the planning and testing.

The applicants have committed to perform an analysis of vibration of reactor internal structures using state-of-the-art methods. The applicants' scaled model, preoperational, and startup testing proposal has been reviewed, including plans to correlate the results of the analysis and tests. The CRBR is classified as a "prototype" with respect to testing requirements.

Elevated temperature effects in dynamic analysis are covered in Section 3.9.9 of this SER.

3.9.2.4 Evaluation Summary

The dynamic testing and analysis of systems, components, and equipment are acceptable and meet the relevant requirements of CRBR PDC 1, 2, 5, 12, and 13 and 10 CFR 50.55a. This conclusion is based on the following:

(1) The applicants have committed to meet the relevant requirements of CRBR PDC 12 and 13 with respect to the design and testing of safety-related piping systems and reactor internal structures to ensure that there is a low probability of rapidly propagating failure and of gross rupture and to ensure that design conditions are not exceeded during normal operation including anticipated operational occurrences by having an acceptable vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high- and moderate-energy piping, and all associated systems, restraints, and supports. The tests will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests will provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports

during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation.

(2) The applicants have committed to meet the relevant requirements of CRBR PDC 2 with respect to demonstrating design adequacy of all seismic Category I systems, components, equipment and their supports to withstand earthquakes by meeting the regulatory positions of RGs 1.61 and 1.92 and by providing acceptable seismic systems analysis procedures and criteria. The scope of review of the seismic systems analysis included the seismic analysis methods of all seismic Category I systems, components, equipment and their supports. It included review of procedures for modeling, inclusion of torsional effects, seismic analysis of seismic Category I piping systems, seismic analysis of multiple-supported equipment and components with distinct inputs, justification for the use of constant vertical static factors and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of nonseismic Category I piping with seismic Category I piping. This review has also included criteria and seismic analysis prodecures for reactor internals and seismic Category I buried piping outside containment.

The system analyses will be performed by the applicants on an elastic basis. Modal response spectra multidegree of freedom and time history methods form the bases for the analyses of all major seismic Category I system, components, equipment and their supports. When the modal response spectra method is used, governing response parameters will be combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses will be used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses will be used in accounting for three components of the earthquake motion for both the time history and response spectra methods. Floor spectra inputs to be used for design and test verifications of systems, components, equipment and their supports will be generated from the time history method. A vertical seismic system dynamic analysis will be employed for all system, and components, equipment and their supports where analyses show significant structural amplification in the vertical direction.

(3) The applicants have committed to meet the relevant requirements of CRBR PDC 1 and 5 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects by meeting the positions of RG 1.20 for the conduct of preoperational vibration tests and by having a preoperational vibration program planned for the reactor internals which provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test evaluation will provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity

of the reactor internals in service is essential to ensure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

(4) The applicants have committed to meet the relevant requirements of CRBR PDC 2 and 5 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the safe shutdown earthquake (SSE) by having a dynamic system analysis to be performed which provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of normal and accident conditions, and the SSE. The analysis will provide adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired.

(5) The applicants have committed to meet the relevant requirements of CRBR PDC 1 with respect to systems and components being designed and tested to quality standards commensurate with the importance of the safety functions to be performed by the proposed program to correlate the test measurements with the analysis results. The program will constitute an acceptable basis for demonstrating the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The staff reviewed the components, component supports, and core support structures as detailed in Section 3.9.3.1 according to the guidance and recommendations provided in SRP Section 3.9.3 (NUREG-0800).

3.9.3.1 Area of Review

The review covered the information presented in the applicants' preliminary safety analysis report (PSAR) concerning the structural integrity of components, their supports, and core support structures which are designed in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1 (hereinafter "the Code") and CRBR PDC 1, 2, 5, 12, and 13. PSAR Section 3.7 (Appendix A), 3.9, and portions of Sections 5.2 through 5.6 contain material pertinent to this review. The Standard Review Plan (SRP) was determined to be applicable to the low temperature (less than 800°F for austenitic material and 700°F for ferritic materials) areas of review. High-temperature aspects are discussed in Section 3.9.9 of the PSAR.

The review covered the following specific areas.

(1) Load Combination, System Operating Transients, and Stress Limits

The design and service load combinations (e.g., design and service loads, including system operating transients, in combination with loads calculated to result from postulated seismic and other events) specified for Code-constructed items designated as Code Classes 1, 2, 3 (including Classes 1, 2, and 3 component support structures), and CS core support structures were reviewed to determine that appropriate design and service limits have been designated for all loading combinations. This review ascertained that the design and service stress limits and deformation criteria comply with the applicable limits specified in the Code. Service stress limits which allow inelastic deformation of Code Class 1, 2, and 3 components, component supports, and Class CS core support structures were evaluated as were the justifications for the proposed design procedures. Piping which is "field run" was included. Internal parts of components, such as valve discs and seats and pump shafting subjected to dynamic loading during operation of the component were included.

(2) Design and Installation of Pressure Relief Devices

The design and installation criteria applicable to the mounting of pressure relief devices (safety valves and relief valves) for the overpressure protection of components were reviewed. The review included evaluation of the applicable loading combinations and stress criteria. The design review extended to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system. The design of safety and relief valve systems was reviewed with respect to the load combinations imposed on the safety or relief valves, upstream piping or header, downstream or vent piping, and system supports. On the reactor and intermediate coolant boundaries, the only relief capability required was for the cover gas. The steam system required relief/safety valves typical of nuclear plant steam system.

(3) Components Supports

The review of information submitted by the applicants included an evaluation of Code Classes 1, 2 and 3 component supports. The review included an assessment of design and structural integrity of the supports. The review included the following types of supports: plate and shell, linear, snubbers, and component standard types. Component supports are those metal supports which are designed to transmit loads from the component to the building structure.

3.9.3.2 Acceptance Criteria and Basis

Acceptance criteria are based on meeting the relevant requirements of the following regulations:

(1) CRBR PDC 1 as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed

(2) CRBR PDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions

(3) CRBR PDC 5 as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions

(4) CRBR PDC 12 as it relates to the reactor coolant boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

(5) CRBR PDC 13 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the design conditions are not exceeded

(6) CRBR PDC 31 states that the intermediate cooling system shall be designed with sufficient margin to meet its safety functions under all plant conditions

Specific criteria necessary to meet the relevant requirements of 10 CFR 50.55(a) and CRBR PDC 1, 2, 5, 12, and 13 are as follows:

(1) Loading Combinations, System Operating Transients, and Stress Limits

The design and service loading combinations, including system operating transients, and the associated design and service stress limits considered for each component and its supports should be sufficiently defined to provide the basis for design of Code Classes 1, 2, and 3 components, component supports, and previously "CS components," now Class CS core support structures for all conditions.

The acceptability of the combination of design and service loadings (including system operating transients), applicable to the design of Classes 1, 2, and 3 components, component supports, previously "CS components," and now Class CS core support structures, and of the designation of the appropriate design or service stress limit for each loading combination, was judged by comparison with appropriate standards developed by professional societies and standards organizations.

The design criteria for internal parts of components such as valve discs, seats, and pump shafting should comply with applicable ASME Code or Code Case criteria. In those instances for which no ASME criteria exist, the design criteria are acceptable if they ensure the structural integrity of the part in such a way that no safety-related functions are impaired.

(2) Design and Installation of Pressure Relief Devices

The applicants should use design criteria for pressure relief stations specified in Appendix O, ASME Code, Section III, Division 1, "Rule for the Design of Safety Valve Installations" and Code Case N-100, "Pressure Relief Valve Design Rules, Section III, Division I, Class 1, 2, or 3." Stress computations and stress limits must be in accord with applicable rules of the Code and/or Code Case.

(3) Component Supports

The component support designs should provide adequate margins of safety under all combinations of loading. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination should meet the criteria in RGs 1.124 and 1.130.

Component supports of active pumps and valves should be considered in context with the other features of the operability assurance program. If the component support affects the operability requirements of the supported components, then deformation limits should also be specified. Such deformation limits should be compatible with the operability requirements of the components supported and incorporated into the operability assurance program defined in SRP Section 3.10. In establishing allowable deformations, the possible movements of the support based structures must be taken into account.

3.9.3.3 Review Evaluation

The specified loading combinations and stress limits employed by the applicants in the design of Code Classes 1, 2, and 3 components, component supports, and Class CS core support structures have been reviewed to determine if they are appropriate. The applicants have committed to design and analyze components, pressure relief devices, and component supports in accordance with the criteria outlined herein. Analysis methods will be accomplished using state-of-the-art methods. Final details of component supports cannot be supplied until analyses have been completed. However, the applicants state that they shall be designed and analyzed in accordance with the ASME Code, Section III, Subsection NF, to withstand the specified loading combinations. The applicants have established a structural evaluation plan (SEP) to ensure that manufacturers will conform to the requirements of SRP Section 3.10 for design and analysis of components.

3.9.3.4 Evaluation Summary

The specified design and service combinations of loading, and associated limits, as applied to ASME Code Classes 1, 2, and 3 components are acceptable and meet the requirements of CRBR PDC 1, 2, 5, 12, and 13 and 10 CFR 50.55a. This conclusion is based on the following:

(1) The applicants have committed to meet the requirements of CRBR PDC 1, 2, and 5, with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Classes 1, 2 and 3 components, by ensuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of normal operation as well as postulated accident conditions and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code Classes 1, 2, and 3 components in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site or other service loadings that result from postulated events or

system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

(2) The applicants have committed to meet the requirements of CRBR PDC 1, 2, and 5, with respect to the criteria used for design and installation of ASME Code overpressure relief devices by ensuring that safety and relief valves and their installation will be designed to standards which are commensurate with their safety functions, and that they can accommodate the effects of discharge resulting from normal operation as well as postulated accident conditions including the dynamic effects resulting from the safe shutdown earthquake. The relevant requirements of CRBR PDC 12, 13, and 31 will also be met with respect to ensuring that the reactor and intermediate coolant boundary design limits for normal operation including anticipated operational occurrences are not exceeded. The criteria used by the applicants in the design and installation of ASME safety and relief valves will provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

(3) The applicants have committed to meet the requirements of CRBR PDC 1, 2, and 5 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Classes 1, 2, and 3 component supports by ensuring that component supports important to safety are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated accident events and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, will meet the positions and criteria of RGs 1.124 and 1.130 and are in accordance with NUREG-0484. The specified design and service loading combinations used for the design of ASME Code Classes 1, 2 and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings from postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.4 Control Rod Drive Systems

The staff reviewed the control rod drive systems as detailed in Section 3.9.4.1 according to the guidance and recommendations provided in SRP Section 3.9.4 (NUREG-0800).

3.9.4.1 Area of Review

The primary and secondary control rod drive systems (CRDS) consist of the control rods and the related mechanical components which provide the means for mechanical movement. CRBR PDC 24 and 25 require that the pair of CRDS provide one of the independent reactivity control systems. The rods and the drive mechanisms shall be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences, or under postulated accident conditions. A positive means for inserting the rods shall always be maintained to ensure appropriate margin for malfunction, such as stuck rods. Portions of the CRDS are a part of the reactor coolant boundary, and the systems shall be designed, fabricated, and tested to quality standards commensurate with the safety functions to be performed, so as to ensure an extremely high probability of accomplishing the safety functions either in the event of anticipated operational occurrences or in withstanding the effects of postulated accidents and natural phenomena such as earthquakes. Section 4.2.3 of the PSAR and Appendix C of the PSAR contain the major portions of the applicants submittal which discuss the CRDS, and which form the basis for this review.

3.9.4.2 Acceptance Criteria and Basis

Acceptance criteria are based on meeting the requirements of the following regulations:

(1) CRBR PDC 1 as it relates to CRDS, requires that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

(2) CRBR PDC 2 as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform their safety functions.

(3) CRBR PDC 12 as it relates to CRDS, requires that the reactor coolant boundary portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

(4) CRBR PDC 24 as it relates to CRDS, requires that the pair of CRDS be one of the independent reactivity control systems and are designed with appropriate margins to ensure their reactivity control functions under anticipated normal operation conditions.

(5) CRBR PDC 25 as it relates to CRDS, requires that the CRDS be designed with appropriate margins to be capable of controlling reactivity and cooling the core under postulated accident conditions.

(6) CRBR PDC 58 as it relates to CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to ensure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

(1) Sufficient descriptive information is provided.

(2) Construction (as defined in NCA-1110 of Section III of the ASME Code) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

(a) Portions of Equipment Classified as Quality Group A, B, C (RG 1.26)

Section III of the ASME Code, Class 1, 2, or 3 as appropriate.

(b) Portions of Equipment Classified as Quality Group D (RG 1.26)

- Section VIII, Division 1 of the ASME Code for vessels and pump castings.

- Applicable to piping systems (ANSI):

B16.5	Steel Pipe Flanges and Flanged Fittings
B16.9	Steel Butt Welding Fittings
B16.11	Steel Socket Welding Fittings
B16.25	Butt Welding Ends
B31.1	Piping
SP-25	Standards
B16.34	Valves

(c) Other Equipment (Non-ASME Code)

Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than those for other plants of similar design having a period of successful operation. Justification of any decreases should be provided.

(3) For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, the stress limits applicable to all portions of the CRDS should be met.

(4) The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and overcoming a stuck rod meet system design requirements.

3.9.4.3 Review Evaluation

The applicants have provided details of the design and testing of the primary and secondary CRDS in Section 4.2.3 of the PSAR and Appendix C of the PSAR. The two designs are similar to and based on LWR and FFTF CRDS since operating experience with these types of CRDS has proven that malfunctions are rare, and the performance requirements for the CRDS are essentially no different in the FFTF and LWRs, and the CRBR. Consequently, it was determined that the LWR Standard Review Plan and its references are almost entirely applicable to the CRBR.

The load combinations, stress and deformation limits, and operability under these conditions have been reviewed for adequacy and are consistent with SRP Section 3.9.3. The system quality group classification has been reviewed and found to be consistent with SRP Section 3.2.2. The applicants submitted a detailed component and life cycle testing program in PSAR Section 4.2.3 and Appendix C. The review consisted of ensuring that the testing program was designed to demonstrate that the CRDS will function properly during and after normal operating occurrences, seismic events, and postulated accident conditions over the full range of system conditions and misalignments expected in service. It was determined that the testing program will include the proper functional tests such as rod insertion and withdrawal, latching operation, scram operation and time, ability to overcome a stuck rod, and wear.

3.9.4.4 Evaluation Summary

The design of the CRDS are acceptable and meet the requirements of CRBR PDC 1, 2, 12, 24, 25, and 58 and 10 CFR 50.55a. This conclusion is based on the following:

(1) The applicants have submitted a detailed program of design, analysis, testing, and operational and shutdown checks to ensure the reliability and safety of the CRDS. The program is designed to demonstrate that under the worst cases of tolerances, misalignment, and loadings (such as seismic events), the systems will perform their functions of inserting the control rods into the reactor. The two independent systems have different designs, each of which meets the stress criteria of the appropriate ASME and ANSI Codes under the proposed loading conditions.

(2) The applicants have met the requirements of CRBR PDC 1 with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the CRDS are in conformance with the requirements of appropriate ANSI and ASME Codes.

(3) The applicants have met the requirements of CRBR PDC Criteria 2, 12, and 24 with respect to designing the CRDS to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure their reactivity control functions and with extremely low probability of leakage or gross rupture of the reactor coolant boundary. The specified design transients, design and service loadings, combination of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes.

(4) The applicants have met the requirements of CRBR PDC 25 and 58 with respect to designing the CRDS to ensure their capability of controlling reactivity and cooling the reactor core with appropriate margin, in conjunction with the reactor protection system. The operability assurance program is

acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stu[ck] rod.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Area of Review

The reactor pressure vessel internals consists of all of the structural and mechanical elements within the reactor vessel. PSAR Section 4.2.2 of Appendix G contain material pertinent to this review. The term "reactor internals" includes core support structures and other internal structures with the exception of the following:

(1) reactor fuel elements and the reactivity control elements out to the coupling interfaces with the drive units (the fuel system design is covered in SRP Section 4.2, but the structural aspects of reactor fuel assemblies are reviewed with the reactor internals)

(2) control rod drive elements (the drive elements inside the guide tubes are covered in SRP Section 3.9.4, but the guide tubes are reviewed with the reactor internals)

(3) in-core instrumentation (in-core instrumentation support structures are reviewed with the reactor internals)

The review covered the following specific areas:

(1) the design or physical arrangements of all reactor internals structures, which includes positioning, support, and provisions to accommodate dimensional changes that result from thermal or other effects

(2) the loading conditions that determine the design basis of the reactor internals

(3) the design basis for the mechanical design of the reactor vessel internals, which includes maximum allowable stresses, stability, deflection, cycling, and fatigue limits

(4) the combination of design and service loadings with respect to the allowable design or service limits or other appropriate criteria

3.9.5.2 Acceptance Criteria and Basis

The acceptance criteria are based on meeting the requirements of the following regulations:

(1) CRBR PDC 1 and 10 CFR 50.55a, as it relates to reactor internals, require that the reactor internals shall be designed to quality standards commensurate with the importance of the safety functions to be performed.

(2) CRBR PDC 2, as it relates to reactor internals, requires that the reactor internals shall be designed to withstand the effects of earthquakes without loss of capability to perform its safety functions.

(3) CRBR PDC 5, as it relates to reactor internals, requires that reactor internals shall be designed to accommodate the effects of and to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated loss-of-coolant accidents.

(4) CRBR PDC 8, as it relates to reactor internals, requires that reactor internals shall be designed with adequate margins to ensure specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences.

Specific criteria necessary to meet the relevant requirements of the regulations identified above are presented below.

The following requirements are applicable for service temperatures of less than 800°F for austenitic materials and 700°F for ferritic materials.

(1) Requirements for loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in SRP Section 3.9.3.

(2) The design and construction of the core support structures should conform to the requirements of Subsection NG, "Core Support Structures," of the ASME Code, and SRP Section 3.9.3.

(3) The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

(4) The applicants should establish deformation limits for reactor internals and present them in their safety analysis report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are discussed in SRP Section 3.9.2.

Specific criteria for high-temperature ($T > 800$°F for austenitic materials, $T > 700$°F for ferritic materials) applications are addressed in Section 3.9.9.

3.9.5.3 Review Evaluation

The applicants have provided details of the reactor internals in Section 4.2.2 of the PSAR. The lower internals structure positions and restrains the reactor assemblies. The upper internals structure stabilizes the control rod drivelines, supports in-vessel instrumentation, and provides redundant mechanical hold-down of reactor assemblies. A passive core restraint system controls the interactions of all reactor assemblies and upper and lower internals structures. The reactor internals concept at CRBRP is similar to the FFTF design and performs essentially the same duties as comparable LWR structures. Consequently, it was determined that the LWR Standard Review Plan and its reference are almost entirely applicable to the CRBR.

The review consisted of ensuring that the design and analysis of the reactor internals will provide adequate support, positioning, and restraint of the reactor assemblies during and after normal operating occurrences, seismic events, and postulated accident conditions over the full range of system conditions and misalignments expected in service. It was determined that the selection of many of the CRBR design features was based on operating experience gained at FFTF. The CRBR core support concept is based on the FFTF design, although the CRBR core basket has been simplified to facilitate construction. The review has determined that the design of the reactor internals does provide the proper features to ensure safe operation, such as core support structures, sufficient coolant flow, bypass flow modules to preclude major flow blockage, transfer and storage of vessel, control of core motion, and coolant mixing to control flow stratification.

The applicants stated that the design and analysis of the reactor internals will be in accordance with Subsection NG of the ASME Code, Section III. In addition, the applicants stated that final evaluation of each of the reactor internal components will include displacement calculations for all operating conditions and will be shown to satisfy the design limits.

Several related reviews are not covered by this section. Dynamic testing and analysis of the reactor internals are reviewed in Section 3.9.2, and the requirements for loads and loading combinations are reviewed in Section 3.9.3 of this SER.

3.9.5.4 Evaluation Summary

Based on the review of PSAR Section 4.2.2, the staff concludes that the design of reactor internals is acceptable and meets the requirements of CRBR PDC 1, 2, 5, and 8 and 10 CFR 50.55a. This conclusion is based on the following:

(1) The applicants have met the requirements of CRBR PDC 1 and 10 CFR 50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.

(2) The applicants have met the requirements of CRBR PDC 2, 5, and 8 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated loss-of-coolant accidents with sufficient margin to ensure that capability to perform their safety functions will be maintained and the specified acceptable fuel design limits will not be exceeded.

The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting defections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 Inservice Testing of Pumps and Valves

The staff reviewed inservice testing of pumps and valves according to the guidance and recommendations provided in SRP Section 3.9.6.

3.9.6.1 Area of Review

This review covered the inservice testing of certain safety-related pumps and valves typically designated as Classes 1, 2, and 3 under Section III of the ASME Code. Other pumps and valves not categorized as Code Classes 1, 2, or 3 may be included if they are considered to be safety related by the staff. Appendix G of the PSAR contains material pertinent to this review. Compliance with the Code will ensure conformance with the CRBR PDC 51, 40, 45, and 37 and of 10 CFR 50.55a.

(1) Inservice Testing of Pumps

(a) The descriptive information in the PSAR covering the inservice test program was reviewed for those ASME Code Classes 1, 2, and 3 system pumps whose function is required for safety, and in addition include pumps not categorized as Code Classes 1, 2, or 3 but which are considered to be safety related.

(b) Procedures for testing for speed, fluid pressure, flow rate, vibration amplitude, lubricant level or pressure, and bearing temperature at normal pump operating conditions were reviewed.

(c) The pump test schedule was reviewed.

(d) The methods described in the PSAR for measuring the reference values and inservice values for the pump parameters above were reviewed.

(2) Inservice Testing of Valves

The descriptive information in the PSAR covering the inservice test program was reviewed for those ASME Code Classes 1, 2, and 3 valves whose function is required for safety. This review did not include those nonsafety-related valves exempted by the Code.

(3) Relief Request

10 CFR 50.55a(g) requires a nuclear power facility to periodically update its inservice testing program to meet the requirements of future revisions of Section XI of the ASME Code. However, if it proves impractical to implement these criteria, the applicants are allowed to submit requests

for relief from Section XI requirements of a case-by-case basis. Accordingly, any request for relief would be reviewed by the staff to determine if the proposed exceptions to Section XI will degrade the overall plant safety.

3.9.6.2 Acceptance Criteria and Basis

The acceptance criteria are based on meeting the relevant requirements set forth in the CRBR PDC 51, 40, 45, and 37 and 10 CFR 50.55a. The relevant requirements are as follows:

(1) CRBR PDC 51, as it relates to periodic functional testing of the containment atmospheric cleanup systems to ensure the leak-tight integrity and the performance of the active components, such as pumps and valves.

(2) CRBR PDC 40, as it relates to periodic functional testing of the cooling water system to ensure the leak-tight integrity and performance of the active components.

(3) CRBR PDC 45, as it relates to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation valves and determine valve leakage acceptability.

(4) 10 CFR 50.55a, as it relates to including pumps and valves whose function will be required for safety in the inservice inspection program to verify operational readiness by periodic testing.

(5) CRBR PDC 37 states that the reactor residual heat extraction systems shall be designed to permit appropriate periodic functional and inspection testing.

Specific criteria necessary to meet the relevant requirements identified herein are indicated below.

The ASME Code Section XI, Division 3, provides rules for inspection and testing of components of liquid-metal-cooled plants. However, Division 3 presently does not contain subsections for inservice testing of liquid-metal-retaining pumps and valves as compared with subsections IWP and IWV of Division 1. The staff has determined that conformance to the requirements of ASME Code Section XI, Division 1 is satisfactory with the provision that any variance, deletions, and additions to the Division 1 inservice inspection requirements will be clearly stated and justified in an acceptable manner by the applicants.

(1) Inservice Testing of Pumps

(a) The scope of the applicants' test program is acceptable if it is in agreement with IWP-1000 of Section XI of the Code and in addition includes pumps not categorized as Code Classes 1, 2, or 3, but which are considered to be safety related. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in IWP-3000, the establishment of a reference set of parameters and a consistent test method is a basic criterion of the program.

(b) The pump test program is acceptable if it meets the requirements for establishing reference values and the periodic testing schedules of IWP-3000 of Section XI of the Code. The allowable ranges of inservice test quantities, corrective actions, and bearing temperature tests are established by IWP-3000 and IWP-4000. The pump test schedule in the plant Technical Specifications is required to comply with these rules.

(c) The test frequencies and durations are acceptable if the provisions of IWP-3000 of Section XI of the Code are met.

(d) The methods of measurement are acceptable if the test program meets the requirements of IWP-4000 of Section XI of the Code with regard to instruments, pressure measurements, temperature measurements, rotational speed, vibration measurements, and flow measurements.

(2) Inservice Testing of Valves

(a) To be acceptable, the PSAR valve test list must contain all safety-related Code Classes 1, 2, and 3 valves required by IWV-1100 except those nonsafety-related valves exempted by the Code and in addition includes valves not categorized as Code Classes 1, 2, or 3 but which are considered safety related. The PSAR valve list must include a valve categorization which complies with the provisions of IWV-2000 of Section XI of the Code. Each specific valve to be tested by the rules of Subsection IWV is listed in the PSAR by type, valve identification number, Code class, and IWV-2000 valve category.

(b) The valve test procedures are acceptable if the provisions of IWV-3000 of Section XI of the Code are met with respect to preservice and periodic inservice valve testing.

(3) Information Required for Review of Relief Requests

(a) Identify component for which relief is requested:

- Name and number as given in PSAR
- Function
- ASME Section III Code Class
- For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

(b) Specifically identify the ASME Code requirements that has been determined to be impractical for each component.

(c) Provide information to support the determination that the requirement in Item (b) is impractical; that is, state and explain the basis for requesting relief.

(d) Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

(e) Provide an explanation as to why the proposed inservice testing will provide an acceptable level of quality and safety and not endanger the public health and safety.

(f) Provide the schedule for implementation of the procedure(s) in Item (d).

Requests for relief from Section XI requirements will be granted by the staff if the applicants have adequately demonstrated either of the following:

(1) Compliance with the Code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety, and noncompliance will provide an acceptable level of quality and safety.

(2) Proposed alternatives to the Code requirements or portions thereof will provide an acceptable level of quality and safety.

3.9.6.3 Review Evaluation

The guidelines of SRP Section 3.9.6 ensure that the applicants have provided sufficient information in the Safety Analysis Report to demonstrate that the safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant. This requirement is met by conformance to a comprehensive program of testing and inspection. The testing and inspection program for pumps and valves is a major safety concern with direct applications to all reactor plants. Consequently, it was determined that the LWR SRP Section 3.9.6 and its references are almost entirely applicable to the CRBR.

The applicants have committed to an inservice inspection program, presented as Appendix G of the CRBR PSAR, which will be in accordance with the requirements of the ASME, Section XI, Divisions 1 and 3. Since at the time of this review Division 3 does not include subsections for the inservice testing of pumps and valves, the applicants have committed to follow the intent of Section XI, Division 1, Subsections IWP and IWV, respectively. This commitment is taken to mean that the applicants shall identify and provide suitable justification for any variance from the requirements of Subsections IWP and IWV. Subject to a determination by the staff that the requested variances are acceptable, full compliance with the requirements of Subsections IWP and IWV of Division 1 will render the pump and valve test program acceptable and will satisfy the relevant requirements set forth in the CRBR PDC 37, 40, 45, and 51 and 10 CFR 50.55a.

3.9.6.4 Evaluation Summary

The applicants have committed to an inservice inspection program which will comply with the guidelines of the ASME Code Section XI, Division 1 and 3. Subject to the satisfactory resolution of the requested variances, the staff concludes that the applicants' pumps and valves test program is acceptable and meets the requirements of the CRBR PDC 37, 40, 45, and 51 and 10 CFR 50.55a. This conclusion is based on the applicants provision of a test program to ensure that safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant. This program includes baseline preservice testing and periodic inservice testing. The program will provide for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress. The applicants have also formulated their inservice test program to include all safety-related Code Classes 1, 2, and 3 pumps and valves and to include those pumps and valves which are not Code Classes 1, 2, and 3 but are considered to be safety related.

3.9.7 Applicability of ASME Code Edition and Addenda and of ASME and ANSI Code Cases (SRP Section 5.2.1)

3.9.7.1 Area of Review

The following areas of the PSAR were included in this review: Tables 2.1-1 and 3.2-5, pertinent portions of Sections 3.2.2 and 5.3.1.1, and sections in Chapters 3, 4, and 5 which discuss Code cases.

In order to establish that safety-related components of the reactor coolant boundary and other fluid systems important to safety or nuclear power plants are in compliance with the Codes and Standards Rule, 10 CFR 50.55a, the applicants are required to provide a table in their safety analysis report (SAR) identifying vessels, piping, pumps and valves, and the component code, Code edition, applicable addenda, and component order date (where applicable) for each component. 10 CFR 50.55a requires that components of the reactor coolant boundary be designated as Class 1 components and constructed* in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1 (hereinafter "the Code"), except for components which meet the exclusion requirements of Footnote 2 of the rule. Components of the reactor coolant boundary which meet the exclusion requirements of Footnote 2 may be classified as Quality Group B in accordance with RG 1.26 and constructed as Class 2 components in accordance with the Code.

For construction permit (CP) applications, the review will determine the acceptability of the information presented in the PSAR, to ensure that the applicants are in compliance with the rules of 10 CFR 50.55a.

In the event there are cases where conformance with the Codes and Standards Rule would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, the applicants must provide a complete description of the circumstances and the basis for proposed alternate requirements. The applicants must describe how an equivalent and acceptable level of safety and quality will be provided by the proposed alternate requirements. The SAR should identify differences between the specific portions of the Code and Code Addenda to which each component has been constructed and that which is required for conformance with 10 CFR 50.55a.

In addition to ensuring compliance with the Codes and Standards Rule, the review determined the acceptability of American Society of Mechanical Engineers (ASME) and American National Standards Institute (ANSI) Code case interpretations specified in the safety analysis report (SAR). These Code cases must be approved

*"Constructed," as used herein, is an all-inclusive term comprising materials certification, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.

before being applied to ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB-Class 1 Components, as stated in 10 CFR 50.55a (a)(2)(ii).

3.9.7.2 Acceptance Criteria and Basis

Acceptance criteria are based on meeting the relevant requirements of the following regulations:

(1) CRBR PDC 1, as it relates to the requirement that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

(2) 10 CFR 50.55a as it relates to establishing minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of components within the reactor coolant boundary and other fluid systems important to safety of nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

To meet the requirements of CRBR PDC 1 and 10 CFR 50.55a, the following regulatory guides are used:

(1) RG 1.26 describes an acceptable method of determining quality standards for Quality Groups B, C, and D water- and steam-containing components important to the safety of nuclear power plants.

(2) RG 1.84 lists those Section III ASME Code cases oriented to design and fabrication which are acceptable to the staff for implementation in the licensing of nuclear power plants.

(3) RG 1.85 lists those Section III ASME Code cases oriented to materials and testing which are acceptable to the staff for implementation in the licensing of nuclear power plants.

(4) RG 1.87 discusses the Section III ASME Code cases used in the design of the elevated temperature components.

(5) RG 1.147 lists those Section XI ASME Code cases which are acceptable to the staff for use in the inservice inspection of nuclear power plants.

3.9.7.3 Review Evaluation

The following areas of the PSAR have been reviewed for compliance with the Codes and Standards Rule, i.e., 10 CFR 50.55a.

(1) 3.1.1, Definitions and Explanations

This subsection contains an acceptable definition of the reactor coolant boundary.

(2) Table 2.1-1, Components Which Comprise the Reactor Coolant Boundary
This table contains a list of components comprising the reactor coolant boundary.

(3) 3.2.2, Safety Classifications

This subsection contains a commitment by the applicants to meet the quality requirements of 10 CFR 50.55a as published September 30, 1974.

(4) Table 3.2-5, Preliminary List of ASME Code Classification for Seismic Category I Mechanical System Components

This table provides a comprehensive list of CRBR safety-related components with the ASME Code classes, editions, addenda, and applicable Code cases. It includes all of the items (reactor coolant boundary components) contained in Table 3.1-1.

(5) 5.3.1.1, (PHTS) Performance Requirements

This subsection indicates that the PHTS will be designed to the 1974 Edition of the ASME Code, with addenda to summer of either 1974 or 1975, depending on the component. This meets the edition and addenda requirements of 10 CFR 50.55a.

(6) Code standards and Code cases associated with the elevated temperature design are reviewed elsewhere in this report.

3.9.7.4 Evaluation Summary

The staff concludes that system components are in compliance with 10 CFR 50.55a and meet the requirements of CRBR PDC 1. This conclusion is based on the following:

The applicants have met the requirements of 10 CFR 50.55a and CRBR PDC 1 with respect to the construction of structures, systems, and components important to safety and to quality standards. The requirements have been met by ensuring that the components of the reactor coolant boundary as defined by the rules of 10 CFR 50.55a, have been properly classified in Table 3.1-1 of the SAR as ASME Section III, Class 1 (Quality Group A) components except for those reactor coolant boundary components which meet the exclusion requirements of Footnote 2 of the rule. These reactor coolant boundary components are classified Quality Group B in accordance with the guidance provided in Regulatory Position C.1 of RG 1.26 and are constructed as ASME Section III, Class 2 components. Table 3.1-1 identifies the component Code, Code edition, and Code addenda for each Quality Group A component such as; reactor vessel, reactor coolant pumps, control valves, block valves, other reactor coolant boundary valves and each Quality Group B component such as interconnecting piping and valves of the reactor coolant boundary which meet the exclusion requirements of Footnote 2 of the rule.

The review of the component Code, Code editions and addenda, as applied to each of these reactor coolant boundary components, indicates they will be constructed in accordance with the requirements of the applicable Codes and addenda that are specified by the rules of 10 CFR 50.55a. The review of Quality Group B (ASME Section III, Class 2) and Quality Group C (ASME Section III, Class 3) components of other fluid systems important to safety is discussed in Section 2.2.2 of this SER.

The specified ASME and ANSI Code cases whose requirements will be applied in the construction of ASME Section III, Division 1, Class 1, Class 2, Class 3, and Class SC components are in accordance with the rules of 10 CFR 50.55a and the guidance provided in RGs 1.84, 1.85, 1.87, and 1.147. It is concluded that compliance with the requirements of these Code cases will result in a component quality level commensurate with the importance of the safety function of these components and constitutes an acceptable basis for satisfying the requirements of CRBR PDC 1.

3.9.8 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

The staff reviewed the determination of rupture locations and dynamic effects associated with the postulated rupture of piping according to guidance and recommendations provided in SRP Section 3.6.2.

3.9.8.1 Area of Review

CRBR PDC 5 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures. Section 3.6 of the PSAR and Reference 2 to Section 1.6 of the PSAR, "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment" (WARD-D-0185) were reviewed according to this criterion.

The applicants have committed to provide information concerning break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping, including "field run" piping, inside and outside of containment in their Safety Analysis Report (SAR). This information was reviewed in accordance with SRP Section 3.6.2, to confirm that requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe rupture will be met.

The review covered the following specific areas:

(1) the criteria used to define break and crack locations and configurations

(2) the analytical methods used to define the forcing functions, including the jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related structures, and components

(3) the dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices, under postulated pipe rupture loads

3.9.8.2 Acceptance Criteria and Basis

The acceptance criteria are based on meeting the requirements of CRBR PDC 5 as it relates to structures, systems, and components important to safety being designed to accommodate the dynamic effects of postulated pipe rupture, including postulation of pipe rupture locations; break and crack characteristics; dynamic analysis of pipe whip; and jet impingement loads.

Specific criteria necessary to meet the relevant requirements of CRBR PDC 5 are as follows:

(1) Postulated Pipe Rupture Locations Inside Containment

Acceptable criteria to define postulated pipe rupture locations and configurations inside containment are specified in BTP MEB 3-1, Revision 1 (attachment to SRP Section 3.6.2).

(2) Postulated Pipe Rupture Locations Outside Containment

For protection against postulated pipe ruptures outside containment, BTP MEB 3-1, Revision 1, provides acceptable criteria to define postulated rupture locations and plant layout considerations.

(3) Methods of Analysis

Detailed acceptance criteria covering pipe whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement, are included in Subsection III of SRP Section 3.6.2.

Although these criteria are generally applicable to a sodium design, some variations were encountered because of features of the design. These are discussed in the review evaluation which follows.

3.9.8.3 Review Evaluation

Section 3.6 of the PSAR and Reference 2 to Section 1.6 of the PSAR, "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment" (WARD-D-0185), were reviewed according to the acceptance criteria described above. Significant variations were found.

First, all sodium piping has been classified as moderate-energy piping even though it is operated at temperatures requiring a high-energy piping classification according to the acceptance criteria. Boiling temperatures for sodium and NaK are significantly higher than their normal operating temperatures in CRBR applications. The saturation temperature at atmospheric pressure is 1,630°F. The saturation temperature for NaK at ambient pressure is 1,450°F. Therefore, in comparison with conventional water systems, no CRBR sodium or NaK systems operate with any significant amount of internal fluid-stored energy. Operating pressures in essentially all CRBR sodium systems depend solely on the discharge head of the system pump. Since the criteria are intended for water piping, and since the sodium piping is to be operated at temperatures well within the subcooled region for sodium at atmospheric pressure, the moderate-energy piping classification is acceptable.

Second, leak rates for sodium piping in containment are substantially less than those required by the acceptance criteria. This is based on the "leak before break" concept that is presented in the piping integrity report described above. The reduced rate is acceptable for components at low temperatures because: (1) the "leak before break" argument was found to be conservative and acceptable for that temperature range, (2) the three-loop redundant sodium piping in containment allows isolation of leaks without loss of safety function, and (3) placing all sodium piping in containment in lined inerted cells mitigates and isolates the consequences of a sodium leak.

BTP MEB 3-1, Revision 1, is the current document specified in the PSAR for defining break locations.

3.9.8.4 Evaluation Summary

On the basis of the leak before break concept for components above 800°F, the staff's evaluation concluded that the pipe rupture postulation and the associated effects are adequately considered in the plant design, and therefore, are acceptable and meet the requirements of CRBR PDC 5. This conclusion is based on the following:

(1) The proposed pipe rupture locations have been adequately assumed and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement will provide adequate protection to the integrity and functionality of safety-related structures, systems, and components.

(2) The proposed piping and restraining arrangement and applicable design considerations for high- and moderate-energy fluid systems inside and outside of containment, including the reactor coolant boundary, will provide adequate assurance that the systems and components important to safety that are in close proximity to the postulated pipe rupture will be protected. The design will be of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside of containment.

(3) For the sodium piping system in the containment, the applicants propose as a design basis the occurrence of a reference 4-in. crack. The size of this reference crack is evaluated in Section 6 of WARD-D-0185 report based on fatigue crack propagation calculated for postulated extreme conditions.

The systems outside the containment are designed based on the criteria in SRP Section 3.6.2; therefore they are acceptable and meet CRBR PDC 5.

3.9.9 Elevated-Temperature Mechanical Integrity

3.9.9.1 Area of Review

The CRBR systems, components, and supports operating at elevated temperature were reviewed with regard to potential failure modes. Design for operation at elevated temperatures requires consideration of additional failure modes, the increased cyclic thermal loads associated with the higher temperatures, and the reduced material strength and ductility in the creep regime. Time-dependent nonlinear response of the material at elevated temperatures must be considered in the analysis.

Systems and components in service at elevated temperatures are subjected to larger temperature variations and differentials than LWR hardware. Moreover, the materials have lower strength at elevated temperatures. The resulting higher thermal strain ranges and increased inelastic strain concentrations tend to accelerate fatigue damage. In addition, the materials are susceptible to creep-rupture damage that results from both applied and residual stresses persisting after transient conditions. Relaxation of such stresses tends to cause ratcheting on subsequent load cycles. The effective microscopic ductility of many of the materials and product forms is reduced by concentration of creep strains in grain boundaries. Consequently, cracking can occur at accumulated strain levels that would cause no problems at temperatures below the creep regime.

The review covers the mechanical engineering aspects of safety for CRBR systems, components, and supports that will operate at elevated temperatures and are included in Chapters 3, 4 (portion), and 5 of the PSAR.

3.9.9.2 Review Evaluation and Acceptance Criteria

3.9.9.2.1 Finding No. 1--Weldment Safety Evaluation Confirmatory Program Required

Potential cracking problems in weldments of the materials of interest operating at the elevated temperatures of interest are a cause for concern. A number of important factors apparently have not been included in the CRBR application for weldments in service at elevated temperatures. The structural integrity of weldments in service at elevated temperatures has not yet been satisfactorily demonstrated by the applicants. The following additional factors must be taken into account:

(1) consideration of crack initiation in the heat-affected zone (HAZ) of the weldment exposed to cyclic sodium temperatures at the inside surface

(2) consideration of the creep-fatigue and creep-rupture damage peculiar to the material property variations or metallurgical notch effects at weldment

(3) consideration of time rate, cyclic rate, and hold-time effects on the HAZ of the weldment in the presence of long shallow cracks

(4) consideration of the enchanced creep in the remaining uncracked wall thickness caused by residual stresses and thermal cycling

(5) evaluation of stability of remaining uncracked wall ligament for operation in the creep regime

A confirmatory program of test and analyses is required to provide quantitative evaluation of the above open questions for the parameters of interest in the CRBRP. Quantitative results are required before an operating license is issued.

Resolution

Resolution consists of the applicants carrying out a program to confirm the structural adequacy of the CRBRP design with regard to weldment integrity. This program will encompass the major elements described herein to quantify the safety margins of the weldments in service at elevated temperatures and hot-leg piping.

The applicants in their February 10, 1983 letter (J. R. Longenecker to J. N. Grace) agreed to provide additional confirmatory testing during the OL review. The details of the confirmatory testing program have not been finalized but the staff has provided a typical confirmatory testing plan in EG&G report EA-6150 (January 1983). The staff finds this acceptable for the CP review.

3.9.9.2.2 Finding No. 2--Elevated-Temperature Seismic Effects

ASME Code Case 1592 imposes limits on various inelastic strains accumulated within the life of a component. The life history is described by grouped cycles of limited intensities. The consequence of varying the loading sequence is not important below the creep regime, and stresses are classified into stress-controlled primary and strain-controlled secondary values. These stress values are then used to perform structural analyses of the cyclic life of the structure.

Seismic events impose high short-term primary stresses on the structure. The seismic loads affect the inelastic strain accumulation by changing the residual stresses that produce enhanced creep. Seismic loads also produce plastic strain accumulation generated within each motion if the intensity of the shake is great enough to cause plastic ratcheting. The relaxation of high residual stresses that exist after a seismic event produces enhanced creep during subsequent operation at elevated temperatures. Consequently, the sequence of loading becomes important in the creep regime.

Resolution

The applicants are committed to take into account any enhanced creep (ratcheting) and any creep-rupture damage resulting from residual stresses at local stress raisers following seismic events. This necessarily includes consideration of the sequence of the seismic events with respect to the operating transients. Since the methods used by the applicants have been supplemented by DOE's RDT, F9-4T, "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components," and F9-5T, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components," this issue is considered resolved with the NRC review of RDT F9-4T and F9-5T as described in Section 3.9.9.2.3, and the resolution of any relevant findings resulted from the review.

3.9.9.2.3 Finding No. 3--Design Analysis Methods, Codes and Standards (Open--Subject to NRC Review of RDT F9-4T and F9-5T Design Methods and Criteria and Resolution of Findings by Applicants)

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

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

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

The applicants in the PSAR and in the report WARD-D-0185 indicate that full inelastic analysis will be used for locations where elastic analysis results do not meet Code limits. The inelastic analysis will be performed in accordance with the RDT F9-5T. Since both RDT F9-4T and F9-5T have not had the benefit of independent review as national consensus standards nor review by the staff, they could not be treated as validated acceptance criteria for the conduct of this review.

Resolution

The proposed resolution consists of the following actions:

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

(2) The staff will conduct a review of RDT F9-4T and F9-5T and will identify any revisions or further technical justification that may be necessary to meet

national consensus and NRC safety standards. On the basis of an initial review, the applicants' use of average material properties alone does not appear to provide sufficient justification of structural safety margins. The applicants' calculated creep-rupture damage may be too low when compared with the considerable strain and cyclic hardening that occurs during fabrication and operation. The applicants' calculated fatigue damage and accumulated strains may be too low if the actual yield strength will be below the average value used in the design analyses.

3.9.9.2.4 Finding No. 4--Elastic Followup in Elevated-Temperature Piping

At elevated temperatures, during creep relaxation, a portion of the elastic train is converted to creep strain. Areas of piping that are more highly stressed are subjected to additional cyclic strain and strain accumulation resulting from elastic followup. To provide for adequate safety margins, the Code requires that stresses with elastic followup be classified as "primary stresses." The applicants, however, performed an inelastic analysis for thermal loading and did not include any portion of the thermal expansion stresses as primary. This approach circumvents the demonstration of adequate safety margins for creep-rupture and creep-fatigue damage.

Resolution

The staff and the applicants agreed to satisfactory methods and criteria for quantifying thermal expansion stresses in piping systems that must be considered to be primary (meeting of November 22, 23, and 24, 1982 held at Westinghouse-ARD). The resolution of this item consists of the applicants using these methods and criteria.

3.9.9.2.5 Finding No. 5--Notch Weakening

The basic allowable stress limits of the Code are based on unnotched creep specimen test data. Stress raisers influence the creep behavior of the entire wall in two basic ways. They introduce a constraint against inelastic flow by inhibiting slip line development. This is manifested in a reduction in the average stress intensity in the net section (a notch strengthening effect). Stress raisers also introduce a site where creep-rupture damage could cause early crack initiation and more rapid crack progagation (a notch weakening effect). Although the combined effect is notch strengthening in most cases, an evaluation is needed to determine what geometric, loading, and material parameters could cause significant notch weakening, particularly for long-term loading at elevated temperatures. Loading conditions such as transverse shear do not introduce any notch strengthening and have contributed to weldment cracking at structural discontinuities.

The applicants should commit to an acceptable program for conducting a parametric study of geometric notches, loading conditions, and material properties in the CRBRP design that is needed to quantify the extent and seriousness of the problem before a construction permit is issued. This study should examine long-term loadings where the material ductility may be minimized by prior cyclic and monotonic straining and thermal aging. Geometric configurations with low inelastic flow constraint and high local stress concentrations should be considered in this evaluation.

Resolution

Resolution consists of the applicants carrying out a program to confirm their creep-fatigue and creep-rupture damage criteria for geometric notches, including local stress and strain concentration effects at structural discontinuities. The applicants in their February 10, 1983 letter (J.R. Longenecker to J. N. Grace) agreed to provide additional confirmatory testing during the OL review. The details of the confirmatory the testing program have not been finalized but the staff has provided a typical confirmatory testing plan in EG&G report (EA-6150, January 1983). The staff finds this acceptable for the CP review.

3.9.9.2.6 Finding No. 6--Creep-Fatigue Evaluation

The applicants have modified the creep-fatigue damage rules given in Code Case 1592 when applied to austenitic stainless steel types 304 and 316 for components that are not Code stamped. These rules assume that the compressive hold, the creep damage is 20% as damaging as that caused by the same sustained stress in tension. The applicants have presented test data showing that compressive stresses have little damaging effect for austenitic stainless steels. By the same token, however, there are some studies that indicate that shear stress is a valid creep-rupture criterion for stainless steels. If the latter were the case, compressive stresses would cause more damage than that obtained by the applicants' modified rules. Thus, the applicants should provide documented justification for arriving at the 20% factor, rather than using 100% damage as required by the Code.

Stainless steel materials subjected to high cycle thermal fluctuations and flow-induced vibrations require a fatigue strength evaluation beyond the Code Case 1592 curve limit of 10^6 cycles. Thus, for the high cycle fatigue evaluation of stainless steels beyond the Code Case limit of 10^6 cycles, the applicants have extrapolated the fatigue curve using a slope of -0.12 on cycles for load-controlled situations. The applicants have also developed special purpose high cycle fatigue criteria for strain-controlled situations. For temperatures below 800°F, ASME Code Committees adopted a new high cycle fatigue design curve up to 10^{11} cycles in 1982. The allowable stress for 10^9 cycles is reduced to 14,000 psi, whereas previously the Code did not go beyond 10^6 cycles, where the allowable stress was 28,200 psi. Similar data on cycles beyond 10^6 also are available for temperatures above 800°F. The applicants should confirm that Code safety margins are met with the new fatigue design curves as extended to 10^{11} cycles.

For 2¼% Cr-1% Mo new fatigue design curves that account for environmental effects have recently been approved by ASME Code Committees. The applicant should ensure that Code safety margins are met with the new fatigue design curves.

Resolution

The applicants have provided sufficient technical justification to resolve this issue but have not yet submitted the required documentation of

(1) the use of the 20% (or less) damage factor for compressive hold times in types 304 and 316 stainless steels

(2) the evaluation of the effects of the reduced fatigue design curves for austenitic materials cycled beyond 10^6 cycles, which will be issued in the Winter 1982 Addenda of the ASME Code

(3) the evaluation of the effects of the 2¼ Cr-1% Mo elevated-temperature fatigue design curves currently proposed by ASME Code Committees

This issue is considered resolved subject to the applicants' completion of the required documentation.

3.9.9.2.7 Finding No. 7--Plastic Strain Concentration Factors

This issue concerns the use of the plastic strain concentration factors, K_e, in performing fatigue evaluations. The simplified methods of the ASME Code, used by the applicants (e.g., in the core support structure--support cone weld analysis), allow this factor to be unity until the primary plus secondary stress rang exceeds 3 S_m. Actually, this factor begins to exceed unity when the local maximum stress range, including the elastic stress concentration factor, exceeds 2 S_y. Moreover, strain multipliers for the concentration of plastic strain on the weaker side of a product form or materials interface is not included in existing formulas for K_e in the Code. The lack of conservatism in the simplified elastic-plastic method of the ASME Code has been pointed out in the published literature.

Resolution

The applicants are committed to performing a satisfactory evaluation of actual or conservative plastic strain concentration effects and the resulting fatigue design life wherever the local maximum stress range exceeds 2 S_y.

3.9.9.2.8 Finding No. 8--Intermediate Heat Transport System Transition Weld

The transition joints of the intermediate heat transport system (IHTS) were analyzed in accordance with the ASME Code and applicable RDT standards. A detailed inelastic analysis showed that the hot joint could meet the ASME Code criteria for only a 15-year life. Also, the applicants' conclusion is based on an anticipated minimum carbon content that does not fall below 0.05%. This joint is expected to see 936°F. The variation in expansion properties between the different materials may be critical, and the difference in properties should be carefully examined. The increased creep-rupture damage resulting from the higher yield properties produced by hardening in a multipass welding process should be evaluated.

Resolution

Resolution consists of the applicants commitment to perform analyses using the methods and criteria to be developed in the confirmatory program described in Section 3.9.9.2.1 in order to evaluate the structural integrity of the critical IHTS transition joints for 30 years' service. If these transition welds cannot be shown to be adequate for 30 years' service, the applicants must provide an acceptable plan for earlier replacement before an operating license is issued.

3.9.9.2.9 Finding No. 9--Steam Generator

The steam generator design is supported by existing test results and by several planned tests, which are outlined in the PSAR. Tests of mechanical properties are included in the planned program to verify and supplement methods in the ASME Code and RDT standards and design information for ensuring the structural adequacy of the steam generator. Prototype and other steam generator tests are included in the program. These test programs are designed to verify assumptions and to provide quantitative data to confirm the adequacy of design analyses.

Simplified elastic design methods for steam generator tubesheets are given in Section III of the ASME Code. These methods are only applicable where plastic deformations are not significant and temperatures remain below the creep regime. The CRBR steam generator tubesheets will be subjected to severe thermal cycling beyond the elastic regime. Significant thermal stresses will arise in the outer region of the perforated area of the steam generator tubesheet adjacent to the rim. Creep rupture damage combined with fatigue resulting from relaxation of high residual stresses limits the life of the component. The ASME Code does not provide similar simplified methods for the design of perforated plates in service at elevated temperatures.

The applicants have stated that Code Case 1592 supplemented by RDT F9-4T will be used for the tubesheet. Although the general criteria for elevated-temperature design may be found in Code Case 1592, the application of these criteria to the three-dimensional stress variations in ligaments operating at elevated temperatures is difficult. The steam generator tubesheet is a complex and unique structure that requires special design analysis methods to handle creep effects. Elastic followup is known to occur in ligaments subjected to cyclic straining under creep conditions. Thermal stresses with elastic followup must be classified as "primary stresses."

During thermal transients, the temperature of the ligaments closely follows the temperature of the fluid in the tubes. The thermal response of the unperforated rim, however, is significantly delayed. Significant in-plant stresses arise between the rim and perforated part of the tubesheet. Moreover, the transient results in severe radial temperature gradients at the interface the rim and perforated part of the tubesheet. Thus, the outer ligaments remain at significantly higher temperatures than those in the remaining area of the tubesheet. These higher temperatures result in lower strength, which makes the outer ligaments potentially more vulnerable to inelastic deformation. Contraction of the perforated portion of the tubesheet may be sufficiently high to cause unacceptable inelastic flow in the outer ligaments. Thus, special analysis methods are needed to bound the deformation, strain ranges, a maximum stress in order to obtain reliable fatigue and creep-rupture damage evaluations.

The applicants have indicated that a detailed inelastic finite element analysis will be performed for a sector of the tubesheet. However, difficulties arise with such an analysis in the modeling of ligaments because of the complex thermal structural interaction with the rim and the tubes. The elastic response of the perforated plate is isotropic; the inelastic response is anisotropic. Moreover, the tubesheet being a discrete structure is particularly sensitive to the use of minimum versus average properties for inelastic analysis. Thus, the boundary conditions and effective properties of the perforated region for inelastic analyses under creep conditions have to be very carefully modeled in order to achieve the required structural integrity.

The tube-to-tubesheet junction is a critical location in the steam generator. The design methods should account for the local stresses at the junction of the tubsheet plate and standpipes.

Resolution

The resolution of this item consists of the following actions:

(1) The applicants are committed to perform test of mechanical properties tests of mechanical properties to verify and supplement the methods in the ASME Code and the RDT standards and design information for ensuring the structural adequacy of the steam generator. Prototype steam generator tests will be run to verify certain performance characteristics. Hydraulic test model, large-leak tests, few-tube tests, DNB, (departure from nucleate boiling) tests, tube support wear tests, modular steam generator tests, single-tube performance tests, stability and interaction tests, tube-to-tubesheet weld tests, scale hydraulic model feature tests, and flow-induced vibration tests will also be conducted. The tests are needed to confirm the structural adequacy of the tubes.

(2) The applicants will carry out a program to confirm the adequacy of the methods and criteria used to ensure the structural adequacy of the tube-sheet for its intended lifetime. The specific tasks involved are:

    (a) Develop effective properties of perforated region for use in inelastic design analyses.

    (b) Evaluate effects of thermal gradients and equivalent material property variations on ligaments near the periphery of the perforated region.

    (c) Extend existing Appendix A-8000 Code methods for calculating the linearized membrane, shear, and in-plant bending* stresses in the ligaments using the equivalent solid plate stresses. Include all of these nominal stresses in the comparison with allowable primary membrane plus bending, and primary plus secondary allowables.

    (d) Develop methods of evaluating local cyclic plastic strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

    (e) Develop methods of evaluating local cyclic creep strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

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*In-plant bending occurs on either side of a minimum ligament section creating a "kinking" type of failure mechanism.

(f) Evaluate elastic followup into outermost ligaments

- Reclassify portion of discontinuity stresses caused by pressure and mechanical loads as "primary" in accordance with the associated amount of elastic followup that occurs during thermal transients.

- Reclassify portion of thermal stresses as "primary" in accordance with the amount of elastic followup that occurs during thermal transients.

(g) Develop ratcheting evaluation methods for outermost ligaments based on elastic equivalent solid plate stresses reclassified according to Item 2(f) and including nominal membrane, shear, and in-plant bending stresses.

(h) Develop creep rupture damage evaluation methods for outermost ligaments based on equivalent solid plate stresses. The effects of elastic followup will reduce the amount of stress relaxation and increase the creep-rupture damage.

(i) Perform detailed tube-to-tubesheet joint analysis for tubes in high radial thermal transient region at periphery of the perforated region and include local thermal effects.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment and Mechanical Systems and Components

3.10.1 Area of Review

The staff review has considered the applicants' approach in addressing the following areas:

(1) Identification of all mechanical and electrical equipment required to perform the functions associated with the emergency reactor shutdown, containment isolation, reactor core cooling, reactor and containment heat removal, as well as those required to prevent significant release of radioactive materials to the environment. Also included is the equipment that will be used by the operator to perform these functions manually and where failure can prevent the satisfactory accomplishment of one or more of the above safety functions.

(2) Identification of the design bases for safety-related mechanical and electrical equipment identified, including the definition of all loads during the normal, abnormal, accident, and postaccident conditions.

(3) Requirements for documentation of the qualification tests and analyses that have been or will be performed on the equipment to meet the design bases.

(4) Demonstration of the adequacy of the seismic and dynamic qualification programs.

3.10.2 Acceptance Criteria and Basis

The acceptance criteria for the area of review are based on meeting the relevant requirements of the following regulations:

(1) PDC 1 and 12 as they relate to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed

(2) PDC 2 and Appendix A to 10 CFR 100 as they relate to qualifying equipment to withstand the effects of natural phenomena such as earthquakes

(3) PDC 5 as it relates to qualifying equipment as being capable of withstanding the dynamic effects associated with external missiles, internally generated missiles, pipe whip, and jet impingement forces

(4) GDC 13, Appendix A, 10 CFR 50 as it relates to qualifying equipment associated with the reactor coolant boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

(5) Appendix B to 10 CFR 50 as it relates to qualifying equipment using the quality assurance criteria provided

Even though the following specific criteria, regulatory guides, and industry standards are applicable to light-water reactors, they provide information, recommendations, and guidance and in general describe a basis acceptable to the staff that may be used to implement the requirements of the regulations above.

(1) RGs 1.89, 1.100, and 1.148
(2) ANSI B16.41, N41.6, N45, N278.1-1975, N551.4
(3) ANSI/ASME N551.1 and N551.2
(4) IEEE 323-1974 and 344-1975

3.10.3 Review Evaluation

The staff has reviewed the PSAR and responses to requests for additional information and finds the applicants' seismic and dynamic program for qualification of safety-related mechanical and electrical equipment acceptable with regard to the construction permit licensing criteria.

Four concerns that the staff identified have been resolved satisfactorily. In particular, Project Management Corporation retains the responsibility to maintain all relevant documentation on equipment qualification in an auditable manner. Total margin was established by including the 10% margin required by IEEE 323-1974 using the required response spectrum (RRS) derived at the mounting of the equipment for the worst location of the equipment on the supporting structure, that is, the location which gives the highest response spectrum. If the final design input should be different from that for which a test was performed, a comparison will be made between the test response spectrum (TRS) and the final design RRS to ensure that the TRS also envelopes the new RRS. A combination of test and analysis may also be used if the final design of the equipment is modified.

Equipment that is subject to the simultaneous occurrence of seismic and other dynamic loads was analyzed for the combination of both loads. The more conservative absolute linear summations methodology was in general used for this combination. The square root of the sum of the squares (SRSS) methodology was used for deriving the three components of an earthquake. Nonlinear material response and analysis methodology have been satisfactorily described in several references and sections within the PSAR. The nonuniqueness of the seismic time history should be considered by using multiple time histories of motion whose response spectrum envelope the design response spectrum.

3.10.4 Review Summary

The staff concludes that the information reviewed meets the intent of the current criteria for seismic and dynamic qualification of safety-related mechanical and electrical equipment.

3.11 Environmental Qualification of Safety-Related Electrical and Mechanical Equipment

3.11.1 Area of Review

The staff review has considered the applicants' approach in addressing the following areas:

(1) Identification of all mechanical and electrical systems required to perform the functions associated with emergency reactor shutdown, containment isolation, reactor core cooling, reactor and containment heat removal as well as those required to prevent significant release of radioactive materials to the environment. Also included in the equipment that will be used by the operator to perform these functions manually and where failure can prevent the satisfactory accomplishment of one or more of the above safety functions.

(2) Identification of environmental design bases for the equipment identified including the definition of the normal, abnormal, accident and postaccident environments.

(3) Requirements for documentation of the qualification tests and analyses that have been or will be performed on the equipment to meet the design bases.

(4) Demonstration of the adequacy of the environmental qualification (EQ) programs.

3.11.2 Acceptance Criteria and Basis

10 CFR 50.49 specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment located in a harsh environment. In accordance with 10 CFR 50.49, the acceptance criteria for CRBRP are specified in Category I of NUREG-0588. For safety-related mechanical equipment located in harsh environments, SRP Section 3.11 states that the NUREG-0588 requirements are general in nature and also could be applied to mechanical equipment.

3.11.3 Review Evaluation

The applicants referenced Topical Report WARD-D-0165, "Requirements for Enviro mental Qualification of Class 1E Equipment," in PSAR Section 3.11.

In addition to this, the applicants have provided information by letters dated December 29, 1982, and January 27 and February 1983 to supplement the information contained in Section 3.11 of the PSAR.

3.11.3.1 Completeness of Safety-Related Systems

The applicants were directed to (1) establish a list of systems and components that are required to prevent or mitigate an accident and (2) identify components needed to perform the function of safety-related display instrumentation, postaccident sampling and monitoring, and radiation monitoring.

The applicants' list of systems for the EQ program, as presented in WARD-D-165, Revision 6, was comparable to the list contained in PSAR Section 3.2 and is acceptable to the staff for the EQ program.

Display instrumentation that will provide information to the reactor operators to aid them in the safe handling of the plant was included in the program. The acceptability of qualification for display instrumentation required by RG 1.97, Revision 2, will be determined during the operating license.

At the operating license review stage, the applicants should provide the list of all nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment. The nonsafety-related equipment identified must be included in the EQ program.

3.11.3.2 Service Conditions

The applicants, in their submittal, have identified the environmental service conditions for all the equipment required to function foir the particular accident. The review and evaluation of the adequacy of these environmental conditions are described below:

(1) Temperature, Pressure, and Humidity Conditions Inside Containment

The applicants' design philosophy, in general, includes total separation between the equipment of the different channels required for the prevention and mitigation of the accident. A severe temperature condition encountered by the equipment for more than one channel will result from primary sodium tank failure during maintenance. The peak value resulting from this event will be much smaller than that normally encountered in the LWR during LOCA or main steamline break events.

The staff has reviewed the profiles in the submittal and finds them acceptable for use in equipment qualification; that is, there is reasonable assurance that the actual pressures and temperatures will not exceed those profiles anywhere within the specified environmental zone (except in the break zone which may be in the proximity of the pipe or at most extend to the cell surrounding the pipe).

(2) Temperature, Pressure, and Humidity Conditions Outside Containment

The applicants have provided the temperature, pressure, and humidity conditions associated with high-energy line breaks outside containment. The criteria used to define the size and location of high-energy line breaks are described in PSAR Section 3.6. The following areas outside containment have been addressed:

(a) control building
(b) steam generator building
(c) diesel generator building
(d) reactor service building

The staff has used a screening criteria of saturation temperature at the calculated pressure to verify that the parameters identified by the applicants are acceptable.

(3) Submergence

The applicants' design philosophy, in general, includes complete separation between the equipment located inside containment and that used for the different channels. Hence, this equipment is no required to be qualified for the submergence as its failure will not result in a common mode failure. The effects of flooding on safety-related equipment in the auxiliary building were presented by the applicants in Section 3.6 of the PSAR and in the applicants' EQ program. All safety-related equipment subjected to submergence will be qualified for submergence for the time duration necessary for it to complete its safety function.

(4) Chemical Environment

Although equipment will not be subjected to severe chemical environments in CRBRP buildings, it may be subjected to sodium aerosol concentrations up to and including 15 mg/m^3 (8.85 mg/m^3 equivalent sodium) suspended equivalent sodium peroxide and 0.1 g/m^2 (0.059 g/m^2) equivalent sodium peroxide as a result of sodium spills. This sodium aerosol environment will be the result of a sodium spill in an intermediate heat transport system (IHTS) cell being ingested by other CRBRP air-filled cells. The resulting sodium aerosol environment will leave the building containing the spill in an air exhaust ducting and will be will be ingested by surrounding air-filled buildings for 10 sec before the ingesting building air intake vent is isolated by closure of the air intake vent dampers. This will result in a nonsevere sodium environment for these buildings. The applicants have committed to qualify all Class 1E equipment for the sodium aerosol environment to which it is exposed. At present, no qualification testing has been completed for equipment located in nonsevere or severe sodium aerosol environments. The applicants are planning to conduct equip-ment qualification testing, including testing of the sodium aerosol environment, in the near future. The applicants should submit the results from these tests to the staff for review and evaluation. Because it is feasible either to qualify the equipment or protect the equipment by mounting it in the enclosure, the staff finds that the applicants' approach to qualifying the equipment for a sodium aerosol environment is acceptable.

(5) Aging

The aging program requirements for CRBRP electrical equipment are defined in Section 4, Category I of NUREG-0588. The degrading influences of temperature, radiation, vibration, and electrical and mechanical stresses should be considered and included in the aging program. Any justifications for excluding preaging of equipment in type testing should be established on the basis of equipment design and application or on state-of-the-art aging techniques. A qualified life is to be established for each equipment item.

In addition to the above, a maintenance/surveillance program should be implemented to identify and prevent significant age-related degradation of electrical and mechanical equipment. The applicants have committed to follow the recommendations in RG 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)," which endorses ANS-3.2/ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." This standard defines the scope and content of a maintenance/surveillance progrm for safety-related equipment. Provisions for preventing or detecting age-related degradation in safety-grade equipment are specified and include (a) utilizing experience with similar equipment, (b) revising and updating the program as experience is gained with the equipment during the life of the plant, (c) reviewing and evaluating malfunctioning equipment and obtaining adequate replacement components, and (d) establishing surveillance tests and inspections based on reliability analyses, frequency and type of service, or age of the items, as appropriate.

(6) Radiation (Inside and Outside Containment)

The Class 1E equipment will be qualified to the worst-case radiation environment on the basis of normal service exposure plus the most severe radiation environment predicted to occur before and during those portions of the specific accident transients for which the component is required to perform its safety function. The applicants have identified three different design-basis source terms for which the equipment is required: (1) the site suitability source term, (2) cover gas release with the reactor at power, and (3) sodium storage tank failure during maintenance. All three source terms are of nonmechanistic origin and only the sodium storage tank failure is considered a design-basis event.

The applicant also has defined the design methodology used to calculate the radiation environments on the basis of the different source terms. The staff has reviewed the methodology and finds it acceptable for use in the qualification of equipment.

3.11.3.3 Documentation

The applicants are committed to meet the documentation requirements identified in IEEE 323-1974 and have the qualification data packages, consisting of the documentation that demonstrates qualification of safety-related equipment, in a systematic and auditable form. The staff finds the applicants' plan for documentation acceptable and in accordance with 10 CFR 50.49.

3.11.3.4 Qualification Method

Detailed procedures for qualifying safety-related electrical equipment in a harsh environment are defined in 10 CFR 50.49 and NUREG-0588. Type testing of equipment in a sequence consisting of preaging (thermal, radiation, and mechanical), seismic and dynamic loading, and exposure to accident conditions (where applicable) is the principal method of qualification.

Although there are no detailed requirements for mechanical equipment, CRBR PDC 1 and 5; Appendix B to 10 CFR 50, Sections III and XVII; and SRP Section 3.11, Revision 1, contain the following requirements and guidance related to equipment qualification:

(1) Components shall be designed to be compatible with the postulated environmental conditions.

(2) Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

(3) Design control measures shall be established for verifying the adequacy of design.

(4) Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

On the basis of the applicants' commitment to meet these requirements, the staff finds the applicants' approach for qualification of mechanical and electrical equipment acceptable.

3.11.4 Evaluation Summary

The staff has reviewed and evaluated the CRBRP program for the environmental qualification of safety-related equipment. This review has included the systems selected for qualification, the environmental conditions resulting from design-basis accidents, and the methods used for qualification. On the basis of these considerations, the staff concludes that satisfactory completion of the program will ensure conformance with the requirements of 10 CFR 50.49 as specified in NUREG-0588, relevant parts of CRBR PDC 1 and 5, and Sections III, XI, and XVII of Appendix B, 10 CFR 50, for safety-related equipment.

Table 3.1 Compliance with principal design criteria (PDC)

PDC	SER sections that discuss compliance with principal design criteria
1	3.2, 3.8, 3.9, 3.11, 4.2.4, 4.4.7, 4.5, 5.2, 5.3, 5.5, 7.1, 9.10, 17
2	2.3, 2.4, 3.2, 3.3, 3.4, 3.5, 3.7, 3.8, 3.9, 4.2.4, 5.6.2, 7.2.3, 7.3, 7.4, 7.5, 7.6, 8.3.3, 9.6, 9.7, 9.9, 9.10, 9.14, 9.15, 15.7
3	3.8, 9.13.1, 15.7
4	3.2, 5.2, 5.3, 8.3.3, 9.2, 9.3, 9.5, 9.6, 9.7.3, 9.8, 9.13.2, 10.4.5, 15.3, 15.6, 15.7
5	2.3, 3.2, 3.5, 3.8, 3.9, 3.11, 5.2, 5.3, 5.6.2, 5.6.3, 5.6.5, 7.2.3, 7.3, 7.4, 7.5, 7.6, 9.14, 9.15, 10.2, 15.6, 15.7
6	3.8, 5.6.2, 5.6.3, 5.6.5, 9.14, 8.2.1, 8.3.3
7	9.4
8	3.9.5, 4.2, 4.3, 4.4, 5.2, 7.5, 15.2, 15.3, 15.4
9	4.3
10	4.3
11	4.3, 5.2, 5.3, 5.6.3, 5.6.5, 7.3, 7.4, 7.5, 7.7, 15.2, 15.3, 15.4
12	3.9, 4.2.4, 4.5, 5.2, 5.3, 5.5, 5.11, 9.8, 10.4.6, 15.3
13	3.9, 4.4.7, 5.2, 5.5, 9.3.6, 10.4.6, 15.3
14	3.8, 6.2, 15.6
15	5.6.3, 5.6.5, 8.2.2, 8.3.1, 8.3.2, 8.3.3, 9.14.1, 9.14.2, 9.14.3, 9.14.4, 9.14.5, 15.7
16	8.2.3, 8.3.3
17	5.6.2, 6.3, 6.5, 7.3, 7.4, 7.5, 7.7, 8.3.3, 9.6
18	4.3, 5.6.3, 7.2.3, 7.3, 7.7, 15.2, 15.3
19	7.2.3, 7.3
20	4.2.4, 7.2.3, 7.3
21	4.2.4, 7.2.3, 7.3
22	7.2.3, 7.3
23	4.2.4, 4.3, 7.2.3, 7.3, 7.7
24	3.9.4, 4.2.4, 4.3, 4.5, 7.2.2.5, 7.2.3, 7.5
25	3.9.4, 4.2.4, 4.3, 15.2
26	5.2, 5.3, 5.4, 5.11, 10.4.6, 15.3
27	5.2, 5.3, 5.4, 15.3
28	5.2, 5.3

Table 3.1 (Continued)

PDC	SER sections that discuss compliance with principal design criteria
29	5.2, 5.3, 5.5
30	5.2, 5.3
31	5.4, 15.3
32	5.4, 15.3
33	5.4
34	9.5, 9.8, 15.7
35	5.6.2, 5.6.3, 5.6.5, 9.3, 7.3, 7.4, 15.3
36	5.6.2, 5.6.3, 5.6.5, 7.4
37	3.9.6, 5.6.2, 5.6.3, 5.6.5, 7.4
38	5.6.5, 9.7.2, 9.9.2, 9.16, 9.9.4, 9.14.2
39	9.7.2, 9.16, 9.9.2, 9.9.4, 9.14.2
40	3.9.6, 9.7.2, 9.16, 9.9.2, 9.9.4, 9.14.2
41	3.8, 6.2, 8.3.3, 15.6
42	5.11.9
43	6.2
44	6.2
45	3.9.6, 6.2, 9.3, 9.5
46	6.2, 9.3
47	6.2, 9.3, 9.5
48	6.2, 9.3, 9.5
49	6.5, 7.3
50	6.5
51	3.9.6, 6.5
52	9.6.2, 9.6.3, 9.15, 10.4.2, 10.4.3, 11.4.1, 11.4.2, 11.4.3, 11.4.4, 15.3 15.5, 15.6, 15.7
53	6.2, 6.5, 9.1, 9.3, 9.6.3, 11.4.1, 11.4.2, 15.5
54	9.1
55	9.1, 9.3, 11.4.4
56	6.2, 6.5, 10.4.2, 10.4.3, 11.4.4, 15.7
57	4.3, 15.2
58	3.9.4, 4.2.4, 4.3, 7.3, 15.3
59	7.5.2.5, 15.4
60	4.4, 15.4

3A APPENDIX: SUPPLEMENT TO SECTIONS 3.2 AND 3.9 OF THE SAFETY EVALUATION REPORT

3A.1 Summary Statement

During the course of the mechanical design review of Chapters 3, 4 (portion), and 5 of the PSAR, 77* specific questions, comments, or concerns were developed. All of these items are contained in Tables 3A.1, 3A.2, 3A.3, and 3A.4 (arranged chronologically with respect to resolution) along with the status and/or the basis of resolution for each item. Next to each item in the margin is a symbol (R, R*, or 0) denoting the status of the item as follows:

R This item is considered to be resolved or closed on the basis of an agreement formulated between the reviewers and the applicants, which has been docketed. The documentation is given in the "Basis for Resolution" for each item.

R* This item is considered to be resolved on the basis of an agreement reached between the reviewers and the applicants and contingent on the receipt of additional information.

0 This is an open item where negotiation is under way but no agreement has been reached between the reviewers and the applicants.

As shown in the tables, all but 12 items (Items 26, 50, 70, and 72 pertaining to low temperature and Findings, 1, 2, and 4-9 pertaining to elevated temperature) are in the resolved, R category. These 12 items are in the R* category. Of the eight items pertaining to elevated temperature in the R* category, completion of confirmatory programs are required for the resolution of Findings 1, 5, 8, and 9. Only one item (elevated-temperature Finding 3) is considered to be an open item, 0 category.

*Fourteen of these items were reduced to nine findings in the area of the review pertaining to elevated temperature. They are Items 55-68.

The 23 items (1, 5-9, 11-13, 24, 26-30, 33, 35-39, 45, and 52) from the portion of the PSAR review pertaining to low temperature, shown in Table 3A.1, are considered to be resolved and classified as Category R or R* as indicated.

Table 3A.1 Mechanical design review of the CRBR PSAR summary list of resolved open items (Oct. 8, 1982)

[References appear at end of table.]

Item, reference, and basis for resolution	Status
(1) Two items have been omitted from the list of seismic Category I mechanical system components (Table 3.2-2) (a) reactor core and internals (b) reactor shutdown systems (Ref. (a), Item 1, p. 3.2.1-4) BASIS FOR RESOLUTION: The required additions have been made to Table 3.2-2 of the PSAR in Amendment 71. (Ref. (b), Encl. 1, Item 1)	R
(5) In general, the general the fluid system boundaries are not clearly indicated on the piping and instrument drawings. (Ref. (a), Item 3, p. 3.2.2-4) BASIS FOR RESOLUTION: The current set of the piping and instrument drawings does not indicate the change in system boundaries. The earlier concern was based on the review of an out-of-date set of drawings, Amendment 59, December 1980, or older. (Amendment 70, August 1982)	R
(6) In Table 3.2-5, the applicants present the selected ASME Code classifications for the principal system components of seismic Category I. The applicants should explain more completely the footnote, "Classified 2, Designed and Constructed to Class 1 requirements." (Ref. (a), Item 4, p. 3.2.2-4) BASIS FOR RESOLUTION: The applicants have amended in a satisfactory manner the footnote in Table 3.2-5. (Ref. (b), Encl. 2, Item 6)	R

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(7) In PSAR Section 3.2.2.2, the applicants list examples of safety class 2 fluid system components, which include the intermediate heat transport system (IHTS) piping extending from the intermediate heat exchanger (IHX). However, this section of piping has been footnoted in Table 3.2-5 as being designed and constructed to Class 1 requirements. The applicants should clarify the discrepancy. (Ref. (a), Item 5, p. 3.2.2-4) BASIS FOR RESOLUTION: The applicants have satisfactorily amended the footnote to Table 3.2-5 to clarify the apparent discrepancy. (Ref. (b), Encl. 2, Item 7)	R
(8) Table 3.2-2 notes that the containment annulus cooling system and cleanup system shall meet the safety Class 3 requirements, but are not classified as safety Class 3. Table 3.2-5 does not list the containment annulus cooling system but does note that portions of the cleanup system shall meet ASME Code Class 3 and RG 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Features Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." The applicants should clarify this apparent discrepancy. (Ref. (a), Item 6, p. 3.2.2-5) BASIS FOR RESOLUTION: The applicants have satisfactorily amended Table 3.2-5 to clarify the safety classification of this system. (Ref. (b), Encl. 2, Item 8)	R
(9) Similarly, the containment annulus filtration system is listed as Class 3 in Table 3.2-2 and as meeting the requirements of RG 1.52 according to Table 3.2-5. The applicants should clarify the safety classification of this system. (Ref. (a), Item 7, p. 3.2.2-5) BASIS FOR RESOLUTION: Applicants have amended Table 3.2-5 in a satisfactory manner to clarify the classification of this system. (Ref. (b), Encl. 2, Item 9)	R

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(11) The pipe whip analysis assumes that the pipe break occurs with the pipe centered in the restraint. This results in an average initial clearance between pipe and restraint. The maximum possible clearance is required.	R
(Ref. (a), Item 1, p. 3.6.2-5)	
BASIS FOR RESOLUTION: Section 3.6.4.4.1 of the PSAR has been amended to state that the maximum clearance will be used (Amendment 71).	
(Ref. (b), Encl. 1, Item 11)	
(12) Detail the relationship between the time variation of the jet thrust forcing function and pressure, enthalpy, and volume of fluid in the reservoir driving the jet. This is required by Section 3.6.2.III.2.C (3) of the Standard Review Plan.	R
(Ref. (a), Item 2, p. 3.6.2-5)	
(13) Table B-1 of Appendix B lists faulted event F-1, whereas in Section B.1.4.1 that transient has been apparently deleted. The applicants should correct Table B-1 to be consistent with the duty cycle description.	R
(Ref. (a), Item 1, p. 3.9.1-9)	R
BASIS FOR RESOLUTION: The applicants have amended Table B-1 to delete event F-1.	
(Ref. (b), Encl. 2, Item 13)	
(24) Are the effects due to local soil settlements, soil arching, and so forth considered in the analysis of Category I buried piping systems? This question concerns Section 3.7.3.12 of the PSAR.	R
(Ref. (a), Item 7, p. 3.9.2-25)	
BASIS FOR RESOLUTION: References to Chapter 2 have been added to Section 3.7.3.12 of the PSAR with Amendment 71. Consideration of soil settlement and arching phenomena is contained in the referenced sections.	R
(Ref. (b), Encl. 1, Item 24)	

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(26) What are the acceptance criteria for FIV tests in PSAR Section 3.9.1? Will there be numerical limits on allowable deformation and/or vibration? (Ref. (a), Item 9, p. 3.9.2-25) BASIS FOR RESOLUTION: Amendment 69 of the PSAR has changed Section 3.9.1.3.5 to include "Acceptance criteria for the UIS accelerometer measurements will be provided in the FSAR." Thus, the applicants have committed to answer the question, but have not provided the criteria yet. (Amendment 69)	R
(27) Part of the test involves installing accelerometers during pre-operational testing. What is the justification that the instrumentation is sufficient and adequate to correlate these test results with the analysis models and fast flux test facility results? What are the acceptance criteria to ensure similarity of results? These questions concern Section 3.9.1 of the PSAR. (Ref. (a), Item 10, p. 3.9.2-26) BASIS FOR RESOLUTION: The scale model tests have shown that the location most susceptible to flow-induced vibration of the reactor internal is the UIS. Thus, the confirmation of acceptable vibration at this location would indicate that vibration elsewhere in the reactor internals also would be acceptable. Criteria for acceptance have been postponed to be included in the FSAR. (Amendment 69)	R
(28) What is the justification that the parameter ratios between the between the model and the CRBRP are adequate to ensure proper modeling (PSAR Section 3.9.1)? What are the acceptance limits for these ratios? (Ref. (a), Item 11, p. 3.9.2-26) BASIS FOR RESOLUTION: A more detailed discussion of the parameter ratios between the model and the CRBRP has been added to Section 3.9.1.3.1.4.d. (Amendment 69)	R

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(29) On page 3.9-1h (Amendment 30), Table 1 is referenced under <u>Vibration Displacements</u>. Where is this table? (Ref. (a), Item 12, p. 3.9.2.26) BASIS FOR RESOLUTION: Table 1 has been changed to Table 3.9-7 in Amendment 71. (Amendment 71)	R
(30) On page 3.9-1h (Amendment 30) under <u>Density Ratios</u>, what is the basis for conservatism? (Ref. (a), Item 13, p. 3.9.2-26) BASIS FOR RESOLUTION: Amendment 71 explains that the effect is conservative since the onset of unstable vibrations would occur in the model at lower frequencies than in the CRBRP. In addition, the driving energies would be higher in the model. (Amendment 71)	R
(33) The loading combination methodology should be consistent with NUREG-0484, "Methodology for Combining Dynamic Loads." Section 8.1.1 of Appendix 3.8-A, "Seismic Design Criteria," of the PSAR does not reference the transients of Appendix B of the PSAR, nor does it explicitly address the methods used to form loading combinations such as a seismic event plus an operating transient. The PSAR should state whether the absolute sum method, square-root-of-the-sum-of-the-squares method, or other methods will be used to form these combinations. (Ref. (a), Item 1, p. 3.9.2-7) BASIS FOR RESOLUTION: This open item was properly addressed in the applicants' response to docketed question CS-210.14. (Amendment 69, May 1982)	R
(35) The sodium-water heat exchangers are of a unique configuration designed to minimize the probability of tube leakage. The applicants should provide a detailed discussion of tube leakage and features included to deal with this potential problem area. (Ref. (a), Item 3, p. 3.9.3-7) BASIS FOR RESOLUTION: Completed tests, and additional tests to be performed, are discussed in Amendment 70. (Amendment 70)	R

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(36) In Table 4.2.47, what is the criterion for the allowable loads on bearings? What is the basis for contact (Hertz) stress between the balls and races? What is the margin for the thrust bearing? (Ref. (a), Item 1, p. 3.9.4-5) BASIS FOR RESOLUTION: Clarified in meeting handout. (Sept. 8-9, 1982, working meeting)	R
(37) Where is Table 4.2-43a referenced, and what is its meaning? (Ref. (a), Item 2, p. 3.9.4-5) BASIS FOR RESOLUTION: Clarified in meeting handout. (Sept. 8-9, 1982, working meeting)	R
(38) On page 4.2-307 what is meant by "stator checks"? Will there be acceptance criteria for shutdown tests? (Ref. (a), Item 3, p. 3.9.4-5) BASIS FOR RESOLUTION: Attachment 2, Ref. (b) clarifies the meaning of stator checks, and states acceptance criteria will be developed for shutdown tests. (Ref. (b), Encl. 1, Attach. 2)	R
(39) What is the basis for determining that the control rod drive system mechanism latching will not chip or otherwise degrade the lead screw so that continued operating will be impaired? (Ref. (a), Item 4, p. 3.9.4-5) BASIS FOR RESOLUTION: Clarified in September 8-9, 1982 meeting handout and Attachment 2, Ref. (b). (Ref. (b), Encl. 1, Attach. 2)	R
(45) There is an apparent inconsistency in specifying the use of RDT F9-4 and F9-5 rather than RDT F9-4T and F9-5T. The applicants should clarify the discrepancy. (Ref. (a), Item 5, p. 3.9.5-6) BASIS FOR RESOLUTION: A statement has been provided to clarify the apparent discrepancy. (Ref. (b), Encl. 1, Attach. 3)	

Table 3A.1 (Continued)

Item, reference, and basis for resolution	Status
(52) Code Cases 1473-1, 1481, 1489, 1521, 1606, and 1607 should be reviewed by the staff to determine acceptability for use in the CRBRP design. Such reviews should include consideration of the unique features of a sodium design. (Ref. (a), Item 4, p. 5.2.1-8) BASIS FOR RESOLUTION: The required reviews have been performed. All listed code cases, except Code Case 1489, have been approved by the staff. The applicants have reviewed the PSAR and determined that Code Case 1489 was referenced, but never used. These references have been removed from the PSAR. (Ref. (b), Encl. 1, Item 52)	R

References:

(a) C. Kido et al., "Clinch River Breeder Reactor Projects, Mechanical Design Review of Chapters 3, 4, and 5 of the Preliminary Safety Analysis Report," EGG-EA-5881, EG&G Idaho, Inc., Idaho Falls, Idaho, July 2, 1982.

(b) J. R. Longenecker letter to P. S. Check, "Meeting Summary for MEB/CRBRP September 8 and 9, 1982, Meeting," September 21, 1982.

The 32 items (2-4, 10, 14-23, 25, 26, 31, 32, 34, 40-44, 46-51, 53, and 54) from the portion of the PSAR review pertaining to low temperature, shown in Table 3A.2, are considered to be resolved and classified as Category R or R* as indicated:

Table 3A.2 Mechanical design review of the CRBR PSAR summary list of resolved open items (Dec. 27, 1982)

[References appear at end of table.]

Item, reference, and basis for resolution	Status
(2) Justification should be provided for not classifying the liquid metal gas leak detection system as a seismic Category I system. (Ref. (a), Item 2, p. 3.2.1-4) BASIS FOR RESOLUTION: The applicants have committed to classifying the liquid metal/gas leak detection system as seismic Category I. (Ref. (j))	R
(3) In PSAR Section 3.2.2, the nonsafety-related components and piping are not clearly identified, nor are the corresponding industry standards for design, construction, and operation clearly presented. (Ref. (a), Item 1, p. 3.2.2-3) BASIS FOR RESOLUTION: The nonsafety-related systems and standards are listed in Table 32-4. (Ref. (b))	R
(4) Do any mechanical systems and components correspond to Quality Group D requirements as contained in RG 1.26? (Ref. (a), Item 2, p. 3.2.2-3) BASIS FOR RESOLUTION: Components that would be classified as Quality Group D are classified as nonsafety-related equipment in the CRBRP Quality Assurance Program. This question does not require a change of the PSAR.	R
(10) It is not clear why the applicants' definition of safety classification presented in PSAR Section 3.2.2 does not include requirements for postaccident containment heat removal and containment atmosphere cleanup systems. BASIS FOR RESOLUTION: The reviewers were directed to Section 6.2 of the PSAR. In the CRBRP no design-basis requirement for these	R

Item, reference, and basis for resolution	Status
systems has been identified. Additional justification is presented on pages 6.2-6 and 6.2-6a.	
(14) Table B-1 indicates zero (0) frequency for upset events U-1b and U-1c. The applicants should correct this apparent omission.	R
(Ref. (a), Item 2, p. 3.9.1-9)	
BASIS FOR RESOLUTION: The applicants have amended Table B-1 to indicate a total frequency of 180 for the three events, U-1a, U-1b, and U1c.	
(Ref. (c))	
(15) The applicants should clarify Footnote 1 of Table B-1 by specifying which events "balance of trips associated with partial decay heat." What is the meaning of the use of "each" associated with events N-4a, N4b, and N-5?	R
(Ref. (a), Item 3, p. 3.9.1-9)	
BASIS FOR RESOLUTION: The applicants have amended the footnote and frequency columns of Table B-1 to clarify the terminology.	
(Ref. (c))	
(16) NUREG-0718 (Rev. 2), January 1982, states that consideration of anticipated transient without scram (ATWS) conditions shall be included in the applicants' test program to qualify reactor coolant system relief and safety valves. In Appendix B of the PSAR, the applicants have not included the ATWS test conditions.	R
(Ref. (a), Item 4, p. 3.9.1-9)	
BASIS FOR RESOLUTION: The applicants directed the reviewers to Appendix H of the PSAR. The ATWS test is adequately addressed in Appendix H.	
(17) One-third of the computer program verification documents reviewed in the PSAR made reference to documents not readily available. A list of the missing documents was sent to the CRBR Project Office in April 1982. Until those documents have been received and reviewed, the adequacy of computer program verification cannot be fully assessed.	R
(Ref. (a), Item 5, p. 3.9.1-9)	

Item, reference, and basis for resolution	Status
BASIS FOR RESOLUTION: The applicants have provided adequate verification documentation. The review has been completed and the documents comply with pertinent requirements of SRP Section 3.9.1.	
(18) The definition of adequate modal content is poorly stated in 6.2 of Appendix 3.7-A, p. 3.7-A8 of the PSAR. It should be rewritten to correspond to that in Section 3.7.2.2.1, p. 3.7-8 of the PSAR.	R
(Ref. (a), Item 1, p. 3.9.2-24)	
BASIS FOR RESOLUTION: The definition has been rewritten and was incorporated in Amendment 70 to the PSAR.	
(19) Are the hydrodynamic loads associated with partially filled tanks (sodium and water) considered in the CRBR design?	R
(Ref. (a), Item 2, p. 3.9.2-25)	
BASIS FOR RESOLUTION: The modifications to page 3.7-8 of the PSAR provide an acceptable response.	
(Ref. (d))	
(20) A more detailed description of the criteria that justify the equivalent static load method of analysis is required. This affects Section 3.7.2.1.2, and 6.1 of the Appendix 3.7-A of the PSAR.	R
(Ref. (a), Item 3, p. 3.9.2-25)	
BASIS FOR RESOLUTION: The proposed PSAR changes adequately identify the desired criteria.	
(Ref. (e))	
(21) The description of simplified analyses should state the floor spectra are valid only for support points that are either explicitly included in the structural analysis or rigidly attached to such a point. This affects the same areas of the PSAR as Item 20 above.	R
(Ref. (a), Item 4, p. 3.9.2-25)	

Item, reference, and basis for resolution	Status
BASIS FOR RESOLUTION: The proposed PSAR changes are acceptable.	
(Ref. (e))	
(22) Is there a maximum permissible length ratio for adjacent elements on a straight run of pipe? The piping models depicted in Figures 4.1-5 and 4.1-7 of WARD-D-0185 appear to have adjacent elements with large length ratios.	R
(Ref. (a), Item 5, p. 3.9.2-25)	
BASIS FOR RESOLUTION: Pages 5.3-39d and 5.3-39da of the PSAR have been modified in Amendment 71 to adequately answer this concern.	
(23) Calculation of displacements for support points not included in the structural models is not discussed. What are the procedures for this calculation? This question concerns Section 3.7.2.7 of the PSAR.	R
(Ref. (a), Item 6, p. 3.9.2-25)	
BASIS FOR RESOLUTION: The proposed PSAR changes are acceptable.	
(Ref. (e))	
(25) Shouldn't the analysis of Category I piping systems be extended beyond the seismic restraints or anchors at boundaries of sufficient distance to ensure accurate support load calculations for the seismic restraints or anchors? This question concerns Section 3.7.3.13 of the PSAR.	R
(Ref. (a), Item 8, p. 3.9.2-25)	
BASIS FOR RESOLUTION: The applicants have modified the text of Section 3.7.3.13 to indicate proper consideration was given to the Category I boundaries.	
(Ref. (c))	
(26) What are the acceptance criteria for FIV tests in PSAR Section 3.9.1? Will there be numerical limits on allowable deformation and/or vibration?	R
(Ref. (a), Item 9, p. 3.9.2-25)	
BASIS FOR RESOLUTION: In Amendment 69 of the PSAR, the applicants committed to providing the criteria in the FSAR.	

Item, reference, and basis for resolution	Status
This item was subsequently reopened. The PSAR also will be revised to indicate an endurance limit for the FIV tests.	
(Ref. (e))	
(31) The applicants should specifically note differences between the testing requirements of RG 1.20 and CRBR testing. The effects of high temperatures on instrumentation should be included.	R
(Ref. (a), Item 14, p. 3.9.2-26)	
BASIS FOR RESOLUTION: The applicants adequately modified the PSAR in Amendment 69.	
(32) The description of the piping startup test program found in Chapter 14 of the PSAR is inadequate. See Section V.1 above for a list of the elements that should be included in an adequate description.	R
(Ref. (a), Item 15, p. 3.9.2-26)	
BASIS FOR RESOLUTION: The applicants will notify the PSAR to indicate the essence of a test program and have committed to supply the detailed test program in the FSAR.	
(Ref. (d))	
(34) In Sections 5.3.2.3.4 through 5.3.3.1.2 of the PSAR, the applicants have not committed to develop and use a snubber operability assurance program as required by Section II-3.b of SRP Section 3.9.3.	R
(Ref. (a), Item 2, p, 3.9.3-7)	
BASIS FOR RESOLUTION: The applicants have modified the PSAR to include a snubber operability assurance program.	
(Ref. (e))	
(40) No tests to determine control rod drive system capabilities to overcome a stuck rod have been included.	R
(Ref. (a), Item 5, p. 3.9.4-5)	

Table 3A.2 (Continued)

Item, reference, and basis for resolution	Status
BASIS FOR RESOLUTION: The applicants have placed a letter with a table on the docket that covers the SCRDM. The PCRDM had been previously addressed. (Ref. (f) and Ref. (g), Encl. 1, Attach. 2)	
(41) The removable radial shielding (RRS) is in a preliminary phase of design stress analysis, taking into account that the effects of environmental conditions have not been completed. The applicants have not provided sufficient information for the staff to complete its evaluation of the RRS component. (Ref. (a), Item 1, p. 3.9.5-5) BASIS FOR RESOLUTION: The applicants have provided a PSAR revision that contains sufficient information on the RRS. (Ref. (b))	R
(42) The applicants should define the "mechanical discrimination features" that are designed into the lower internals components to ensure proper support and alignment and to accommodate thermal expansion. (Ref. (a), Item 2, p. 3.9.5-5) BASIS FOR RESOLUTION: The applicants modified the text of the PSAR to define "mechanical discrimination features" and to state that these features were designed to allow thermal expansion. (Ref. (c))	R
(43) The applicants should specify the criteria for change out of nonpermanent reactor internal components, such as the lower inlet modules (LIMs). Present information is insufficient to conclude that structural interference will not occur during LIM withdrawal. (Ref. (a), Item 3, p. 3.9.5-5)	R
BASIS FOR RESOLUTION: Upon receipt of verbal explanation of the text on page 4.2-120 and upon rereading that page, the staff believes that the PSAR adequately explains this phenomenon. Therefore, this item is resolved.	R

Table 3A.2 (Continued)

Item, reference, and basis for resolution	Status
(44) In Table 5-1 of Appendix G of the PSAR, the applicants have not provided a program of testing and inspection of the reactor internals structures. (Ref. (a), Item 4, p. 3.9.5-6) BASIS FOR RESOLUTION: The applicants adequately modified the PSAR in Amendment 69.	R
(46) PSAR Section 4.2.2.4.2 states that special project structural design rules were used to determine adequacy of the upper internals structure. The applicants should provide a description of, and basis for, the use of these rules. (Ref. (a), Item 6, p. 3.9.5-6) BASIS FOR RESOLUTION: The proposed PSAR revision provides an adequate description of, and basis for, the special rules. (Ref. (g))	R
(47) The applicants should specify the methods of simplified and rigorous inelastic analysis mentioned in PSAR Section 4.2.2.4.2.6. (Ref. (a), Item 7, p. 3.9.5-6)	R
BASIS FOR RESOLUTION: The proposed PSAR revision adequately specifies the methods of inelastic analysis. (Ref. (g))	R
(48) The applicants have not provided sufficient details of the inservice testing program for pumps and valves to allow the staff to complete its evaluation at this time. (a) In those instances where the CRBR inspection and testing requirements are different from ASME Code, Section XI, the applicants should identify those differences and provide justification for the variance. (b) In those instances where requirements have been specified that are not in ASME Code, Section XI, those requirements should be clearly identified. (Ref. (a), Item 1, p. 3.9.6-7)	R

Item, reference, and basis for resolution	Status
BASIS FOR RESOLUTION: The applicants have provided a general description of the inservice testing program for pumps and valves and have committed to providing the detailed program in the FSAR. (Ref. (h))	
(49) Provide an amended version of Table 3.1-1, "Components Which Comprise the Reactor Coolant Boundary," that includes that following for each item in the current table: (a) ASME Code class (b) ASME Code edition (c) ASME Code addenda (Ref. (a), Item 1, p. 5.2.1-8) BASIS FOR RESOLUTION: The above information has been incorporated into Table 3.2-5, which is acceptable. (Ref. (i))	R
(50) Does the reactor coolant boundary design, which was made to code editions and code cases at least 5 years old, provide a level of safety comparable to a similar design made to current code editions and code cases? (Ref. (a), Item 2, p. 5.2.1-8) BASIS FOR RESOLUTION: The applicants have provided an adequate response.	R
(51) A table identifying all ASME and ANSI code cases applied to Section III, Division 1 and 2 components should be included in the PSAR. (Ref. (a), Item 3, p. 5.2.1-8) BASIS FOR RESOLUTION: The modified Table 3.2-5 adequately resolves this question. (Ref. (i))	R
(53) Does that current design of the elevated-temperature portion of the core support structure to Code Case 1592-7 (as supplemented by RDT standards) achieve a level of safety comparable to a design done to the current Code Case N-201, "Class CS Components in Elevated Temperature Service, Section III, Division 1"? (Ref. (a), Item 5, p. 5.2.1-8)	R

Item, reference, and basis for resolution	Status

BASIS FOR RESOLUTION: The applicants responded to the above question in the docketed report, "CRBRP Special Stress and Criteria Consideration," ES-LPD-82-009. The reviewers concur with the applicants' assessment therein that the Code Case 1592-7 provides an equivalent design analysis compared with the current requirements of the elevated-temperature code case for core support structures, N-201. Thus, the above question is answered. However, two exceptions to the requirements of Code Case 1592 were taken. The first one (a reduced creep-damage rule for compressive hold times) will be considered in the resolution of elevated-temperature Finding No. 6. The second exception substitutes progressive liquid penetrant examination for full radiography examination of the weld joining the core support structure forging to the core support plate. The Section III, Class 1, rules (Case 1592-7) would require full radiography, whereas the Section III, Class CS, rules (Subsection NG) permit progressive penetrant testing provided the design factors given in Table NG-3352-1 are applied in the determination of allowable stress intensities and fatigue life. The applicants have stated that these factors have been used in the evaluation of this joint. Therefore, the reviewers concur, in this situation (840°F for 1 hour), that this exception can be safely used.

Since the Code Case 1592-7 rules are supplemented with the RDT standards, this aspect of the question is still under consideration as part of the confirmatory program for elevated-temperature Finding No. 3. The elevated-temperature issues are addressed in Section 3.9.9 of this report.

(54) Section 5.1.2 of the PSAR states that part or all of the auxiliary liquid metal system and the cover gas system is included in the reactor coolant boundary, yet components of neither system are mentioned in Table 3.1-1, "Components Which Comprise the Reactor Coolant Boundary." Clarify this discrepancy. If components of these systems are not to be added, justify this action. R

(Item 6, p. 5.2.109)

BASIS FOR RESOLUTION: These systems have been added to Table 3.1-1.

(Ref. (i))

Table 3A.2 (Continued)

References:

(a) C. Kido et al., "Clinch River Breeder Reactor Projects, Mechanical Design Review of Chapters 3, 4, and 5 of the Preliminary Safety Analysis Report," EGG-EA-5881, EG&G Idaho, Inc., Idaho Falls, Idaho, July 2, 1982.

(b) J. R. Longenecker letter to P. S. Check, "Additional Information from November 22-24, 1982, MEB/CRBRP Meeting," HQ:S:82:157, Dec. 22, 1982.

(c) J. R. Longenecker letter to P. S. Check, "Amendment No. 73 to the Preliminary Safety Analysis Report (PSAR) for Clinch River Breeder Reactor Plant (CRBRP)," HQ:S:82:125, Nov. 30, 1982.

(d) J. R. Longenecker letter to P. S. Check, "Additional Information Resulting From the September 8-9, 1982, MEB/CRBRP Meeting," HQ:S:82:109, Oct. 20, 1982.

(e) J. R. Longenecker letter to P. S. Check, "Meeting Summary: November 22-24, MEB/CRBRP Meeting," HQ:S:82:143, Dec. 14, 1982.

(f) J. R. Longenecker letter to P. S. Check, "Meeting Summary for Reactor Mechanical Shutdown Systems Working Meeting, October 14, 1982," HQ:SD:82:107, Oct. 15, 1982.

(g) J. R. Longenecker letter to P. S. Check, "Meeting Summary for MEB/CRBRP, September 8 and 9, 1982, Meeting," HQ:S:82:093, Sept. 21, 1982.

(h) J. R. Longenecker letter to P. S. Check, "Additional Information on In-Service Testing," HQ:S:82:104, Oct. 13, 1982.

(i) J. R. Longenecker letter to P. S. Check, "Additional Information Resulting from the September 8-9, 1982, MEB/CRBRP Meeting," HQ:S:82:128, Nov. 23, 1982.

(j) J. R. Longenecker letter to P. S. Check, "Meeting Summary for the SER Open Item Meeting, December 21, 1982," HQ:S:82:156, Dec. 30, 1982.

The nine items (69-77) shown in Table 3A.3 are the result of discussions at various meetings between the applicants, Advisory Committee on Reactor Safeguards (ACRS) working groups, and the NRC staff.

Table 3A.3 Mechanical design review of the CRBR PSAR summary list and status of additional open items (Dec. 27, 1982)

[References appear at end of table.]

Item, reference, and basis for resolution	Status
(69) The applicants should review the current version of BTP MEB 3-1 (Rev. 1) to ensure that other documents used for specifying pipe break locations provide an equivalent level of conservatism. (Ref. (a), p. 3.6.2-5) BASIS FOR RESOLUTION: The applicants have committed to delete all references to the O'Leary letter and substitute BTP MEB 3-1, Revision 1. The applicants have reviewed the PSAR and will ensure an equivalent level of conservatism. (Ref. (b))	R
(70) The information on load combinations and corresponding limits in PSAR Sections 3.9.2/3.7A is incomplete. Coverage equivalent to that in current safety analysis reports (e.g., the Byron PSAR) should be provided. Emphasis should be placed on (a) general support buckling, (b) bolt criteria in faulted conditions, Section NF support bolts, and so forth, and (c) amplification of load combinations and limits. STATUS: Additional information is to be provided later.	R*
(71) The applicants state that the design and analysis of the reactor internals will be in accordance with Subsection NG of the ASME Code, Section III. The applicants should clarify whether or not this subsection is applied to construction. (Ref. (a), p. 3.9.5-4) BASIS FOR RESOLUTION: The proposed PSAR revision adequately resolves this item. (Ref. (b))	R
(72) The applicants should provide additional justification on the integrity of the core support structure--support cone weld. STATUS: The applicants will supply additional information. The reviewers will address this question with the elevated-temperature Issue 7.	R*

Table 3A.3 (Continued)

Item, reference, and basis for resolution	Status
(73) The applicants should address an optimization of the number of snubbers so that any effects of snubber failure will be minimized. BASIS FOR RESOLUTION: The applicants presented their procedures for snubber optimization at the November 22-24, 1982 Waltz Mill meeting. The presentation was acceptable and so this item was resolved. (Ref. (c))	R
(74) The applicants are to provide data on the selection of the plant duty cycles. BASIS FOR RESOLUTION: The applicants presented data on duty cycles and selection and combination procedures at the November 22-24, 1982 Waltz Mill meeting. The presentation was acceptable and so this item was resolved. (Ref. (c))	R
(75) The applicants are to provide data on the seismic safety margins. BASIS FOR RESOLUTION: The applicants presented data on seismic margins at the December 1, 1982 meeting of the ACRS working group. This item was resolved at that meeting.	R
(76) The applicants are to provide a discussion on the design of the ex-vessel storage tank (EVST). BASIS FOR RESOLUTION: The applicants led a discussion on the EVST at the November 22-24, 1982 Waltz Mill meeting. All questions were answered and so this item was resolved. (Ref. (c))	R
(77) The applicants are to discuss the design of elevated-temperature pipe clamps and their effect on the piping. BASIS FOR RESOLUTION: The applicants presented their stress analysis of the pipe clamps and the effects of the clamps on elevated-temperature, thin-wall, large-diameter piping. The analysis appeared to be very thorough and accurately represented the loading conditions. With the acceptance of that presentation and supporting data and calculations, this item was resolved. (Ref. (c))	R

Table 3A.3 (Continued)

References:

(a) C. Kido et al., "Clinch River Breeder Reactor Project, Mechanical Design Review of Chapters 3, 4, and 5 of the Preliminary Safety Analysis Report," EGG-EA-5881, EG&G Idaho, Inc., Idaho Falls, Idaho, July 2, 1982.

(b) J. R. Longenecker letter to P. S. Check, "Meeting Summary: November 22-24, MEB/CRBRP Meeting," HQ:S:82:143, Dec. 14, 1982.

(c) J. R. Longenecker letter to P. S. Check, "Meeting Summary: November 22-24, MEB/CRBRP Meeting," HQ:S:82:143, Dec. 14, 1982.

The nine major elevated-temperature findings, the SER section in which each finding is discussed, and the status of each finding appear in Table 3A.4.

Table 3A.4 CRBR mechanical design review summary list of elevated-temperature findings and status (Dec. 23, 1982)

Finding	SER section	Status
Finding No. 1: Weldment Safety Evaluation (defined confirmatory program required)	3.9.9.2.1	R*
Finding No. 2: Elevated-Temperature Seismic Effects (resolved based on resolution of Finding No. 3)	3.9.9.2.2	R*
Finding No. 3: Design Analysis Methods, Codes, and Standards (open--subject to NRC review of RDT F9-4T and F9-5T design methods and criteria and resolution of findings by applicants)	3.9.9.2.3	O
Finding No. 4: Elastic Followup in Elevated-Temperature Piping (resolved based on applicants' commitment to perform additional defined analyses)	3.9.2.2.4	R*
Finding No. 5: Notch Weakening (defined confirmatory program required)	3.9.9.2.5	R*
Finding No. 6: Creep-Fatigue Evaluation (resolved subject to receipt of docketed response from applicants)	3.9.9.2.6	R*
Finding No. 7: Plastic Strain Concentration Factors (resolved based on applicants' commitment to perform additional analyses or tests)	3.9.9.2.7	R*
Finding No. 8: Intermediate Heat Transport System (IHTS) Transition Weld (resolved based on applicants' commitment to evaluate IHTS transition welds using methods and criteria of the Finding No. 1 confirmatory program)	3.9.9.2.8	R*
Finding No. 9: Steam Generator (defined confirmatory program required)	3.9.9.2.9	R*

4 REACTOR

4.1 Summary Description

The CRBRP will use a mixed (Pu-U)-oxide-fueled, sodium-cooled fast reactor having a total thermal output of 975 MWt. The reactor system will consist of the removable core components (fuel, blanket, control, and shield assemblies) and associated structures located inside the reactor vessel. These structures will consist of the core support structure, lower flow modules, core barrel, upper internals structure, core former rings, radial shield, thermal baffles, and a vortex suppressor plate. The core support structure will consist of the core support plate, the lower inlet modules, and the support cone, which form a boundary inside the vessel between the high-pressure inlet sodium and the low-pressure outlet sodium. The fuel, control blanket, and removable shield assemblies will be supported by the core support plate, which also will support a fixed radial shield. Each of these reactor assemblies will have two load pad areas that match the elevation of the core former rings. The rings will be supported by the core barrel, which is welded to the core support plate.

The upper internals structure, located above the core, will be supported from the intermediate rotating plug of the vessel closure and keyed to the upper core former ring permitting vertical motion while restraining lateral and rotational motion. The structure laterally will stabilize primary and secondary control rod shroud tubes. In case of a loss of hydraulic balance on any core assemblies, the upper internals structure will act as a secondary holddown device. The four support columns of the upper internals structure will have jacks for lifting the upper internals structure with its keys clear of the core former ring and reactor assemblies for refueling.

A vortex suppressor plate will be provided just below the sodium pool surface to minimize gas entrainment in the sodium leaving the outlet plenum. Fuel transfer and contingency storage positions will be provided in the annulus formed between the core barrel and the reactor vessel thermal liner.

The active length of the core will be 36 in. and the equivalent diameter will be 79.5 in. The fuel region will consist of a single enrichment zone with a total fissile plutonium loading of ~1,500 kg. The reactor control systems will include nine primary and six secondary control rods. Either system will be capable of shutting down the reactor from full power to hot standby conditions.

4.2 Fuel System Design

The staff and its consultant (Los Alamos National Laboratory, LANL) reviewed the design and performance of the fuel, blanket, and absorber rod for the CRBRP. The mechanical design of the control assemblies was also evaluated.

Area and Scope of Review

The area of review was defined in the statement of work from the CRBR Program Office to Los Alamos National Laboratory (LANL) (letter, Dec. 17, 1981) as follows:

> Perform a review of the CRBR fuel design described in Chapter 4 and related appendices of the CRBR Preliminary Safety Analysis Report (PSAR). Also take into consideration previous NRC staff evaluations of this design. Based on this review provide a draft technical evaluation report which indicates the basis for approval of the proposed design criteria, any deficiencies in the proposed design, recommendations for additional design criteria or features, and the potential for achieving an effective and reliable design based on proposed and recommended design criteria or features.

The review was to include identification of necessary changes to the Standard Review Plan (SRP) (NUREG-0800) and general design criteria (10 CFR 50, Appendix A) for light-water reactors (LWRs).

In accordance with the instructions, SRP Section 4.2 was followed closely.

Approach to Evaluation

The approach to evaluation was delineated as follows:

(1) General guides including the general design criteria, applicable portions of the SRP, regulatory guides, and branch technical positions were reviewed.

(2) Specific guides were distilled from the general guides for use as yardsticks in this specific review.

(3) The acceptance criteria proposed by the applicants were reviewed for conformance with the specific guides.

(4) The applicants' predictions for the fuel system performance were reviewed for conformance with the acceptance criteria.

In practice, the last two items were reviewed primarily from the viewpoint of whether or not the applicants' testing and development plans could reasonably be expected to yield reliable acceptance criteria and performance models in time for the review of the Final Safety Analysis Report (FSAR) and the application for an operating license, and whether or not the applicants' fuel design could be expected to succeed, or there were reasonable fallback positions.

Figure 4.1 displays the anticipated flow of information and the review process.

Application of the Standard Review Plan

SRP Section 4.2 is almost wholly devoted to aspects of fuel rod performance. Relatively detailed and specific guidance is provided that is oriented toward LWRs. It was attempted to observe the spirit of the SRP where the guidance was obviously inappropriate for a liquid metal fast breeder reactor (LMFBR). The SRP was revised for Section 4.2 in accordance with this principle.

The principal changes that were made were as follows:

(1) All references to hydriding of zircaloy were deleted, as were all references to the criteria associated with the emergency core cooling system (ECCS), and where appropriate replaced with corrosion criteria relevant to the LMFBR.

(2) All references to departure from nucleate boiling (DNB) were deleted, and criteria relevant to the LMFBR were substituted.

(3) The guidance on pellet cladding interaction (PCI) and reactivity insertion accidents (RIAs) were combined, and an enthalpy criterion analogous to the LWR criterion for cladding failure (NUREG-0800 was proposed for LMFBR fuel rods.

(4) Coolable geometry criteria were revised as appropriate for LMFBRs. An enthalpy criterion analogous to the LWR 280 cal/gm criterion (NUREG-0800) was proposed for LMFBR fuel rods under RIA conditions based on LMFBR data and analysis methods.

Application of the Principal Design Criteria

Only one PDC (PDC 8, analogous to LWR GDC 10) is directly applicable to Section 4.2. This criterion requires the establishment of "specified acceptable fuel design limits" (SAFDLs).

PDC 8 is:

> CRITERION 8 - Reactor Design. The reactor and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

4.2.1 Fuel and Blanket Rods

This section contains a description of the fuel and blanket system, a presentation of design limits and guidelines, and the evaluation of fuel and blanket rod performance and performance models.

4.2.1.1 Description of Fuel and Blanket System

CRBR fuel will be similar to LWR fuel. The fuel will consist of cylindrical, sintered, ceramic pellets encased in tubes, and will be called an element, a rod, or a pin. Space for fission gas generated during operation and released from the fuel will be provided by a plenum formed by an extension of the cladding above the fuel. A holddown spring in the plenum will ensure proper positioning of the fuel in the cladding. The rods will be grouped into bundles of 217 rods each; each bundle will be enclosed in a closed hexagonal tube (or duct as it is usually called) that internally will measure 4.335 in. across the flats with a 0.12-in. wall thickness. The bundle with its duct and associated upper and lower hardware is called an assembly.

There are three major differences between CRBR and LWR fuel.

(1) CRBR fuel will consist of mixed (uranium-plutonium) oxide, 33% plutonium (Pu) by weight, as compared with uranium dioxide (UO_2) slightly enriched in U-235 in LWRs.

(2) CRBR cladding will be 20% cold-worked type 316 stainless steel as compared with Zircaloy in LWRs.

(3) CRBR will be designed to "breed" nuclear fuel; this will be accomplished by exposing fertile material (depleted UO_2) at either end of the fuel column (axial blankets) and in separate assemblies both on the periphery of the core (radial blanket) and in the core (internal blanket).

Other differences from LWR fuel include the following:

(1) CRBR rods will be arranged within the bundle in a triangular array rather than in the square array for LWRs.

(2) Spacing between rods and between rods and duct wall in CRBR will be maintained by a 0.056-in.-diameter wire that will be wrapped in a helical spiral on each rod and will have a pitch of 11.9 in.

(3) Relative to the fuel volume, the plenum volume to accommodate released fission gas will be much larger for CRBR rods than for LWR rods, being approximately 20% greater than the volume of fuel (active plus blanket).

(4) CRBR rods will be smaller than LWR rods; the former will have an outer diameter of 0.23 in. as compared with about 0.5 in. for the latter.

(5) The CRBR fuel column will be much shorter; it will be 3 ft long as compared with about 12 ft for those in most LWRs.

(6) CRBR smear density will be about 85%; the smear density of LWR fuel is about 90 to 91%. Smear density is the fraction of total volume inside the cladding that is occupied by fuel at 100% theoretical density.

The overall rod length in CRBR, including axial blankets, plenum, and end hardware, will be 114 in.

The design of radial and internal blanket rods is identical, and is similar to that of fuel rods. The principal differences are that they will be larger in diameter (0.5 in. compared with 0.23 in.) and the mixed-oxide fuel column will be replaced with depleted UO_2. Other parameters for blanket rods are shown in Table 4.1. Blanket rods will be grouped into bundles of 61 rods and will be encased in the same hexagonal duct as will be used for fuel rod bundles. Operating characteristics for blanket rods are different from fuel rods in that blanket rods initially generate very low power that increases monotonically with irradiation as the plutonium content builds up. Fuel rods, on the other hand, start irradiation at the maximum power they will experience, then the power slowly decreases as the fissile (plutonium) content "burns" out and as blanket rods generate an ever greater portion of the heat load.

The CRBR core will consist of 156 fuel assemblies, 76 internal blanket assemblies, 126 radial blanket assemblies, and 6 assemblies that may be either fuel or internal blanket assemblies. Fifteen positions will be reserved for control assemblies, nine for the primary control system, and six for the secondary control system. The overall core dimensions will be 79.5 in. diameter (equivalent) by 36 in. active length. The axial blanket will extend 14 in. above and below the active core.

The CRBR fuel system will be similar to that used in the FFTF, just now beginning operation. A comparison between the two systems is provided in Table 4.2. CRBR fuel design is based on the FFTF fuel design and on experience gained in the LMFBR development program. FFTF operating experience as well as test results from the FFTF will be the basis for the final evaluation of CRBR fuel performance.

4.2.1.2 Design Bases and Limits

This section is devoted to a review of the applicants' design bases and to an evaluation of their design limits and acceptance criteria.

4.2.1.2.1 Design Bases

The primary functions of a fuel assembly (as proposed in the PSAR) are as follows:

(1) to provide, protect, and position the nuclear fuel of the fast breeder reactor core to produce heat for the reactor heat transport system

(2) to provide neutrons for breeding plutonium in the core and surrounding blankets

The primary functions of the proposed radial and internal blanket assemblies are as follows:

(1) to provide, protect, and position the fertile material around and in the core for conversion to plutonium

(2) to produce heat for the primary heat transport system

In addition to these functional requirements, the CRBR PSAR lists operational and design requirements (PSAR Sections 4.2.1.1.2.1 and 4.2.1.1.2.2). Because of the preliminary nature of the PSAR analyses, a design life of 328 full-power days (80,000 MWd per ton peak burnup), corresponding to the first core conditions, was used as a design basis for fuel and internal blanket assemblies. The equilibrium goal residence time for fuel is two cycles (550 full-power days or 110,000 MWd per ton peak burnup). Equilibrium design life was not considered in the PSAR as far as fuel performance is concerned, but may be addressed in the FSAR. The equilibrium goal was not considered in this SER. The initial radial blanket assemblies were expected to have a residence time of four to five cycles, or up to 1,100 full-power days.

The fuel (and blanket) rod design is required to accommodate a combination of loading mechanisms that may be encountered over the normal, anticipated (upset), unlikely (emergency), and extremely unlikely (faulted) operational conditions of the CRBR "Design Duty Cycle" (described in PSAR Appendix B). These loading mechanisms include the following:

(1) Fuel and cladding loading caused by differential growth between components. Sources of differential growth are fuel and cladding irradiation-induced swelling and creep, thermal expansion and thermal creep, and elastic and plastic strain.

(2) Cladding pressure loading caused by gas released from the fuel during burnup, with additional release and fuel expansion during overpower transients.

(3) Secondary stresses generated by radial, axial, and circumferential thermal and flux gradients in the cladding. These stresses, which occur during steady-state operation, are caused primarily by variations in the local swelling rate, but are relaxed by fuel rod bowing and irradiation creep. During transients, thermal stresses caused by unequal heating or cooling rates may be generated. At shutdown, partial load and refueling conditions, residual stresses associated with these relaxation mechanisms will be present.

(4) Interaction forces generated between the cladding and the wire wrap by differential axial and radial growth of the two components.

(5) Cyclic cladding and wire wrap stresses produced by fuel and blanket flow-induced vibration. Low-cycle fatigue stresses may also be generated during certain transients, seismic events, and nonoperational shock loadings.

(6) Accident loadings such as fission gas jet impingement or sodium-fuel interaction as described and analyzed in PSAR Section 15.4.

As noted in the PSAR, consideration of the above loading mechanisms was intended as a minimum requirement for evaluation of the design adequacy of the fuel and blanket assembly rods. It was acknowledged that additional loading mechanisms, while unexpected, could be discovered during future irradiation testing.

The staff concludes that the foregoing list of loading mechanisms and requirements is as complete as possible at this stage, is consistent with the Standard Review Plan, and provides a sound basis for design of the fuel system.

4.2.1.2.2 Design Limits

For each operational condition of the "Design Duty Cycle," fuel damage severity limits were established. For normal operation and anticipated events the objective was for no reduction of effective lifetime below the design values (that is, no fuel failures before reaching design burnup). For "unlikely events" (emergency conditions) the design objective was for generally no more than a reduction in burnup capability with at most a "small fraction" of fuel failures. Maintenance of coolable geometry was the requirement for "extremely unlikely" events (faulted conditions). The postulated duty cycle events, number of each event over the 30-year lifetime of the plant, and fuel damage severity limits associated with each event are provided in Tables 4.3, 4.4, 4.5, and 4.6. These "damage severity limits" are patterned after and are consistent with applicable portions of Section III of the ASME Code and with Reactor Development and Technology (RDT) standards (USAEC, 1969).

The previously described damage severity limits are statements of desired objectives. To ensure the attainment of those objectives, a set of "design limits" was defined in specific quantitative terms. The design limits for the fuel and blanket at steady state were as follows:

(1) For fuel and internal blanket rod cladding, the primary and secondary thermal creep strain in the circumferential (hoop) direction was proposed to be less than 0.2% for a 2-to-1 stress biaxiality ratio. For radial blanket rod cladding, the maximum allowable steady-state ductility was set at 0.1%. The basis for distinguishing between internal and radial blanket cladding is the effect of the lower creep rate expected for the latter on the failure strain, as inferred from Dounreay fast reactor (DFR) data (Bagley et al., Vol. 1, pp. 87-100, 1974).

(2) The peak fuel or blanket power was proposed to be less than the minimum value for incipient fuel or blanket melting with a 15% overpower margin.

(3) No cladding plastic strain was to be allowed during normal operation.

(4) The cladding primary equivalent stress requirement was that it must remain below the proportional elastic limit, or alternatively, 90% of the yield strength.

(5) Flow channel closure caused by fuel-rod bowing and differential radial growth of the fuel rods, wire wrap, and duct was proposed to be limited by the flow area reduction used to derive the thermal hot spot factors of PSAR Section 4.4 (Thermal and Hydraulic Design); that is, the effect at 3σ confidence of the statistic combination of fuel rod bowing, nominal geometry variations and differential radial growth variations along the length of the subchannel were not to produce a variation in the hot subchannel coolant temperature rise at the outlet greater than 2.8% of the nominal value.

(6) The fuel designers believed that irradiation-induced creep and swelling are super plastic phenomena and are therefore not ductility limited. Hence, no limits were provided for these phenomena.

For fuel and blanket transient design limits, it was proposed in the PSAR that the total thermal creep and plastic strain accumulated during steady-state operation, all anticipated events, and the worst minor incident (unlikely event), would be less than 0.3% at a 2-to-1 stress biaxiality ratio. Both thermal creep and plastic strain accumulation would be calculated using the ductility limited strain (DLS) model. It was also proposed that the total accumulated damage fraction calculated using the cumulative damage function (CDF) model (Jacobs, 1976) for steady-state operation, all anticipated events, and the worst unlikely event should not exceed 1.0.

Fatigue was recognized to be a possible damaging mechanism (PSAR, pp. 4.2-14 and 4.2-63). However, only the form of a limit was proposed. The fraction of number of stress cycles referred to the fatigue endurance limit was to be added to the CDF fraction with the whole not to exceed 1.0. Based on preliminary investigation, the applicants concluded that fatigue caused by high frequency flow vibrations should be acceptable because of the low amplitude of vibration (PSAR, p. 4.2-65), and proposed to address treatment of this damage mechanism in the PSAR.

No temperature limit for the CRBR cladding has been developed for either steady-state or these transient conditions.

For extremely unlikely conditions, the design requirement was to maintain coolable geometry. The corresponding design limit was that clad melting should be precluded.*

In summary, the fundamental fuel and blanket rod design requirement presented for the CRBR was that cladding integrity be ensured during steady-state operation combined with a specified number of anticipated upset events as well as a single unlikely event. In other words, the ultimate design lifetime is that point in time at which an unlikely event would just cause failure. The determination of this point must reflect all aspects of the prior mechanical and environmental histories as well as all analytical and operational uncertainties.

The staff's evaluation of the adequacy of the design limits is given in Section 4.2.1.2.4.

4.2.1.2.3 Application of Design Limits

In its treatment of the combined effects of normal, anticipated, and unlikely events on the fuel rod cladding performance, the CRBR PSAR separates the anticipated and unlikely events of the design duty cycle into three categories: decreasing severity transients, increasing severity cooling transients, and reactivity insertion transients. Because significantly higher loading can be accommodated during reduced temperature operation, as a preliminary assessment, no additional cladding damage was assigned to decreasing severity transients. Moreover, instead of fully characterizing and evaluating each damaging increasing severity transient according to its effect on cladding performance, the approach used for the PSAR was to "umbrella" these events. That is, based upon preliminary cladding temperature calculations for selected transients that were expected to produce the highest cladding temperatures, transient envelopes were selected and applied in place of each particular transient (that is, the frequency is preserved) for the undercooling and reactivity insertion events. Importantly, the maximum cladding temperature for these envelopes was based upon the expected performance of the plant protection system (PPS); that is,

*No coolant boiling is listed as a design requirement in PSAR Section 4.2, but unlike the other listed design requirements, is treated like a design guideline rather than as a design limit. PSAR Section 15.1 lists avoidance of cladding melting as the means of ensuring coolable geometry with no boiling as a limit to preclude cladding melting. However, both are considered to be design guidelines. See Section 4.2.1.2.3, of this SER, for an explanation of the distinction between limit and guideline. The staff has chosen to consider avoidance of cladding melting as a design limit.

upon the preliminary PPS trip subsystems, their settings, and anticipated (design) response times.

As a further aid to design evaluation, a system of "design guidelines" was constructed. These design guidelines were temperatures to be defined conservatively to ensure fuel performance within the design limits. To avoid confusion, definitions for "design limit" and "design guideline" are presented now. As used in reactor licensing, "design limit" is a parameter value that should not be exceeded during the design lifetime of the component. It is not necessarily physically impossible to exceed the design limit; the limit, instead, may be a calculated value, which might require some remedial action if exceeded. The term "design guideline," as defined by the applicants, is "a policy, rule, or practice to be followed as a matter of course when a component is being designed, or when its potential operating conditions are being considered; this policy, rule, or practice is considered violable so long as the restrictions set forth by any relevant design limit are upheld."

The design guidelines were established and used as follows.

Calculations were performed with both the CDF and the DLS models (both described in Section 4.2.1.3.3 and 4.2.1.3.4, respectively) to predict performance for the duty cycle using the more severe umbrella events. A generalized approach using "transient limit curves" (TLCs) was a prominent part of the process. A TLC, generated using the CDF model, is a graphic representation of a fuel rod's propensity for failure during a given category of unlikely events (Jacobs, p. 10, 1976). A TLC expresses the time dependence of the value of a variable relevant to the event that corresponds to cladding failure. Failure is presumed if the TLD value of the variable is exceeded by the actual in-transient value of the variable. An example is the peak cladding temperature to cause failure during a transient, plotted versus time in reactor.

Using this information, design guideline peak cladding and fuel temperatures were defined for each damage severity limit to serve as assuredly safe indicators for screening use. To evaluate a particular event, a thermal calculation was performed to determine the peak cladding and fuel temperatures during the event. The peak temperatures then were compared with the design guidelines for that damage severity limit. If the peak cladding and fuel temperatures both were less than the design guidelines, the damage severity limit was presumed to be met without further analysis. If either peak temperature exceeded its design guideline, then detailed analyses of the event were performed with both the CDF and the DLS models to determine whether the design limits were met.

The design guidelines thus defined (PSAR, Vol. 13, p. 15.1-53) were as follows:

(1) anticipated events	-	cladding, 1500°F
	-	fuel, < solidus
(2) unlikely events	-	cladding, 1600°F
	-	fuel, < Solidus
(3) extremely unlikely events	-	no boiling

The no boiling guideline for extremely unlikely events was stipulated on the basis that departure from nucleate boiling was not a signficant risk to heat transfer continuity in the low-pressure, high-conductivity sodium coolant system. Significant disruption of heat transfer would be very unlikely before the onset of boiling. Hence, cladding melting would be precluded and coolable geometry would be preserved by a no-boiling guideline (PSAR Section 15.1.2.2).

The design limits and guidelines were intended to preclude loss of cladding integrity. That is, the limits and guidelines were meant to provide a safe margin against cladding failure. As an example, the selection of a value of 0.2% thermal creep strain to cover steady-state operation was based primarily on DFR data. These data were interpreted to show that a creep strain of 0.7% would be obtained on 20% cold-worked stainless steel irradiated at approximately 1,000°F to a total fluence of 3 x 10^{22} n/cm^2. For the radial (not internal) blanket cladding, the maximum allowable steady-state ductility limited strain was set at 0.1%. The radial blanket cladding strain limit was set lower because the DFR data set used for the fuel and internal blanket strain limit indicated that at the lower strain rate estimated for the radial blanket, failure would occur at one-half the strain.

It is important to note that the calculated cladding strain proposed for design purposes is not directly related to the real cladding strain. Cladding strain is not measured in reactor. Instead the cladding plastic and thermal creep strain is calculated as a function of time to determine the time for the cladding to reach the given strain limit. This time is considered to be the maximum fuel rod lifetime as determined by the strain criterion for the given conditions. To verify this performance method it was proposed by the applicant that calculated fuel rod lifetimes would be compared to the actual lifetimes of fuel rods exposed to a particular environment.

4.2.1.2.4 Conclusions on Application of Design Limits

The staff concurs with the approach taken by the applicants to base the design limits on mechanical loading phenomena. The CRBR fuel-coolant system is such that mechanical loading is far more likely to be the dominant failure mechanism than the heat transfer and corrosion mechanisms that are the basis for LWR design limits. The bases for this finding are as follows:

(1) Heat transfer disruption occurs at coolant temperatures in the LWR system that are much lower in general and in particular much closer to normal operating conditions than is true for the sodium-cooled CRBR LMFBR system.

(2) Type 316 stainless steel is weaker than Zircaloy above 1,300°F, the temperature range involved in anticipated transients in the CRBR.

(3) Type 316 stainless steel appears to be more susceptible to loss of ductility and load bearing capability in the LMFBR environment than is Zircaloy in the LWR environment, barring unusual hydriding conditions.

Both the cumulative strain limit and the limit of 1.0 on the CDF are so intertwined with the models to which they apply that the staff cannot judge them separately. They are reviewed as a part of the two fuel models in Sections 4.2.1.3.3 and 4.2.1.3.4.

The second limit cited in Section 4.2.1.2.2 of this SER requires that there be no fuel or blanket melting up to 15% overpower. The staff concludes this limit is prudent because it precludes fuel mobility during normal operation and minimizes differential expansion loading of the cladding.

The third limit cited in Section 4.2.1.2.2 of this SER requires that there be no cladding plastic strain. The staff concludes this limit is a prudent requirement in general agreement with broad industry practice.

The fourth limit cited in Section 4.2.1.2.2 of this SER requires the equivalent stress to be less than the proportional elastic limit, or alternatively, to be less than 90% of the yield strength of the cladding. This limit ensures that the third limit will be observed.

The fifth limit cited in Section 4.2.1.2.2 of this SER requires that the fuel assembly be designed so as to not invalidate the hot channel factor based on flow area reduction. The staff concludes that such a limit is reasonable to ensure that the cited hot channel factor is conservative.

All of these limits are auxiliary to the CDF and DLS fuel evaluation models and are acceptable.

As indicated in Section 4.2.1.2.2, the applicants recognize fatigue to be a damaging mechanism, and proposed to limit fatigue damage by requiring the sum of the CDF fraction and the fraction of stress cycles referred to the endurance limit to not exceed 1.0. The applicants also proposed to address definition and implmentation of the fatigue limit in the FSAR. The staff concurs that deferral of the fatigue limit implementation to the FSAR is acceptable because a fallback position is available (reduction of residence time) if a fatigue problem should develop, and because there have been no indications of a fatigue-based failure problem with the Experimental Breeder Reactor Number II (EBR-II) steady-state irradiation program. The staff emphasizes that the applicants must address fatigue including low-cycle fatigue and fatigue of irradiated materials as might be caused by load-following operation and reactor scrams and shutdowns. This issue is to be addressed for any fuel model to be used independently for the FSAR. The applicants have committed to this requirement.

The staff concludes that the design guideline approach is acceptable provided (1) the applicants demonstrate that the guidelines are indeed conservative for all events, and (2) that the significant deficiencies noted later have been removed by the time the FSAR is submitted for either fuel evaluation model that is to be used for the FSAR. The applicant has committed to address these deficiencies.

The use of a no-melting guideline for fuel for normal operation, anticipated events, and unlikely events should preclude fuel failure and significant fuel damage during overpower conditions for step reactivity insertions large enough to initiate a scram, and for reactivity ramps faster than about 50 cents per second. For lesser reactivity steps and slower ramp rates, this guideline precludes the possibility of expulsion of molten fuel.

Effectively, there are no nonviolable limits to ensure coolable geometry. Avoidance of cladding melting serves as a design limit to ensure coolable geometry, along with a no-boiling guideline to ensure that cladding melting does not occur. The staff believes that these guidelines are inadequate of themselves to ensure coolable geometry for the reasons which follow.

First, the staff is very dubious that coolable geometry would not be affected if cladding temperatures ever did approach melting, even if available data indicate that the probability of cladding ballooning (Hunter and Fish, 1974) or slumping* is small. Secondly, the staff believes that in transient overpower (TOP) events molten fuel can be expelled well below coolant boiling; such has been observed in several tests. Although coolable geometry would not necessarily be compromised by molten fuel expulsion, it is very difficult to predict the course of events once expulsion occurs. It is far more prudent and simple to preclude the possibility of expulsion during design events.

The staff recommends and believes that the applicants can select a nonviolable cladding temperature limit for which there are data to confirm that coolable geometry would be preserved, yet which provides ample design flexibility. Such a limit should address boiling and dryout phenomena if coolant boiling is allowed by the limit selected. To protect against molten fuel expulsion, the staff recommends a nonviolable limit on fuel enthalpy, or some other property or process that is more relevant to TOP conditions than coolant or cladding temperature. The staff has performed analyses that indicate such a limit would be feasible.

The applicants have committed to provide firm coolable geometry limits and bases, therefore, for NRC review before submittal of the FSAR.

On the basis of (1) information provided in the PSAR (Chapters 4 and 15) about the thermal conditions reached in design-basis events and the overpower trips used, and (2) the response of fuel rods to transient overpower conditions as seen in transient testing (Hunter and Fish, 1974; Baars, 1980; Henderson et al, 1981), the staff emphasizes that it is very unlikely that molten fuel will be expelled or that heat transfer from the cladding to the coolant will be disrupted durign design-basis events. Available data on response of cladding to extreme conditions (see Hunter and Fish, 1974, and footnote on this page) shows that coolable geometry would not be affected until significantly more severe conditions were reached than are predicted to be reached in design-basis events. In addition, if the temperature limit finally confirmed cannot be demonstrated to be conservative, the fallback of adopting a firm, nonviolable no-boiling limit to ensure coolable geometry is available. Therefore, the staff concludes that firm limits and bases that address these concerns can be

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*The evidence on slumping comes from a series of experiments on gas cooled fast reactor fuel assemblies that was conducted at Los Alamos National Laboratory (LANL) wherein 7 and 37 rod clusters encased in a hexagonal or circular duct were subjected to decay power levels under helium natural circulation cooling conditions. Cladding melting was observed in several of these experiments; the relevant aspect of the experiments was that there was no discernible evidence that any cladding slumped that did not actually melt. These observations are the author's own, based on visual examination of experiment remains. These experiments are reported in LANL Nuclear Reactor Safety Quarterly Progress Reports LA-NUREG-6934-PR, LA-NUREG-6842-PR, and NUREG/CR-1201.

deferred beyond the construction permit, as provided in the commitment by the applicants to supply the limits and bases (FSAR).

The staff had reservations about the adequacy of the no-boiling guideline to ensure coolable geometry for undercooling conditions. This concern was over the possibility that large numbers of the high plenum pressure rods in an end-of-life (EOL) assembly might fail before coolant boiling, with attendant release of fission gas causing more prolonged disruption of cooling under loss-of-flow conditions than under full-flow conditions, possibly leading to cladding dryout and melting. In response to Question QCS490.23 on this concern, the applicants cite the results of an analysis of the safe shutdown earthquake (SSE) event in which there is not only a flow coastdown but also a power excursion. When several burnups were reviewed, it was assumed that all the rods in hot assemblies would fail. It was found that cooling was disrupted for about 0.2 sec, cladding melting was not approached, and coolant could reenter all channels after the disruption. Subject to a confirmatory review of the analysis, which the staff believes can be postponed until the FSAR is submitted, the staff concludes that the concern has been resolved.

The staff was also concerned that the no-boiling stipulation is a design guideline rather than a limit. It is not clear how the applicants intend to deal with an evaluation in which there was an onset of boiling; that is, how they would further evaluate the event, and what basis they would use to judge whether cladding would melt beyond the onset of boiling. However, as previously stated, the applicants have agreed to supply nonviolable limits that will address boiling phenomena, if boiling is allowed.

In summary, the staff concludes that the no-boiling limit is necessary but <u>not</u> sufficient to ensure coolable geometry for reasons discussed previously. The staff believes that the addition of a fuel enthalpy limit (or other limit more directly related to overpower phenomena than cladding temperature), which is based on transient overpower data for molten fuel expulsion, would provide the needed protection against that eventuality without loss of design flexibility. To ensure coolable geometry under loss-of-flow conditions, the staff strongly recommends that the applicants replace the cladding melting criterion with a nonviolable cladding temperature limit that provides substantial margin to cladding melting and also to any irreversible path such as cladding dryout that could cause cladding temperatures to approach melting. The applicants have committed to do so before submittal of the FSAR, including a detailed basis backed up with data for the limit selected. Deferral to the FSAR is acceptable on the basis that available data indicate that conditions now predicted for design-basis events are far less severe than those in which coolable geometry would be in jeopardy, and on the basis that the no-boiling guideline can be converted to a nonviolable limit as a fallback.

4.2.1.3 Fuel and Blanket Design Evaluation

This section is devoted primarily to evaluation of the applicants' fuel performance models.

4.2.1.3.1 Operational and Design Requirements

The general design philosophy, as applied to operational and design requirements, is outlined in Section 4.2.1.2 of this SER and is presented in more detail in PSAR Section 4.2.1.1.2. In terms of operating conditions the applicants address the following subjects:

(1) fuel design life (burnup)

(2) the indications from the LMFBR Base Technology Program that this design life could be achieved in CRBR

(3) thermal-hydraulic input into calculations of component temperatures

(4) shipping and handling loads

(5) rod loading mechanisms for operational and accident conditions

The applicants' design basis of 328 full-power days (FPDs), 80,000 MWd per ton peak burnup, is founded on FFTF technology (design analyses effort), on DFR irradiation experience, and on data from test rods irradiated in the EBR-II and in the General Electric Test Reactor (GETR). As noted previously, the FFTF and CRBR designs are similar except for differences associated with the CRBR blanket assemblies (which have no parallel in FFTF), and except for the change from 20% and 25% Pu for FFTF fuel to 33% Pu for CRBR fuel. The DFR irradiation experience, the FFTF fuel-design-analysis effort, the irradiation tests conducted in support of the FFTF program, and the experience gained with the FFTF fuel system, all will be as applicable to the CRBR as to the FFTF with one possibly major exception: that is the manifold changes in fuel performance that may accrue from the enhanced Pu content of the fuel for CRBR.* The 80,000 MWd per ton design basis is thus basically dependent for support on the DFR, EBR-II, and GETR data base until direct data from FFTF driver fuel and FFTF tests become available. The staff's evaluation of the adequacy of this data base is given in Section 4.2.1.3.7 of this SER.

The design basis is relatively modest in comparison with the projected equilibrium peak burnup objective of 110,000 MWd per ton, more than 500 mixed-oxide fuel rods clad in stainless steel have been irradiated to more than the initial peak fuel burnup goal of 80,000 MWd per ton, and additional valuable experience will be available from FFTF operation before the FSAR submittal.

As noted previously, the PSAR separates fuel rod loadings into two categories: (1) shipping and handling loads, and (2) operational loads. The shipping and handling load design values were based on 6-g axial and 2-g lateral accelerations, the same as selected for the FFTF fuel assembly design. Subsequent shipping tests, described by Henderson et al. (1981), have shown that these values are conservative and that shipping and handling loads experienced by the test assembly caused no discernible damage to the fuel assembly (HEDL-TME 74-8, 1974).

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*Several issues involved in this discrepancy between the data base and the CRBR design were posed in Question QCS490.1. The applicant in responding to this question indicated that data on performance of enhanced Pu content (30% to 40%) fuel rods were obtained from test ANL-08 conducted in EBR-II. The data are still being evaluated and are not now available to the staff; however the applicants indicate that current methods adequately predicted performance.

Current operational loading requirements are listed in Section 4.2.1.2 of this SER. The staff agrees with the applicants that additional loading mechanisms could be discovered during future irradiation testing and/or technology programs. The effect of any such additional loadings on design adequacy must be analyzed and reported in the FSAR.

Thermal-hydraulic input to component temperatures is evaluated in Section 4.4 of this SER.

4.2.1.3.2 Environmental and Material Considerations, Fuel and Blanket

PSAR Section 4.2.1.1.3 contains a discussion of environmental and material considerations. Certain "design basis material properties" were presented and asserted to be used for assembly design and analysis on a conservative basis. These considerations for cladding included the following:

(1) irradiation-induced creep and swelling
(2) stress-rupture properties and thermal creep
(3) tensile properties
(4) cladding wastage

Other considerations not wholly confined to the cladding included the following:

(1) fission gas release
(2) operation with failed fuel
(3) redistribution of nongaseous fission products
(4) blanket rod environment considerations

These properties and considerations were not measured in an in-reactor environment, but were determined by postirradiation examination (PIE) and testing. The full range of prototypic CRBR conditions for cladding includes the following:

(1) irradiation temperature--700° to 1,400°F

(2) fluence--0 to 13 x 10^{22} n/cm^2, E > 0.1 MeV (fuel)
--0 to 17 x 10^{22} n/cm^2, E > 0.1 MeV (blanket)

(3) burnup--0 to 80,000 MWd per ton

(4) fueled and unfueled environments

(5) both plenum gas pressure and fuel cladding mechanical interaction (FCMI) loadings

(6) transient conditions

(a) temperature, 700° to 1,800°F
(b) event duration, < 1 sec to 600 sec
(c) strain rate, up to 10% per second

A significant data base exists; nevertheless, the staff is concerned about the adequacy of the data base because of the following:

(1) atypical factors (relative to CRBR design parameters and test conditions) in much of the data base, particularly for transient data and because the fuel plutonium content lies outside the data base plutonium content

(2) incomplete coverage of the entire range of CRBR conditions by the data base

Because of these concerns the staff cannot now guarantee that the design life goal will be achieved. Further information directed at indicated concerns is needed to remove this equivocation. Specific limitations of the current data base and the additional information required are detailed in Section 4.2.1.3.7 of this report. The applicants plan extensive additional testing as detailed in the responses to Questions QCS490.1, 490.2, 490.6, 490.9, 490.12, 490.15, 490.17, and 490.18, which the staff believes address the truly significant issues.

4.2.1.3.2.1 Irradiation-Induced Creep and Swelling

The applicants asserted that irradiation-induced creep and swelling are superplastic phenomena and are not ductility limited. At this time no unique limit or set of limits has been identified for total strain, swelling-induced strain, or inelastic strain (total strain minus that caused by swelling). The applicants' position that irradiation-induced creep and swelling are not damaging is supported by the following points:

(1) It has not been possible to correlate failures of EBR-II rods (Cantley et al., 1975; and Olson, Walter, and Beck, pp. 134-151, 1976) clad with 300 series stainless steel with total or inelastic strain at failure.

(2) Axial failure locations in those instances did not correspond to axial locations of maximum strain, either total or inelastic.

(3) Recent ex-reactor test results (Haguard, 1976) showed that failure strength for irradiated unstressed specimens did not differ significantly from irradiated stressed specimens with up to 1% irradiation creep strain.

Irradiation-induced creep and swelling are nevertheless important design considerations because they are responsible for most in-reactor deformation and dimensional change of the cladding and ducts. One very important phenomenon discussed in PSAR Section 4.2.1.1.3 is the behavior of the fuel cladding gap. Changes in the gap thickness (including closure of the gap) have a major influence on the heat-transfer conductance between fuel and cladding. Fuel-cladding mechanical interaction (FCMI) loading occurs when the gap closes.

Irradiation-induced cladding swelling and creep tend to increase the gap thickness; the various processes that cause fuel growth plus differential thermal expansion between fuel and cladding tend to close the gap. In general, the gap tends to close early in exposure, then to open again late in life. The applicants propose to calculate the magnitude of the fuel-cladding gap and the steady-state FCMI loading on gap closure using the LIFE-III code (Billone et al., 1977) (or LIFE-IV, if available).

The staff has had no opportunity to review the code, its models, or the qualification of the code. The staff has concluded, however, that the code need not be reviewed for the construction permit on the basis that there are fallback positions available (primarily reduction of power) should there be problems using the code's predictions. The code must be reviewed and demonstrated to be satisfactorily conservative for CRBR conditions if it is to be used for the FSAR.

A second important phenomenon related to irradiation-induced swelling and creep is dilation and bowing of the ducts. Swelling of the duct causes duct dilation, and because of sensitivity of swelling to temperature, contributes to duct bowing. Differential thermal expansion between rods and duct coupled with swelling of rod cladding causes the clearance between duct and rods to close. Altogether, these phenomena can result in interference between the rod bundle and the duct, putting bending stresses on both rods and duct. Irradiation creep acts to minimize these stresses, and the applicants maintain that significant additional loads are not likely. Nevertheless, there are two potential problems.

(1) Significant discharge loads at refueling may be encountered because the distortion and dilation of the ducts may result in interference between assemblies and/or between assemblies and core restraint structures.

(2) Significant interference between the rod bundle and the duct could also cause cladding hot spots.

The applicants have established a design limit to control the second potential problem to an acceptable level (see design limit (5) in Section 4.2.1.2.2). However, by the very nature of the problems, there are no directly relevant data with which to verify calculations of the dilation, distortion, and rod bundle-duct interactions that may occur under prototypic conditions. Basically, verification of design, analysis methods, and design limits for this area must await receipt of data from the FFTF to qualify the calculation methods and determine the adequacy of the CRBR design in this area. Largely for this reason, the staff has not attempted to review in detail the methods being used in this area. However, for the FSAR, a detailed review of the methods and FFTF data will be required. Should these phenomena become real problems, the fallback position would be to reduce in-reactor residence time.

4.2.1.3.2.2 Stress Rupture, Thermal Creep, and Tensile Properties

Different properties were used for the two fuel evaluation models. Hence, these properties are best reviewed with the individual evaluation model.

4.2.1.3.2.3 Cladding Wastage

Differences also existed between the treatments given cladding wastage in the two evaluations models. However, similar comments apply to both, and such comments are presented in this section.

Cladding wastage predictions for both fuel evaluation models presume that the effect of a given depth of penetration of the cladding, by whatever cause, reduces the cladding thickness by that much. However, there is the possibility that stress intensity factors associated with such mechanisms as intergranular corrosion may make the cladding weaker for a given attack depth than would be predicted by the "net section" approach. The applicants have concluded (Jacobs, pp. 173-177, 1976) that some Hanford Engineering Development Laboratory (HEDL) tests demonstrated that use of the "net section" method gives satisfactory results. The staff has reviewed the data in Olson, Walter, and Beck (pp. 134-151, 1976) and concludes that the applicants' assertion is supported with regard to wear marks. The data presented also support the applicants with regard to most corrosion attack. However, the data for intergranular attack do not appear to be conclusive. Apparently only 4 out of the 14 data points presented exhibited intergranular corrosion, and of those, only 2 showed attack depths greater than 0.5 mils (see Table 10 and Figures 57 through 63 in Olson, Walter, and Beck, 1976).

In any event, none of the data used to develop wastage models reflect the increased corrosion rates that may apply for (1) the increased plutonium content of the current fuel design (33% versus 25% for the data base), nor (2) the predominantly plutonium fission product spectrum in the CRBR as compared with the EBR-II data base, the fuel for which had 30% or more enrichment in U-235 (Baars, 1980; and Johnson and Hunter, 1978).

For these reasons the staff cannot now confirm that the wastage models conservatively represent the effects of prototypic corrosion of the cladding. The applicants do indicate in their response to Question QCS490.1 that preliminary results from the ANL-08 test show fuel cladding chemical interaction (FCCI) to be similar or less severe than for the 25% Pu rods in the data base. The ANL-08 test was an EBR-II test with fuel rod Pu content ranging from 30% to 40%. The test has not been fully evaluated and the results were not available to the staff for review. Data from planned FFTF tests and surveillance of FFTF fuel (responses to Questions QCS490.1 and 490.15) should adequately resolve the effect of the atypical fission product spectrum in EBR-II tests.

The applicants claim that because application of the fuel evaluation models to the EBR-II test data yielded conservative results, the wastage models embedded therein were proven to be conservative. The staff does not accept this argument. Such results prove only that the <u>overall</u> model was conservative for those data.

Fuel adjacency effect (Hunter and Johnson, 1979; Lovell, Christensen, and Chin, 1979) is the term that has been applied to the severe reduction of load-bearing capability and ductility that occurs in cladding irradiated next to fuel at temperatures less than about 1,050°F to 1,100°F. The effect now generally is accepted as real. There seems to be a growing body of evidence that the effect may be caused at least partially by metallurgical phenomena (Hunter and Johnson, 1979) or interaction with fission products (Adamson et al., 1981; Duncan, Panayoyou, and Wood, 1981) rather than caused by mechanical phenomena, a testing artifact, or strictly irradiation damage. Lovell, Christensen, and Chin (1979) suggest that grain boundary embrittlement caused by segregation of impurities to grain boundaries may be responsible for fuel adjacency in combination with helium embrittlement. Adamson et al. (1981) and Duncan, Panayoyou, and Wood (1981) discuss experimental evidence for embrittlement of prototypic cladding by liquid mixtures of cesium and tellurium fission products and evidence for temperature sensitivity of the effect that appears to agree well with the effect as shown by fuel-cladding transient tester (FCTT) tests (Johnson and Hunter, 1978). Vaidyanathan and Adamson (1981) and DeMelfi and Kramer (1981) propose approaches to modeling the observed fission product interaction.

Fuel adjacency may not be a strictly wastage phenomenon, but with the implication of fission product interaction this seems to be a logical classification for the effect. A good understanding of this phenomenon and identification of the relevant operational parameters that define its behavior are essential to an adequate fuel evaluation model.

In summary, the staff's primary concern is that the fuel adjacency effect be adequately understood so that it can be clearly demonstrated that the effect is properly accounted for in the design evaluation models.

4.2.1.3.2.4 Fission Gas Release

The applicants assert that fission gas released from the fuel flows through cracks and microcracks in the fuel to the plenum during steady-state operation, and that little or no pressure difference is required to support this small flow. Substantial data are cited to support that view; see Weimar and Sebening, 1973; Zimmerman, 1974; Longest et al., pp. 149-163, 1975; and Longest et al., pp. 213-215, 1974.

The staff is more concerned about the basis for estimating how much fission gas is released from the fuel to the plenum for predicting plenum gas pressure. Plenum gas pressure is the principal load on the cladding during undercooling conditions, and is predicted to be the most damaging loading during steady state. Predictions for the FFTF assumed that all fission gas produced was released to the plenum. This assumption was not necessarily grossly conservative. The data (Leggett et al., 1979) show very high release fractions (nearly 100%) for rods approaching 80,000 MWd per ton.

The applicants proposed to use an empirical correlation to data (PSAR, p. 4.4 - 40) that predicts the fractional release of fission gas as a function of local burnup and linear heat rating. The correlation has not been confirmed for fuel rods with a plutonium content exceeding 25%, except in a preliminary way (see response to Question QCS490.1). The staff finds no indication that allowance was made for enhanced gas release caused by elevated fuel temperatures during overpower anticipated events. Some unknown allowance for additional release because of overpower events may be implicit in the operating history of the EBR-II test fuel rods used to calibrate the correlation, but there is no way to determine whether this is adequate allowance for CRBR fuel evaluation.

The staff would prefer an across-the-board assumption of 100% release for design purposes, particularly for blanket rods. However, the correlation does predict 100% release for high burnup, high-power rods and high fractional releases for other high-burnup rods. For this reason, the staff accepts the correlation.

The applicants assert in the answer to Question QCS490.25 that the use of 100% release of fission gas for all rods would result in an orificing pattern that would overcool cold rods (which may experience less than 100% release of fission gas) when that flow would be better used for the hottest, highest power rods that would experience 100% release of fission gas. However, in view of the highly nonlinear dependency of both thermal creep and stress-rupture on temperature, a reduction of relatively few degrees in operating temperature would probably correct any significant impact that use of 100% release for all rods would cause. Thus, it appears that the use of the correlation buys, at most, a few degrees in operating temperature, and the staff is inclined to question the aggressiveness of this aspect of the design.

Although questioning the justification for departing from an assumption of 100% release of fission gas for design purposes, the staff concludes that the departure proposed by the applicants is not by itself significantly nonconservative and is therefore acceptable.

4.2.1.3.2.5 Fission Product Migration

Cesium is expected to be the principal agent of fission product redistribution. With the low oxygen-to-metal ratio design of the CRBR fuel rods, it is expected that the cesium will tend to migrate to the blanket regions of the rod where it may react with blanket material to form Cs_2UO_4 (cesium uranate). Cesium uranate has a relatively low density compared with the UO_2 it replaces (specific volume about 3.5 times UO_2). However, with adequate void space available, it is expected to cause no problem. It was for this reason that a diametral gap of 10 mils was provided in the axial blanket region. Strain caused by this mechanism has been observed (Karnesky, 1977), but no cases of fuel rod failure caused by the phenomenon have been identified.

The staff concludes that this phenomenon is not likely to be a major influence on CRBR fuel performance.

4.2.1.3.2.6 Operation With Failed Fuel

The CRBR was designed with the intention of operating with failed fuel. It is not expected that rods exhibiting only fission gas leaks will be removed before goal burnup. The applicants have committed to remove all failed rods exhibiting fuel-sodium contact at the first refueling outage after evidence of contact until sufficient data have been collected to ensure that safe operation is feasible (PSAR, Vol. 6, p. 4.2-27, and Vol. 23, p. Q241.76-2). In any event, a limit is to be established on the maximum delayed neutron rate acceptable before the reactor must be shut down and the subject assembly removed.

Test results to date (Washburn et al., 1979) have shown that events develop on a day time frame rather than on a minute or second time frame. That is, there would be time to evaluate appropriate action before a decision on action would be required. It appears that swelling of defected rods can be locally concentrated, and this is now considered to be the major roadblock to demonstrating safety with failed rods. The applicants state in their response to Question QCS490.18, in which they were asked to update the PSAR discussion on operation with failed fuel, that the United Kingdom experience at DFR was that operation could be safely continued for 60 days (Sloss et al., 1979) and that preliminary U.S. data indicate that operation can be safely continued for at least 22 days. Deterioration appears to be primarily related to shutdown-startup cycles.

The applicants do not address whether a "gas leaker" would be expected to progress into fuel-sodium contact, and if so, how rapidly. No information is yet available on how failed rods, either "gas leakers" or those showing evidence of fuel-sodium contact, respond to anticipated events or design-basis accidents.

The staff concludes, on the basis of available data, that operation can continue to the next outage with a "gas leaker" failure, provided the delayed neutron detector does not show a generally increasing trend (indicating fuel-sodium contact). Should a generally increasing trend be detected on the delayed neutron signal, a controlled shutdown should be initiated at once for the purpose of removing the offending assembly. At such time as data are available on behavior of failed rods in design events, and limits have been established for operation with fuel-sodium contact failures, more relaxed requirements can be considered for operation with failed fuel.

4.2.1.3.2.7 Blanket Rod Environmental Considerations

The differences between fuel and blanket assemblies noted by the applicants include the following:

(1) Both internal and external corrosion rates should be less for blanket rods than for fuel rods, the former because of the low plutonium content in blanket material, and the latter because the sodium velocity is slow enough to enter the velocity-sensitive region where corrosion rates are lower albeit variable.

(2) Postfailure swelling should be less for the low plutonium content blanket material.

(3) FCMI predictions by the LIFE code may not apply to blanket rods as the LIFE code was calibrated only to fuel rod data.

The staff would add (1) the different power history that will be experienced by blanket rods, and (2) the much longer residence time for radial blanket assemblies. Blanket rods start at very low power and gradually build up to their peak power as U-238 is converted to Pu-239. Restructuring of the oxide, release of fission gas, and thermal performance may all be affected by this power history. At least one EBR-II assembly (PNL-17) (Porten, pp. 13-23, 1976) experienced a power increase during its residence wherein restructuring was expected at the new higher power but was apparently inhibited by the prior operation.

The much longer residence time for radial blanket rods most probably affects primarily cladding wastage. Irradiation effects most probably will saturate well short of the 2×10^{23} n/cm^2 ($E > 0.1$ MeV) fluence that 4 to 5 cycles' residence time for the radial blanket assemblies represents. Nevertheless, there are no data now to confirm this position.

The LIFE code is available to predict thermal performance and other phenomena such as restructuring for blanket rods. It has been noted that the code has not been qualified to blanket rods. However, with the use of appropriate material properties coupled with a full set of appropriate models, the code should provide generally valid results. Blanket rods are, after all, effectively fuel rods with a power history that is different from other rods.

Because there is a tool available to predict blanket rod performance that should be adequate by and large, and there is a reasonable fallback position available (reduction of in-reactor residence time), the staff concludes that experimental demonstration of acceptable performance and qualification of a performance prediction model for blanket rods can be deferred until submittal of the FSAR.

4.2.1.3.3 Review of Cumulative Damage Function (CDF) Fuel Evaluation Model

The CDF model (Jacobs, 1976) assumes that there exists a damage function for fuel and blanket cladding so that at failure:

$$L(t)dt = 1.0$$

where t is time, and L is the damage function. It was also assumed that the following would hold true:

(1) The damage function can be divided into independent increments, each representing the damage caused by a particular segment of operation.

(2) Separate and independent damage increments can be defined for thermal creep and plastic deformation.

Although the life fraction approach is used routinely in other applications, there is no theoretical basis for the practice. It is thus an empirical approach and must be qualified by typical data throughout its range of application.

The CDF evaluation model purports to account for damage to the cladding during all of the transient events to which the fuel rod is exposed and to all increments of steady-state operation between those events. Environmental effects (irradiation damage, internal and external corrosion, and loss of interstitials) are included as are uncertainties both of environmental effects and of mechanical properties.

Basically, the CDF model evaluates loads less than the proportional elastic limit (PEL) on a stress-rupture basis, employing a life fraction approach to account for varying temperature-load histories. For loads exceeding the PEL, the CDF value depends on the extent to which the load penetrates the plastic deformation region. Loads greater than the ultimate strength yield a CDF of 1.0 regardless of prior history. The ultimate strength is reduced by (1) prior creep and (2) prior plastic loading. Correlations to predict the effect of both prior creep and plastic loads on ultimate strength are provided as is the correlation relating the CDF fraction to a particular plastic load. As with material correlations, these correlations are formulated so as to take into account uncertainties.

The staff has no problem with the basic approach, provided, as mentioned before, the model is adequately qualified to typical data. The staff does have significant reservations about the model, as follows:

(1) The model must account for the fuel adjacency effect (see discussion under Section 4.2.1.3.2.3) if it is to be a credible means for evaluation of fuel performance. The applicants state in their response to Question QCS490.10 that "with respect to the utilization of FCTT data from fuel adjacent cladding, the applicants' current methodology does, indeed, incorporate formulations derived from the entire data base." However,

Jacobs (Westinghouse, CRBRP-ARD-0115, p. 143) explicitly states that data from fueled cladding would not be included in the CDF validation study until they were fully understood. The staff therefore remains concerned that fueled cladding data be incorporated into the CDF. The applicants have committed to provide explicit documentation on how the fuel adjacency effect is included in the CDF model for the FSAR.

(2) The model did not recognize the possibility of transient FCMI during slow overpower events (Westinghouse, CRBRP-ARD-0115, pp. 11-13). Apparently, the fuel (or blanket) was assumed to respond to slow overpower events similar to the way it responded to undercooling events. Yet, it has been shown that the response to all overpower events, slow or fast, is fundamentally different from the response to undercooling conditions (HEDL-TI 75001-12.2).

The applicants state in their response to Question QCS490.10 that transient FCMI was included in analyses reported in the PSAR via loads calculated by the PCON code. This contradicts the passage cited in Jacobs (1976). The staff therefore remains concerned that transient FCMI is not considered in the CDF model. However, the staff believes the problem is principally one of updating the PSAR documentation, and this will be corrected by the FSAR.

(3) The CDF model was not verified against integral fuel rod in-reactor data, either steady-state or transient. Although such data have serious atypical flaws as discussed later, it would be highly desirable to qualify the model against them in view of previous comments on fuel adjacency and transient FCMI.

The applicants in their response to Question QCS490.10 cite tests used to calibrate and validate LIFE-IV-T, demonstrate that loads calculated by the PCON code are greater than those calculated by LIFE-IV-T, and then offer this response to demonstrate that the CDF model has effecively been evaluated against integral rod tests. The staff does not accept this as a meaningful evaluation of the method. The evidence offered may demonstrate conservative loads, but does not demonstrate how the CDF model evaluates the cladding response as compared with the tests. The applicants have now unequivocally committed to qualifying the CDF model against such tests and will document same for the FSAR.

(4) The CDF model focuses entirely on cladding response; no procedures were specified to be followed in determining the cladding load-temperature history for analyzing an event. The load-temperature history is at least as important to the validity of the model as are the cladding response characteristics, and in the case of LMFBR technology, are likely to be a major source of uncertainty. To be complete, the model should clearly specify how the load-temperature history is to be determined. The applicants have committed to provide such definition for the FSAR.

(5) The applicants should reassess their position on fluence dependency and conservatism of the tensile properties used in the CDF model, in view of the data in Fish, Cannon, and Wire (1979).

(6) PDC 8 (analogous to GDC 10) requires that specified acceptable fuel design limits (SAFDLs) be observed with "appropriate margin." The CDF model purports to predict whether or not fuel rod cladding will be breached. The auxiliary correlations that provide parameters to the CDF model are applied using confidence limits on the true mean predictions of the correlations. Confidence limits on the correlation prediction refer only to the uncertainty of where the true mean was derived. As expressed in their answer to Question QCS490.11, the applicants believe that confidence limits adequately cover the true variance of the properties (as opposed to the apparent property variance caused by measurement error, etc.). In so doing the applicants have effectively assumed a partition of the overall data variance into that caused by the true property variance and that caused by measurement error and/or inability to conduct precise tests, but have provided no supportable basis for that particular partition. The staff does not deny that a substantial portion of the overall data variance may be caused by measurement error, and so forth, but the staff cannot accept the applicant's partition without some supportable basis.

The staff therefore believes that the applicants' treatment of the model uncertainties is not now appropriate, and does not satisfy the "appropriate margin" requirement of PDC 8. The staff believes that the applicants must do one of the following to make the treatment of uncertainties acceptable and to comply with the requirements of PDC 8 (analogous to GDC 10):

(1) Demonstrate that the use of confidence limits (as opposed to tolerance limits) for dealing with uncertainties of mechanical properties is generally accepted with wide application.

(2) Provide a supportable basis for what portion of data variability from the mean is caused by variability of the property from the mean, and determine tolerance limits to cover that portion of the variability.

(3) Adopt tolerance limits to cover the data, even though this forces them to compensate for measurement and/or test error.

The staff suggests that application of either Bayesian techniques (Lloyd and Lipow, pp. 190-197, 1962) or analysis-of-variance techniques (Bowlar and Lieberman, Chap. 10, 1972) could be used as a means for developing bases for partitioning the overall data variability into two parts. The staff also suggests the use of Monte Carlo analysis methods to minimize excessive conservatism, rather than the use of all parameters at the extreme of the level of coverage specified.

The applicants have committed to justify the adequacy of confidence limits for mechanical properties for the FSAR.

Statistics experts at LANL state (based on Ang and Jang, pp. 196-199, 1979; and Lloyd and Lipow, pp. 190-197, 1962) that the error propagation method used by the applicants (Equation E.2 on p. 203 of Jacobs, 1976) is valid only for functions whose arguments are statistically independent and that are linear in all arguments or are monotonic in a single variable. The applicants do not attempt to establish the independence of all arguments. More importantly, it appears that the key variables in the CDF are decidedly nonlinear. The key variables themselves are correlated by functions that are linear in their arguments, but the variables are used in a nonlinear fashion in the CDF. For instance, the Larson-Miller Parameter (LMP), used in the CDF to correlate stress-rupture data, is correlated as a simple linear function of the stress. However, in the CDF, what is used is the time to rupture, which is an exponential function of the LMP divided into the time increment to obtain that portion of the CDF. Similarly, although details are not given in Jacobs (1976) on just how the CDF is calculated, the other key variables are undoubtedly incorporated into the CDF in a nonlinear fashion. Furthermore, the LANL experts advise that Equation E.5, of Jacobs (p. 204, 1976)* used to determine uncertainty bounds, is applicable only to simple linear relations (Montgomery and Peck, p. 127, 1982; and Draper and Smith, p. 89, 1982). Hence, presuming the staff is correct that the CDF depends on multiple, effectively nonlinear parameters, the uncertainty bounds are incorrect for that reason alone.

LANL experts advise that there are no known analytical formulations to determine tolerance limits for multiple, nonlinear problems such as the CDF model, and recommend Monte Carlo simulation methods for this purpose. To generate input for the Monte Carlo calculations, the LANL experts recommend use of tolerance limits for the CDF parameters determined by Equation 2.45, of Montgomery and Peck (p. 32, 1982) for simple linear relations and by Equation 4.48, p. 141 of the same reference, for multiple linear relations.

The applicants provide a Monte Carlo analysis that addresses the above concerns in their response to Question QCS490.11. The applicants have committed to satisfactorily address the problem of nonlinearity for the FSAR.

In summary, the staff concludes that the CDF fuel evaluation model is not acceptable at this time. The applicants are aware of the major deficiencies (neither the fuel adjacency effect nor transient FCMI is addressed), and the staff believes the applicants intend to correct the former deficiency by the FSAR and may have already corrected the latter deficiency although this has not been reflected in Jacobs (1976). The staff concludes that verification of the CDF model should be extended to integral rod performance tests. Finally, the staff concludes that the statistical treatment, or provision for appropriate margin, is not adeuqate because it is based on confidence rather than tolerance limits. Ways to resolve this issue were identified. The applicants have committed to address all of these concerns for the FSAR.

4.2.1.3.4 Review of the Ductility Limited Strain Fuel Evaluation Model

The ductility limited strain (DLS) fuel evaluation model was derived from the model used to evaluate FFTF fuel (Sim and Veca, 1971, Rev. 1972) and known variously as "FCF-213," the "design recipe," and the "design procedure." Many features of the FFTF model were retained, but several significant changes were made. Unfortunately, the description in the PSAR was very sketchy, no detailed description was available for review, and the FRST computer code used to apply the model was not available. Nevertheless, enough information was provided

*Also the portion of Equation II.B.4 on p. 9 of Jacobs (1976), to the right of the "±" sign.

that the staff could reach the conclusion that the model did differ significantly from the FFTF model. A meeting with the applicants provided additional information.

Features of the DLS model that were retained from the FFTF model include the following:

(1) Use of solution-annealed type 316 stainless steel tensile and thermal creep properties for the cladding.

(2) Physical and thermal properties of both fuel and cladding.

(3) Use of a constant fuel-cladding gap conductance for both steady-state and transient conditions dependent only on the as-fabricated gap.

Features of the DLS model that were significantly changed from the FFTF model include the following:

(1) Use of 0.3% (rather than 0.7%) cumulative thermal creep and plastic strain as the cladding failure criterion (cladding integrity limit). This was partitioned into 0.2% strain (thermal creep only) for steady-state operation and 0.1% strain for both anticipated and unlikely events.

(2) Inclusion of irradiation-induced cladding swelling and creep in the model.

(3) Inclusion of steady-state FCMI in the model.

(4) Recognition of the axially variable cladding loading that would be the result of both steady-state and transient FCMI.

(5) Strain was calculated based on the equivalent stress rather than the hoop stress.

(6) Plenum pressure was not based on 100% release of fission gas.

Changes (1) and (3) would clearly be conservative and would not affect application of the FFTF model qualification to the DLS model. The net effect of Changes (2), (4), and (5) is not clear. Change (6) would affect results nonconservatively.

The applicants assent in their response to Question QCS 490.13 that temperature uncertainties were accounted for in the DLS model but nominal temperatures were used in the FCF213 model, thereby contributing definitely more conservatism to the DLS model. This is not true. Temperature uncertainties (Porten, Aug. 1976) were always used in the FCF 213 model when fuel performance was being evaluated. Nominal values were used only in the qualification work, where the use of the high temperatures would have produced a nonconservative evaluation.

The DLS model, like the CDF model, addresses the combined effects of both steady-state and transient operation. This was purportedly accomplished through the use of the cumulative strain cladding integrity limit. Cladding wastage was included. Steady-state FCMI loads were calculated using the LIFE-III code and input to the FRST code for evaluation along with plenum gas pressure loading for steady-state operation.

Transient undercooling events were modeled assumimg only plenum gas pressure loading. The cladding failure criterion that was used for unlikely undercooling events was based on burst pressure, the same as for the FFTF model. The burst pressure was determined in a manner that depends almost entirely on the ultimate strength of the cladding, and is very insensitive to small strains such as the maximum strain used in this model.

Transient FCMI loading was considered for transient overpower events as well as plenum gas pressure loading, but the treatment of transient FCMI was different from that used for FFTF fuel evaluation. Transient FCMI loads were calculated with the LIFE-IV code and input to the FRST code. The model for cladding response to loading in overpower events was similar to the FFTF model. Below the proportional elastic limit (PEL), only thermal creep was considered, and above the PEL, only plastic deformation was considered. The ultimate strength was assumed to coincide with a plastic strain equal to the strain criterion (0.3%), plus the elastic strain at the PEL. Strain was accumulated as the evaluation proceeded, with failure presumed at 0.3% strain. If the plastic region was entered with little or no accumulated thermal creep strain, then failure occurred when the ultimate strength was reached. If there was prior thermal creep strain, then failure would correspond to some stress intermediate between the PEL and the ultimate strength.* That stress would be displaced toward the PEL from the ultimate strength in proportion to the fraction of the strain criterion represented by accumulated thermal creep.

The net effect of these models was that the cumulative strain criterion acted to limit the acceptable severity of normal operation and anticipated events. As the staff understands how the model was defined and used, the cumulative strain had no effect whatever on the failure point for unlikely undercooling events and only a moderate effect on the failure point for unlikely overpower events. In fact, the failure point as modeled depends most heavily on the ultimate strength of the cladding, not on cumulative strain.

The staff has the following comments and reservations about the model:

(1) The applicants assert that the use of solution-annealed type 316 stainless steel properties for the cladding conservatively represents the behavior of 20% cold-worked irradiated type 316 stainless steel cladding. PSAR Figure 4.2-2B is offered as proof of the conservatism. The referenced figure compares the ultimate strength used for the model with ultimate strength data stated to be for 20% cold-worked irradiated stainless steel and obtained at strain rates of 3×10^{-2} per second and 0.44×10^{-2} per second. The staff believes that this comparison gives a false picture of the conservatism of the model ultimate strength because the comparison data were for unrealistically high strain rates. For a model in which strain rate is not included, conservatism dictates that data for a relatively slow strain rate would be more appropriate. The same comparison was made with data from Fish, Cannon, and Wire (1979) taken at fluences high enough for saturation of fluence effects and at a strain rate of 4×10^{-5} per second (about the same strain rate as the data used for the CDF ultimate strength correlation). This comparison indicated the model data to be nonconservative by 8 to 10 ksi in the 1,200-1500°F temperature

*This feature is different from the FFTF model and adds conservatism.

range. It is also noted that the data from Fish, Cannon, and Wire (1979) do not reflect the fuel adjacency effect. Nevertheless, the proof of such an empirical model lies in how well it is qualified to data.

(2) Comparison of the yield strength from Fish, Cannon, and Wire (1979) with the PEL used in the DLS model (assuming it was the same as used in the FFTF model) shows the PEL values to be quite conservative, except at 1,500°F. The predicted values of plastic strain for anticipated events should be conservative. However, yield strength conservatism is unlikely to contribute much to conservatism of failure predictions because, as already noted, the failure point is heavily dependent on the ultimate strength.

(3) The inclusion of irradiation-induced swelling and creep would have removed a great deal of conservatism from predictions for reactivity addition events had the FFTF model for response to such events been retained. The applicants asserted that irradiation-induced swelling and creep were incorporated so as to affect only the gap closure process and thus had a much lesser impact than it would have with the FFTF model.

(4) There are no plans to address fatigue mechanisms for the DLS model--at least, none are stated.

(5) The staff has reservations about the conservatism of the cladding wastage models (see Section 4.2.1.3.2.3 of this SER).

The staff concludes that the DLS model as it now stands is not acceptable. This conclusion is based primarily on the model being so changed from the FFTF model that it must be requalified to both undercooling and overpower data; the applicants have committed to requalify the model for the FSAR. This deficiency is easily correctable, and with the availability of fallback positions noted at the end of Section 4.2.3, the correction can be deferred to the FSAR. The staff cautions the applicants that they must specifically demonstrate that the model provides an "appropriate margin" as required by PDC 8 (analogous to GDC 10).

4.2.1.3.5 Review of Fuel System Response to Seismic Events

This section is devoted to evaluation of the fuel system response to seismic loads; that is, to loads generated directly by a seismic event, as opposed to loads indirectly caused by a seismic event such as an insertion of reactivity caused by a seismic event.

The staff believes that response of the fuel system to a seismic event is fundamentally determined by the response of the duct assembly. If the duct assembly remains intact, then the fuel or blanket rod bundle need bear only the loads generated by its own weight, and it is unlikely that control assemblies could not be inserted. Based on an approximate calculation, the staff believes it unlikely that either fuel or blanket rod bundles would exceed acceptable damage limits if the duct did not fail. Misalignment may slow insertion of the control assemblies, but this has been accounted for in evaluating the response of the reactor to a seismic event.

A preliminary model for response of the reactor core and its components that is described in PSAR Section 3.7.15 was used to determine the loads reported in the PSAR. A more sophisticated model described in PSAR Section 3.7.2 is planned for FSAR analyses. The loads determined by the preliminary model were considered in evaluation of the assembly design. Details of the analyses are contained in a Westinghouse document (Prevenslik et al., Jan. 1978).

The staff briefly reviewed this reference, which evaluated fuel assembly performance during steady-state operation and duty cycle transients as well as the response to seismic loads. The evaluation employed strain criteria to judge failure modes of crack initiation and excessive deformation and a life fraction criterion to evaluate creep-fatigue interaction. These criteria were based on the ASME Code, Code Case N-47 (previously designated Case 1592) and the RDT draft criterion for breeder reactor core components, June 1976, with appropriate additions to account for irradiation-induced loss of ductility and for the deformation caused by irradiation-induced swelling and creep. Evaluations were performed at the shield block (just below the assembly), core midplane (CMP), above core load pad (ACLP), top load pad (TLP), assembly attachment, and orifice plate location. The CMP, ACLP, and TLP locations are of the most interest as these locations see the highest temperature (ACLP and TLP), and the most irradiation (CMP). Assembly failure was most closely approached at the TLP at the beginning of life where the crack initiation criterion was approached within 30%. A peak stress of 46,000 psi (about 90% of yield strength at peak temperature) was predicted at the beginning of life at the ACLP during an SSE event.

It appears to the staff, based on comparisons of temperature information in the aforementioned reference and temperatures supplied in PSAR Section 4.4, that nominal thermal hydraulic design value (THDV) temperatures were used in the evaluation with no allowance for temperature uncertainties. If this conclusion is correct, then with the addition of temperature uncertainties, predictions of close approach to the criteria could have become predictions for failure. Therefore, the predictions in the referenced evaluation may not be conservative. However, the fallback of lowering the operating temperature is available.

The applicants assert informally that temperature uncertainties were considered in their analysis, but that those considerations are documented in references that are not available to the staff.

The staff concludes that as long as the ducts remain intact, acceptable fuel damage limits would likely not be exceeded, and that the control assemblies could be inserted into the core in the event of a seismic event. The staff further concludes that the analyses in Reference 171 to the PSAR may not be conservative, and should be repeated with appropriate allowance for temperature uncertainties, or documentation provided demonstrating that they were considered in Reference 171 to PSAR Section 4.2. There is available the fallback of reducing the operating temperature if problems should be encountered; therefore this may be deferred until the FSAR is submitted. The applicants should include an analysis of rod bundle (especially blanket) response to seismic loads, and should carefully address the likelihood of widespread contact between assemblies at midplane and how this would change performance in seismic events for the FSAR. The applicants have committed to provide an evaluation of bundle-duct interaction in seismic events for the FSAR.

4.2.1.3.6 Review of Plant Protective System Speed of Response Requirements

PDC 8 for LMFBRs (analogous to GDC 10 for LWRs) requires that "The reactor and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any conditions of normal operation including the effects of anticipated operational occurrences." This criterion requires (1) that specified acceptable fuel design limits (SAFDLs) be established, and (2) that the plant protective system act fast enough to preclude exceeding any SAFDL during any design-basis event. This section evaluates compliance with the second requirements, namely, plant protective system (PPS) speed of response.

The required speeds of response of the PPS were specified in PSAR Figures 4.2-93 and 4.2-94. Figure 4.2-93 addressed the primary system requirements for operational basis earthquakes (OBEs), safe shutdown earthquake (SSE), normal operation at 100% flow, and startup conditions (40% flow, 0% power). Figure 4.2-94 addressed requirements for the secondary system for the SSE at full flow, the OBE and nonseismic events at 40% flow, and the OBE and nonseismic events at 100% flow. These requirements were stated in terms of the required reactivity insertion as a function of time after current interruption to the primary control rod drive mechanism stator, and time after current interruption to the secondary system solenoid valve. The instrumentation lag times were not included.

The requirements were then compared with reactivity insertion predicted as a function of time for the beginning of cycle 4 (BOC-4) and BOC-5 for the primary system for nonseismic events in PSAR Figure 4.2-114, and for the SSE in Figure 4.2-119. Similar information was supplied for the secondary system for nonseismic events in Figure 4.2-122. The comparison showed substantial margins in time and inserted reactivity for all cases.

According to the applicants, the requirements presented in PSAR Figures 4.2-93 and 4.2-94 were based on iterative evaluations of transients, with results for anticipated transients being particularly influential. No details were given on how these iterations were performed, how many and what events were included in the iterations, what trip points and instrumentation delays were assumed, specifically how the requirements were weighted toward the anticipated events, and how the applicants were assured that the most demanding event had been included. However, the applicants did use the required reactivity insertion curves to perform the analyses reported in Chapter 15 of the PSAR, for those events that are sensitive to the speed of response of the PPS. Therefore, if the most demanding transient was not evaluated against the speed-of-response curves, the problem is one of accident delineation, not one of defining the speed of response of the PPS.

To evaluate whether the predicted performances of the PPS as shown in PSAR Figures 4.2-114, 4.2-119, and 4.2-122 conservatively represented the PPS capability, access to the CRAB code and its qualifications is required. Consequently, it was not possible to evaluate whether the predicted performance would be conservative and whether the margin between the requirements and the predicted performance would satisfy the requirement of PDC 8 (GDC 10) with the methods used.

The staff concludes that verification of the methods reviewed can be postponed until the FSAR evaluation because of the following:

(1) There are available fallback positions (changed trip points) if major problems in PPS response are found.

(2) Analysis have been performed to define the speed of response, and analyses show the expected speed of response exceeds the requirements. What is needed is the auditing of those methods.

4.2.1.3.7 Review of Fuel Rod Performance Data Base

The data base reviewed is relevant to calibration and qualification of the design fuel evaluation models and to assessing design margins. The experimental program from which the data base was obtained included the following types of tests:

(1) steady-state irradiations of fuel rods in the EBR-II facility and in the GETR (Cantley et al., 1975)

(2) transient tests of irradiated and unirradiated fuel rods in the transient reactor test (TREAT) facility (Baars, 1980; Hanford, HEDL-TME 75-47; and Barts et al., 1975) and in the sodium loop safety facility (SLSF) (Henderson et al., pp. 2-55 to 2-66, 1981; and Henderson et al., pp.2-67 to 2-78, 1981)

(3) ex-reactor tests on specimens of irradiated and unirradiated fuel rod cladding in the fuel cladding transient tester (FCTT) (Johnson and Hunter, 1978)

(4) tensile, stress-rupture, and immersion density tests on cladding and structural material specimens irradiated in unfueled capsules in the EBR-II facility

The data base extracted from the experimental program consists of the following:

(1) data on cladding corrosion, both inside and outside

(2) fuel rod failure thresholds under steady-state conditions

(3) fuel rod failure thresholds under transient conditions

(4) fuel microstructure changes and fission gas release under both steady-state and transient conditions

(5) cold fuel-cladding gap behavior as a function of steady-state irradiation parameters

(6) cladding response to and failure threshold during constant gas pressure loading while undergoing temperature ramps of 1° to 500°F per second (almost all at 10° and 200°F per second)

(7) mechanical properties for irradiated and unirradiated cladding and duct material

(8) swelling and irradiation creep behavior for cladding and duct material

The range of conditions anticipated in CRBR was reviewed in Section 4.2.1.3.2 of this SER. Ideally, the testing program and its data would have covered the entire range of CRBR conditions with a statistically designed matrix to yield a maximum of prototypic information on behavior and uncertainties from which it would not be necessary to extrapolate to evaluate the fuel design. The idea is achieved rarely in the real world, and the CRBR is no exception. There do appear to be sufficiently serious shortcomings in the data base that additional data are required before either of the fuel evaluation models can be satisfactorily qualified, and before clear margins to failure can be established for the fuel design.

Shortcomings that are applicable to both steady-state and transient data include the following:

(1) There are few steady-state data and no transient data directly applicable to blanket rod performance. (See applicants' response to Question QCS490.9.)

(2) Data on cladding irradiated at temperatures above 1,000°F are very sparse (Johnson and Hunter, 1978), as are data on cladding irradiated to fluences $> 8 \times 10^{22}$ n/cm^2 ($E > 0.1$ MeV). CRBR conditions extend up to temperatures of 1,400°F and fluences of 1.3×10^{23} n/cm^2 ($E > 0.1$ MeV) for fuel cladding, and 1.7×10^{23} n/cm^2 ($E > 0.1$ MeV) for blanket cladding.

(3) Significant atypical factors in the data base are as follows:

(a) All data upon which the CRBR fuel rod design was based were obtained for plutonium contents of 20 to 25% whereas the current design calls for 32% plutonium.

(b) Virtually all EBR-II tests were on fuels in which the U-235 was enriched to between 30% and 65% to achieve the desired power. Therefore, the fissile content was higher than typical, and the ratios of fluence to burnup and of flux to power were lower than typical. Fluence effects may saturate at a higher fluence under typical conditions, probably at a higher level of cladding damage. The fission product spectrum will be different under typical conditions, possibly causing more severe corrosion of the inside cladding surface.

(c) Virtually all data were obtained on short fuel rods (13.5-in. test versus 36-in. typical fuel column length). The axial component of cladding loading may have been significantly downgraded in the short rods.

(d) The axial power profile in EBR-II test rods was flatter than is expected in the CRBR (peak to average 1.1 for EBR-II compared with 1.2 for CRBR). The fuel microstructure will vary more widely within a rod; microstructure has been identified as a significant factor in fuel rod overpower failure threshold.

Deficiencies in the transient data base include the following:

(1) There are no data at all for overpower conditions on fuel rod response and failure thresholds for reactivity insertion rates between about 0.005 cent per second* and 5 cents per second, and very few data between 5 cents per second and 50 cents per second. This gap covers the range of design overpower events. The staff regards this as one of the most serious deficiencies in the entire data base. Data from the W-2 test (a 5-cents-per-second overpower test conducted recently by HEDL in the SLSF) indicate that cladding breaches occurred far earlier than expected.

(2) There are virtually no data on failure thresholds for undercooling conditions that are relevant to the end of life. Most FCTT tests were conducted at too high pressures (2,500 psi or more), or at low fluences and burnups, or both. Only four loss-of-flow tests have been conducted on irradiated fuel rods (W-1 (Henderson et al., pp. 2-55 to 2-66, 1981), L2, L3, and L4 (Deitrich et al., 1974)), and all of these were for low fluence and burnup rods.

In the applicants' response to Question QCS490.20, they indicate that tests in the relevant range of conditions have been conducted. The release of this data for staff review and a schedule for that review are now being considered.

(3) Atypical factors in the transient data base include the following:

(a) All transient integral fuel rod tests were subject to significant radial power depressions. The ratio of maximum (outer surface) to minimum (inner surface or centerline) power ratio ranged from 1.5 to 2.5 for the most relevant tests. In addition, most Argonne National Laboratory seven-rod tests were subject to significant circumferential power variations in the outer rods (variations of 2 to 1 or so). This deviation involves more than just correcting the temperature profiles for the power depression. The processes proceeding within a fuel rod are quite complex, particularly for overpower conditions. The upset of normal temperature relationships may well upgrade the importance of some mechanisms and downgrade others, thus giving a false picture of fuel pin response. Attempts were made to match calculated mean fuel and cladding temperatures in test design, but the staff believes it is important to match the entire radial fuel temperature profile as closely as possible. The staff believes that good matches are possible only for flowing sodium tests in which the power changes relatively slowly (50 cents per second or slower).

(b) Essentially all of the integral rod tests were conducted in TREAT. It is not possible to precondition fuel rods in TREAT before initiating the test transient because TREAT is capable of providing only several seconds of power, at most. The issue at stake is that cracks that opened in the fuel at the last shutdown before the test would not be healed by the time a few seconds at power in TREAT had passed, at which time the transient is initiated. It has been asserted that unhealed cracks would make the fuel weaker in an overpower transient than if the cracks were healed (fused together).

*The rate cited is approximately that of the power-to-melt tests.

(4) Almost all data on the transient response and failure threshold of irradiated cladding, particularly fueled cladding, comes from FCTT tests. These data are all for gas pressure loading and are not necessarily applicable to transient overpower conditions where FCMI may be present. Data are needed that closely simulate transient FCMI conditions (1) as to load distribution caused by friction between fuel and cladding, (2) as to strain rate conditions, and finally (3) as to local loading conditions because of outer surface fuel cracks or misplaced chunks of fuel. The staff acknowledges that FCTT data for fueled cladding are likely conservative for the first two aspects listed. However, FCTT data would not necessarily be conservative for the latter type of loading.

The staff concludes that the following additional information is required if deficiencies noted in the fuel evaluation models are to be corrected and if assessment of the margin to failure or unacceptable performance is to be possible for the FSAR. Information needed on cladding behavior is as follows:

(1) Data on the fuel adjacency effect, including definition of the range of operating conditions where the effect is encountered; definition of the mechanism of cladding failure when the effect is present; and definition of the parameters and magnitude thereof controlling cladding failure.

Resolution: Probably best addressed in out-of-reactor tests using fission product simulants, a continuation of work reported in Hunter and Johnson (1979). A reexamination of past FCTT tests plus performance of additional FCTT tests as needed are required to define further the temperature boundaries that appear to delimit the region where the fuel adjacency effect appears.

(2) FCTT data to confirm the response to loss-of-flow conditions of high fluence cladding under gas pressure loading of 1,000 to 1,500 psi. The cladding irradiation temperature range should cover 900° to 1,400°F.

Resolution: Conduct conventional FCTT tests on fueled cladding irradiated in the cited temperature range pressurized to the 1,000 to 1,500 psi range. A matrix of 25 to 30 tests is suggested. A statistically designed matrix of tests should be considered to maximize the confidence and applicability of the data.

(3) Data on how fueled irradiated cladding responds to overpower type loading.

Resolution: Out-of-reactor testing of fueled irradiated cladding should maximize the amount of information on this subject. Effort should be concentrated on expanding mandrel tests wherein the mandrel surface simulates the fuel surface as closely as possible. A careful study should be made of the feasibility of simulating fuel surface cracks or displaced fuel chunks out-of-reactor.

Information required on integral rod performance includes the following:

(1) Data on fuel response to overpower events in the range of ramp rates from the power-to-melt tests to 10 cents per second.

Resolution: Perform single pin tests at power ramp rates ranging from 0.1 cent per second to 10.0 cents per second in EBR-II using planned techniques. Estimate 6 to 9 tests would be a minimal program. EBR-II offers advantages of preconditioning, flat radial power profile, and capability for ramps slower than 10 cents per second.

(2) Data to evaluate the effects of atypical factors in the transient data base, including short (13.5-in. fuel column) versus long (36-in. fuel column) rods, radial power depression; and lack of preconditioning.

Resolution: Conduct comparative EBR-II single-pin tests (where preconditioning and flat radial power profiles are possible) and FFTF-TREAT single-pin tests (where pins of typical length can be tested). A minimum program would consist of six tests, two EBR-II (short) pins tested in EBR-II, two EBR-II pins tested in TREAT, and two FFTF (long) pins tested in TREAT, all at 10 cents per second. This program would provide a means to evaluate the combined significance of preconditioning and a flat radial power profile against the significance of typical length fuel pins. There is simply no way to conduct transient tests that are typical in all respects at this time.

(3) Data on atypical factors in the steady-state data base, said factors including 25% plutonium content versus CRBR typical 33% plutonium, altered power-to-flux ratio (because of use of EBR-II test rods enriched in U-235), altered fission product spectrum (also because of use of enriched UO_2), and short versus long rods.

Resolution: Surveillance of FFTF fuel will provide information on all but the plutonium content question. Irradiation of one or more assemblies with CRBR plutonium content in FFTF with appropriate post-test examination and appropriate check transient tests should establish what, if any, significant differences are caused by the high plutonium content in the CRBR.

(4) Data on all aspects of the performance of blanket rods, both steady-state and transient.

Resolution: The irradiation of two assemblies of blanket rods (one under internal blanket conditions and one under radial blanket conditions) in FFTF would address this need. Transient tests of selected rods from those assemblies would then address the need for confirmation of transient response data. At least two loss-of-flow and two overpower transient tests should be conducted.

(5) Data on response of "gas leaker" failed rods to a shutdown-startup sequence, and to define if and how such failures progress into the sodium-fuel contact phase, if such rods are to remain in reactor until normal goal exposure.

(6) Data on the response of failed rods, both gas leakers and those showing fuel-sodium contact, to design basis and anticipated event transients, if there is to be any operation with failed rods.

In addition to these data, information is required on the following:

(1) Data on the distortion of rods and ducts, and on rod bundle-duct interaction under prototypic conditions.

Resolution: This data need should be satisfied by surveillance of FFTF assemblies and by FFTF refueling experience.

(2) Data on fatigue endurance limits for irradiated cladding and structural materials, both low and high frequency.

In summary, the staff concludes that although the LMFBR experimental program is massive, there are serious shortcomings in certain areas that must be addressed by the time the FSAR is submitted. A testing program to provide the required data is identified. The staff ackowledges that some of the data identified as being needed may exist now, but are not yet evaluated or are otherwise not available, and that the applicants plan to obtain the balance as evidenced by the responses to Questions QCS490.1, 490.2, 490.6, 490.9, 490.12, 490.15, 490.17, and 490.18. Nevertheless, the staff has stated its position on the additional data required, based on what is now available so that the staff's position can not be mistaken.

4.2.2 Control Assemblies

This section is wholly devoted to an evaluation of the mechanical design of the control assemblies extending from the coupling with the control drive mechanism down into the core.

4.2.2.1 Description of the Control Assemblies

The control assemblies consist of absorber rod bundles encased either in a hexagonal inner duct (primary system) or a circular "guide tube" (secondary system), which is moved up or down in the outer hexagonal duct to accomplish the function of the control system. Tables 4.7 and 4.8 display other design parameters for the primary and secondary assemblies, respectively.

The absorber rods are approximately twice as large as fuel rods, and about the same as blanket rods and have both upper and lower plena. A spacer in the lower plenum and a hold-down spring in the upper plenum maintain positioning of the B_4C pellet column.

4.2.2.2 Design Bases

Control assemblies were to be designed for 328 full-power days (FPDs), with the mechanical design to be adequate for 550 FPD in the event that it is later shown that the nuclear characteristics of the absorber rods are capable of operation for at least two cycles.

Performance was to be evaluated at plant expected operating conditions plus upper 2σ confidence level hot channel factors for normal operation and anticipated events. For unlikely and extremely unlikely events, thermal hydraulic design value conditions plus 3σ upper confidence level hot channel factors were to be used.

Loading mechanisms to be considered for absorber pins included the following:

(1) plenum gas pressure loading

(2) primary and secondary cladding stresses generated by three-dimensional thermal and flux gradients

(3) interaction forces between cladding and wire wrap and between bundle and duct caused by differential growth

(4) fatigue caused by flow vibration and transient operation

Control assembly loadings to be considered were as follows:

(1) loading caused by general distortion of assemblies or seismic events transmitted through the core system

(2) pressure loading

(3) secondary loading caused by steady-state and/or transient operation

(4) loadings applied during refueling by refueling equipment, including a shipping load of 6 g both axially and radially

(5) insertion and withdrawal loading, including scram arrest loading

4.2.2.3 Design Limits

Design limits for absorber pins are given in Table 4.9 and for the balance of assembly in Table 4.10.

In addition, thermal hydraulic limits included the following:

(1) There is no centerline absorber melting at an upper 3σ confidence level for either steady-state with 15% overpower, or for anticipated or unlikely events.

(2) There is a design guideline (see Section 4.2.1.2.3 of this SER for definition of design guideline) of 1,600°F maximum cladding midwall temperature shall be adopted for unlikely events.

(3) No coolant boiling is to be allowed, based on local saturation temperature.

(4) Hot channel factors shall include allowance for the differential radial growth between rod bundle and inner duct or guide tube.

Clearance requirements are also specified as identified in Table 4.2-36 in the PSAR on pp. 4.2-383 through 4.2-386. These requirements have the standing design limits (except for minor aspects). One of the requirements in the referenced table is that there should be enough clearance between absorber pellets and cladding to preclude mechanical interaction between the two.

4.2.2.4 Evaluation of Design Bases and Limits

The listed loading mechanisms for absorber rods and for the balance of assembly account for all of the loadings treated for fuel and blanket rod assemblies, plus the special loadings applicable to insertion and withdrawal of the control assemblies. The staff concludes that these provide a satisfactory bases for design of the control assemblies.

The staff concludes that specifying a clearance requirement to preclude pellet-cladding mechanical interaction provides substantial conservatism to the absorber rod design.

The limits cited in Table 4.9 for absorber pins are based primarily on ASME Code Case 1592. Most of the limits as cited in that table appear to be defined in the code case. This particular code case was redesignated Code Case N-47, and is listed in the 1980 Edition of the ASME Code.

Code Case N-47 is specifically applicable to high-temperature applications where thermal creep may be significant, but makes no allowance for irradiation effects. The applicants explicitly address effects of irradiation in design limits for the balance of assembly as set forth in Table 4.10, but for the absorber rods as set forth in Table 4.9, address irradiation effects only through the CDF adaptation to control assemblies (see next paragraph). The applicants indicated, however, that allowance for irradiation was made in the design of absorber rods, but Table 4.9 as constructed inadvertently omitted specifications in that regard.

One limit cited in Table 4.9 that is not related to the ASME Code is that the CDF modified to include appropriate material properties shall not exceed a value of 1.0. Because the applicants intend to avoid absorber cladding mechanical interaction, the staff concludes that application of the CDF model to absorber rods should be acceptable provided that significant interaction between the B_4C absorber pellets and the cladding does not occur in the absence of contact pressure between the two. The material properties used for the absorber rod version of the CDF model were not available for review, but the staff concludes that their review can be postponed until the FSAR is prepared.

The design limits for the balance of assembly shown in Table 4.10 appear to be reasonable and represent prudent engineering practice. However, for uniform strains >5% when the temperature is >800°F, the limits are based on ASME Code Case 1592. Also, for strains >5% and temperatures <800°F, fatigue evaluation based on unirradiated material modified to reflect ductility and fracture data for irradiated material was deemed acceptable. The staff concurs with this approach for purposes of PSAR evaluation. For the FSAR, the applicants should demonstrate that this approach does in fact conservatively reflect fatigue performance of irradiated material.

The staff concludes that the limits based on the ASME Code and Code Case N-47 as modified to account for irradiation effects are acceptable, but for the FSAR should be modified to reflect high and low frequency fatigue data on irradiated material, or demonstrate that the limits used conservatively represent the effects of irradiation.

4.2.2.5 Review of Control Assembly Performance Evaluation

As was previously indicated, the material properties that were used in the control assembly version of the CDF model were not availabe for review, thus precluding a definitive evaluation of the CDF model in this application. However, subject to verification that the properties used were conservative and presuming that absorber cladding mechanical interaction will be avoided, the staff believes that the CDF model is acceptable for absorber rod performance evaluation.

The information provided in the PSAR on the details of how absorber rod and balance of assembly performance evaluations were conducted was minimal, and the staff is unable to evaluate the methods and procedures that were used. However, a substantial data base exists (although it was not available for review), and additional significant testing is planned. This includes the irradiation of a primary control assembly as an actual rod in the FFTF under nearly prototypic conditions. Further, the conditions of irradiation are less severe for the absorber rods than for the fuel rods (no mechanical interaction loads, less exposure to high temperature). Hence, the staff believes that there is a reasonable expectation that the control assemblies will perform satisfactorily in the CRBR.

4.2.3 Liquid Metal/Materials Compatibility

Introduction

In a nonisothermal flowing sodium/stainless steel system, iron, chromium, and nickel are dissolved from the high-temperature regions and deposited in the lower temperature regions because of supersaturation. Included in this process of mass transfer is the formation and decomposition of various transition metal and sodium double oxides. In addition, the formation of a double oxide would also affect the oxygen activity in the sodium system is the double oxide is thermodynamically more stable than the Na_2O. The fuel and blanket assemblies of CRBR should be designed to withstand the high-temperature sodium environment. Various cladding degradation mechanisms that have been identified include:

(1) selective leaching of chromium and nickel from austenitic stainless steel surface resulting from high-temperature sodium exposure

(2) degradation of material strength resulting from loss of interstitials such as carbon and nitrogen

(3) loss of material resulting from fretting and wear

In the design of fuel rods, cladding wastage rates are used in the various analytical models which are used to evaluate fuel rod performance. The sodium corrosion equations used for PSAR analysis are based on in-pile and out-of-pile experimental data obtained in operating LMFBRs. Fretting wear is important because it was observed in a few EBR-II test assemblies and a wastage allowance is considered for the CRBRP fuel and radial blanket rod strain analysis. Fretting wear is not well understood at this time but the approach is to design pin bundles with sufficient tightness to preclude fretting wear and to provide a design allowance for it in the event fretting does occur.

Areas of Review and Acceptance Criteria

The primary functions of a fuel assembly are (1) to provide, protect, and position the nuclear fuel of the CRBR to generate heat and (2) to provide neutrons for breeding plutonium in the core and surrounding blankets. The primary functions of the radial blankets assembly are (1) to provide, protect, and position the fertile material around the core conversion to plutonium and (2) to produce heat for the primary heat transport system.

For the CRBRP, cladding allowance for the fuel elements, fuel blankets, and control rods should be designed to limit the consequence of sodium chemical reactions. Means to detect sodium reaction products and to control the sodium purity should be provided to limit and control the extent of sodium attack to ensure the cladding integrity and the functions of components important to reactor safety. Means should be provided to assess the cladding integrity as a function of CRBR operating conditions and to limit the release of fission products and sodium reaction products to the environment to protect plant personnel and to avoid undue risk to the public health and safety.

Cladding alloy should be resistant to sodium corrosion which includes selective leaching of particular alloy constituents and the formation of double oxides. Furthermore, the cladding alloy should be resistant to loss of strength resulting from interstitial transfer, fretting, and wear. The corrosion behavior of the cladding alloy, including general and localized attack, must be characterized. In addition, experimental data and analytical models should be included and discussed for cladding wastage allowance calculations.

Safety Evaluation

The environmental and material consideration addressed here pertains to "cladding wastage." As defined by the applicants, cladding wastage denotes both material loss and degradation of properties resulting from the high temperature, high energy neutron flux, and sodium exposure.

The corrosion of stainless steel in sodium proceeds by two mechanisms: (1) selective removal of chromium and nickel by diffusion and (2) physical removal of the surface material. In addition, depletion of carbon and nitrogen from the cladding affects the strength of the cladding material. Sodium-cladding interaction is defined as any cladding degradation mechanism caused by the sodium.

(1) Sodium Corrosion

The formulation describing the surface corrosion of type 316 stainless steel in flowing sodium was developed (Bagnall and Jacobs, May 1975). A total of 64 data points were evaluated with temperatures ranging from 914°F to 1,337°F and velocities ranging from 2.8 to 40 ft per second.

The oxygen contents associated with the above data were reported in terms of the various measurement techniques available: vacuum distillation, mercury amalgamation, plugging meters, cold-trap temperature, and vanadium wire equilibration. The data, in terms of the reported oxygen contents, show considerable scatter. However, in Whitlow's paper (Oct. 1977) all oxygen concentrations were adjusted to a vanadium wire equilibration (VWE) value thus providing an internally consistent oxygen parameter.

As corrosion relates to the sodium velocity and VWE oxygen content the above study shows that, with velocity greater than 10 ft per second, the corrosion rate is independent of velocity and linearly dependent on the oxygen content. At velocities of less than 10 ft per second the corrosion is linearly dependent on velocity.

The data with velocities greater than 10 ft per second (52 data points) were fit by regression techniques to the Arrhenius equation:

$$R_c/C_o = 6.68525 \times 10^{-7} \text{ Exp } (-18{,}120/T_k) \tag{1}$$

where

R_c = corrosion rate (mils/yr)

C_o = VWE oxygen concentration

T_k^o = temperature (°K)

Although Equation (1) does not account for the influence of the axial thermal gradient that would exist in an operating fuel rod, under the condition of an axial gradient the solubility of the oxygen in sodium increases; therefore, the corrosion rate might be greater than that predicted. However, Weber (1967) reports that fuel rods with extremely high axial gradients had the same corrosion rate as out-of-pile tests with no thermal gradients. Thus it can be concluded that Equation (1) can be applied to fuel rods with large axial gradients and is conservative for predicting corrosion wastage of fuel cladding in CRBRP.

(2) Depletion of Chromium and Nickel

Dissolution of alloying constituents into liquid sodium is the basis of the corrosion of stainless steel in high-purity sodium. At constant temperature, the process proceeds until the chemical potential for a given alloying element reaches equilibrium between the two phases. This simple dissolution process may be extended to explain mass transfer in a forced-circulating, nonisothermal sodium/stainless steel coolant system.

Mass transfer in the CRBR system will be controlled mainly by two factors: (a) the existence of a concentration (actually the activity) gradient of an element in the system and (b) the presence of a thermal gradient. The driving force for the concentration-gradient effect is the tendency of the system to reach equilibrium with respect to the chemical potential of the element involved. For the temperature-gradient effect, the driving force is the temperature-dependence of the solubility of the element in sodium.

It is found for the 300 series austenitic stainless steels that the most pronounced effect is the selective depletion of nickel, chromium, and manganese resulting in the change from an austenitic to a ferritic surface layer. As the layer of depleted ferrite grows thicker, the rates of removal of nickel and chromium decrease until a steady state is reached. At this stage the thickness of the depleted zone remains constant and the composition of the material removed by the sodium at the sodium-stainless steel interface is approximately equal to the composition of the alloy. Since there are differences in the chemical compositions of the various austenitic stainless steels, variations in depletion depth and composition gradients near the sodium-alloy interface are expected.

The depth of the substrate depleted of chromium and nickel is dependent on the relative rates of the two associated competing processes, namely, atomic diffusion and sodium corrosion. Given a constant outer surface, the distribution of the diffusing substitutional elements may be approximated by

$$C(x,t) = C_s + (C_o - C_s)\ \mathrm{erf}\ [X/2(Dt)^{\frac{1}{2}}] \qquad (2)$$

where

C_s = concentration at the constant sodium interface
C_o = initial concentration
D = diffusion coefficient
t = time
X = distance from the alloy-sodium interface

The solution to the diffusion equation is valid so long as the physical system is infinite. Since the type 316 stainless steel fuel cladding thickness is significantly thicker than the diffusion zone, Equation (2) can be used to estimate the degree of chromium and/or nickel depletion at the cladding surface.

In CRBRP-ARD-0147 (Travis, Oct. 1977), the depth of chromium- and nickel-depleted substrate relative to the receding sodium-cladding interface is given as:

$$\chi_n = \chi(C_o) - R_c t \qquad (3)$$

where

χ_n = depth of the depleted substrate
C_o = concentration at the inner boundary of the depleted zone
R_c = corrosion rate
t = time of exposure

(3) Carbon Transport

The direction of carbon movement in the CRBR coolant system will depend on the temperature gradients and the materials present. Interstitials tend to transfer from ferritic to austenitic stainless steels and from unstabilized to stabilized grades, and also from hot-leg regions to cold-leg regions. The loss of chromium and nickel from the cladding or piping surface is treated as a component to the cladding or piping wastage, whereas the changes in carbon and nitrogen concentrations are treated as a loss of strength of the cladding.

In general, experimental results show that high interstitial sodium exposure increases the strength and decreases the ductility of stainless steels when compared with helium-exposed controls. Furthermore in a sodium environment, the interstitial effect is a time variant because the interstitial concentrations are continually changing via diffusion to or from the sodium. Therefore, as discussed in CRBRP-ARD-0147 (Travis, Oct. 1977), the model describing the sodium strength modification has two component features. The first deals with the relationship between strength and interstitial content and the second treats the kinetics of the changes in concentration.

The normalizing function describing the relative isochronous rupture strength is given by:

$$F_c = \delta r(c)/\delta r(C_o) \tag{4}$$

where

C_o = reference interstitial content
C = interstitial content of interest
$\delta r(C_o)$ = reference isochronous strength
$\delta r(c)$ = rupture strength with C

In the case of CRBR the reference interstitial content (C_o) will be taken to be 0.06 wt%. In this case Equation (4) becomes

$$F_c = 0.55754 = 9.15817\ C - 29.730\ C^2 \tag{5}$$

(4) Fretting and Wear

Because of differential thermal expansions and flow-induced vibration, fretting and wear of the CRBR fuel rod cladding might be caused by relative movement between the wire wrap of one rod and the cladding of the adjacent rod. In WARD-D-0166 (Gorkisch, Dec. 1977), it is concluded that flow-induced vibrations of the fuel rods are not expected to be significant and will not cause damaging fretting or wear. Furthermore, the applicants indicated that the CRBR fuel assembly rod bundles will have porosity sufficiently small to eliminate or prevent the potentially damaging vibrations. Therefore, fretting and wear of the CRBR fuel cladding will be caused mainly by differential thermal expansions.

The basis for the wastage allowance for fretting and wear was addressed in the PSAR. A 2.5-mil wastage at end of life (0.5 mil at beginning of life plus 2.0 mils linearly applied during life) was proposed. The magnitude of the fretting wear wastage allowance was based on type 316 stainless steel pin-disc tests performed at Westinghouse (Bowen, May 1970). However, these results were obtained under geometrical conditions nonprototypic of the CRBR cladding and wire wrap combination. At the operating license stage, the applicants should provide more supportive data typical of the FFTF operating experience and materials to confirm the 2.5-mil fretting and wear allowance.

Summary

As discussed in Section 4.2.1 of this SER, two analyses are performed for the fuel and radial blanket cladding: a strain analysis and a cumulative damage function (CDF) analysis. Wastage resulting from sodium corrosion is accounted for in both analyses. However, no allowance is made for chromium, nickel, carbon, and nitrogen depletion for the strain analysis. Because the fretting and wear correlation generally results in a greater cladding wastage than the chromium- and nickel-depleted zone, it would be conservative to consider just the wastage caused by fretting and wear. As indicated by the applicants in the PSAR, wastage allowances for sodium corrosion are consistent with values used in the LIFE-III code. Although the staff accepts this analysis for the present construction permit review, it will require that the FFTF operating experience and materials data be factored in to verify this allowance for the subsequent operating license review.

Chromium carbide coatings in the CRBRP will be on removable assembly load pads as described in PSAR Section 4.2.1.2.2 and on the secondary driveline/control assembly coupling described in PSAR Section 4.2.3.1.7. There are no design criteria and no "limits" identified or established for carbon transfer in the primary coolant system that relate specifically to carbide coatings present on certain parts of the primary control rod system (PCRS). The staff agrees with the applicants' contention that the concern over the possibility that carbide coatings would produce carburization of core component arises, not because of the chromium carbide itself, but mainly because of free carbon that may be present in the binder material. However, because the coating contains a large amount of free chromium from the nichrome binder which tends to soak up the free carbon during high temperature sodium exposure, it will not result in significant carbon release to the sodium, and because the surface area of the PCRS carbide coating is a negligible fraction of the total surface area exposed to the sodium in the primary system, carburization of core components (including the fuel cladding) by the presence of chromium carbide in the PCRS will not be significant. The staff accepts this conclusion for the present construction permit review. The applicants should provide the FFTF operating experience and materials data to confirm this analysis for the subsequent FSAR review for the CRBR at the operating license stage.

In the PSAR, the applicants proposed a 2.5-mil wastage allowance for fretting and wear. This allowance for fretting and wear was based primarily on results obtained from tests carried out at Westinghouse Electric Corporation on type 316 stainless steel combinations of the pin-disc design. Similar tests also were conducted in Germany. Although the allowance is acceptable for review of the CRBR construction permit, the applicants should provide the applicable FFTF operating experience and materials data for verification at the operating license stage.

4.2.4 Summary and Conclusions on Fuel Design

The staff has identified its primary concerns as well as summarized the results of the review in this section. If these concerns are not adequately addressed in the FSAR, it may be necessary to resort to one or more of the fallback positions that are identified.

The staff concludes that the design bases and the system of design limits for fuel and blanket assemblies are acceptable with two exceptions, provided the deficiencies of the fuel evaluation models and the data base are addressed successfully, and that the use of umbrella events and thermal screening guidelines are demonstrated to be conservative. The two exceptions are the applicants' assertions (1) that coolability is guaranteed as long the cladding does not melt, and (2) that observation of a no-boiling guideline is sufficient to preclude attainment of whatever ultimate limit is adopted to ensure coolable geometry. The staff maintains that a fuel enthalpy limit (or other limit more relevant to TOP conditions than coolant or cladding temperature) is also required to preclude the expulsion of molten fuel in overpower events, which tests have shown generally occurs before boiling and sometimes below the cladding temperature design guideline of 1,600°F. The staff believes that the current design coupled with PPS scram limits now proposed would easily comply with such a limit. Although the staff did not identify specific evidence (in a brief review of data) that coolable geometry would be impaired for loss-of-flow conditions short of cladding melting, the staff is very dubious that coolable geometry would not be seriously affected if cladding temperatures ever did approach melting. The staff strongly recommends that the applicants replace the cladding melting criterion with a cladding temperature limit that provides a significant margin to melting and to any irreversible path that could lead to melting.

The staff concludes that for the construction permit the applicants need only to commit to a fuel enthalpy limit (or other limit related to overpower phenomena) to preclude expulsion of molten fuel and to a cladding temperature limit that provides substantial margin to cladding melting and ensures that irreversible paths leading to cladding melting are precluded. The provision of actual limits and of detailed bases for both limits is deferrable to the FSAR. The applicants have committed to provide such limits and the bases therefore for NRC review prior to submittal of the FSAR.

Deferral of the limits and bases to the FSAR is based on the existance of substantial data bases for both limits, on the fallback position of converting the non-boiling guideline to a nonviolable limit, and on the fact that the PPS overpower trips essentially preclude fuel melting for design-basis events. Also, to address other needs, the applicants plan (and the staff is calling for them) to obtain other data that will be relevant to the bases for these limits. See Section 4.2.1.2.4 for a detailed presentation of staff concern over the cladding melting criterion and the need for an additional provision besides a no-boiling guideline.

The staff concludes that the CDF model is not acceptable now as a means to evaluate the fuel and blanket rod performance. The staff reached this conclusion because the CDF model does not now reflect the fuel adjacency effect (see Section 4.2.1.3.2.3 for a discussion of fuel adjacency),it has not been qualified against integral fuel pin tests, and because property uncertainties (particularly mechanical properties) are treated with confidence bands on the mean rather than tolerance limits that cover the data. The last named point is discussed in detail in Section 4.2.1.3.3. The staff also concludes that the DLS model is not acceptable now as a means to evaluate fuel and blanket rod performance because of inadequate qualification to data and because of failure to demonstrate a clear margin to failure under the full range of CRBR conditions. The staff has also noted reservations on some aspects of auxiliary models and correlations, most notably, in the section on cladding wastage (Section 4.2.1.3.2.3). If the fuel system cannot be shown to be acceptable with revised models, there are fallback positions available as noted at the end of this section. Because there are fallback positions available, the staff believes that it need not be shown that the models have been made acceptable until the FSAR is being prepared. The applicants have committed to address the enumerated fuel model issues in the FSAR.

The staff concludes that the data base is not complete enough to qualify the evaluation models to all design events or to the full range of CRBR conditions and is not complete enough to allow a full evaluation of the margin to failure. Also, the data base contains atypical factors that cannot be evaluated adequately now. The staff is also particularly concerned that experimentation be conducted as necessary to understand the "fuel adjacency" effect, to evaluate atypical factors in the data base, and to understand how fuel and blanket rods respond to "slow" TOP events. Additional experimentation was identified in Section 4.2.1.3.7 to satisfy the data base deficiencies, much of which is already planned by the applicants. The additional data specified should provide an adequate basis for correcting modeling deficiencies, to allow full qualification of the performance models, and to fully assess the margin to failure. The applicants plan an experimental program as detailed in the answers to submitted questions that substantially addresses the issues of concern.

The CRBR was designed to operate with failed fuel. An experimental program is under way in EBR-II and in TREAT (transient tests) to investigate the behavior of failed fuel under continued operation; some results are now available. Continued operation with failed fuel for at least a limited period may be very beneficial economically because of the sizable outage time required to cool down, position the refueling equipment, remove the offending assembly, then prepare for operation and start up. The staff believes nevertheless that the applicants should remove "gas leaker" failures at the first outage occurring for any reason after detection of the failure. A controlled shutdown should be initiated at once and the offending assembly should be removed upon indications of a fuel-sodium contact failure as evidenced by a generally increasing delayed neutron signal. These restrictions should remain in effect until all questions surrounding operation with either type of failure have been resolved (see Section 4.2.1.3.2.6) and the applicants have determined firm limits on operation with failed fuel that the NRC has accepted.

With regard to design of control assemblies, the staff was unable to make a complete review of methods of evaluating performance; neither was the staff able to review the data base on irradiation behavior thus far, although the applicants committed to supply test documents to support the FSAR. Nevertheless, it is clear that the design is conservative. This is based on the avoidance of absorber pellet-cladding mechanical interaction, and on the low cladding end-of-life stress caused by released helium. Further, irradiation of a full CRBR control assembly as a control rod is to take place in the FFTF. The staff concludes that there are reasonable expectations for absorber rods to perform well under CRBR conditions for the present design life of 328 full-power days and that further review is not required prior to issuing a CP.

With regard to the response of the core components to seismic events, the staff concludes that the applicants demonstrated that duct walls would not fail for either the OBE or the SSE. The staff found no evidence that the applicants had analyzed fuel or blanket rod assembly response to such events, but conclude based on an approximate analysis conducted by the staff that no significant damage was likely as long as the duct remained intact. The staff also concludes that as long as the duct walls would remain intact, there would be little chance that the control assemblies could not be inserted. The applicants should provide a detailed analysis of rod bundle response to seismic events for the FSAR, and have committed to do so. In its review, the staff did not find evidence that the applicants had considered temperature uncertainties, although they maintain that uncertainties were considered. The applicants should either provide documentation showing they have considered uncertainies, or repeat their evaluation, considering uncertainties, and document this evaluation. Reduction of operating temperatures is a reasonable fallback if acceptable performance can not be demonstrated with more conservative temperatures. Therefore, the staff concludes that clarification of this area can be deferred to the FSAR.

The staff concludes that there is a reasonable basis to expect that the fuel design will prove successful. This is based on data that show the following:

(1) The cladding failure threshold for irradiated rods is conservative under loss-of-flow conditions with respect to the design guideline (1,600°F) out to an exposure of 50,000 MWd per ton by a margin of at least 200°F (Baars, HEDL-TME 75-40).

(2) The thermal conditions of fuel and cladding at failure thresholds for irradiated rods for rapid overpower events as determined in TREAT tests are far above the peak thermal conditions allowed in design-basis events by the CRBR PPS (Baars, 1980).

(3) Cladding failures for EBR-II test rods were very sparse; most of those that did occur were caused by atypical factors such as reconstitution of rods into assemblies for continued irradiation after interim examinations (Weber, Almasey, and Karnesky, 1979).

In addition, substantial experience will be gained with the FFTF fuel system (almost the same as the CRBR system) before FSAR submittal.

Although there is a reasonable basis to expect the fuel system to be successful, there is concern that the emergence of significant fuel performance problems is not precluded. These could include the following:

(1) Most aspects of performance affected by plutonium content may extrapolate smoothly from the 25% in the data base to the 33% of CRBR design, but it is not guaranteed that all will. Further, the combined effects of several properties affected by the change may become significant.

(2) There may be unanticipated surprises within the slow overpower event range for which there are no data now.

(3) Blanket rods may be subject to unanticipated behavior because of their unique power history (continuously increasing power to a maximum at the end of life). (See Section 4.2.1.3.2.7.)

(4) There may be presently unanticipated implications on fuel rod behavior once the fuel adjacency effect is fully understood.

If any of these potential problems are realized and other approaches do not resolve the problem, fallback positions include the following:

(1) reduction of goal exposure
(2) reduction of peak power

(3) lower operating temperature
(4) adjustment of PPS trips

The staff concludes that with the execution of the identified experimental program, all outstanding issues in the fuel design area can be resolved and the CRBR fuel system will perform well; significant amounts of radioactivity will not be released; and neither accidents nor earthquake-induced loads will render the system unable to cool the fuel, or to insert the control rods.

4.2.5 Reactivity Control

4.2.5.1 Introduction

Section 4.2.3.1 of the PSAR contains information pertaining to the CRBR protection and reactivity control systems, their design bases, their test program, and their acceptance criteria.

4.2.5.2 Acceptance Review

The staff has reviewed the applicants' proposed design, design criteria, test program, and design bases for the reactivity control system for CRBRP. SRP Section 3.9.4 provides general acceptance criteria and review procedures for the staff review. These acceptance criteria include the following principal design criteria: PDC 1, 2, 12 20, 21, 23, 24, 25, and 58.

In addition, the regulatory guides identified in Section VI of the SRP also apply to CRBRP.

4.2.5.3 Description of Applicants' Proposed Design

4.2.5.3.1 Reactivity Control

Reactivity control will be accomplished by means of two independent control rod systems, called primary and secondary, using boron carbide as the neutron absorber, inserted axially into the core from the top.

4.2.5.3.2 Primary Control Rod System

The primary control rod system will consist of nine individual control rods each containing 5.57 kg (nominal) of boron carbide pellets stacked 36 in. vertically within tubular pins in bundles of 37 pins.

4.2.5.3.3 Secondary Control Rod System

The secondary control rod system will consist of six individual control rods each containing 5.0 kg (nominal) of boron carbide pellets stacked 36 in. vertically within tubular pins in bundles of 31 pins.

4.2.5.3.4 Control Rod System Mounting

Both control rod systems will be mounted on the reactor closure head. The control rod drive mechanisms for both primary and secondary systems will be mounted in nozzles in the reactor head.

4.2.5.3.5 Control Rod Driveline Location

Individual control rod driveline assemblies will extend within shroud tubes from their respective control rod drive assemblies to the proximity of the top of the reactor core. The guide tubes will be restrained laterally by weldments to the nozzle extensions at el 8.125 in. and by weldments to the upper internal structure at el 341.65 in. The drivelines for both primary and secondary control rod systems will terminate in mechanical connectors which attach to the control rods.

4.2.5.3.6 Control Rods

The control rods will be housed within individual control assemblies configured as hexagonal ducts, mounted on the core support plate among the fuel elements. The control assemblies will extend above the fuel elements and the control rods will be moved vertically within the control assemblies.

4.2.5.3.7 Control Rod Movement

When located in the full-out, parked position, the boron carbide absorber will be a nominal 1½ in. above the top of the fuel plane of the core. At the full-in position, the boron carbide pins will extend equidistant above and below the core midplane.

4.2.5.3.8 Primary Control Rod System Capability

The primary control rod system is specified to be capable of effecting hot shutdown with eight of the nine rods functioning. Further, the primary control rod system is specified to be capable of effecting hot shutdown and maintaining control without benefit of the secondary control rod system.

4.2.5.3.9 Secondary Control Rod Capability

The secondary control rod system is an independent redundant and diverse shutdown system. During normal reactor operation, the secondary control rod system is specified to be in the parked position; that is, raised out of the core to a point where the bottom of the control rods are a nominal 1½ in. above the top of the core. The secondary control rod system is specified to be capable of scram and maintaining hot shutdown condition with five of the six control rods operating and without any benefit from the primary control rod system. The applicants stated at the ACRS meeting September 30, 1982, on instrumentation and control, the secondary control rod system will be capable of shutting the reactor down to cold shutdown.

4.2.5.3.10 Primary Control Rod system Based on FFTF

The primary control rod system will be similar to the fast flux test facility (FFTF) reactivity control system. Changes have been made to the FFTF configuration to accommodate the CRBRP requirements and configuration. The primary control rod drive mechanism will employ a single lead screw, roller-nut design powered by a six-phase, four-pole, variable reluctance, dc stepper motor. The primary control rod drive motor stator will be cooled with nitrogen gas at 2-15 psig. The motor tube itself will be pressurized with argon gas at 15 psig.

4.2.5.3.11 Secondary Control Rod System Tailored for CRBR Plant

The secondary control rod drive mechanism will employ a dual lead screw, ballnut drive mechanism, driven by an argon-cooled ac electric motor.

4.2.5.3.12 Applicable Codes and Standards

The applicants have stated the control rod drive mechanisms will be designed to conform to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition, together with applicable Code Case 1592, supplemented by appropriate reactor development technology and industrial work standards for welding materials and processes and materials for service in a liquid sodium environment.

4.2.5.3.13 Control Rod System Materials Chosen Based on FFTF and Light-Water Experience

The applicants have stated that the materials specified for the control rod drive mechanisms will be selected on the basis of FFTF experience and applicable light-water reactor experience.

The applicants have stated that as a result of development tests performed to verify critical wear and friction applications such as bearing, guide bushings, and the torque-restraint device the materials specified include: 300 series stainless steels: 304, 316, Inconel 600; Inconel 718; Inconel 750; type 17-4 ph stainless; 400 series stainless steels: 410, 440C; and Haynes Alloy No. 25. The design life is 30 years.

4.2.5.3.14 Control Rod Drive Mechanism Mounting and Bellows Isolation Systems

The control rod drive mechanisms will be installed in reactor head nozzles and will be isolated from the sodium environment by bellows. The motor side of the bellows will be pressurized with argon gas at 15 psig. The bellows systems will progress from the nozzle to each of the three concentric shafts on the secondary control rod system. The bellows on the primary control rods essentially will isolate the drive line shaft threads and the scram spring from the sodium atmosphere.

The primary control rod drive motor system will be pressurized for cooling the motor using nitrogen at 5 psig (nominal). A separate pressure system using argon will pressurize the upper drive line inside the bellows system. The lead screw will be protected from sodium vapor by a bellows welded to the upper bellows support in the nozzle. The lower end of the bellows will be welded to the lower bellows support.

The secondary control rod system drive line will be pressurized internally with argon at 60 psig (nominal) between the drive shaft and sensing tube, from the upper bellows between the sensing tube and the drive shaft to the lower bellows between the sensing tube and the drive shaft. The bellows system in the condary control rod upper and lower drive lines consists of five individual bellows: the belows connecting the upper sensing tube and the drive shaft; the main shaft bellows; the bellows connecting the upper tension rod with the sensing tube; the bellows connecting the lower tension rod with the sensing tube; the bellows connecting the lower sensing tube and the drive shaft.

4.2.5.4 Staff Evaluation

The staff finds that the proposed primary and secondary reactivity control systems are of largely different design principle and that either system can reliably respond to abnormal conditions including appropriate margin for malfuntions such as a stuck rod. In addition, the staff finds that the proposed two systems have incorporated sufficient functional diversity and diversity in design to ensure that abnormal conditions will not produce the loss of the protection function. The applicants have provided a large degree of independence in the design of the primary and secondary reactivity control systems.

The applicants have committed to and have initiated an extensive test program for both the primary and secondary reactivity control systems. The testing includes perofrmance tests, tests during plant startup, and acceptance tests. The staff has reviewed the testing during plant startup, and acceptance tests. The staff has reviewed the testing and finds that all tests on the primary rods have been successfully completed to date and that most tests in the test program have been completed. All tests on the secondary rods have also been successfully completed to date, but the testing program for all secondary rods is somewhat behind the program for the primary rods. The staff review at that time will ensure that the testing and test results are sufficient and equivalent for both the primary and secondary reacitivity control systems. The staff will require diversity of maintenance for the primary and secondary reactivity control systems.

Seismic qualification testing of the primary rods is currently under way and should lead to the successful seismic qualification of the primary rods. The staff will confirm this during the operating license review. The applicants have committed to seismically qualify the secndary control rod system. The staff finds this acceptable and will confirm the results of the seismic qualification program during the operating license review.

4.3 <u>Nuclear Design</u>

The review of the nuclear design of the Clinch River Breeder Reactor (CRBR) was based on information supplied by the applicants in the PSAR, in documents referenced in that report, and in meetings. This Safety Evaluation Report was based in large part on a review conducted by Los Alamos National Laboratory (LANL) under contract with the NRC.

4.3.1 Design Bases

The nuclear design was reviewed against the CRBR Principal Design Criteria (PDC) which were developed by the NRC CRBR Project Office for use in this review. Part of the nuclear design review consisted of an evaluation of these criteria and the design bases derived from them by the applicants. Each criterion which applies to the nuclear design is discussed below.

<u>PDC 8--Reactor Design</u>

This criterion requires that the reactor be so designed that specified acceptable fuel design limits are not exceeded during normal operation and anticipated transients. This implies the establishment of specified acceptable fuel design limits. These are discussed in Section 4.2 above. The nuclear design basis following from this criterion is that the power distribution peaking factor shall be limited so as to prevent violation of limits. This is an acceptable design basis.

PDC 9--Inherent Reactor Protection

This criterion requires that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity (emphasis added). In light-water reactors, this has been interpreted to mean that the fuel temperature coefficient (doppler) must be negative at all power levels for which significant reactivity feedback occurs. The moderator temperature coefficient is not required to be negative. However, it is generally true that the total power coefficient is negative throughout the power range from essentially zero power to full power.

The CRBR meets the doppler coefficient requirement, but at low power the total power coefficient may be positive. This is acceptable if account is taken for this effect in the analysis of core stability and low power transients.

PDC 10--Suppression of Reactor Power Oscillations

This criterion requires that power oscillations that can result in conditions exceeding specified acceptable fuel design limits be not possible or be readily detected and suppressed. This is an acceptable design basis.

PDC 11--Instrumentation and Controls

This criterion requires that instrumentation be provided to monitor various parameters and systems that impact on reactor safety along with controls to ensure that the parameters and systems are maintained within prescribed operating ranges. This is an acceptable design basis for CRBR.

PDC 18--Protection System Function

This criterion requires that the protection system be automatically initiated to prevent exceeding specified acceptable fuel design limits during anticipated operational occurrences and to sense accident conditions and initiate the operation of systems and components important to safety. This is an acceptable design basis for CRBR.

PDC 23--Protection System Requirements for Reactivity Control Malfunctions

This criterion requires that the protection system be designed so that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system such as accidental withdrawal of control rods. For light-water reactors, control rods have been defined to be the assemblage of control elements that are moved by a single control rod drive mechanism. The same definition applies to the CRBR. It should be noted that the control rod withdrawal event is given as an example. Other reactivity control system events must be considered and the rod withdrawal event may be excluded from analysis if more than a single failure is required to initiate it. For example, for PWRs, the single-failure event is the accidental withdrawal of the control rod bank. Withdrawal of a single rod requires multiple failures, and exceeding specified acceptable fuel design limits is acceptable for the event.

This criterion, as applied to light-water reactors, is interpreted to mean that the primary scram system (the rods) must be able to prevent exceeding specified acceptable fuel design limits initiated by a single failure in the reactivity control system (rods, boron bleed and feed system, flow controller, etc.). No action by the secondary system is assumed.

The design basis for the CRBR--that either system be capable of bringing the reactor to hot standby temperature (with a stuck rod and failure of the other system to scram) in the presence of the limiting reactivity insertion event--meets the requirement of PDC 23 and is acceptable.

PDC 24--Reactivity Control System Redundancy and Capability

This criterion requires two independent reactivity control systems of different design principles to be provided. Both systems are required to reliably sense and respond to off-normal conditions and to contain margin for malfunctions such as a stuck rod. For anticipated operational occurrences, one system, acting alone, must prevent the violation of specified acceptable fuel design limits and the other system acting alone, must ensure that the capability to cool the core is maintained. Each system must be capable of bringing the reactor to hot shutdown with allowance for the maximum reactivity associated with any anticipated operational occurrence or postulated accident. One system must be capable of holding the reactor subcritical for any temperature lower than the hot shutdown temperature.

This criterion is more conservative than GDC 26 of 10 CFR 50 and is acceptable. The discussion of diversity and redundacy of the two systems is presented in Section 4.2.4.3 of this SER.

PDC 25--Combined Reactivity Control Systems Capability

This criterion requires that the reactivity control systems have a combined capability of reliably controlling reactivity changes to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained. The control system design basis (see discussion under PDC 23) satisfies this criterion and is acceptable.

PDC 57--Reactivity Limits

This criterion requires that the reactivity control system be designed with appropriate limits on the amount and rate of increase of reactivity to ensure that during postulated accidents only limited damage to the core boundary occurs and the capability to cool the core is not impaired. The staff concludes that this is an acceptable criterion for CRBR.

PDC 58--Protection Against Anticipated Operational Occurrences

This criterion requires that the reactivity control and protection systems be designed to ensure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. The staff concludes that this is an acceptable criterion for CRBR and that the design bases of the reactivity control and protection system meet this criterion and are acceptable.

In summary, the staff concludes that the PDC discussed above are appropriate for the CRBR. The staff further concludes that the nuclear design bases meet the requirements of these criteria and are acceptable.

4.3.2 Nuclear Design Description

The CRBR will be a mixed (Pu-U) oxide-fueled, sodium-cooled, fast reactor. The reactor design has been based on the design performance parameters presented in Table 4.11. A brief description of the nuclear-design-oriented aspects of the reactor follows.

The nuclear design is concerned with four major components: the fuel, the blankets (both inner and radial blankets as well as axial blankets), the primary reactivity control system, and the secondary reactivity control system. Each component will consist of hexagonal assemblies containing either fuel (mixed plutonium-uranium dioxide), blanket material (depleted uranium dioxide) or control material (enriched boron carbide). The reactor will contain 156 fuel assemblies, 82 inner blanket assemblies (76 plus 6 alternate fuel/blanket assemblies), 126 radial blanket assemblies, and 15 reactivity control assemblies (9 primary control assemblies and 6 secondary control assemblies).

The inner blanket assemblies will be dispersed heterogeneously throughout the central region of the core. Each fuel assembly will contain 217 fuel rods and each blanket assembly will contain 61 blanket rods. The primary and secondary reactivity control assemblies will contain 37 and 31 absorber rods, respectively. Fuel, blanket and absorber rods will be clad with stainless steel. The active core height will be 36 in. with 14-in. upper and lower axial blankets. The coolant flow will be upwards through the core. Table 4.12 presents pertinent characteristics of each of these assemblies (dimensions and volume fractions are based on room temperature conditions).

The design calls for annual refueling. At equilibrium conditions, batch replacement of all fuel and inner blanket assemblies will be performed every 2 years, with a planned midterm interchange of six inner blanket assemblies for six fresh fuel assemblies designed to add sufficient excess reactivity to the core to complete the 550 full-power-day burnup. The radial blanket assemblies in the first and second rows will be replaced as a batch at 4- and 5-year intervals, respectively.

Reactivity control will be provided by the primary and secondary reactivity control systems. The primary control system will provide shim control for normal operational power changes and will be the primary system for normal and scram shutdown of the reactor. Both systems will be designed to shut down the reactor in either the normal or scram mode with the other system inoperable. The plant protection system design provides instrumentation to monitor reactor conditions and provides the appropriate interfaces to activate the reactivity control systems in the event of off-normal conditions. Inherent reactivity control and nuclear stability will be provided primarily by the doppler coefficient and to a lesser degree by negative feedback from fuel axial expansion effects. Power peaking has been minimized through the nuclear design. Table

4.13 presents CRBR values for the reactivity coefficients and axila and radial power peaking and compares them to similar values for the FFTF.

4.3.2.1 Power Distributions

Section 4.3 of the PSAR gives the following design bases for the power distribution.

(1) The arrangement of the fuel and blanket assemblies shall be selected to minimize the power peaking factors.

(2) The resulting radial and axial power distributions shall be compatible with the maximum allowable linear power rating in the fuel and blanket assemblies.

A more explicit statement of the power distribution limit is contained in Section 3.1.3.1 of the PSAR:

> The power distribution limits are derived from the maximum allowable peak heat generation rates for nominal and anticipated operational conditions which, when combined with the rod mechanical and thermal design parameters, assure that incipent fuel melting does not occur in the fuel pellet with peak power.

The determination of a maximum allowable heat generation rate is subject to constraints set by the fuel and blanket mechanical/thermal design. For this reason this limit is actually generated in Section 4.4. Determination of this adequacy of the power distribution is dependent on the design bases as contained in Section 4.4.1. In particular, these design bases include requirements for preservation of fuel, blanket, and control rod integrity; no centerline fuel, blanket, or control rod absorber melting and no sodium boiling are included.

The suitability of these design bases and the manner in which they are implemented is discussed in Sections 4.2 and 4.4 of this SER.

Explicit, quantitative values for the maximum allowable heat generation rates in the fuel and blanket and control assemblies have not been identified in the PSAR. In response to a request for further information the applicants have stated that there are no maximum allowable heat generation limits per se. Each situation is evaluated separately by fuel and thermal/hydraulic analyses to ensure that no incipient melting will occur at 15% overpower with thermal hydraulic design conditions and 3 σ uncertainties applied to the calculated power distributions. Design limits of 16 kW per foot in the fuel and 20 kW per foot in the blankets have been used. The design described in the PSAR meets these limits at 15% overpower with 3 σ uncertainties. This is acceptable.

To ensure a conservative power envelope, the applicants have applied both statistical and nonstatistical (direct) uncertainties to the calculated nominal values. Statistical uncertainties include experimental uncertainties (fission rates and gamma heating), criticality and control rod insertion rate uncertainties, and local fuel fissile content uncertainties. Nonstatistical uncertainties considered include method/ modeling uncertainties, control rod banking uncertainties, power level uncertainties and a spatially dependent power tilt uncertainty. In addition, the power envelope is determined at 15% overpower.

The staff does not find that the effects of fuel densification and resulting power spikes therefrom have been included in either the determination of the power distribution or the evaluation of its adequacy.

The PSAR contains the results of design calculations for the CRBR power distribution. These results, in the form of tables and figures, include the following;

(1) fuel and inner blanket power fractions at the beginning and end of the first six cycles

(2) power normalization factors for the axial blanket above and below the fuel and inner blankets and for the radial blankets

(3) radial peaking factors by assembly and for the peak rod

(4) normalized axial peaking factor curves

(5) power distribution by assembly and at the peak rod for the nominal calculation and also with uncertainties and 15% overpower factored into the calculation.

For this PSAR evaluation, the staff has not fully reviewed all of the basic data, methods, codes, and calculated values used in the power distribution analyses. However, what follows is the staff's assessment of the analyses, the results obtained, and the potential that a satisfactory power distribution can be obtained for the CRBR.

The applicants have not fully described the radial and axial models used in the analyses. In particular, the following are not fully described: the homogenizations used; their extent and upon what basis they were determined; how the control assemblies, including the partially inserted assemblies were treated in the models; and how the effects of streaming were accounted for. In an integral sense the adequacy of how well the applicants have handled these modeling problems surfaces in the uncertainties associated with the power distribution. Likewise the adequacy of the methods used shows up in the uncertainties. However, in order for this to be true, the uncertainties determined must be representative of those for the CRBR.

Many of the uncertainties are determined by applying the CRBR design methods to ZPPR (zero power plutonium reactor) critical experiments and then comparing the calculated values with those from the experiment. This is a reasonable approach and one used throughout Section 4.3 of the PSAR. However, it does not automatically yield the correct uncertainties or verify the methods for the following reasons:

(1) ZPPR critical experiments are performed at room temperature and thus calculated-to-experiment ratios to be used at some other temperature must reflect the additional uncertainty introduced in the ratio in going from room temperature to the temperature of interest, that is, operating temperature.

(2) ZPPR critical experiments are made at zero power with "clean" fuel. Thus, calculated-to-experiment ratios are valid only at BOC1 and should include the uncertainty of extrapolating from zero to full power.

(3) Because of differences in geometry and composition between the ZPPR criticals and the final CRBR design, a direct extrapolation of uncertainties from the ZPPR critical experiments to the CRBR can be misleading.

These three considerations, which might be thought of as uncertainties in the uncertainties, are not necessarily secondary effects. It is quite conceivable that these considerations could significantly impact the uncertainties used and consequently the reported power distributions. The applicants have not clearly discussed these considerations nor indicated (quantitatively) their potential effect. The PSAR does indicate that additional experimental verification will be undertaken using more prototypical CRBR configurations.

Combining two-dimensional diffusion theory with spatial synthesis techniques is a common approach used in the industry to calculate power distributions. The deficiencies of this approach are readily seen in the C/E ratios near the control assemblies, radially at the fuel-blanket interfaces, and at the interface between the central core and the axial blankets. The use of supplemental transport theory calculations might substantially reduce some of the calculational uncertainty introduced through the use of diffusion theory.

The applicants indicate that there are no plans to provide in-core instrumentation to monitor the power distribution. In response to staff requests they have submitted a rationale supporting this decision. They argue that there are several features in CRBR which permit safe operation without detailed in-core power monitoring. These include:

(1) the fast neutron spectrum which produces a relatively flat power distribution without local peaks

(2) the absence of xenon effects because of the fast spectrum

(3) the batch refueling scheme which makes very unlikely a misloading event which would affect the power distribution significantly

(4) the presence of two independent rod position indicator systems which permits the rods to be banked to close tolerances

(5) the presence of exit thermocouples which would detect any gross flux tilts that might develop in the core

The staff concludes that the applicants have adequately shown that additional in-core monitoring is not necessary.

A comparison of the power peaking factors for CRBR and FFTF is provided in Table 4.13. The staff assumes that the FFTF data are design (calculated) data. The staff assumes these data were provided to illustrate the adequacy of the CRBR design. The staff encourages the applicants to use the FFTF experimental data to verify and qualify CRBR design methods and uncertainties.

In summary, the staff concludes:

(1) The design bases for power distributions meet the requirements of PDC 8 and, as stated, are acceptable for a CP review.

(2) Sufficient data for cycles up to the equilibrium cycle are presented to permit the conclusion that the CRBR design can probably meet the design bases. This is acceptable at the CP stage.

(3) Verification of the codes, methods, and uncertainties associated with the power distributions should rely upon more typical CRBR configurations. The applicants have committed (in a letter of December 8, 1982) to such a program to provide data for the FSAR review. This is acceptable at the CP licensing stage.

(4) The effects of fuel densification have not been factored into the power distribution analyses. The applicants have committed (see above reference) to include such effects in the FSAR analyses. This is acceptable.

4.3.2.2 Reactivity Coefficients

The reactivity coefficients relate the effect on core reactivity of changes in core conditions, for example, fuel and blanket temperature, coolant density, and so forth. CRBR PDC 9 specifically addresses this aspect of the design by requiring that in the power-operating range the net effect of the prompt inherent nuclear feedback should tend to compensate for a rapid reactivity increase.

The applicants have proposed the following design bases to satisfy CRBR PDC 9.

> The reactor and associated coolant system shall be designed so that in the normal operating range, including anticipated overpower transients, the net effect of the prompt inherent nuclear feedback characteristics mitigate the effects of a rapid increase in the reactivity.

The staff interprets this to mean that the applicants evaluate all significant inherent reactivity coefficients and determine which ones can contribute a prompt reactivity feedback. Then it must be demonstrated that the net prompt feedback is negative.

The applicants have considered and evaluated the following reactivity coefficients for cycles 1 to 4 of the CRBR:

(1) doppler
(2) sodium void
(3) sodium density
(4) uniform axial expansion
(5) uniform radial expansion
(6) fuel assembly bowing

Of these coefficients, only the doppler effect results in an instantaneous feedback. Therefore, a secondary requirement of the applicants is that the fuel temperature doppler coefficient shall be strongly negative when the reactor is critical.

Although both fuel and blanket doppler effects are instantaneous with a change in temperature, a rapid reactivity increase does not necessarily bring about an instantaneous, uniform increase in the temperature throughout the core. In addition, because of the larger size of the blanket rods relative to the fuel rods, the blanket rod temperature would not rise as quickly as that for the fuel rods. Also, most of the power generation occurs in the fuel rods. Therefore, use of a combined fuel and blanket doppler coefficient could possibly result in an overestimate of the prompt reactivity coefficient in the case of an overpower transient brought about by a rapid reactivity insertion. Definitive arguments or analyses have not been provided to verify a uniform temperature increase with time in the fuel and blankets which would, therefore, justify use of a total core doppler coefficient. On the basis of the reasons cited, the staff concurs with the applicants' requirement that the fuel temperature doppler coefficient should be strongly negative when the core is critical.

In the analyses of the accident conditions of Chapter 15, the applicants indicate that conservative values of the doppler coefficient are used. Conservative value is defined to be the nominal value less 2σ or 3σ uncertainty. This is an acceptable design procedure. However, the applicants failed to specify the quantitative value for the doppler coefficient, whether it is for the fuel alone or the entire core, and how the value used was derived from the data presented in PSAR Section 4.3.

A value of -0.0019 (-0.0027 + 30%) is given in Section 15.2 of the PSAR for what is called the doppler coefficient. However, no units are given, so what is presumably meant is the doppler constant. No discussion is presented of the fuel-blanket weighting which leads to this value and no overall value is given in Section 4.3 of the PSAR. Therefore, no conclusion can be made with respect to the suitability of the Section 15.2 value.

The CRBR will have, in addition to a negative doppler coefficient, a negative prompt power (sum of doppler and axial fuel expansion) coefficient over the whole range from zero to full power. Further, in the power-operating range (defined to be 40% to 100% of full power) the average power coefficient (sum of all reactivity components) will be negative. This meets the design basis and is acceptable.

The ultimate requirement for the prompt reactivity coefficient is that it should provide sufficient inhibition to rapid power rises to permit the protection system to terminate anticipated transients and accidents before unacceptable consequences occur. A further requirement is that it should suppress power oscillations that cannot be controlled by the reactivity control system (PDC 10). On the basis of the discussion in the PSAR the staff concludes that the CRBR meets these requirements.

The reactivity coefficients have been determined and the uncertainties assigned primarily on the basis of the application of the CRBR design methods to ZPPR critical experiments which deviated substantially from the current heterogeneous CRBR design. As an example, many of the ZPPR criticals were homogeneous mockups. In the case of the doppler coefficient, the SEFOR experiments provided the basis. Comparisons of calculations using CRBR methods were made to the experimentally determined value and to the value calculated by General Electric (GE). The good agreement among the three values permitted the use of the uncertainty determined by the GE analysis for the CRBR. This is acceptable at the CP review stage.

The staff understands that further verification and evaluation of the doppler coefficient will be performed in engineering mockup experiments in ZPPR. The applicants have committed to perform and document such experiments so that data will be available for the OL review (letter, Dec. 8, 1982). This is acceptable.

A similar situation exists for other reactivity coefficients. A first-order perturbation theory code is used to calculate the various coefficients with diffusion theory codes used to compute forward and adjoint fluxes. Verification depends chiefly on experiments performed in ZPPR using core mockups which differ in varying degrees from the heterogeneous CRBR design. Diffusion theory and first-order perturbation theory include significant modeling approximations and adequate verification is essential. The discussion in the PSAR is sufficient to permit the conclusion that CRBR can probably meet the design bases in the area of reactivity coefficients. This is acceptable for the CP review stage if coupled with a commitment to provide the following:

(1) Descriptions of the codes, techniques, and procedures used to obtain the coefficients. These may be in the form of topical reports.

(2) A planned experimental program to verify the techniques and procedures in time for the FSAR review.

The applicants have made such a commitment (letter, Dec. 6, 1982).

4.3.2.3 Reactivity Control System

The reactivity control system must meet the requirements of the CRBR PDC 23, 24, 25, 57, and 58. These criteria are presented in Section 4.3.1 above.

The reactivity control system will consist of a primary and a secondary control system. The functional requirements of these two systems are as follows:

Primary System

(1) provide the primary shutdown system for off-normal conditions

(2) provide normal operational control

(3) provide normal reactor startup and shutdown control

(4) provide additional margin for control in the event of any anticipated reactivity fault

Secondary System

(1) provide the secondary shutdown system for off-normal conditions

(2) provide reactor shutdown independent of the primary system

(3) provide additional margin for control in the event of any anticipated reactivity fault

The primary system will consist of nine control rods, which will be grouped into two banks according to their assigned mode of operation. The three row 4 control assemblies are startup assemblies and will be parked above the core during normal operation. The six row 7 corner assemblies are the burnup operating group that will be partially inserted in the core during operation as required by excess reactivity requirements. Both the fully withdrawn row 4 and partially inserted row 7 corner assemblies will be used to obtain the primary shutdown margin. The secondary system will consist of the six row 7 flat control assemblies. These rods will be parked above the core during normal operation and provide backup shutdown capability. Both the primary and secondary systems will use fully enriched B_4C (92% boron-10).

The following design bases, limits, guidelines and functional requirements have been specified in Section 4.3 of the PSAR.

(1) Two independent, diverse, reactivity control systems, that is, a primary and a secondary system, are provided by the design.

(2) The primary system is to have both an operational and shutdown capability. It is designed to meet fuel burnup and load follow requirements, and also compensate for criticality and refueling uncertainties.

(3) The primary system must have sufficient worth at any time in the operating cycle, assuming the failure of any single active component (e.g., a stuck rod), to shut down the reactor from any operating condition and to maintain subcriticality over the full range of coolant temperatures expected during shutdown. Allowance must be made for the maximum reactivity fault associated with any anticipated occurrence.

(4) The secondary system must have sufficient worth at any time in life (that is, reactor cycle), assuming the failure of any single active component (e.g., stuck rod), to shut down the reactor from any planned operating condition to the hot standby temperature of the coolant, that is, hot standby condition. Allowance must also be made for the maximum reactivity fault associated with any anticipated occurrence.

(5) The primary system must maintain the fuel cladding within the limits defined in Section 4.2 for anticipated, unlikely, and extremely unlikely events (assuming the secondary system fails to scram). The secondary system must maintain core coolable geometry for anticipated, unlikely, and extremely unlikely events (assuming the primary system fails to scram).

The applicants have also specified as a design basis that the maximum controlled reactivity insertion rate from control rod withdrawal at the design maximum speed of 9 in. per minute (4.1¢ per second) shall not result in a violation of the fuel limits. This reactivity insertion rate is that for the highest worth control rod. In the case of accidental control rod withdrawal, a system of rod blocks is designed to terminate the rod withdrawal because of misalignment with the bank or if the withdrawal causes power to exceed an overpower limit. These protections are in addition to the operator's capability to respond to the visual indications of rod motion and the audio alarm and occur before the operation of the protection system.

Both the primary and secondary reactivity control systems use control rods. The independence of these two systems is based on the differences of the mechanical design of their components, for example, the control rod drive train and the control rod mechanism. The evaluation of the redundancy and diversity of these systems is presented in Section 4.2.4.3 in this SER.

The control system criterion states that the control system must perform its safety function for any single malfunction of the system. The applicants have interpreted this to mean failure of the highest worth rod in the system which responds to the trip. In the primary system, this is taken to correspond to a row 7 corner rod running out and not responding to the scram. For the secondary system, this corresponds to the case where the stuck secondary rod is adjacent to the faulted primary rod which is running out.

In determining the control requirements for the primary and secondary reactivity control systems the applicants have used the following approach. The primary system is designed to control the following effects: power defect, maximum reactivity fault, reactivity excess, criticality margin, and fissile tolerance. The power defect includes the reactivity effects of doppler, radial and axial core contraction, assembly bowing, and sodium density change in going from hot full power (including 3σ uncertainties and 15% overpower) to hot refueling temperature, that is, 400°F hot refueling temperature minus 25F° uncertainty. The maximum reactivity fault (in any anticipated operational occurrence) is defined to be the withdrawal of the highest worth control rod from its furthest inserted position to the full-out position. Reactivity excess uncertainty is included because the fuel enrichment requirements are for guaranteeing hot, full-power criticality at the end of each burnup cycle. The primary system must also include the uncertainty in the cold criticality prediction uncertainty. Control margin is included for the batch fissile content tolerance in the fuel. Fuel stack height and impurity uncertainties are not included, since they are included in the fuel enrichment. Also, burnup reactivity swing uncertainties are considered to be one sided, that is, that the core will be in a higher-than-expected reactivity state at end of life because of an overprediction of the burnup reactivity defect.

The secondary system requirements are based on only the power defect and the maximum reactivity fault. However, for the secondary system the power defect considered is for going from hot full power to the hot standby condition, that is, 550°F minus 50F° temperature uncertainty.

The PSAR contains tables of the primary and secondary control assembly requirements and worths at beginning and end of cycle for cycles 1 through 6. Curves are presented showing the control assembly worth as a function of depth of the control assembly bank in the core and fraction of control assembly worth withdrawn versus fractional withdrawal of the control assembly. Also curves of the primary control assembly bank withdrawal histories are presented for cycles 1 through 6 and the conditions of expected worth/nominal requirement and minimum worth/maximum requirements. Minimum subcritical shutdown at zero power and refueling temperature has been discussed as well as uncontrolled withdrawal of the highest worth rod. The staff has reviewed these data and the calculational methods and its comments follow.

Calculations of the control assembly worths have been based on 2-D diffusion theory calculations using the code 2DB. Qualification of the calculational method has relied on comparisons of calculated to experimental results for ZPPR-7 and ZPPR-8. These comparisons have raised questions about the use of diffusion theory and the mesh structure in particular. In fact, the applicants have decided not to use the biases derived from this comparison in the PSAR control rod design analysis, but to rely on forthcoming engineering mockup critical (EMC) experiments to resolve the differences between calculated and measured quantities. The applicants have committed to document the codes, techniques, and procedures used to obtain control rod worths and to verify their adequacy against experiment for the FSAR review (letter, Dec. 6, 1982).

The staff concludes that the data presented are acceptable at the CP review stage when coupled with this commitment.

4.3.2.4 Instrumentation and Control and Protection System Functions

CRBR PDC 11 and 18 specify requirements for instrumentation and control and the design of the protection system. Portions of these requirements impact on the neutronics design and, therefore, have been considered in this review. In particular, CRBR PDC 11 requires that instrumentation and controls be provided to monitor variables and systems that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions, and to maintain the variables and systems within prescribed operating ranges. CRBR PDC 18 requires automatic initiation of the reactivity control systems to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and also to ensure the automatic operation of systems and components important to safety under accident conditions.

The staff's review has consisted of determining whether the design includes appropriate instrumentation to monitor those parameters and systems which impact on the fission process and automatic initiation of the reactivty control systems for off-normal conditions. That staff has not reviewed the instrumentation and associated electronics designs (see Section 7 of this SER). A brief review of the instrument trip points has been made.

The plant protection system (PPS) will include the reactor shutdown system (RSS), the containment isolation system (CIS), and the shutdown heat removal system (SHRS). Of these three subsystems, only the RSS will be directly involved in protecting the core during operation. The RSS will be designed to initiate and carry to completion a trip of the reactivity control assemblies to prevent the results of fault conditions from exceeding the allowable fuel limits.

The RSS will rely on the primary control rod system (PCRS) and the secondary control rod system (SCRS) to provide the reactor shutdown as required to protect the fuel, that is, ensure that fuel limits are not exceeded. Associated with the RSS is instrumentation to sense off-normal events and, if necessary, provide the appropriate signals to activate the PCRS and/or the SCRS, that is, scram the control assemblies. Fuel limits have been specified in the PSAR Section 4.2 along with design bases for the PCRS and SCRS. In particular, design bases and requirements on control assembly scram times, reactivity insertion and withdrawal rates, and misalignment have been presented together with additional mechanical, material, and environmental requirements. These all serve to ensure that the PCRS and SCRS can meet their safety objectives of ensuring that fuel design limits are not exceeded. However, the ability of these systems to perform and meet their safety requirements is predicated on the presence of instrumentation and controls to sense off-normal conditions and then to activate the appropriate systems, for example, the PCRS and SCRS.

The RSS will provide instrumentation to monitor neutron flux, reactor inlet plenum pressure, sodium pump speed, sodium flow, reactor vessel sodium level, undervoltage, steam flow, feedwater flow, intermediate heat exchanger primary outlet sodium temperature, stream-drum level, evaporator outlet sodium temperature, and sodium-water reactions. Design bases and requirements for this instrumentation are delineated in PSAR Section 7.2.1.2 and its subsections. Off-normal plant conditions (design requirements) which result in trips (scrams) of the PCRS and SCRS and the instrumentation used to sense these conditions are presented (PSAR Table 7.2-2). In addition, specific design requirements for the instrument system are presented. Essential performance requirements for the PPS equipment are presented along with the general functional requirement: "The Plant Protection System is designed to automatically initiate appropriate action to prevent unacceptable plant or component damage or the release or spread of radioactive materials." Table 15.1.3-1 of the PSAR provides PPS subsystem trip levels or trip equations which are designed to terminate the postulated events (off-normal conditions).

The PSAR contains descriptions of the PPS instrumentation which will be used in monitoring plant parameters and systems and sensing off-normal conditions. A table of performance (accuracy and speed of response) requirements for the PPS instrumentation has been provided. Separate shutdown instrumentation for the PCRS and SCRS is also described. The applicants state that the instrumentation associated with the PCRS and SCRS will provide monitoring of the flux over the top three decades of the power range. Both systems will have this capability. The staff concludes that, coupled with the source range monitoring system, this system meets the instrumentation requirements.

The flux monitoring instrumentation providing input to the PPS will be located at three places around the circumference of the core and will be calibrated to monitor the core power. The spatially dependent (radial) power distribution will not be an input to the PPS. This is consistent with light-water-reactor practice. Monitoring of the spatial distribution is done in order to ensure that the assumptions regarding initial conditions assumed for transients and accidents are valid. Any distortions in power shapes resulting from the transient must be accounted for in the accident analyses to ensure that acceptance criteria for the event are met. The applicants indicate that outlet thermocouples will be used for backup monitoring of radial power shapes. The acceptability of these instruments for this purpose is discussed in Section 4.4 of this SER.

The path followed from fuel limits to instrumentation and trip limits to the PCRS and SCRS appears to be, in principle, correct. However, in practice it is very difficult to trace this path. The reason for this difficulty is that fuel limits and resultant cladding and coolant limits have not always been presented in terms of fixed quantitative values. It is not clear from the

PSAR how the trip limits and equations were derived and to what fuel, cladding, and coolant quantities and values they correspond. These relationships and background information should be provided for the FSAR review.

PSAR Section 4.2 provides analyses (safe shutdown earthquake) to support the thesis that the rate of reactivity insertion provided by the PCRS and SCRS is sufficient to ensure that fuel limits are not exceeded. Other applicable scenarios have yet to be analyzed. The staff notes that the analyses are based on the use of B_4C control rods which are fully enriched in boron-10. If the applicants use a lesser boron enrichment in the control rods (this possibility has been suggested in PSAR Section 4.3), the entire analyses will have to be updated for the new control assembly insertion depths and assembly worths. Furthermore, the reactivity insertion scenarios which rely on the doppler coefficient to mitigate the transient will have to be examined regarding the value of the doppler coefficient used, as was discussed earlier in Section 4.3.2.2 of this report. This doppler consideration is applicable regardless of the boron enrichment used in the control rods.

The applicants have provided design bases and requirements for the instrumentation used to monitor the systems and parameters important to the fission process and for automatic initiation of the reactivity control systems to preclude exceeding the fuel limits.

The PPS design provides no way of monitoring the reactor power spatial distribution. This is acceptable provided that account is taken in accident analyses of changes in power distribution and monitoring of initial condition for transients is performed.

The path from fuel limits to instrument setpoints needs to be clearly described. In particular, the quantitative fuel limits which were used should be specified, together with a clear explanation as to how the instrument setpoints were derived from these limits, including any margin for uncertainties, and so forth. This can be done at the OL stage.

The staff concludes that a reasonable potential exists that the design can meet the requirements of CRBR PDC 11 and 18.

4.3.2.5 Reactor Stability

CRBR PDC 10 requires that the reactor system be designed to ensure that power oscillations which lead to exceeding fuel limits are not possible or can be reliably and readily detected and suppressed.

In Section 4.3.1.3 of the PSAR the applicants list as a design basis CRBR PDC 10. Toshow compliance with this design basis, the applicants rely on linear systems techniques. Two different criteria were considered for determining the stability based on the set of first-order differential neutron and thermal equations:

(1) The system is stable if the real part of all roots of the characteristic equation $\underline{X} = \underline{A}\ \underline{X} + \underline{b}\delta k$, where _ indicates a matrix, is negative, that is, the real part of the eigenvalues of the matrix A is negative. Conversely, if the real part of any root is positive then the system is unstable.

(2) The system is stable if the system output is bounded for a given bounded input; the system is unstable if the system output is unbounded for a given bounded input. The system is stable in the practical sense if the system output is bounded within acceptable parameter levels.

The first criterion for stability indicated above is predicated on the validity of using the linear systems approach, that is, linear first-order differential equations to describe the neutronics and temperature behavior of the reactor system. However, the PSAR provides no quantitative indication of the validity of the linearized approach. The approach used considers reactivity feedback effects from reactor assembly bowing, doppler, fuel expansion, and sodium density changes.

Stability analyses have been performed using the two stability criteria stated at the beginning of this section. The first analyses, and presumably the second, were performed using an early model of the CRBR heterogeneous core and thus will have to be updated. These analyses illustrate the stability of the reactor which is dominated by the doppler feedback, even when it is taken at one-half its nominal value. During startup, at low power-to-flow ratios (P/Fs) the system response will be influenced by the positive assembly bowing feedback. The PSAR indicates that the maximum assembly bowing reactivity insertion will occur during reactor startup at the 9% power, 40% flow point. For this case the analyses indicate a power increase to the point where the bowing reactivity becomes negative and all parameter responses change slowly to approach a new equilibrium state. Thus, in the startup range the staff does not expect that the positive reactivity feedback caused by assembly bowing will result in conditions that will exceed the fuel limits. Above a P/F of 0.7 the bowing feedback becomes negative. In the power operating range bowing is not expected to influence the stability.

The applicants have provided design bases and criteria for meeting CRBR PDC 10. Demonstration of stability is provided through the use of linear system techniques which are valid for small reactivity perturbations. Applicability of this approach for large reactivity changes and nonlinear feedback mechanisms has not been justified in the PSAR. Updating of the calculations to the current CRBR design is needed. The applicants have committed (letter, Dec. 6, 1982) to performance of analyses of FFTF stability tests to verify their methods and to confirm that power oscillations do not occur in CRBR. They will document the verification in the FSAR. In addition, core stability must be verified during initial startup testing of the plant. This is acceptable for the CP review.

4.3.3 Analytic Methods

There are no specific criteria which the analytic methods used in the design analyses must satisfy. However, light-water-reactor practice requires the applicants to use state-of-the-art methods which have been verified by comparison with measured data.

The staff has discussed the methods used for neutronics design in the preceding sections of this report. In general, the staff's comments centered on the use of diffusion theory versus transport theory and the validity of the first-order perturbation theory applications. The staff concludes that substantial qualification/verification work must be performed for the methods because most of the earlier work was done based on critical experiments which differed significantly from the CRBR. As noted in the discussion above, the applicants have committed to provide more details of the method used and to verify that method against experimental data. These details and verifications will be documented in the FSAR.

The staff has not reviewed the codes and methods in detail. In addition, the staff has not performed a detailed review of how these codes and methods were applied in the various design analyses, for example, mesh structure, modeling approximations, and input parameters. The codes and methods used appear to be state of the art, but verification using experimental mockups closely approximating the CRBR design will be required for the operating license review stage. Commitment to perform such verification is acceptable at the construction permit stage.

4.4 Thermal and Hydraulic Design

Introduction

This section addresses the thermal and hydraulic design of the reactor vessel internals including the core support structure, fuel assemblies, blanket assemblies, control assemblies, shield assemblies, core barrel, and upper internals structure. Its purpose is to describe and evaluate the design criteria, analysis methods, development testing, and instrumentation associated with the above.

The reactor vessel (Figure 4.2) will be a single-walled vessel which will enclose the reactor core, coolant, and vessel internal structures. These internal structures will provide positioning, support, and flowpaths for the sodium coolant. More detail on the reactor vessel design criteria can be found in Section 5.2 of this SER.

The reactor vessel will be filled with sodium up to a level approximately 65 in. below the bottom of the closure head. It will be connected to three primary heat transport loops (Section 5) via three 24-in.-diameter inlet nozzles at the bottom of the vessel and three 36-in.-diameter outlet nozzles $\sim$16 ft below the top of the sodium pool. The sodium level will be maintained constant in the vessel by continously adding sodium via a makeup pump and draining any excess sodium out of the vessel via an overflow line.

Inside the reactor vessel the lower internals structure (whose major components are the core support plate, core barrel and support cone) will provide a pressure boundary between the inlet and outlet plenums. The core support plate will provide support for the following major in-vessel components: the lower inlet modules (LIMs), the core barrel, and the fuel, blanket, control, and radial shield assemblies. Upper and lower core former rings and the bypass flow module will be supported by the core barrel which will be welded to the core support plate. Above the core, the upper internals structure, which will be attached to the intermediate rotating plug of the reactor vessel closure head, will provide guidance and stability for the control rod drive mechanisms and will house core outlet temperature monitoring instrumentation. The upper internals structure will be keyed to the upper core former ring to prevent lateral movement. The upper internals structure also will act as a secondary holddown device for the core assemblies if they lose hydraulic holddown. A vortex suppressor plate that will be placed just below the sodium pool surface will serve to reduce cover gas entrainment in the sodium exiting the outlet plenum.

4.4.1 Description

Each of the major in-vessel components is described in more detail in the sections that follow and the flowpaths and flow allocations within the vessel are described in Section 4.4.1.4.

4.4.1.1 Lower Internals Structure

The lower internals structure will serve the functions of support, location, and restraint for the reactor fuel, blanket, control, and removable radial shield assemblies. The main lower internals structure components will be the core support plate, core support cone, core barrel, horizontal baffle, fixed radial shield, lower inlet modules, and fuel transfer and storage assembly.

The combination of the core support plate and support cone will form the pressure boundary for the reactor inlet plenum separating the high pressure inlet sodium from the rest of the reactor vessel. The core support cone and support plate will be welded together and the support cone will be welded to the reactor vessel. Small holes in the support cone will prevent gas entrainment under the core support structure. The primary function of the core support plate will be one of support for the reactor assemblies.

The core barrel will extend upward from the core support plate to the top of the core assembly outlet nozzles. It will be a thick-wall right-circular cylinder that will surround and provide lateral support for the reactor core. The core barrel will serve as support for the core former rings (which will be the lateral contact points for the core assemblies) and, also, will provide the attachment point for the fixed radial shield. Between the core barrel and the reactor vessel there will be a horizontal baffle which will be attached to the top of the core barrel. The horizontal baffle will serve to channel flow to the space between the reactor vessel wall and its thermal liner, to separate the bypass flow around the core barrel from the hot sodium in the outlet plenum region, and to protect the components in the cooler regions below the core outlet from thermal shock and striping.

Lower inlet modules will be inserted into lined holes in the core support plate (Figure 4.3). Each inlet module will hold and distribute flow to the inlet nozzles of seven assemblies. In addition, some lower inlet modules will provide flow to the bypass modules and subsequently to the radial shield assemblies. There will be a total of 61 inlet modules. The lower inlet modules also will serve other important functions:

(1) They will be equipped with flow strainers to filter out any particles greater than 1/4-in. diameter, thus preventing them from entering the core assemblies.

(2) They will be designed to preclude blockage from large objects by providing multiple flowpaths to each assembly inlet.

(3) They will be designed to preclude insertion of any fuel, blanket, control, or shield assembly into a region of the core for which it is not properly orificed. This will be accomplished by physical features (called discriminators) which will be provided in each LIM and which will mate with a corresponding feature on the inlet nozzles of the core assemblies.

Holddown of the core assemblies will be achieved through a combination of hydraulic pressure balance and the weight of the assemblies. The lower inlet modules and core assembly inlet nozzles will be designed to provide inlet plenum pressure on the interior of the inlet nozzles to counteract forces produced by the upward flowing sodium. The upper internals structure will provide a backup hold-down mechanism if the hydraulic balance should be lost.

4.4.1.2 Upper Internals Structure

The upper internals structure (UIS) (Figure 4.4) will provide stability and guidance for the control rod mechanisms. It will support in-vessel instrumentation and will act as a backup system for holddown of the core assemblies. The structure will consist of four support columns, two transverse interconnected plates, four shear webs, flow chimneys, shroud tubes, and instrumentation posts. The upper internals structure will be attached to the intermediate rotating plug of the reactor vessel upper closure assembly. The flow chimneys will be designed to promote mixing and thus reduce flow stratification (and temperature stratification) in the upper plenum. Each chimney will direct the flow from several core assembly outlet nozzles to the reactor outlet plenum.

4.4.1.3 Core Assemblies

In the reactor core region there will be 156 fuel assemblies and 76 inner blanket assemblies. Also, there will be 6 assemblies that will be alternated between fuel and inner blanket. In addition, there will be 126 outer radial blanket assemblies and 312 removable radial shield assemblies. The 15 control assemblies, 9 primary and 6 secondary, will also be part of the core region. The arrangement of the assemblies in the core is shown in Figure 4.5. Each of these assemblies is discussed in greater detail below.

4.4.1.3.1 Fuel Assemblies

Each fuel assembly will be made up of 217 fuel pins, an outer duct or can, a lower shield, and inlet and outlet nozzles (Figure 4.6). The outer can will be a hexagonal tube that will form the flowpath for the assembly; the flowpath for each fuel pin will be formed by its surrounding pins and by a wire-wrap spacer. From 156 to 162 fuel assemblies, depending on the particular fuel cycle, will be in the core at any one time. The design of the fuel assemblies is very similar to that at FFTF, and Section 4.2 of this SER presents a detailed comparison of the FFTF and CRBR fuel assembly design parameters.

In the lower portion of each fuel assembly there will be a combination shield-orifice region. The shield portion will have vertical flow passages that will have a large length-to-diameter ratio and offset flow holes, therefore, providing protection from radiation damage for the lower inlet modules and the core support plate. The orifice portion will be located at the bottom of the shield and its design will determine the coolant flow in the assembly.

At the bottom and top of the fuel assemblies there will be inlet and outlet nozzles, respectively. The inlet nozzle will be cylindrical and designed to contribute to the hydraulic balance system and mate with the discrimination feature in the LIMs. The inlet nozzle will contain six elongated flow inlet holes designed to preclude flow blockages if the assemblies should lose hydraulic holddown and raise against the UIS. The outlet nozzle will be hexagonal and will serve as the handling portion of the assembly.

Each assembly will have two areas, called load pads, that will serve to maintain interassembly gaps and the desired duct bowing profile. The lower load pad will be located just above the core and will consist of a region of the duct that will be rasied or thickened around the duct circumference. The upper load pad will be part of the outlet nozzle.

4.4.1.3.2 Blanket Assemblies

The inner and outer radial blankets will be of identical design (except for flow allocation) and will provide the fertile material for nuclear breeding (Figure 4.7). There will be 76 inner blanket assemblies and 126 outer radial blanket assemblies. In addition, there will be 6 assembly locations that alternate between fuel and blanket. The blanket assembly flowpaths will be similar to the fuel assemblies; however, they will contain fewer, larger diameter pins (61 versus 217). Detailed design data for blanket assemblies is given in Section 4.2 of this SER.

4.4.1.3.3 Removable Radial Shield Assemblies

Radial shield assemblies will provide neutron and gamma radiation shielding for the core former rings and the core barrel. These assemblies will have the same outside shape as the fuel and blanket assemblies. Flow will be provided to these assemblies via the lower inlet modules (for the inner shield assemblies) and via the bypass flow module (for the outer shield assemblies).

4.4.1.3.4 Control Assemblies

Control assemblies will consist of pins with boron carbide (B_4C) pellets inside stainless steel cladding, inner and outer ducts, and inlet and outlet nozzles on the outer duct. The inner duct will be sized to slide into the outer hexagonal duct. There will be a total of 15 control assemblies--9 for the primary control system and 6 for the secondary control system. The primary control system assemblies will contain 37 pins each; the secondary control system assemblies will contain 31 pins each. The design of the primary control assemblies is similar to that of the FFTF control rods. Flow will be provided to the control assemblies via an inlet nozzle similar to that for fuel assemblies but orificed for a smaller flow rate.

4.4.1.4 Reactor Coolant Flowpath

4.4.1.4.1 Inlet Plenum

Three equally spaced inlet nozzles will supply coolant to the reactor inlet plenum. The inlet nozzles will direct the flow downward at a 60° angle toward the bottom of the reactor vessel to enhance mixing of the flow from the three loops. This design provides sufficient mixing to prevent the buildup of significant thermal stresses in the core support structure, the lower inlet modules, and the reactor vessel during steady-state and transient operation.

4.4.1.4.2 Lower Internals Structure

The flow will proceed upward from the inlet plenum to the 61 lower inlet modules. The lower inlet modules then will distribute the flow to fuel, blanket, control, and inner radial shield assemblies. The design of the lower inlet modules also will minimize the potential for flow blockage by providing multiple flowpaths to the core assemblies and by straining out any particles greater than 0.25-in. diameter. Six peripheral lower inlet modules also will provide flow through the core support plate to six bypass modules and their corresponding outer radial shield assemblies. In addition, the core support plate will contain a low-pressure manifold that will distribute leakage and orificed bypass flow from the lower inlet modules to the space between the core barrel and reactor vessel.

4.4.1.4.3 Fuel, Blanket Shield, and Control Assemblies

Flow through fuel, control, inner blanket, and part of the outer blanket assemblies will be determined by orifices located in each of these assemblies. Flow for the remaining blanket assemblies and inner radial shield assemblies will be controlled by orifice plates in the corresponding lower inlet modules. Flow to the outer radial shield assemblies will be determined by orificing in the bypass flow module. There will be a total of 12 orificing zones for the fuel and blanket assemblies, each with a different flow rate. Control assemblies will be orificed within the assembly and the radial shield assemblies will have two orificing zones. Flow rates to the various components are shown in Tables 4.14 and 4.15 as a function of orificing zone.

4.4.1.4.4 Outlet Plenum

The coolant flow from the fuel, blanket, control, and some of the radial shield assemblies will pass through the upper internals structure on its way to the outlet plenum. Coolant flow from the remaining assemblies will go directly into the outlet plenum. After mixing in the outlet plenum the coolant will flow out of the reactor vessel through three 36-in.-diameter equally spaced outlet nozzles.

Scope of Review

PSAR Chapter 4.4 and SRP Section 4.4 were reviewed along with Westinghouse documents WARD-D-0050, Revision 3, "CRBR Core Assemblies--Hot Channel Factors," and WARD-D-0210, Revision 1, "Steady State Thermal/Hydraulic Performance of Fuel and Blanket Assemblies." In addition, PSAR Sections 4.2, 4.3, 15.1, 15.2, and 15.3 were reviewed for their interfacing with the thermal/hydraulic design.

RG 1.68 was also reviewed for its applicability since it is referenced in SRP Section 4.4.

The CRBR principal design criteria were reviewed; PDC 8 and 60 apply to Section 4.4.

Evaluation Criteria

The reactor thermal and hydraulic design was reviewed for:

(1) Compliance with PDC 8 and 60

(2) Compliance with applicable sections and intent of SRP Section 4.4, including RG 1.68. In general, all items in SRP Section 4.4 and RG 1.68 were considered applicable except those unique to LWRs.

In general, the design was evaluated for

(1) acceptability of the applicants' design criteria
(2) adequacy of analysis methods and codes
(3) adequacy of development testing
(4) adequacy of uncertainty evaluation
(5) adequacy of instrumentation provided
(6) adequacy of design features included to preclude flow blockage or reduced flow

4.4.2 Design Bases

The thermal and hydraulic design of the reactor is based upon providing adequate flow to all vessel components during steady-state and transient conditions to enable them to meet their design objective (i.e., lifetime), yet provide adequate margin for safe operation. To ensure this will be accomplished, a set of criteria were established which the design had to meet. The criteria address both replaceable components (fuel, blanket, control, and radial shield assemblies) and permanent 30-year lifetime components (vessel, core support plate, core barrel, UIS, etc.). For the replaceable components the criteria are intended to maintain fuel, blanket, and control pin integrity for design-basis events up to and including unlikely events and to maintain coolable geometry for extremely unlikely events. For the structural components (both replaceable and permanent) the criteria are intended to maintain the component temperatures and hydraulic forces within those values assumed in the component design and safety analysis. Two sets of plant conditions were defined to be used in the criteria and in the design analyses, depending on whether the component being evaluated was replaceable or permanent.

These conditions are:

(1) Thermal/Hydraulic Design Values (THDVs)

Used for analysis of permanent components and as the starting point for all unlikely and extremely unlikely transient events. The temperatures and flow rate associated with THDV are:

(a) reactor inlet temperature--730°F
(b) reactor outlet temperature--995°F
(c) total reactor flow rate--41.446×10^6 lb/hr
(d) total reactor ΔP (nozzle to nozzle) 126 psi

(2) Plant Expected Operating Conditions (PEOCs)

Used for steady-state and anticipated transient analysis for all replaceable core components. These values represent those that are actually expected for the CRBR first core. The temperatures and flow rate associated with PEOC are:

(a) reactor inlet temperature--704°F
(b) reactor outlet temperature--954°F
(c) total reactor flow rate-43.93 x 10^6 lb/hr
(d) total reactor ΔP (nozzle to nozzle)--∿115 psi

The criteria and limits used in the thermal/hydraulic analysis are summarized below.

4.4.2.1 Cladding Temperature

The maximum fuel, blanket, and control pin cladding temperature for both steady-state and transient conditions must be consistent with fuel burnup objectives, blanket and control assembly lifetime objectives, and with the fuel, blanket, and control pin cladding integrity criteria defined in Sections 4.2 and 15.1. Table 4.16 summarizes the cladding integrity criteria and other constraints upon which the thermal/hydraulic analysis of these components is based. It should be noted that some of these criteria may change as a result of the evaluation in Section 4.2 of this report. Any changes must be reflected in the final thermal/hydraulic design.

4.4.2.2 Pin Centerline Temperatures

No fuel, blanket, or absorber material melting is allowed at 115% power, PEOC conditions, at the 3σ confidence level for fuel assemblies, blanket assemblies, and absorber pins.

4.4.2.3 Coolant Temperature

The sodium temperature shall be less than the boiling point during normal operation, anticipated, unlikely, and extremely unlikely transient conditions.

4.4.2.4 Coolant Velocities

The coolant velocities shall be less than:

(1) 30 ft/sec for nonreplaceable components
(2) 40 ft/sec for replaceable components in the high temperature region
(3) 50 ft/sec for replaceable components in the low temperature region

4.4.2.5 Pressure Drops

The total pressure drop shall be within the primary pump head capability and the hydraulic forces on the in-vessel components shall be within their structural capability.

4.4.2.6 Coolant Flows

(1) Adequate coolant shall be provided to ensure structural integrity of the following components (structural integrity is discussed in Section 4.2.2).

  (a) radial shielding

  (b) core barrel

  (c) core former components

  (d) reactor vessel thermal liner (provide sufficient flow to maintain the vessel wall <900°F during normal operation)

  (e) fuel transfer and storage assembly

  (f) upper internals structure

  (g) primary boundary (maintain bulk sodium temperature <1140°F)

(2) Control assembly flow rates have to ensure adequate margin against flotation when the driveline is disconnected and allow the assemblies to satisfy scram insertion requirements.

(3) The orificing design shall provide shielding to the lower inlet structures consistent with their lifetime.

4.4.2.7 Gas Entrainment

The design shall prevent gas entrainment sufficient to cause significant heat transfer or reactivity changes in the core.

4.4.2.8 Flow Blockage

The design shall provide features to minimize the potential for flow blockage of in-core assemblies.

4.4.3 Design Analysis

The preliminary design analysis of the reactor thermal/hydraulics presented in the PSAR was performed using the heat generation data developed via the neutronics and power generation calculations discussed in Sections 4.2 and 4.3. Once the heat generation rates and fluences of the various components were known, the required flow to these components to maintain their temperatures and structural properties within preestablished limits could be calculated. To avoid many iterations of flow calculations, the applicants chose to express each of the limits presented in Table 4.16 in terms of a maximum temperature for each component and then calculate the flow distribution necessary to maintain the components at or below these temperatures. Based on this approach, a preliminary flow distribution for the in-vessel components was calculated (Tables 4.1 and 4.15). This calculated flow distribution is then to be used during final design as input to the detailed component structural analysis. The final analysis may show that some adjustment is required in the flow distribution.

4.4.3.1 Computer Codes

Many computer codes were used in assessing the core thermal/hydraulic performance and in arriving at a preliminary flow distribution. A brief summary of the codes used and their application is offered below.

(1) CATFISH--calculates the flow and ΔP in each core assembly given the orificing and hydraulic resistances. Used for at power conditions only.

(2) COBRA--calculates flow and temperature distribution within a fuel or blanket assembly. Can be used for at power conditions and, in a modified form (COBRA-WC), is to be used by the applicants for natural circulation conditions with the ability to calculate interassembly flow redistribution.

(3) CONROD--calculates absorber pin temperature and pressure. Used in the analysis of secondary control assemblies.

(4) CORINTH--calculates interassembly flow redistribution during undercooling transients. Used for transient conditions only.

(5) CORTEM--calculates intra- and interassembly heat tranfer in a 30° sector of the core. Used in the analysis of secondary control assemblies.

(6) COTEC--calculates flow and temperature distribution within fuel and blanket assemblies. Used for at power conditions only.

(7) CRSSA* - calculates pellet, cladding, and coolant temperature distribution for the primary control rods. Used for at power conditions only.

(8) DEMO--calculates plant thermal response to simulated transients. Models reactor core, PHTS, IHTS, and steam system. Used for at power and natural circulation conditions.

(9) DYNALSS--calculates secondary control assembly hydraulic conditions (flows and pressures) during a scram. Used in predicting scram time.

(10) FATHOM-360--calculates the radial and circumferential temperature profile for fuel, blanket, and absorber pins any place in the core. Used for at power conditions only.

(11) FLODISC--calculates steady-state flow in a series of parallel channels. Used for calculating reactor flow and ΔP.

(12) FORE-2M--calculates reactor power, reactivity feedbacks, and coolant, cladding, fuel, and structural temperatures for steady-state or transient conditions. Used in conjunction with DEMO code for natural circulation analysis.

(13) FULMIX--calculates thermal/hydraulic conditions inside the pin bundle of a removable core component. Used for calculating the coolant temperature distribution within a bundle.

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*Indicates a Westinghouse proprietary code.

(14) HAFMAT--calculates flow distribution for specific complex flowpath areas, which may include heat generation. Used for steady-state conditions.

(15) LIFE-III--calculates fuel and blanket rod thermal state and mechanical behavior as a function of operating history. Used for at power conditions only.

(16) NICER* - calculates the steady-state thermal performance of fuel and blanket assemblies by calculating axial and radial rod temperature profiles, fission gas pressure, mixed mean outlet temperature, and bundle ΔP.

(17) OCTOPUS--calculates the required flow split among the 12 orificing zones for the fuel and blanket assemblies, based upon satisfying thermal constraints on the assemblies in the various zones.

(18) STALSS--calculates required assembly flow rate, flow split, and pressure drop. Used in the analysis of secondary control assemblies.

(19) THI-3D--calculates thermal/hydraulic conditions inside a pin bundle. Is used for primary control assembly thermal/hydraulic analysis.

(20) TRITON--calculates cladding, coolant and duct temperature distribution in fuel, blanket, control and shield assemblies with interassembly heat transfer. Used for at power conditions only.

The sequence in which these codes are used and their interface is shown graphically in Figures 4.8, 4.9, and 4.10 for the fuel and blanket assemblies and primary and secondary control assemblies, respectively. Radial shield assembly temperatures are calculated using the TRITON code. Most of the computer codes used by the applicants have been extensively tested and have been used for many years. This is particularly true for the subchannel codes COTEC, COBRA, and TRITON as well as the LIFE-III code. Empirical correlations for friction and form losses are used as input in the calculation of pressure drops in the various reactor components. The large variety of computer codes, ranging from survey-type codes to detailed design analysis codes, together with the extensive thermal/hydraulic test program is intended to permit detailed and reliable analysis of all thermal/hydraulic phenomena under full-power, full-flow conditions. The applicants have indicated they plan to establish a similarly reliable framework of computational methods for the analysis of low-power, low-flow conditions.

Since CRBR has been designed to have the capability to remove decay heat through the main heat transport system loops via natural circulation, the analysis associated with this condition has been highlighted in the PSAR, along with the analysis for full-power conditions.

4.4.3.2 Full-Power, Full-Flow Analysis

Section 4.4 of the PSAR does not deal in detail with the flow hydraulics and temperatures outside of the core region. Those issues are addressed in PSAR Section 4.2.2.1.3; however, no calculational details are provided.

*Indicates a Westinghouse proprietary code.

Reactor internals pressure drops were calculated for the reactor vessel inlet plenum, lower inlet modules, UIS, and reactor exit nozzle. Those analyses were based on water tests performed in the inlet plenum feature model and the integral reactor flow model and calculations with the CATFISH code. Uncertainties in resistances were taken at their maximum value, that is either 3σ whenever a large enough data base existed, or at their bounding value (generally 1.2 times nominal) when only engineering estimates were evaluated. It was furthermore assumed that all resistances were simultaneously taken at their maximum value.

The maximum hydraulic loads on the core components were calculated considering steady-state, transient, and test operation. Under steady-state conditions, the core support structure is subjected to a maximum load of 7.37×10^6 lb, which corresponds to a conservatively estimated maximum core pressure drop of 160 psi. The maximum core pressure drop assumes that the pump-head flow characteristic is at the maximum and all noncore pressure losses are at their minimum values. The hydraulic loads for other than steady-state conditions were found to be lower.

The core thermal analyses carried out for full-flow, full-power conditions primarily are based on meeting the cladding design criteria in Table 4.16. The following analyses were carried out for fuel, blanket, control, and shield assemblies

(1) temperature calculations

    (a) fuel rod
    (b) cladding
    (c) duct
    (d) coolant (subchannel, mixed mean)

(2) pressure calculations

    (a) hydraulic
    (b) fission gas

(3) power-to-melt

(4) linear power

(5) rod life (using guideline temperatures to stay below the cladding strain and cumulative damage function (CDF) limits)

The uncertainties and conservatisms used in these analyses are discussed in Section 4.4.4 of this SER. The results of these analyses are presented in Section 4.4 of the PSAR.

4.4.3.3 Low-Flow, Low-Power Analyses (Natural Circulation)

At low flow, hydraulic resistance characteristics of the in-vessel components change, loop transport times increase, intersubassembly heat transfer becomes significant in removing heat, and temperature-induced buoyancy effects generally tend to help promote flow to the areas where it is needed most. A limited number of calculations have been carried out for low-flow, low-power conditions.

The intent of these analyses was to show that the sodium temperature remained below its boiling point. In these calculations the effects of intra- and inter-assembly heat and flow redistribution at low flow rates were conservatively neglected. FFTF analysis and test results indicate that the inclusion of those effects during natural convection conditions significantly reduces the temperatures for the hot-test assembly. Other conservatisms are discussed in Section 4.4 of this SER. The analysis reported in the PSAR was performed with the DEMO-FORE-2M code package. Future analysis will take into account intra- and inter-assembly heat and flow redistribution by adding the COBRA-WC code to this package. The results of the natural circulation analysis done to date indicate that if the system works as planned, coolant and cladding temperature will remain well below the sodium boiling temperature.

4.4.3.4 Transient Analyses

The fuel, blanket, and control pin cladding transient design limits are listed in Table 4.16. In the thermal/hydraulic analysis presented in the PSAR, guideline cladding temperatures were chosen (to simplify the calculations) representative of cladding temperatures necessary to stay below the cladding strain and CDF limits. The flow distribution to the core assemblies was then set to maintain cladding temperatures below these guideline values. Transient temperatures for structural components are to be calculated as input to the component structural analysis. Results of these analyses are presented in Section 4.4 of the PSAR.

4.4.4 Uncertainties/Conservatism

The analysis for steady-state and transient conditions includes an allowance for uncertainties and in some cases provides additional conservatism to envelope worst-case conditions. The application of uncertainties and conservatism is as follows.

(1) To account for possible variations in the input and basic assumptions used in the analysis codes, a set of hot channel factors (HCFs) was developed for each type of removable core assembly. These hot channel factors are an attempt to quantify all of the uncertainties which could affect the analysis. As a example, the HCFs for the fuel assemblies are shown in Tables 4.17 and 4.18. The hot channel factors represent a multiplier to be applied to the calculated ΔT for the parameter of interest to give an upper-bound expected temperature. They are broken down by source (inlet flow maldistribution, fissile fuel maldistribution, etc.), by type (direct or statistical), and by parameter of interest (coolant temperature, cladding temperature, etc.). The direct factors for a particular parameter represent conditions which could occur and if so would not be random throughout the core. Each one is applied as a direct multiplier and the values chosen are upper-bound values derived from test data or analysis. The statistical factors are those that could occur randomly anywhere in the core and are to represent an uncertainty at the 3σ level of confidence. They are combined for a given parameter by multiplying each one times the calculated ΔT for the parameter and then taking the square root of the sum of the squares. For example, if the coolant in a particular fuel assembly were calculated to have a nominal ΔT of 150°F in going from the assembly inlet to the core midplane, then the 3σ hot channel coolant

temperature at that location at THDV conditions would be calculated as shown in Table 4.9.

Adding the 37.3°F to the THDV inlet temperature (730°F) and the nominal ΔT (150°F) yields a hot channel coolant temperature of 917°F at core midplane.

The HCF application is based on three major assumptions: (a) that the temperature of interest is affected in a linear fashion by the HCF variables, (b) that all HCFs are independent variables, and (c) that the statistical uncertainties are distributed in a normal fashion. The detailed derivation of the HCFs is discussed in WARD-D-0050, Revision 3, "CRBR Core Assemblies--HCF Analyses."

For final design the applicants have committed (letter, Dec. 6, 1982, #HQ:S:82:139) to a more rigorous application of the HCFs which will include evaluation of the conservatism of the major assumption listed above. In addition, they have committed to establish confidence levels on the HCF values and to ensure consistent application of the HCFs.

(2) In general, the uncertainties in Table 4.20 are applied in the thermal/hydraulic analysis of the in-vessel components.

(3) Worst-case (3σ uncertainty) values for the physical properties were used in the analysis.

(4) The calculational techniques used conservative assumptions as follows:

(a) Power-to-melt analysis used direct plus 3σ statistical HCFs. It also accounted for the uncertainty in the EBR-II generated power to melt test data.

(b) Fuel and blanket plenum pressure calculations used minimum plenum volume, 2σ upper bound values for burnup, fission gas yield, and fission gas release and direct plus 2σ statistical HCFs in calculating plenum temperature.

(c) Control rod flotation used PEOC ΔP plus 3σ uncertainty.

(d) Natural circulation calculations used 3σ uncertainties on decay heat, THDV plus 20°F initial plant conditions, conservative pump coastdown time, 3σ HCFs, 30% extra core ΔP, neglected inter- and intra-assembly heat redistribution, and used full-power, full-flow temperature peaking factors for in-core assemblies.

(e) DHRS heat removal calculations used 3σ uncertainties on decay heat and neglected the heat capacity of the IHTS.

4.4.5 Instrumentation

Permanent in-vessel instrumentation is to be provided consisting of the following:

(1) A single replaceable drywell thermocouple will be located above the core assembly outlet nozzle for the following assemblies (see Figure 4.11 for locations):

(a) 148 out of 156 fuel assembly positions
(b) 72 out of 76 inner blanket assembly positions
(c) 6 out of 6 alternating fuel/blanket positions
(d) 79 out of 132 radial blanket assemblies

The purpose of these thermocouples is to provide surveillance information on reactor core outlet conditions and to provide a control signal to the reactor control system (see Section 7.7.1.2) for controlling reactor outlet temperature. Thirty of the thermocouples have been selected to provide input to the control circuit (see Figure 4.11 for location of these 30 thermocouples).

No safety function is claimed for any of the outlet thermocouples.

(2) Five drywell induction probe sodium level sensors will be mounted through the reactor head to measure Na levels. Three of these will provide input to the reactor shutdown system and have a range of from +6 in. to -24 in. below the Na pool. The fourth will be a spare for this system and the fifth will have a range of from +6 in. to -14 ft below the top of the sodium pool.

(3) Four biaxial accelerometers will be located on the UIS for vibration monitoring during initial startup activities.

(4) Two drywell surveillance thermocouples will be provided on the periphery of the core to measure sodium temperature.

No additional temporary in-vessel instrumentation is planned for monitoring initial startup.

4.4.6 Development Testing

Extensive out-of-reactor development testing has been completed, is under way, or is planned to support the reactor and in-vessel component thermal/hydraulic design. The test program consists primarily of water tests (although some tests using sodium have been run) on full-size core assemblies and reduced-size models of the reactor inlet and outlet plenums (including the components contained therein). Tables 4.21-4.27 summarize the out-of-reactor test program. The test program is structured to look at flow distribution, flow-induced vibration, pressure drop, mixing and thermal striping, core assembly orifice characterization, and confirmation of analysis codes. This test program provides much of the basis for verifying that the in-vessel flow rate, component vibration, ΔP, and thermal limits will be met. For the fuel, blanket, and control assemblies additional in-reactor testing has been completed or is under way in EBR-II and FFTF which will also provide input to the thermal/hydraulic design of these assemblies. These tests are discussed in Section 4.2 of the PSAR.

Summary and Conclusions

The CRBR thermal/hydraulic design is in the preliminary stage of development. The flow allocations selected have to be confirmed by final analysis of the in-vessel components; many of the development tests still need to be completed and/or documented. The design criteria are, however, acceptable. The staff's summary and conclusions regarding the design and criteria are discussed below.

The design criteria developed by the applicants are designed to prevent fuel, blanket, or control pin failure for design-basis events up to and including unlikely events, to maintain coolable geometry for all extremely unlikely events, to keep the temperature and hydraulic loads of the in-vessel components within the limits assumed in their design and safety analysis, to prevent gas entrainment, and to minimize the potential for flow blockage. In addition, criteria on maximum flow velocities are given to preclude excessive erosion, excessive flow-induced vibration, and excessive hydraulic forces on in-vessel components. These criteria are conservative and are consistent with the principal design criteria and with similar criteria used on FFTF. The criteria requiring the maximum sodium temperature to remain below the Na boiling temperature is also considered conservative since it will maintain the in-vessel thermal/hydraulic conditions in a region where the heat-transfer characteristics are well known. The staff considers these criteria acceptable.

With one exception (discussed later) these criteria have been applied and a design has been developed which has the potential of operating as intended with margin to the criteria limits.

The design methods used apply conservatisms in many areas and the key features of the design have been or are to be confirmed by development testing. Extensive out-of-reactor development testing has been completed, is under way, or is planned to support the reactor and in-vessel component thermal/hydraulic design. The test program consists primarily of water tests (although some tests using sodium have been run) on full-size core assemblies and reduced-size models of the reactor inlet and outlet plena, including the components contained therein. Tables 4.21-4.27 summarize the out-of-reactor test program. The test program is structured to study flow distribution, flow-induced vibration, pressure drop, mixing and thermal striping, and core assembly orifice characterization, and to confirm analysis codes. No hydraulic instabilities have been found or are expected since this system is entirely one-phase flow. This test program will provide much of the basis for verifying that the in-vessel flow rate, component vibration, pressure drop, and thermal limits will be met. For the fuel, blanket, and control assemblies, additional in-reactor testing has been completed or is under way in EBR-II and FFTF that will also provide input to the final thermal/hydraulic design of these assemblies. The planned test program appears adequate to confirm the thermal/hydraulic design.

The design and development approach being used on CRBR is very similar to that used on FFTF. Many of the CRBR design criteria are similar to those used on FFTF, and the use of extensive water, sodium, and in-reactor development testing, both full-scale and scale model testing, to verify the design and design methods parallels the FFTF development. In addition, key features of the design will be demonstrated by plant testing during initial startup, similar to what was done on FFTF.

The major conservatisms in the analysis are discussed in Section 4.4.4 of this SER. In general, the analysis contains many areas where conservatisms are applied; the results are a set of calculations which bound expected worst-case uncertainties and which demonstrate that the proposed design is acceptable for a construction permit and has a reasonable expectation of undergoing the final design process with a minimum of change. The conservatisms meet or exceed those required by SRP Section 4.4 and are applied to both steady-state and transient conditions. In addition, the applicants have committed (letter, Dec. 20, 1982, #HQ:S:82:149) to perform (during initial startup) whole-plant tests to demonstrate natural circulation and DHRS performance and to obtain sufficient data to confirm the analysis methods used for these events (i.e., will enable verification of analysis methods for conditions other than the test condition). The staff considers this an acceptable approach at the construction permit stage since calculations done to date indicate that adequate natural circulation will be present in CRBR and, if for some reason adequate natural circulation capability is not demonstrated during initial startup, a fallback position exists of providing additional independent power supplies to the decay heat removal systems.

The design does provide effective measures to preclude assembly inlet flow blockages and core loading errors which could lead to reduced flow. This is accomplished by the following design features:

(1) a lower inlet plenum design which strains out all particles larger than 1/4-in. diameter and precludes flow blockage to one or more assemblies by a series of multiple flowpaths

(2) a fuel assembly design which strains out particles greater than 0.056-in. diameter (0.033-in. diameter for blanket assemblies)

(3) core assembly designs with slotted flow inlets which, even if the assemblies lose hydraulic holddown and rise up and touch the upper internals structure, will still allow adequate flow to the assembly

(4) discriminator feature at the bottom of each core assembly which mates with a corresponding feature in the core support structure to preclude it from being inserted in a core region for which it is not designed

Only flow blockage associated with assembly fabrication or failure to discharge a spent assembly when it reaches its lifetime limit are not covered by the above. As with any reactor these events will be precluded by administrative control, quality assurance, and assembly inspection, such as the airflow test to be done on each removeable assembly subsequent to fabrication.

Assembly outlet flow blockage during refueling is not a concern since analysis (letter, Dec. 23, 1982, #HQ:S:82:155) has been provided demonstrating that radial heat conduction alone is sufficient to remove heat from a blocked assembly. Outlet flow blockage at power is not considered credible since pump flow would tend to remove any object on top of the core and large objects would be detected via the outlet thermocouples in the UIS.

The prevention of gas entrainment in the sodium and the buildup of gas in natural collection areas in the vessel and primary HTS has also been a design consideration. The safety concerns with gas in the primary system are:

(1) the effect gas passing through the core has on heat transfer and reactivity

(2) the potential for trapped gas to expand upon reduced pump discharge pressure and enter the core or cut off flow in a loop

The system provides adequate features to eliminate this problem by providing gas vents at natural collection points (core support structure, top of reactor vessel, top of IHX, pump tank) and by providing a suppressor plate below the top of the Na pool in the reactor vessel to reduce turbulence and stop vortex formation. Even if a gas bubble did enter the core, the effect on reactivity and cladding temperature is predicted to be well within the ability of the plant to accommodate, as documented in PSAR Section 15.2.3.2.

The permanent core instrumentation appears adequate for confirming the vibration characteristics of the in-vessel components and monitoring routine operation, and should provide good diagnostic information on core behavior. The applicants will establish a preoperational and initial startup test program in accordance with RG 1.68 to measure and confirm thermal/hydraulic design aspects.

Safety-grade in-core instrumentation is not considered necessary because the design will include features to minimize the potential for flow blockage and loading errors, to rapidly detect local fuel failures, and to detect and terminate core-wide events. In addition, the power distribution is to be maintained within limits by operation of the control rods in a banked configuration as required by Technical Specification limits.

The detailed modeling, methodology, and verification of the design codes was not reviewed at this stage. The results of the staff's overcheck calculations (discussed later), the fact that the applicants have completed or are planning to complete extensive development testing to verify their predictions, and the fact that the applicants have committed to perform in-plant demonstration tests of natural circulation and DHRS performance provides confidence that all codes to be used in the design will be verified and acceptable calculational techniques used for final design. A more extensive review of the codes is planned at the OL review stage.

Because of (1) the similarity of the CRBR thermal/hydraulic design to the FFTF design (fuel assembly designs are for all practical purposes hydraulically identical and the major in-vessel components have the same basic hydraulic features; i.e., three loops, lower and upper plenums separated by a core support structure, horizontal baffle plates, upper internals structure) and (2) the extensive thermal/hydraulic development testing programs in support of CRBR, there is a high probability that the final design can be shown to meet all of its goals and requirements. It should be noted, however, that it is the staff's opinion that continued FFTF operation is required to confirm the CRBR fuel, blanket, and control assembly design, since this will provide data, under prototypic irradiation conditions, on assembly performance which for a new plant design such as CRBR is considered necessary.

To verify the applicants' predictions of system performance, the staff had performed independent calculations of the applicants' base case steady-state, natural circulation, and DHRS plant performance predictions. The object of these calculations was to overcheck some of the most key and severe plant conditions and to compare independent predictions with those of the applicants. Although exact agreement between the staff's and applicants' calculations was not considered necessary, it was the staff's opinion that the overcheck calculations should show the applicants' analysis to be conservative; the design limits met and the trends observed during the transient events should be similar for the staff and applicants' calculations. With the above shown for several different events, the applicants' analysis methods would be considered sufficient for analyzing the proposed design at the CP stage. The comparison of calculations (applicants vs. staff) was done on a case-by-case basis and a judgment was made on each regarding compliance with the above. The overcheck calculations performed are listed below:

(1)	In-vessel flows and temperatures for the steady-state/full-power condition	Performed by ANL using the COMMIX code
(2)	In-vessel and HTS loop flows and temperatures for the scram to natural circulation from full power	Performed by ANL and BNL using the COMMIX and SSC codes
(3)	In-vessel and primary HTS loop flows and temperatures for the scram to DHRS heat removal from full power.	Performed by ANL and BNL using the COMMIX and SSC codes

Results of the overcheck calculations are summarized as follows:

(1) The calculated steady-state flows and temperatures were very close to those predicted by the applicants and the design predictions in this area are considered acceptable.

(2) The calculated plant response to a scram to natural circulation indicated close agreement with similar values calculated by the applicants. Examples of the results and agreement are shown in Figures 4.12-4.14. In all cases the design criteria (no sodium boiling) were predicted to be met, the calculated trend of the transients were similar, and the applicants' results were close to or more conservative than the overcheck calculations.

(3) The calculated plant response to a scram to DHRS heat removal indicated close agreement with the applicants' prediction (see Figure 4.15). In addition, the sensitivity and correctness of the applicants' assumption on in-vessel flow short circuiting (amount of DHRS vessel inlet flow which goes directly out the DHRS outlet nozzle without mixing in vessel) were investigated. Figure 4.16 shows the results of the sensitivity investigation. The applicants assumed a 20% value for short circuiting in their analysis, and the effect of higher or lower short circuiting is shown in this figure.

A detailed in-vessel thermal hydraulic calculation using the COMMIX Code was run to determine the amount of short circuiting. The results of this calculation implied little or no short circuiting was taking place, thus indicating that the 20% short-circuit assumption in the applicants' analysis is conservative.

In the cases analyzed the design criteria (primary sodium bulk temperature < 1,130°F) were predicted to be met, the calculated trend of the transients was similar, and the applicants' result was close to the overcheck calculation.

From the above results it is concluded that the analysis methods used by the applicants provide a reasonable prediction of system behavior and that it is highly unlikely that the proposed plant design will be capable of meeting its design limits. This conclusion is based upon staff judgment that since the analysis methods used by the applicants are satisfactory for the conditions selected for overcheck, it is reasonable to expect that they are also valid for conditions and events other than those overchecked by the staff. Further over-checks and comparisons with test data will be performed in support of the staff's operating license review to verify the above.

Staff review has identified one area where the CRBR design needs to be modified as part of final design to adequately implement the design criterion on prevention of control rod flotation. This area is discussed below.

Control Rod Flotation

The prevention of control rod flotation from inadvertent start of the primary pumps during refueling is a requirement on the thermal/hydraulic design of the reactor. Since, under worst-case conditions, flotation of six or more control rods during refueling has the potential for causing unprotected critical or prompt critical operation, the importance of this requirement is emphasized. For the secondary control rods, the prevention of flotation is accomplished by a hydraulic holddown feature which is an integral part of the design of the secondary control assembly. This feature ensures that there is always a net downward force on the movable bundle, regardless of the magnitude of flow through the assembly.

For the primary control rods, the prevention of flotation is accomplished by limiting (via control assembly orificing) the flow rate through the assembly to a value which will not cause a sufficient ΔP on the movable absorber bundle to lift it. The applicants have performed a full-scale water test on a prototypic primary control assembly to determine the flow and ΔP necessary to lift the absorber bundle. This flow rate and ΔP were determined to be 55,000 lb per hour and 7.2 psi, accounting for test uncertainties. The nominal flow rate and ΔP of the primary control assemblies at plant expected operating conditions is 49,600 lb per hour and 5.94 psi. With allowance for worst-case uncertainties in the nominal flow rate and ΔP caused by manufacturing tolerances, core inlet flow maldistribution, and so forth, and considering the fact that the primary pumps have an output capability in excess of the plant expected operating conditions, it is conceivable that the rod bundle pressure drop could exceed 7.2 psi and the movable absorber bundles in the primary control assemblies would lift because of an inadvertent start of the primary pumps during refueling conditions. Therefore, the potential for flotation of the primary control assemblies needs to be reevaluated. Since the primary control assemblies are removable components and since a fall-back position of reducing flow in the assemblies exists, resolution of this problem is not considered necessary for issuance of a construction permit. However, during final design the applicants must make modifications to the design, provide a positive means of absorber bundle holddown, or otherwise demonstrate that the primary control assemblies will not lift with sufficient confidence to justify exclusion of this event from the design-basis spectrum.

Additional items resulting from staff review which should be addressed as part of final design are:

(1) The investigation of the unplanned ΔP increase in the FFTF primary system during cycle 1 operation should be documented and the results of the investigation factored into the CRBR design. It should be noted the applicants have committed to do this (letter, Dec. 6, 1982, #HQ:S:82:139).

(2) The results from the latest power-to-melt test in FFTF should be factored into the fuel design. It should be noted the applicants have committed to do this (letter, Dec. 6, 1982, #HQ:S:82:139).

(3) The final design should document the specific thermal and hydraulic conditions assumed in the design analysis of all in-vessel components and document that these conditions are satisfied by the final design. As committed to (letter, Dec. 6, 1982, #HQ:S:82:139), the applicants will consider the use of passive in-vessel dosimetry to characterize the component's environment where the margin between the calculated environment and the environment used for design is not considered adequate.

(4) Analyzing the removable core components to plant expected operating conditions (PEOCs) will limit plant operations to PEOC. This restriction should be reflected in the plant Technical Specifications.

(5) The hot channel factors (HCFs) and their methodology to be used in evaluating natural circulation should be demonstrated to be conservative. Currently, the full-power values of the HCFs are used in the natural circulation analysis along with the many other conservatisms in the analysis. The overall results appear conservative; however, it is the staff's opinion that since the applicants consider the HCFs by themselves to be conservative, this needs to be demonstrated.

With adequate attention during final design to the items listed above, the staff concludes that the thermal/hydraulic design of the core meets the requirements of SRP Section 4.4 and PDC 8 and 60, and is acceptable for issuance of a construction permit.

4.4.7 Loose Parts Monitoring System

The applicants' loose parts monitoring system (LPMS) is described in PSAR Section 7.5.13.

<u>Acceptance Review</u>

SRP Section 4.4 (NUREG-0800) specifies review procedures of the loose parts monitoring system. The SRP requires that the design criteria, instrument types, location, and mounting be reviewed at the construction permit (CP) stage. In addition, the SRP references the following review documents.

(1) RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

(2) PDC 1, "Quality Standards and Records."

(3) PDC 13, "Instrumentation and Control."

Discussion--State of the Art of Noise Signature and Loose Parts Monitoring

Sensors used are generally one of two types: (1) acoustic sensors, which are accelerometers of the piezoelectric type, hardened against radiation and relatively high temperature and (2) ceramic or metal wave guides, which transport sound away from hostile radiation or temperature environments to points where amplifying equipment can be safely mounted. The methods and locations for mounting sensors depend on the data desired, the type of sensor used, reactor configuration, and the ambient radiation and temperature environments. Sensors are mounted on the reactor vessel and major components in the primary system, or on primary system piping, or in the containment building several feet away from the reactor vessel and major components in the primary system.

The piezoelectric sensors produce low-level signals and must be amplified. Signals are carried in shielded, low-noise cable. This cable protects the signal against radiation and high temperature. Signals can also be filtered and sent through discriminators depending upon the source, frequency, calibration problems, and the ultimate use for the data.

The signals are connected to alarms, tape recorders, audio outputs, microprocessors, and jacks for auxiliary output. Sensors are chosen for ability to withstand both the ambient temperature radiation and atmospheric environment, and consideration of the frequency range to be monitored and the reliability predicted. Where noise signature is desired, fewer sensors are required than for loose parts monitoring. The requirement for diagnostic capability to locate potential loose parts also increases the locations to be monitored as well as increases the number of sensors required.

Where diagnostics are specified and are to include determination of location of loose parts, the sensors must be located to sense vibrations from definable geographic locations in the primary system. In addition, sufficient sensors are needed to be able to perform at best, three-dimensional triangulation. If the diagnostics are to include size, weight, and configuration of a loose part, the calibration requirements of the sensor array are very exacting. It is necessary to execute not only credible initial sensor calibration, but it is also necessary to provide access for periodic recalibration, maintenance, and replacement of sensors.

Hanford Engineering Development Laboratory (HEDL), the operator of the fast flux test facility (FFTF), is in the process of testing a loose parts monitor in the upper plenum of the FFTF reactor. The FFTF upper plenum sodium temperatures and flow rates and gamma radiation levels are comparable to those of CRBR. The FFTF monitoring system under test consists of an array of high-temperature sodium-immersible microphones and accelerometers mounted on a standard FFTF reactor component called a post-irradiation open test assembly. The microphones are read from a processing computer which receives signals both directly and processed through a system consisting of a band pass filter, RMS converter, and low pass filter. The arrangement provides level tracking to account for variations in acoustic background and coolant flow. Both the accelerometers and microphones are of the piezoelectric type. Initial quantitative test data are being analyzed. Early reports indicate initial calibration and background noise identification have been accomplished satisfactorily to provide normal operation noise signatures at various flow rates and temperature conditions.

Staff Position

In the staff's opinion, it is feasible and desirable to install in CRBRP a comprehensive noise-surveillance system to include loose parts monitoring and loose parts disgnostic capability. This system should monitor noise signatures of the reactor vessel, primary piping, primary pumps, and intermediate heat exchangers. Noise signatures of normal operation at all specified coolant flow rates and power demand levels are to be determined for each major primary system segment having unique noise characteristics.

For the reactor vessel, primary piping, primary pumps, and the intermediate heat exchanger, calibration of diagnostic noise monitoring equipment should be performed initially, before startup. The noise monitoring sensors, conductors, amplifiers, filters, and other integral component parts of the noise signature and loose parts monitoring system should be capable of being inspected, tested, recalibrated, and replaced as necessary, at the least during refueling operations. The primary system sensors should be of sufficient quantity and so located that three-dimensional triangulation can be performed for location of loose parts.

The applicants have committed to design and install a loose parts monitoring system. The applicants provided a description of the loose parts monitoring system in Amendment 74 of the PSAR. Design requirements are expected to be established within the period November 1982-May 1984. The applicants have determined a list of general design criteria to be followed in creating the design for the CRBRP loose parts monitoring system. The criteria are listed below:

(1) The CRBRP project will utilize RG 1.133 in implementing a loose parts monitoring system except where differences between light-water-reactor (LWR) and liquid-metal-fast-breeder-reactor (LMFBR) technology require different methods.

(2) The loose parts monitoring system should meet the requirements of Position C.1.g. of RG 1.133, Rev. 1.

(3) Sensors shall be located to detect loose parts at natural collection points for each of the components listed in Item (4).

(4) Sensor locations will include, as a minimum, reactor vessel, primary heat transport system pump, intermediate heat exchangers, intermediate heat transport system pump, steam generator modules, reactor vessel head, and/or upper plenum.

(5) Sensors shall be proven state of the art, consistent with LWR technology, modified as necessary for CRBRP environment, and shall be redundant.

(6) Methods of mounting sensors shall be either direct mounting to components/ piping or by attachment to suitable standoff mounting.

(7) Sensitivity (threshold energy) shall be adequate to identify all loose parts that could potentially result in degradation of above components by impacting.

(8) Suitable audible indications/monitoring of the presence of loose parts will be provided in the control room, and at other plant locations as appropriate.

(9) A baseline noise signature will be established for the loose parts monitoring system at the beginning of plant operations. A surveillance plan will be established at the operating license stage.

It is the staff's position that the applicants' commitment and scheduled time frame for delineating design requirements, as listed above, constitute an acceptable approach for adherence to RG 1.133 and applicable design criteria. The staff will review the applicants' documentation on loose parts monitoring as it is generated to ensure continued adherence to and implementation of this acceptable approach. In addition, the staff will review the system sensitivity specifications and operating procedures at the OL stage in accordance with SRP Section 4.4 guidance.

4.5 Reactor Materials

4.5.1 Control Rod Drive Systems Structural Materials

The reactivity control system for CRBR will consist of the primary control rod system and the secondary control rod system. There are three major components in each system: (1) the control rod drive mechanism (CRDM), mounted to the top of the reactor vessel closure head, (2) the control rod drivelines, connecting the drive mechanism to the absorber in the core region, and (3) the control rod assembly, located in the core region. The latter consists of a movable absorber (B_4C) pin bundle and an outer type 316 stainless steel duct assembly.

The staff concludes that the CRDM structural materials are acceptable and meet the requirements of CRBR PDC 1, 12, 24, as well as 10 CFR 50.55a. The staff reached this conclusion because the applicants demonstrated that the properties of materials selected for the control rod drive mechanism components exposed to the reactor coolant satisfy Section III and Parts A, B, and C of Section II of the ASME Code and other applicable documents.

Material choices for control rod systems were based on prior experience and data from FFTF and on the goal of maintaining adequate mechanical properties in an atmosphere of inert gas, sodium vapor, and liquid sodium.

The primary mechanisms utilize type 403 stainless steel in the motor tube and segment arms for its ferromagnetic properties. In addition, both types 403 stainless steel and 17-4 PH will be utilized in highly stressed areas (such as the segment arms and lead screws because of their high strength, respectively). The secondary mechanism will utilize many different types of materials. The major load-carrying members will be made from high-strength materials suitable for the conditions.

Regarding type 17-4 PH material, only the lead screw will be fabricated from this metal and exposed to elevated temperatures. The 17-4 PH material being utilized in the lead crew will be procured and heat treated per ASME Code SA-64 as modified by DOE Std. RDT M7-6. The latter standard permits only two heat-treat temperatures: 1,100°F and 1,150°F. The FFTF mechanism lead screw was purchased and heat treated to the same specifications. In service, the maximum temperature experienced by the lower portion of the lead screw, which is fabricated from 17-4 PH, will be in the temperature range of from 400°F to 450°F. This maximum temperature will be experienced only when the control rods are fully inserted. During reactor operation, with the rods fully or partially retracted, the service temperature of the lead screw will be less than 400°F. This latter condition should account for the major exposure time for this 17-4 PH material.

In selecting the type 17-4 PH material for this application, the early experience with this alloy involving embrittlement was considered, especially when the material is aged at a relatively low temperature such as 950°F and subjected to a service temperature in the 600°F to 800°F range. On the basis of the high aging temperature of 1,100°F coupled with a service temperature of 450°F or lower, embrittlement was not considered a problem. It was concluded that below 550°F the embrittlement is minimal. The position is substantiated by the data presented in the PSAR. The exposure time required to cause room temperature Charpy impact values to fall below 154 ft-lb, together with the exposure time to deplete room temperature impact values by one-half shown, is presented in the PSAR. This indicates that at temperatures below 500°F, essentially no effect on impact strength is observed.

Material choices for the drivelines and bellows were predicated on maintaining adequate margins of strength in the high-temperature sodium and irradiation environment. Inconel 718 will be utilized fo the primarly control rod system to take advantage of its high strength and hardness and its favorble wear and antigalling characteristics. The primary CRDM bellows will consist of a main bellows, a disconnect actuating rod bellows, and a position indicator rod bellows. All of the bellows will be fabricated of Inconel 718, and will be located above the sodium level in a low-level irradiation environment. The main bellows will be cycled as the control rod system is moved. The outer two bellows will cycle only for the refueling or maintenance modes. Before installation in the mechanism, all bellows will be required to pass a 20-cycle breaking test with subsequent leak testing. These tests have been completed without a bellows failure.

In addition, the controls imposed upon the austenitic stainless steel of the mechanisms provide that the delta ferrite requirements and determination for filler classifications E308L, E308, ER308L, and ER308 shall conform to the requirements of Section III of the 1974 ASME Code. The delta ferrite range of 5% to 9% shall be used since these limits are consistent with the Code and RDT standards. These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. Thse controls meet the quality standards requirements of CRBR PDC 1 and 28 and satisfy the requirements of CRBR PDC 12 relative to prevention of leakage and failure of the CRBR control rod drive structural materials.

Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress-corrosion cracking will not occur during the design life of the component. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of ASME Code Section III, NB-2160 and NB-3120. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with staff positions. Surface finish and cleanliness will also meet all requirements of the ASME Code and the other contract documents. Periodic swab tests of stainless steel surfaces during fabrication in the shop will be performed to ensure that potentially harmful substances such as chlorides do not contact the components in concentrations greater than specified in applicable codes and standards.

The controls used provide assurance that austenitic stainless steel components will be properly cleaned on site. The controls satisfy Appendix B of 10 CFR 50 regarding controls for onsite cleaning of materials and components.

4.5.2 Reactor Internal Materials

The lower internal structure positions and restrains the reactor core. The primary components are the core support plate, core barrel, horizontal baffle, core former structure, fixed radial sheld, lower inlet modules, flow bypass modules, and the final transfer and storage assembly. The upper internal structure, located above the core and supported from the intermediate rotating plug of the vessel closure, will stabilize and support the control rod drive lines and in-vessel instrumentation, and will provide mechanical holddown for the reactor core. Conduits in the upper internal structure are designed to mitigate and equalize temperature effects from the effluent from the reactor core. The primary components of the upper internal structure are the support columns, transverse interconnected plates, shear webs, shroud tubes, and instrumentation posts.

CRBR PDC 1 requires that structures, systems, and components important to safety shall be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The general design rule for the selection of materials for the reactor internal and core support structure is an end-of-life minimum residual ductility. The values of 10% and 5% minimum residual elongation as determined on uniaxial tensile specimens are required for specified materials in the permanent components. The 5% minimum residual elongation criterion will be used in the low stress areas of the inlet module and fixed radial shield. Similar to the fast flux test facility design, some of the components of the Clinch River reactor internal and core support structures are designed to be removed for inspection and replacement in the event in-service deterioration of integrity is suspected.

The applicants identified the materials of construction of the reactor internal and core support structures in the PSAR. The major structural materials are types 304 and 316 stainless steel, Inconel alloy 718, and weld-deposited type 308 and type 316 (or 16-8-2) stainless steel. Minor quantities of other metallic materials were specified, such as chromium, chromium carbide, Haynes 273, or aluminized type 316 stainless steel to minimize wear and/or galling tendency. Extensive test programs have been carried out and referenced by the applicants to ensure the end-of-life structural integrity and adequacy of performance in the liquid metal fast breeder environment for which they are specified.

The components for the lower internal structure were designed and analyzed in compliance with the requirements of Section III of the ASME Code, 1974 Edition, including Summer 1975 Addenda. The design temperature for the core support structure is 775°F; however, that temperature may be exceeded during certain transients for short periods of time. In the event Section III requirements cannot be applied, the following supplemental rules will be involved: RG 1.87; RDT Std. F9-4, "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components (Supplement to ASME Code Cases N-47, N-48, N-49, N-50, and N-51)"; RDT Std. F9-5, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components"; RDT Std. E15-2NB, "Class 1 Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III, Subsections NCA and NB)"; the application of special purpose strain controlled high-cycle fatigue criterion to types 304 and 316 stainless steel at temperatures up to 1,100°F (discussed in Section 4.2.2.3.2.2 of the PSAR); the material properties listed in "Nuclear Systems Materials Handbook," TID-26666, and Section 4.2.2.3.3.1 of the PSAR. The horizontal baffle and the fuel transfer and storage assembly are internal components designed to the requirements of the ASME Code, 1974 Edition, including Winter 1976 Addenda; the fixed radial shield is designed to the requirements of the ASME Code, 1977 Edition, including Winter 1977 Addenda.

The support columns and in-vessel transfer machine of the upper internal structure were designed and analyzed to the requirements of Class 1 appurtenances of Section III of the ASME Code, 1974 Edition, including Winter 1974 Addenda, and RDT Std. E15-2NB. Other components of the upper internal structure operating at temperatures below 800°F were designed and analyzed to the requirements of Section III of the ASME Code, 1977 Edition, including Summer 1977 Addenda. The components operating at temperatures in excess of 800°F were designed and analyzed to the requirements of ASME Code, Section III, 1977 Edition, including Summer 1977 Addenda, Code Cases 1592, 1593, and 1594; RDT Std. E15-2NB; RDT Std. F9-4; the "Nuclear Systems Materials Handbook," TID-26666; Technical Bulletin T-39,"Inconel Alloy 718" (International Nickel Company); and the design fatigue curve, Figures 4.2-4.8, in the PSAR.

The applicants have modified the high-temperature creep-fatigue damage rule of RDT Std. F9-4 and Paragraph T-1400 of Code Case 1592. The modification was justified on the basis of a conservative evaluation of the fatigue test data for types 304 and 316 stainless steel. The creep-fatigue damage rules of Paragraph T-1400 of Code Case 1592 consider creep-damage accumulation resulting from stresses which are compressive to be equally as damaging as creep-damage accumulation from tensile stresses. Strain-controlled fatigue test data of types 304 and 316 stainless steel consistently point to compressive residual stresses having little or no deleterious effect. Test data indicate stress concentrations having a less severe effect on stress rupture strength than predicted by the analytical approaches of RDT Std. F9-4 and Code Case 1592 criteria. In the case of type 316 stainless steel, there is a trend toward significant notch strengthening for some geometries, particularly with a service environment and life at the upper limit of those in the upper internal structures.

The modification is applicable only to nonpressure boundary components. In the service life of a component where the three principal stresses are compressive during a hold period, the creep-failure evalution is modified to be 20%, as damaging as that caused by the same stress in tension.

There are two areas of concern about materials behavior in the reactor internal and core support structure. Additional information should be provided to the staff in the FSAR to ensure the integrity of the component in these areas. The concerns are: (1) performance of alloy 718 at extremely high temperatures (~ 1,200°F) where thermal striping may occur, (2) time-temperature effect of thermal gradient on the distortion of the horizontal baffle, and (3) effect of thermal and irradiation gradients on the distortion of ducts in the control rod assembly.

The staff concludes that the materials used for the construction of the CRBRP reactor internal and core support structure have been identified by specification and found to be acceptable to meet the requirements of CRBR PDC 1. The materials are in conformance with the recommendations of RG 1.87 and to the requirements of Section III of the ASME Code.

The applicants have met the requirements of CRBR PDC 1 with respect to ensuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structure are of high quality and adequate for structural integrity.

The delta ferrite requirements and determination for filler classifications E308L, E308, ER308L, and ER308 shall conform to the requirements of Section III of the 1974 ASME Code. The delta ferrite range of 5% to 9% shall be used, since these limits are consistent twith the Code and RDT standards. These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity.

The applicants have met the requirements of the high-temperature ASME Code cases except the creep-fatigue damage rule as conservatively modified by data from research and development of creep-fatigue programs, and, further, the guidelines identified by RDT standards, precursor of the high-temperature Code cases.

The staff concludes that the materials to be used for the construction of the CRBRP reactor internal and core support structure have been identified by specification and found to be acceptable to meet the requirements of CRBR PDC 1. The materials are in conformance with the recommendations of RG 1.87 and the requirements of ASME Code Section III.

The applicants have met the requirements of CRBR PDC 1 with respect to ensuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structure are of high quality and adequate for structural integrity. The controls imposed upon components of austenitic stainless steel satisfsy the applicable requirements.

The applicants have met the requirements of the high-temperature ASME Code cases, except the creep-fatigue damage rule as conservatively modified by data from research and development of creep-fatigue programs, and, further, the guidelines identified by RDT standards, precursor of the high-temperature Code cases.

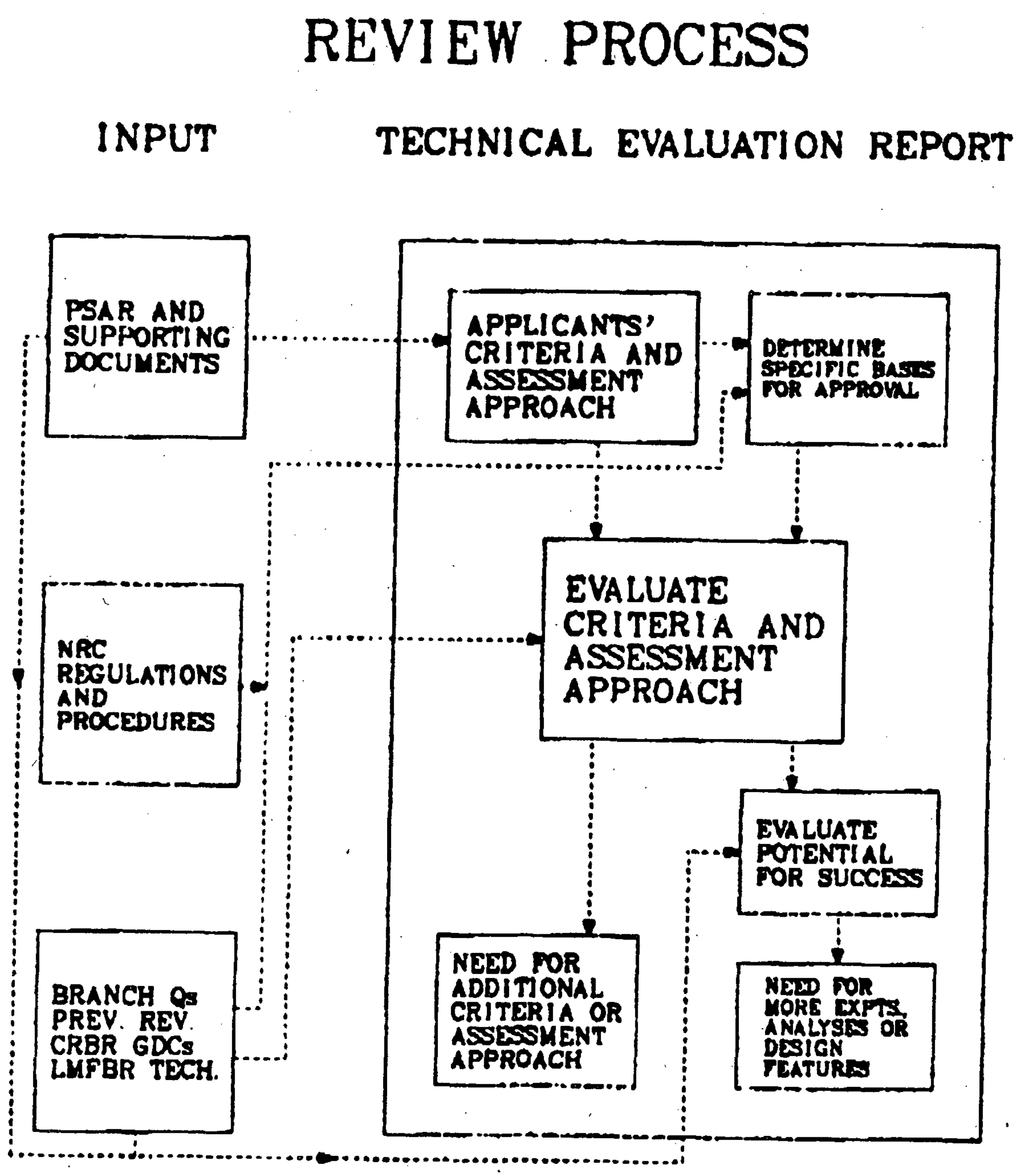

Figure 4.1 Approach to review of CRBR fuel systems

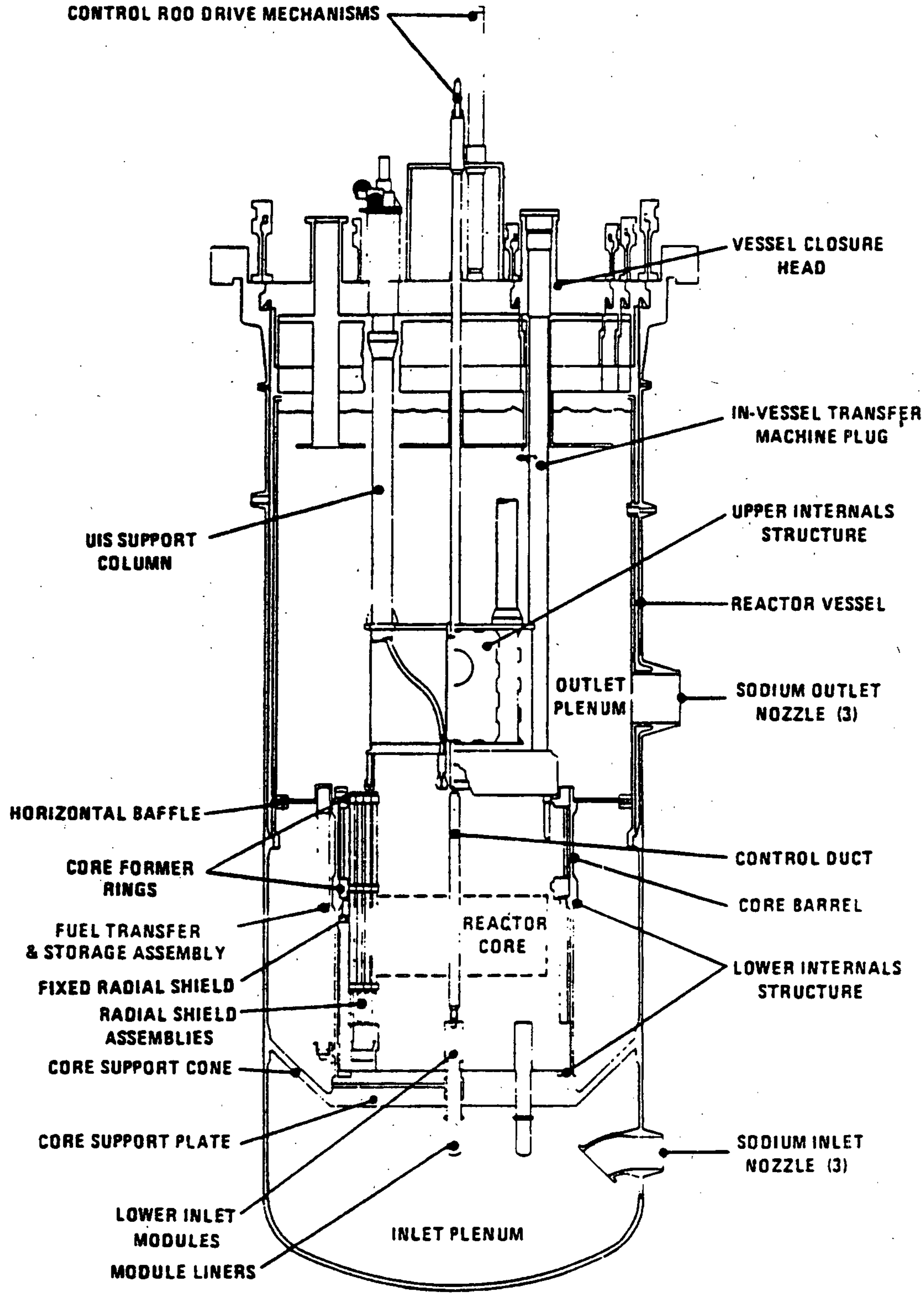

Figure 4.2 Reactor elevation

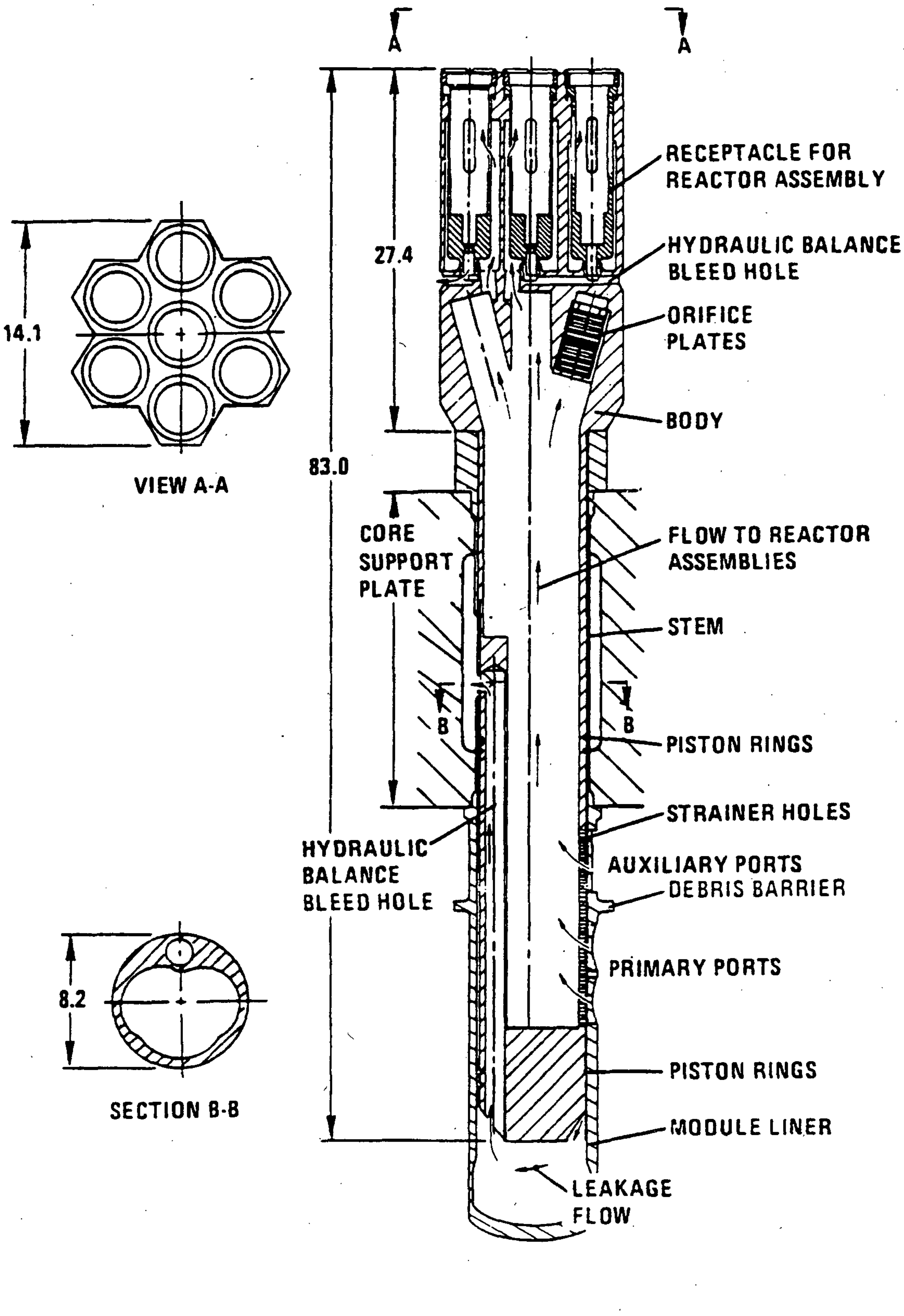

Figure 4.3 Elevation of typical lower inlet module

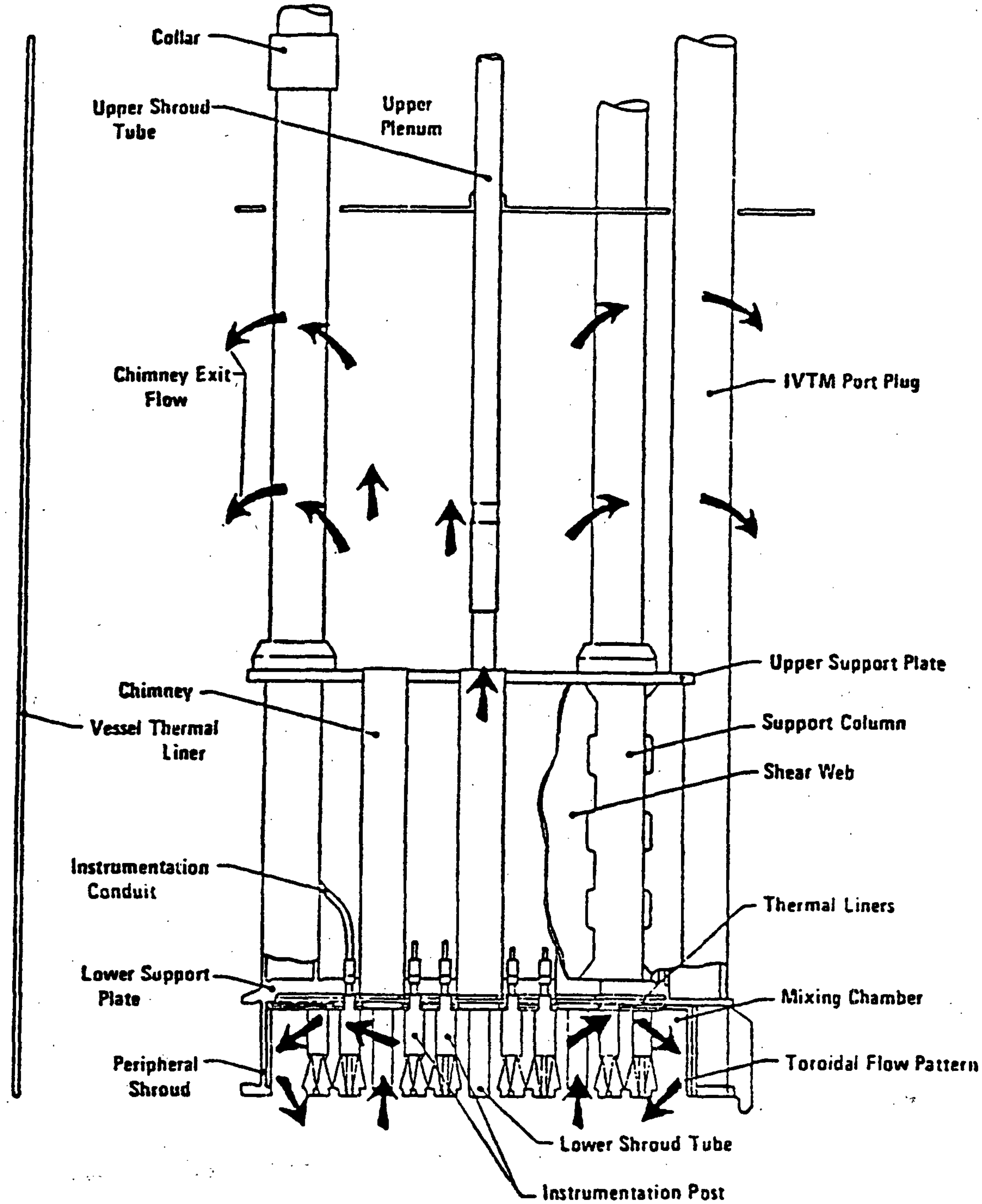

Figure 4.4 Elevation of the upper internals structure in the CRBRP

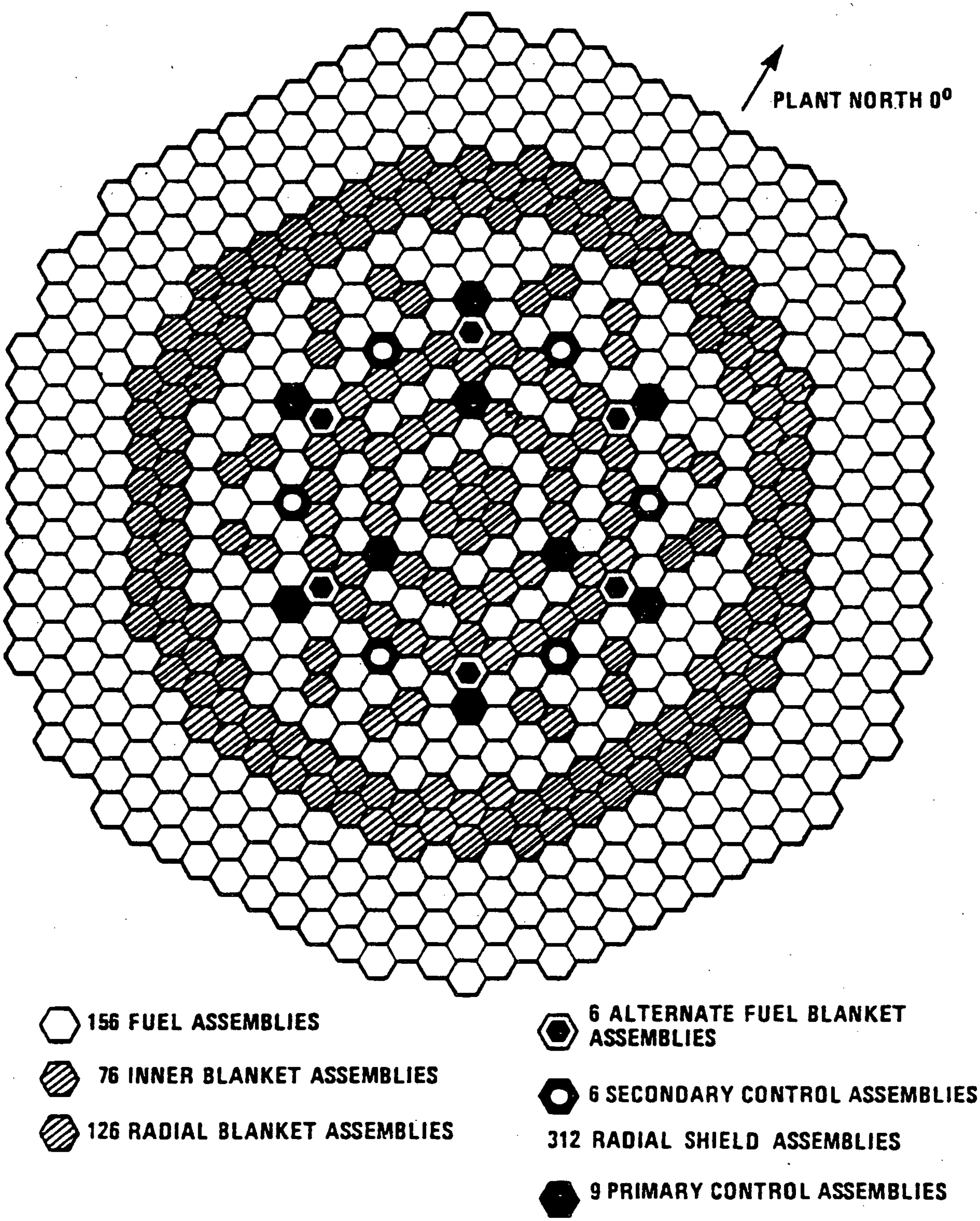

Figure 4.5 Clinch River breeder reactor core layout

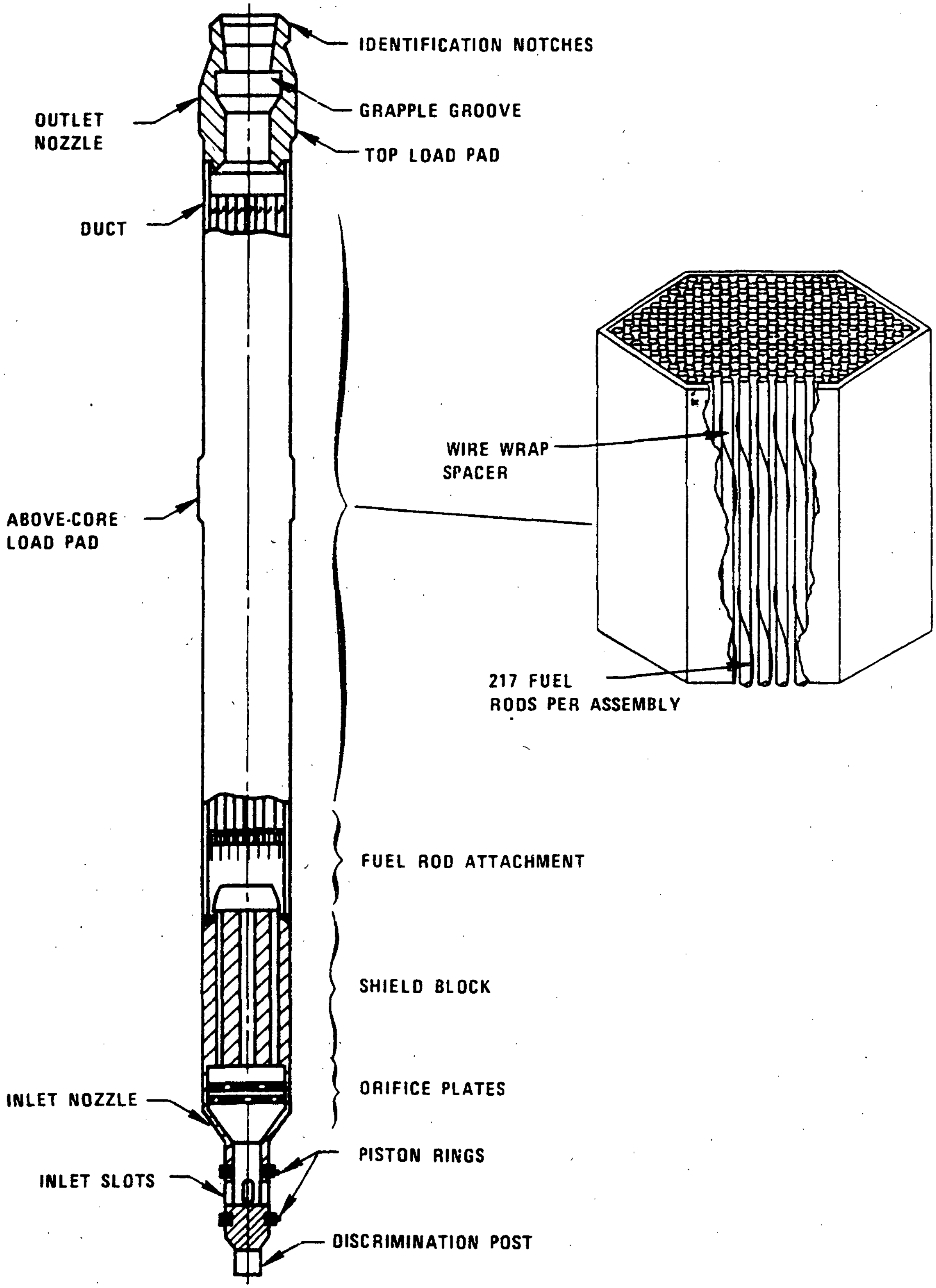

Figure 4.6 Fuel assembly schematic

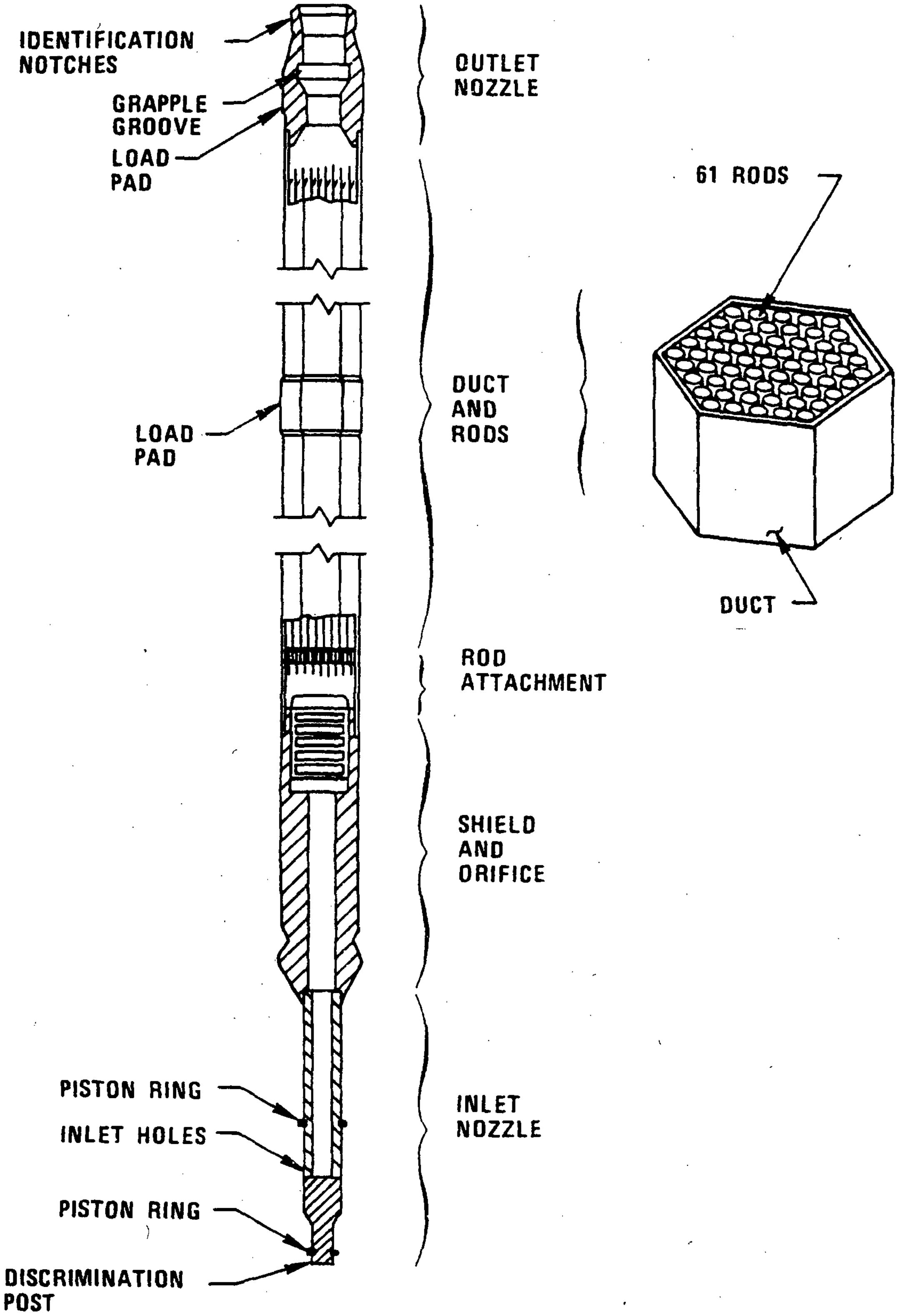

Figure 4.7 Radial blanket assembly schematic

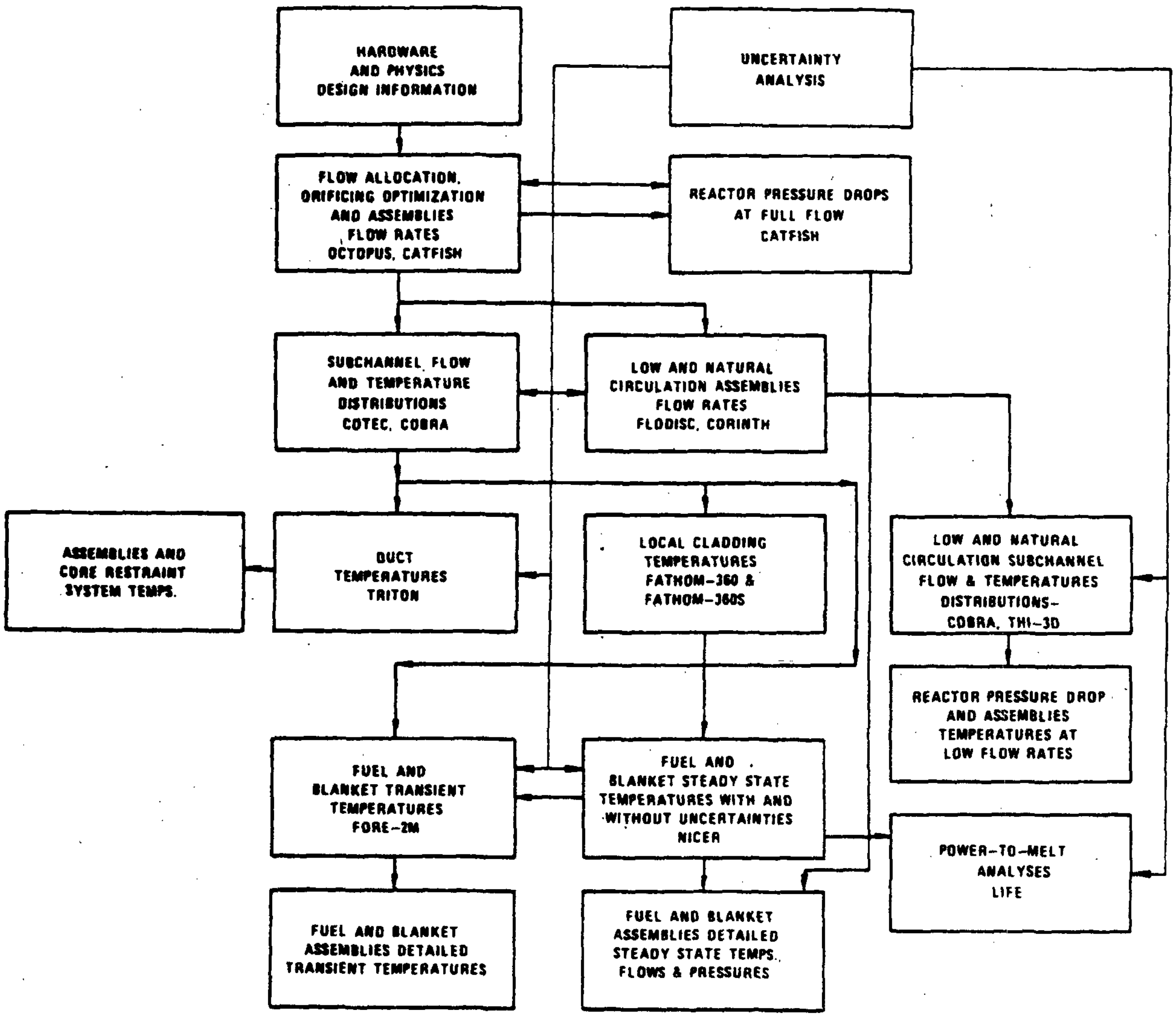

Figure 4.8 Fuel and blanket assemblies--thermal/hydraulic analysis flow diagram

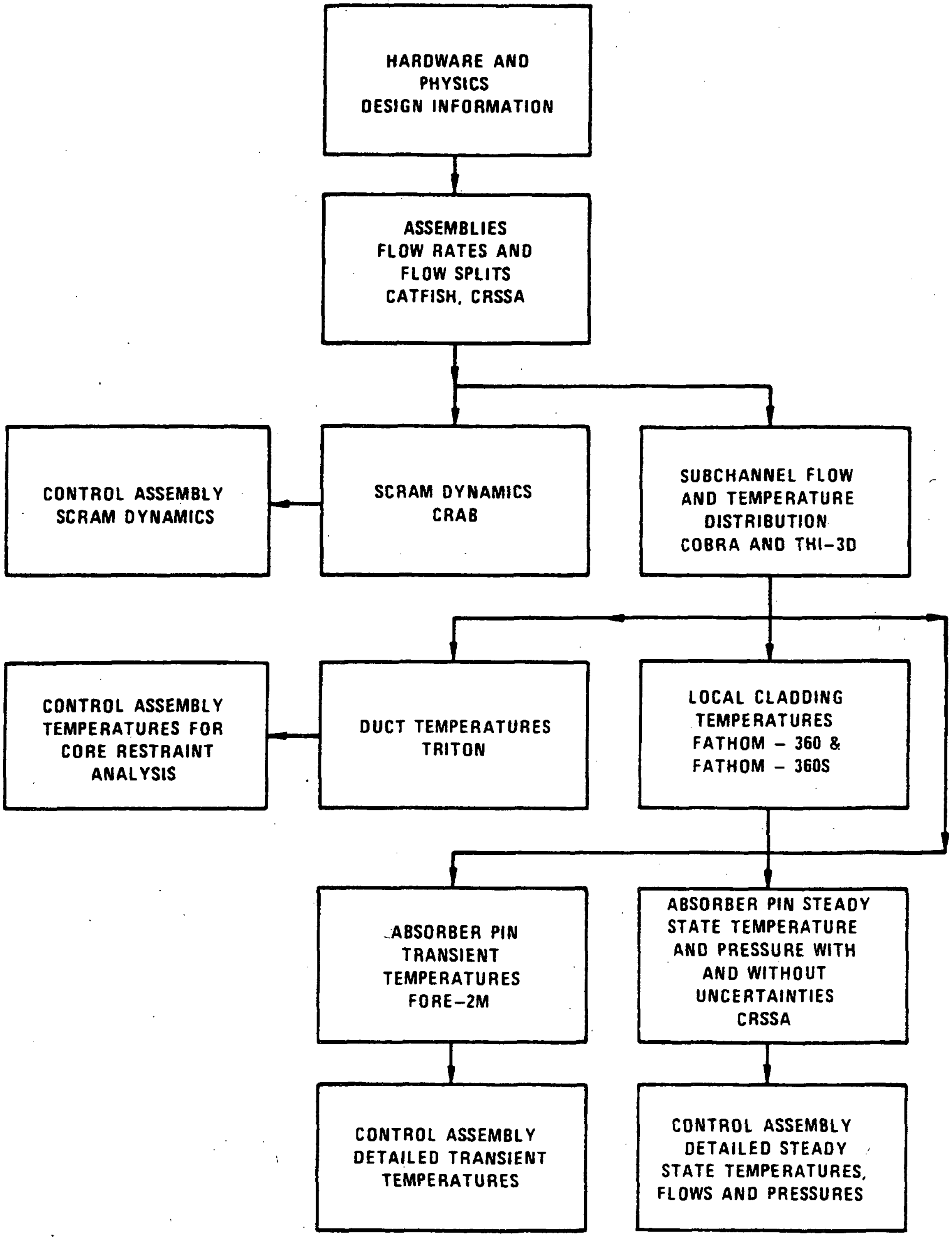

Figure 4.9 Primary control assembly and absorber pin--thermal/hydraulic analysis flow diagram

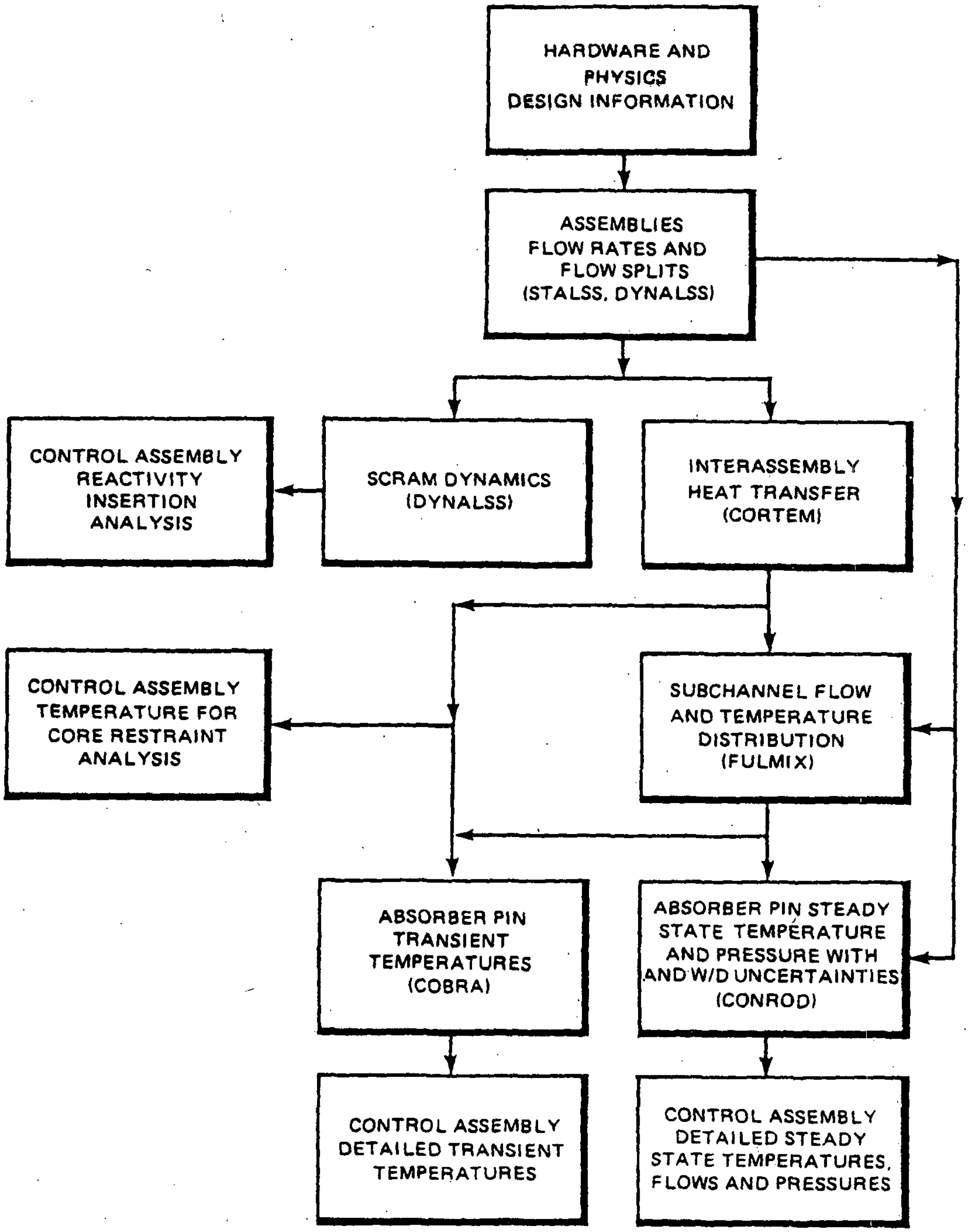

Figure 4.10 Secondary control assembly and absorber pin--thermal/hydraulic analysis flow diagram

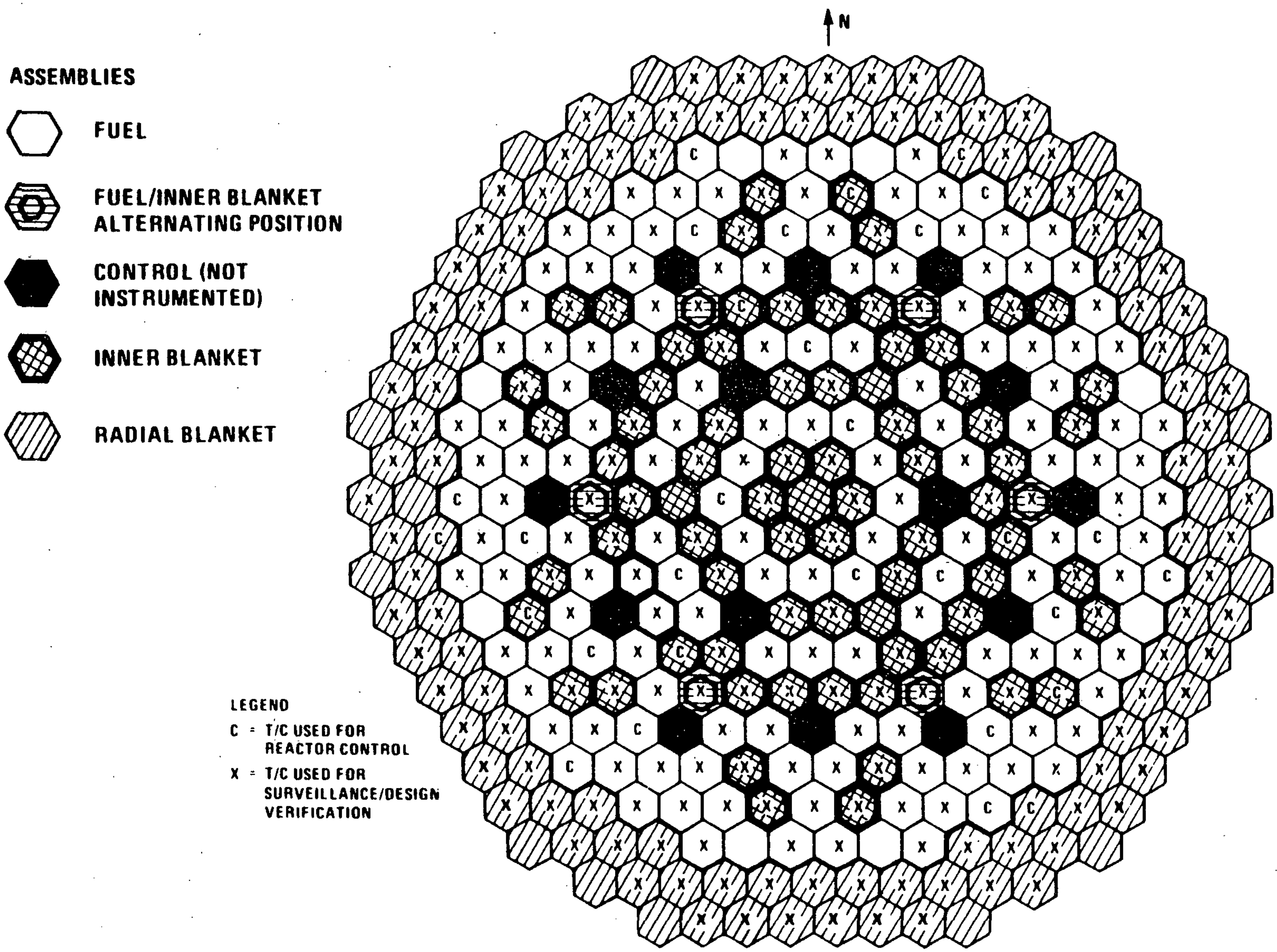

Figure 4.11 CRBRP heterogeneous core exit thermocouple coverage

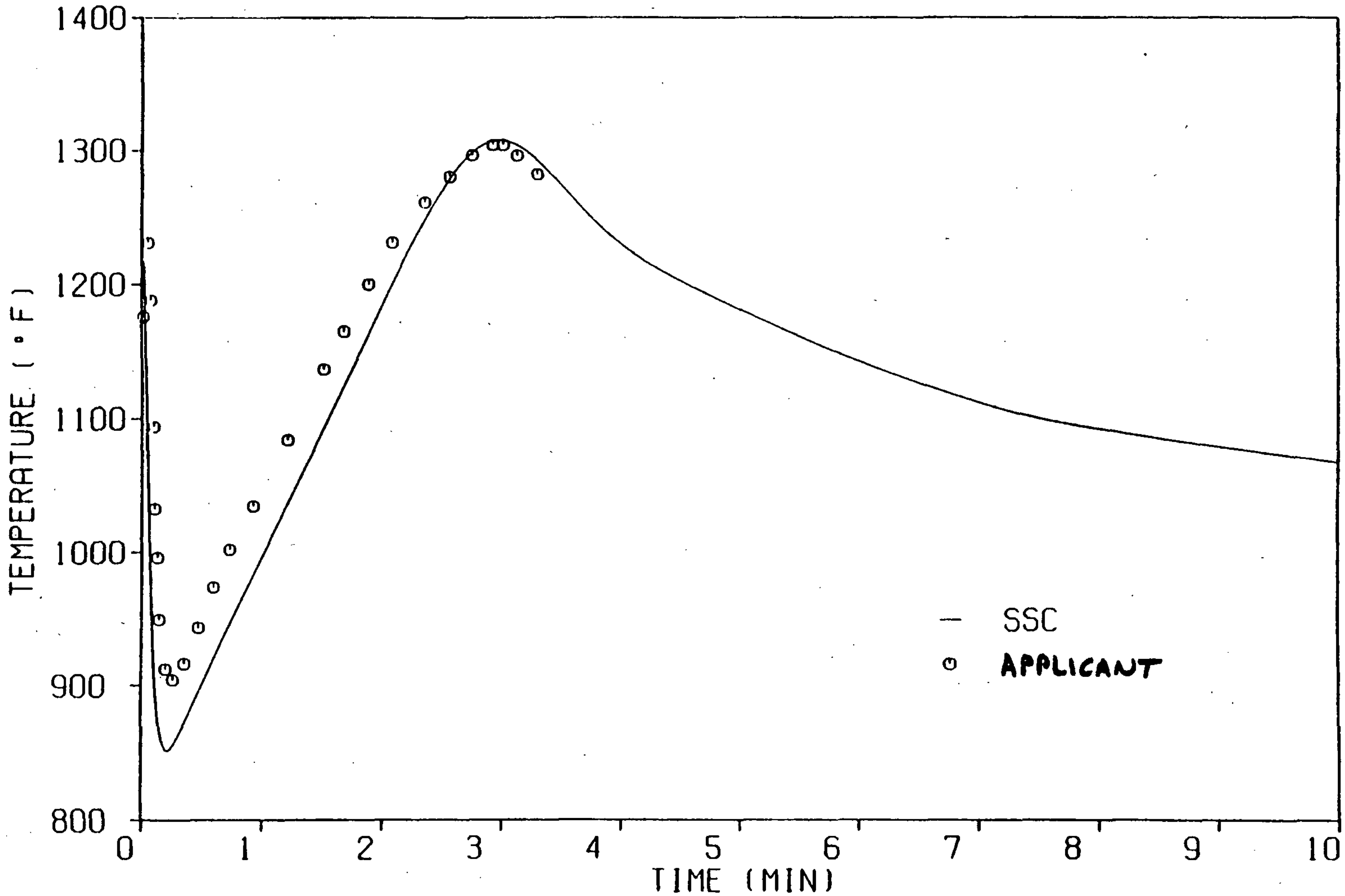

Figure 4.12 Natural circulation coolant temperature for hot fuel pin

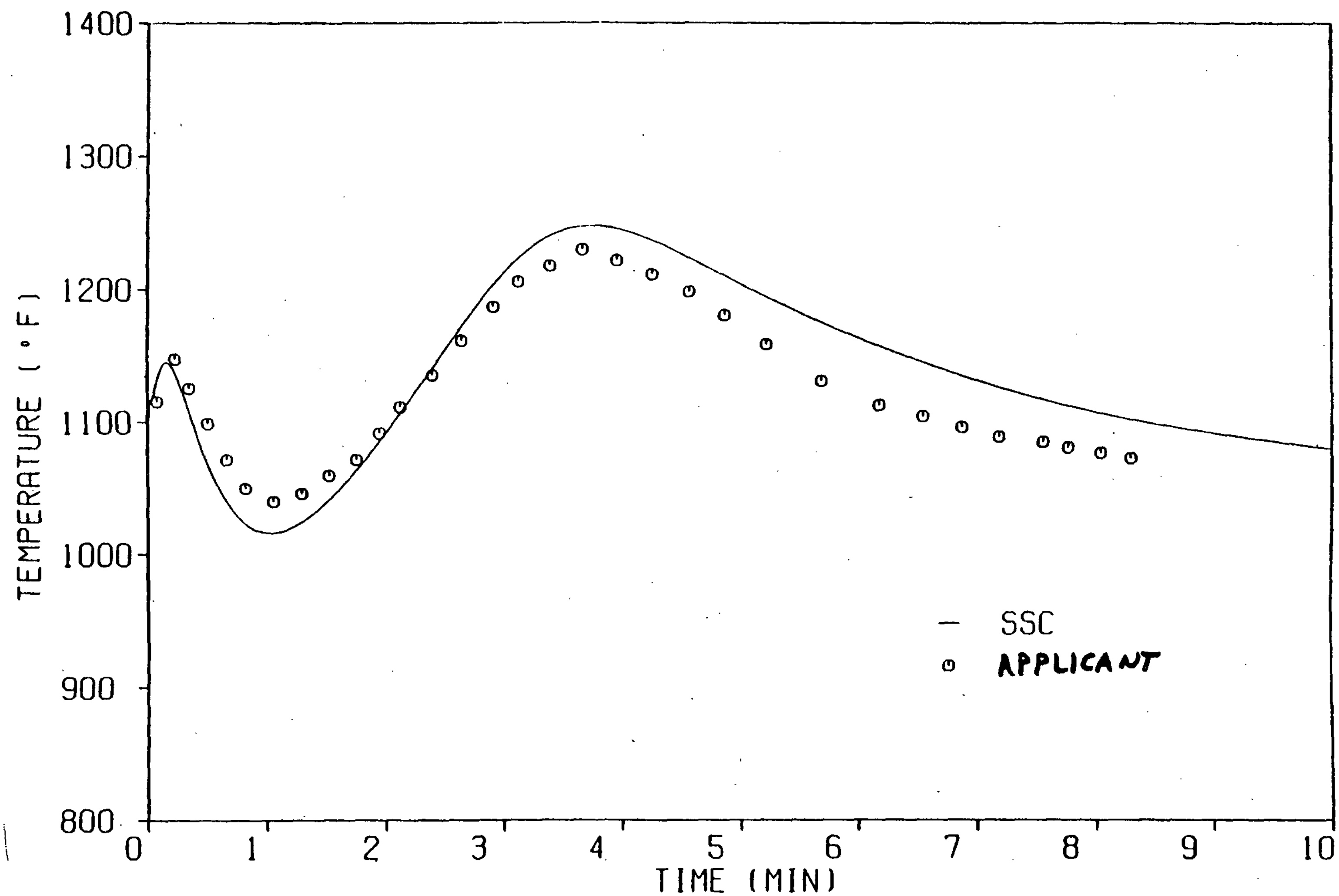

Figure 4.13 Natural circulation coolant temperature for hot inner blanket pin

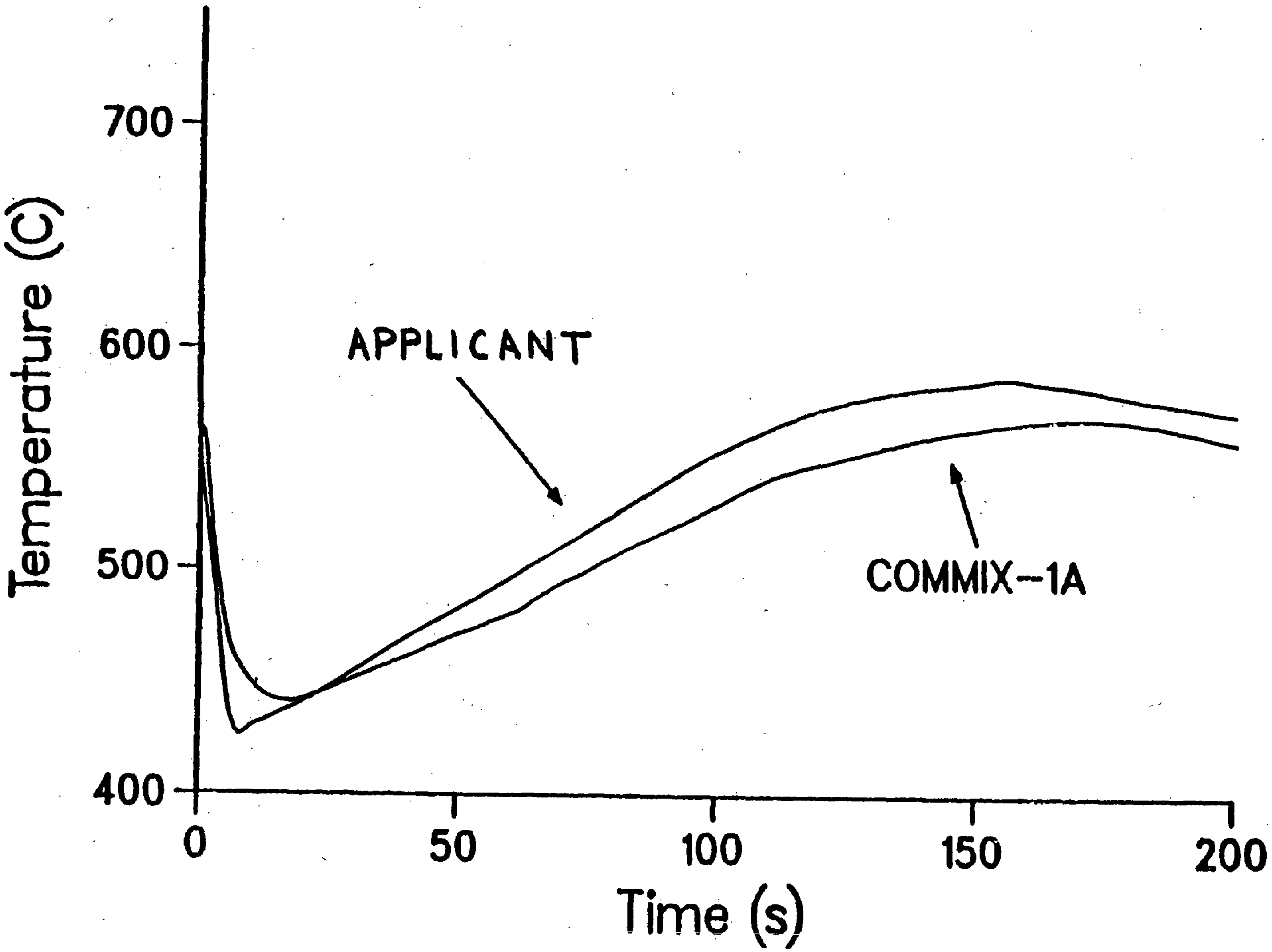

Figure 4.14 Fuel assembly outlet temperature

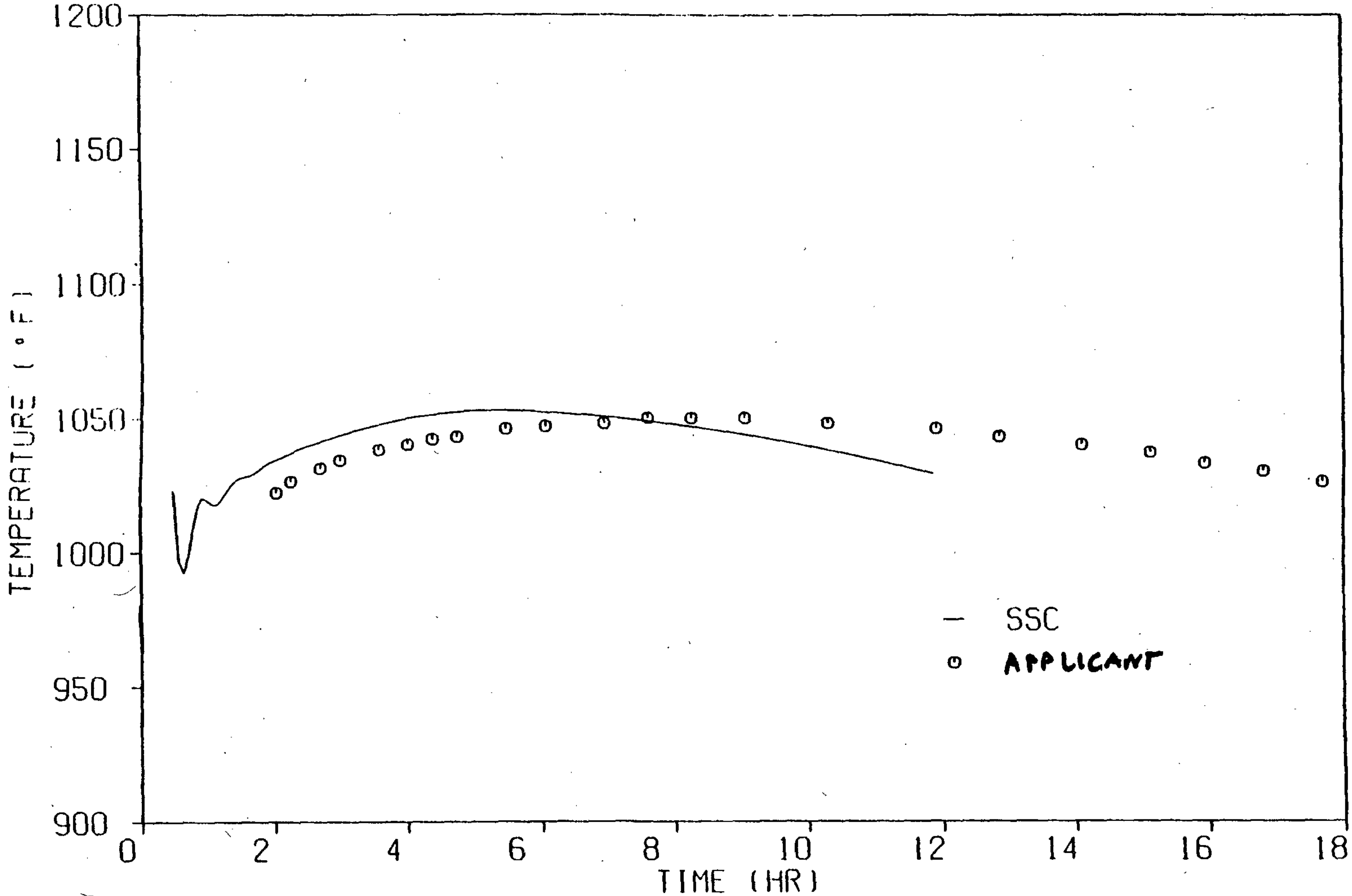

Figure 4.15 Direct heat removal service design-basis accident--bulk upper plenum sodium temperature

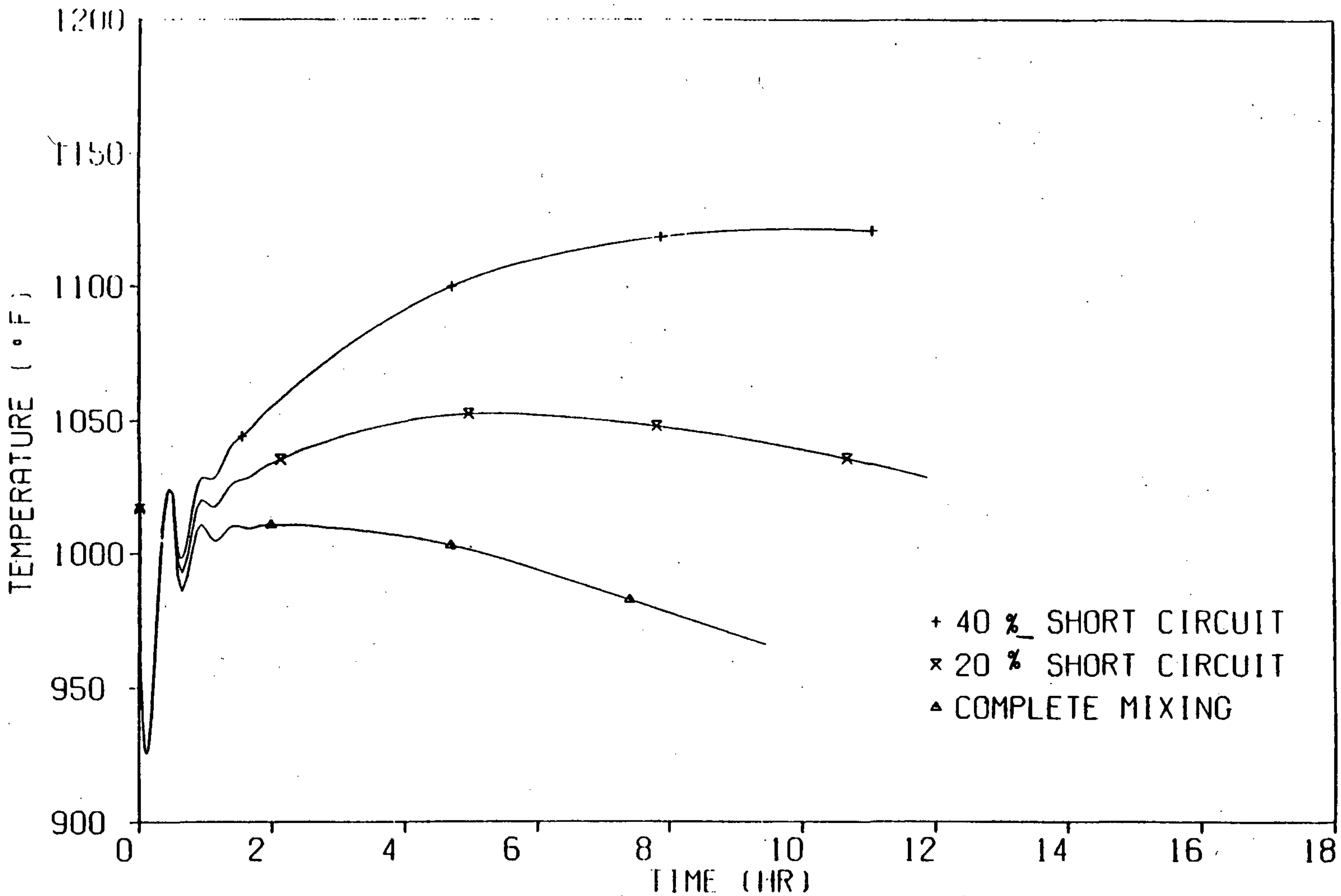

Figure 4.16 Direct heat removal service event--sensitivity of bulk upper plenum sodium temperature to variations in "short circuit" flow fraction

Table 4.1 CRBRP blanket assembly design parameters

Parameter	Value
Assemblies (number)	
Radial	126
Internal	76 to 82
Blanket rods per assembly (number)	61
Flow orificing zones (number)	
Radial	4
Internal	2
Blanket rod pitch-to-diameter ratio	1.072
Blanket rod spacing	Wire-wrap, 0.033-in.-diameter wire with 4-in. pitch
Blanket rod length (in.)	116.5
Cladding and duct material	20% cold-worked type 316 stainless steel
Rod outside diameter (in.)	0.506
Cladding thickness (in.)	0.015
Fuel pellet column (in.) (depleted UO_2)	64
Pellet diameter (in.)	0.470
Pellet length (in.)	0.5 to 1.0 at supplier's option
Pellet density (% of theoretical)	95.6
Smear density (% of theoretical)	93.2 (dish voids not considered)
Pellet material	Depleted UO_2
Diametral gap between cladding and pellet (initial, cold) (in.)	0.006
Distance across flats on duct inside (in.)	4.335
Duct wall thickness (in.)	0.120
Radial blanket rows (number)	2½
Fuel arrangement	Column of cylindrical pellets
Fission gas cold available plenum volume ($in.^3$)	8.12
Fill gas	Helium
Fill gas pressure (atm) (at ambient temperature)	1
Spacer material	20% cold-worked type 316 stainless steel
Assembly pitch (in.)	4.760

Table 4.2 Comparison of CRBRP and FFTF fuel assembly details

Parameter	CRBRP value	FFTF value
Low enrichment assemblies (number)	0	28
High enrichment assemblies (number)	156 to 162	45
Rods per assembly (number)	217	217
Rods (total number)	33,852	15,841
Rod outside diameter (in.)	0.230	0.230
Rod radial spacing	0.056-in. wire wrapped around fuel rod cladding in clockwise helical spiral with pitch of 11.9 in.	0.056-in. wire wrapped around fuel rod cladding in clockwise helical spiral with pitch of 11.9 in.
Rod triangular pitch (in.)	0.2877	0.2877
Clearance between fuel rod assemblies at wires (nominal, in.)	0.0017	0.0017
Clearance between fuel rods (nominal, in.)	0.0577	0.0577
Cladding thickness (nominal, in.)	0.015	0.015
Rod axial support	17 key-shaped rails	17 key-shaped rails
Pellet column length (in.)	64 including two 14-in. axial blankets	49 including two 5.7-in. Inconel 600 reflectors
Fission gas plenum length (in.)	48.0	42.0
Fission gas plenum available volume cold ($in.^3$)	1.287	1.158
Lower axial blanket length (in.)	14	0.8

Table 4.2 (Continued)

Parameter	CRBRP value	FFTF value
Lower axial blanket composition	Depleted UO_2	Depleted UO_2
Core region length (in.)	36	36
Core pellet material	Plutonium-uranium dioxide	Plutonium-uranium dioxide
Core pellet diameter (in.)	0.1935	0.1945
Pellet length (in.)	2.205 to 0.283	0.205 to 0.283
Density		
Nominal (% of theoretical)	91.3	90.4
Cold-smeared (% of theoretical)	85.5	85.5
Plutonium content	32.8	19.8 and 24.2
Diametral gap between fuel cladding and fuel pellet (nominal, in.)	0.0065	0.0055
Upper axial blanket length (in.)	14	0.8
Upper axial blanket composition	Depleted UO_2	Depleted UO_2
Axial blanket pellet diameter (in.)	0.1900	0.1900
Axial blanket pellet length (in.)	0.31 to 0.46	0.4
Nominal density (% of theoretical)	96.0	95.5
Diametral gap between fuel cladding and axial blanket pellets (nominal, in.)	0.0100	0.0100
Inlet nozzle distrimination post maximum diameter (in.)	1.956	1.985
Orifice zones (number)	6	3
Shield blocks (number)	1	3
Shield block total length	20.0	21.5
Shield block effective length (in.)	14.7	13.1

Table 4.2 (Continued)

Parameter	CRBRP value	FFTF value
Shield block outside hex (in.)	4.695	4.665
Load pad, outside dimension across flats (in.)	4.745	4.715
Duct across flats, dimension (inside, in.)	4.335	4.335
Duct wall thickness (in.)	0.120	0.120
Load pad thickness (nominal, in.)	0.205	0.190
Fuel rod growth clearance (in.)	2.1	1.00
Outlet nozzle inside diameter (nominal, in.)	3.60	2.80
Outlet nozzle outside diameter (nominal, in.)	3.90	3.90
Misaligned grapple pickup capability	1.75	1.25

Table 4.3 CRBRP fuel rod duty cycle normal operation

Event*	Name	Number in 30 years
N-2	Normal startup	840
N-3	Normal shutdown	270
N-4	Load following	**
N-5	Step-load changes of +10% of full load	750 each
N-6	Steady-state temperature fluctuations	30×10^6
N-7	Steady-state flow induced vibrations	1×10^{10}

*Description of events given in PSAR, Appendix B, general plant transient data.

**Daily load follow is not a design requirement. Rate of power change is to be determined.

NOTE: Fuel damage severity limit--no significant loss of effective lifetime.

Table 4.4 CRBRP fuel rod duty cycle anticipated (upset) event

Event	Name	Type	Number* in 30 years
U-1	Reactor trip from full power	Overcooling	180
U-2B	Uncontrolled rod withdrawal from 100% power with delayed manual trip	Overpower	10
U-2C(D)	Uncontrolled rod withdrawal from startup with automatic (delayed manual) trip	Overpower	17
U-2E	Plant loadings at maximum rod withdrawal rate	Overpower	10
U-16	Operating basis earthquake (30-cent step reactivity insertion)	Overpower	10

*Number of normal and upset events for fuel rod design life:
(1) Cycle 1 (128 full-power days (FPDs))--1/30 of number in 30 years to nearest whole number
(2) Cycles 1 and 2 (320 FPDs)--2/30 of number in 30 years to nearest whole number
(3) Cycles after 1 and 2 (274 FPDs)--number of transients per cycle--1/30 of number for 30 years

NOTE: Fuel damage severity limit--no reduction of effective lifetime below the design value.

Table 4.5 CRBRP fuel rod duty cycle unlikely (emergency) events

Event	Name	Type	Number
U-18(S)	Loss of all offsite power (with secondary system shutdown)*	Undercooling	1 of worst
E/6	Three-loop natural circulation	Undercooling	
-	60-cent step reactivity insertion (PPS design evaluation transient)	Overpower	

*Concurrent failure of primary shutdown systems shifts fuel damage severity limit to next highest level.

NOTE: Fuel damage severity limit--a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures.

Table 4.6 CRBRP fuel rod duty cycle extremely unlikely (faulted events)

Event	Name	Type	Number
F-1	Safe shutdown earthquake (with a 60-cent step)	Overpower	1

NOTES: Most severe transients in the duty cycle are being tested in reference fuel transient program as indicated in Table 4.2.59 of PSAR.

Fuel damage severity limit--substantial fuel melting and/or cladding distortion in individual fuel rods, but the configuration remains coolable.

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Table 4.5 CRBRP fuel rod duty cycle unlikely (emergency) events

Event	Name	Type	Number
U-18(S)	Loss of all offsite power (with secondary system shutdown)*	Undercooling	
E/6	Three-loop natural circulation	Undercooling	1 of worst
-	60-cent step reactivity insertion (PPS design evaluation transient)	Overpower	

*Concurrent failure of primary shutdown systems shifts fuel damage severity limit to next highest level.

NOTE: Fuel damage severity limit--a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures.

Table 4.6 CRBRP fuel rod duty cycle extremely unlikely (faulted events)

Event	Name	Type	Number
F-1	Safe shutdown earthquake (with a 60-cent step)	Overpower	1 of worst
-	$2/sec reactivity insertion (plant protection program evaluation transient)	Overpower	

NOTES: Most severe transients in the duty cycle are being tested in reference fuel transient program as indicated in Table 4.2.59 of PSAR.

Fuel damage severity limit--substantial fuel melting and/or cladding distortion in individual fuel rods, but the configuration remains coolable.

Table 4.7 Primary control assembly dimensions

Parameter	Value
Control assembly	
Outside distance across flats	4.575 in.
Inside distance across flats	4.335 in.
Overall length	168.0 in.
Inner duct/outer duct clearances (diametral)	
Upper wear pad	0.120 in.
Lower wear pad	0.100 in.
Duct to duct (across flats)	0.257 in.
Control rod	
Outside distance across flats of inner duct (nominal)	4.038 in.
Inside distance across flats of inner duct (nominal)	3.950 in.
Inner duct wall thickness (nominal)	0.044 in.
Number of pins	37
Pin pitch-to-diameter ratio	1.05
Pin outside diameter	0.602 in.
Pin cladding thickness	0.049 in.
Total pin length	78.445 in.
Plenum length (upper/lower)	22.700/14.220 in.
Insulator pellet length	0.700 in.
Active absorber length	36.0 in.
Pellet diameter	0.4500 in.
Pellet density	92% theory
Boron-10 loading (92% enriched) (nominal)	5.57 kg
Wire wrap diameter (interior/edge)	0.030/0.15 in.
Wire wrap pitch	12.75 in.
Pin bundle clearance (diametral)	0.034 in.

Table 4.8 Secondary control assembly dimensions

Parameter	Value
Control assembly	
Outside distance across flats	4.578 in.
Inside distance across flats	4.335 in.
Overall length	168.0 in.
Guide tube	
Outside diameter	4.280 in.
Inside diameter	4.080 in.
Clearance between guide tube and channel	0.0275 in.
Clearance between guide tube and piston	0.070 in.
Control rod	
Outside diameter of control rod duct	3.777 in.
Inside diameter of control rod duct	3.697 in.
Number of pins	31
Pin pitch-to-diameter ratio	1.05
Pin outside diameter	0.552 in.
Pin cladding thickness	0.029 in.
Total pin length	56.80 in.
Plenum length (upper/lower)	8.091/9.750 in.
Insulator pellet length (upper/lower)	0/0.5 in.
Active absorber length	36.0 in.
Pellet diameter	0.470 in.
Boron-10 loading (minimum/nominal)	4.7/5.0 kg
Wire wrap diameter	0.033 in.
Wire wrap pitch	6 in.
Boron-10 enrichment	92%

Table 4.9 Control assembly--absorber rod structural criteria

Conditions	Criteria
Steady State	P_m or $P_m + P_b < S_p$ S_p = Proportional elastic limit. These limits apply only after initial assent to Full Power. ASME Code Section III Subsection NB and Code Case 1592 limits apply during Preoperational Handling and Checkout.
Steady State and Upset	$F_D \leq D$ F_D = Damage factor determined similar or equivalent to procedure of Paragraph T-1411 of Code Case 1592. K' factors do not apply. Factor F_D to be based on average stress across a section. For D refer to Figure T-1420-2 of Code Case 1592.
Steady State, Upset, and Emergency	The following limits on the combined primary (local) membrane plus bending plus secondary stress intensity range shall be satisfied: $\epsilon_u > 1\%$ The combination of $(P_L + P_b/K_t)/\sigma_y'$ and $(Q_R)_{max}/\sigma_y'$ stress intensities shall be restricted to within 80% of the shakedown boundary based on elastic-perfectly plastic material behavior with the flow stress taken to be σ_y. For axisymmetric structures away from discontinuities the Bree elastic-plastic shakedown diagram can be used. $\epsilon_u < 1\%$ $(P_L + P_b + Q)_{max}$ $< 0.75\ S_u$ for steady state and upset conditions $< 0.90\ S_u$ for emergency conditions Notes: S_y values may be used when σ_y' (cyclic yield strength) is not available K_t is defined in Code Case 1592 $(Q_R)_{max}$ is maximum range of secondary stress intensity
Steady State, Upset, Emergency, and one worst case transient event	$$\sum_{\tau} \left[\frac{\epsilon^C}{(\epsilon_u')_\tau} + \frac{\epsilon^P}{(\epsilon_u')_\tau} \right] < 0.9$$ ϵ^C = thermal creep, ϵ^P - plastic strain, τ - time ϵ_u - uniform elongation, $\bar{\epsilon}$ - average strain rate, TF - triaxiality factor (stress state), and $\epsilon_u' = \epsilon_u$ (TF, $\bar{\epsilon}$) CDF ≤ 1.0 CDF - cumulative damage function similar to that applied to the fuel pin (as described in Part 5 of Section 4.2.1.1.2.2) with applicable material properties.

Table 4.10 Control assembly--balance of assembly structural criteria

ε_u	Maximum Temperature $\leq 800^{\circ}F$	Maximum Temperature $> 800^{\circ}F$
$\varepsilon_u \geq 5\%$	A1 - ASME Code Section III, Subsection NB Rules plus: 1. The effect of irradiation swelling and creep deformations shall be evaluated. 2. Design stress levels shall be based on ASME Code Section III, Subsection NB, unless test data in NSMH indicates use of higher allowables. 3. The mean stress correction for fatigue evaluations is to be based upon the cyclic stress-strain curve. 4. The fatigue evaluation shall be based upon test data for irradiated material or upon the ASME Code curve for unirradiated material, modified by the method of characteristic slopes for predicting fatigue failures based on ductility and fracture data from tests on irradiated material.	A2 - Rules of A1 plus: 1. Code Case 1592. 2. RDT Standard F9-4T. 3. The effect of loss of carbon, nitrogen etc. to be considered when determining allowables. 4. Irradiation swelling and creep deformations to be considered in creep collapse analysis. 5. Inelastic strain accumulation must be computed as a function of temperature, stress state, geometry, and time under consideration. No fluence consideration is required.
$3\% \leq \varepsilon_u < 5\%$	B1 - Rules of A1 plus: When the option to invoke justifiable higher irradiated allowables is utilized:	B2 - Rules of A2 and B1.

Table 4.10 (Continued)

ϵ_u	Maximum Temperature ≤ 800°F	Maximum Temperature >800°F
	B1 - Rules of A1 plus: (Cont'd) 1. No primary plus secondary stress is permitted to exceed that associated with strain of $\xi'_u/2$ 2. Primary membrane stresses shall not exceed the lesser of 1/3 UTS or 2/3 of the proportional stress limit. 3. Linearized primary membrane plus bending stresses shall not exceed the lesser of 1/2 UTS or the proportional limit stress.	
$\epsilon_u > 3\%$	C - Rules of B1 and B2 for their respective temperature ranges plus: 1. Brittle fracture shall be considered as a potential failure mode. 2. Where low ductility components are subject to shock loads, failure due to the superposition of transient stress waves shall be considered.	

Table 4.11 Reactor system design performance parameters

Parameter	Value
Total reactor power, MWt	975
Plant capacity factor (availability factor x load factor) ultimate first cycle, %	75/35
Range of operation, (% rated thermal power) (3-loop operation)	40 to 100
Total coolant flow rate (design range), 10^6 lb/hr	41.5 to 47.7
Maximum nozzle-to-nozzle pressure drop (at 41.5 x 10^6 lb/hr, thermal/hydraulic design value), psi	123
Reactor inlet temperature (thermal/hydraulic design value) °F	730
Reactor bulk coolant temperature difference (thermal/hydraulic design value), °F	265
Reactor mixed mean outlet temperature (thermal/hydraulic design value), °F	995
Average breeding ratio (initial cycle)*	1.29
Total fissile Pu inventory, kg (for beginning of life)	1,502
Fuel burnup, capability objective, MWd/MT initial core, peak	80,000
Maximum neutron flux, total, n/cm²/sec	5.5×10^{15}
Cycle lengths: Initial cycle (full power days)	128
Second cycle (full power days)	200
Later cycles (full power days)	275
Pressure drop, psi	
Across core support structure (design)	170
Through reactor vessel (maximum anticipated)	123

*The breeding ratio is the ratio of the production of fissile plutonium (Pu-239 + Pu-241) to the destruction of fissile material (U-235 + Pu-239 + Pu-241). In this definition, Pu-240, Pu-242, and the mass of U-235 up to the weight fraction found in depleted uranium (0.2%) is not considered as fissile material.

Table 4.12 Reactor description

Parameter	Description
FUEL ASSEMBLIES	
Fuel height, m	0.9144
Geometry	Hexagonal
Number in core beginning of cycle 1 (BOC1)	156
Rod array	Triangular
Rods per assembly, no.	217
Rod pitch (fully compacted), mm	7.264
Overall assembly dimensions:	
Flat-to-flat distance outside edges of hexagonal duct (away from load pads), m	0.1162
Flat-to-flat distance inside edges of hexagonal duct	0.1101
Pitch, m	0.1209
Volume fractions in fuel assembly (drawing dimensions):	
Fuel	0.325
Sodium	0.419
Gap	0.022
Structure	0.234
Total heavy metal (Pu and U) weight in fuel (BOC1), kg	5189
FUEL RODS	
Number in plant (BOC1)	33,852
Cladding outside diameter, mm	5.842
Diametral gap, mm	0.165
Cladding material	20% CW-type 316 SS
Cladding thickness, mm	0.381
Pitch/diameter ratio	1.24
Smear density, % of theoretical	85.5
FUEL PELLETS	
Material	Plutonium oxide and uranium oxide
Density, % of theoretical	91.3
Plutonium weight fraction (first core)	0.328
Plutonium weight fraction (typical equilibrium core)	0.330

Table 4.12 (Continued)

Parameter	Description
FUEL PELLETS (cont.)	
Isotopic composition of feed plutonium (low - 240 grade)	
Pu-238, % by weight	0.06
Pu-239, % by weight	86.04
Pu-240, % by weight	11.70
Pu-241, % by weight	2.00
Pu-242, % by weight	0.20
Isotopic composition of depleted uranium	
U-235, % by weight	0.2
U-238, % by weight	99.8
BLANKET THICKNESS AND COMPOSITION	
Top axial (fuel assemblies), m	Sodium, stainless steel plus depleted uranium oxide (0.356)
Bottom axial (fuel assemblies), m	Sodium, stainless steel plus depleted uranium oxide (0.356)
Radial, m	Sodium, stainless steel plus depleted uranium oxide (~0.28)
TOP AND BOTTOM AXIAL BLANKETS	
Geometry	Same as fuel
Blanket Pellets:	
Material	Depleted uranium oxide
Density, % of theoretical	96.0
Total heavy metal (uranium) weight in blanket (upper and lower), kg	4,225
INNER/RADIAL BLANKET ASSEMBLIES	
Geometry	Hexagonal
Number in plant, inner/radial (BOC1)	82/126
Rods per assembly, no.	61
Rod pitch, mm	13.778
Assembly drawing dimensions:	
Flat-to-flat distance outside edges of hexagonal duct, m	0.1162
Flat-to-flat distance inside edges of hexagonal duct	0.1101
Volume fractions in blanket (drawing dimensions):	
Fuel	0.539
Gap	0.014
Sodium	0.278
Structure	0.169
Total heavy metal (uranium) weight in blanket, inner radial (BOC1), kg	8,270/12,707

Table 4.12 (Continued)

Parameter	Description
INNER/RADIAL BLANKET RODS	
Number in plant, inner radial (BOC1)	5002/7,686
Cladding outside diameter, mm	12.852
Diametral gap, mm	0.152
Cladding material	20% CW-Type 316 SS
Cladding thickness, mm	0.381
Pitch/diameter ratio	1.072
INNER/RADIAL BLANKET PELLETS	
Material	Depleted uranium oxide
Pellet density, % of theoretical	95.6
CONTROL ROD ASSEMBLIES	
Geometry	Hexagonal
Number in plant	15
Primary rods (startup, burnup, and load follow)	9
Secondary rods (safety)	6
Neutron absorber	Enriched boron carbide
Fraction of theoretical density, %	92.0
B-10 enrichment in boron carbide:	
(1) Primary rods (all cycles), atom %	92.0
(2) Secondary rods, atom %	92.0
Rods per assembly:	
Primary system	37
Secondary system	31
Cladding material	20% CW-type 316 SS
Cladding outside diameter:	
Primary system, mm	15.291
Secondary system, mm	14.036
Cladding thickness	
Primary system, mm	1.270
Secondary system, mm	0.699

Table 4.13 Reactivity coefficients and power peaking factors CRBR and FFTF

Coefficients and factors	Parameter	CRBR	FFTF
Doppler constant $(-T \frac{dk}{dT})$			
Initial core, BOC1	Fuel	0.0026	0.0050
	Inner blanket	0.0044	-
	Radial blanket	0.0012	-
	Axial blankets	0.0003	-
Initial core, EOC2	Fuel	0.0026	0.0055
	Inner blanket	0.0049	-
	Radial blanket	0.0012	-
	Axial blankets	0.0003	-
Equilibrium core, BOL	Fuel	0.0024	0.0050
	Inner blanket	0.0041	-
	Radial blanket	0.0015	-
	Axial blankets	0.0003	-
Equilibrium core, EOL	Fuel	0.0024	0.0055
	Inner blanket	0.0046	-
	Radial blanket	0.0013	-
	Axial blankets	0.0004	-
Core-average sodium density coefficient (¢/°F)			
First cycle		-0.006	-0.049
Uniform radial expansion coefficient (¢/°F)			
First cycle		-0.177	-0.21
Uniform axial expansion coefficient (¢/°F)			
First cycle		-0.038	-0.038
Power peaking factors (fuel assemblies not adjacent to inserted control rods)			
First core			
Radial, BOC1		1.18	1.36
Radial, EOC1		1.15	1.28
Axial, BOC1		1.28	1.24
Axial, EOC1		1.26	1.23
Equilibrium core			
Radial, BOL		1.18	1.41
Radial, EOL		1.24	1.32
Axial, BOL		1.27	1.23
Axial, EOL		1.19	1.23

NOTE: BOC = beginning of cycle; EDC = end of cycle; BOL = beginning of life; EOL = end of life.

Table 4.14 Core orificing zone flow allocation

Zone	Type	No. assemblies/ zone	Flow (lb/hr)/per assembly Cycles 1,3,5,...		Cycle 2		Cycles 4,6,8,...	
1	Fuel	39	189,990	(201,900)	188,520	(200,340)	187,050	(198,780)
2	Fuel	54	176,790	(187,870)	175,420	(186,420)	174,060	(184,970)
3	Fuel	21	166,900	(177,360)	165,610	(175,990)	164,320	(174,620)
4	Fuel	18	153,400	(163,020)	152,220	(161,760)	151,030	(160,500)
5	Fuel	24	149,480	(158,850)	148,330	(157,630)	147,170	(156,400)
	Fuel	0,3 or 6			178,590	(189,780)	177,190	(188,300)
6	Inner blanket	6,3 or 0	68,790	(73,100)	69,330	(73,680)		
7	Inner blanket	57	88,790	(94,360)	88,110	(93,630)	87,420	(92,900)
8	Inner blanket	19	78,030	(82,920)	77,420	(82,270)	76,810	(81,620)
9	Radial blanket	12	62,300	(66,210)	61,820	(65,700)	61,340	(65,190)
10	Radial blanket	36	48,300	(51,330)	47,930	(50,930)	47,550	(50,530)
11	Radial blanket	48	35,090	(37,290)	34,820	(37,000)	34,540	(36,710)
12	Radial blanket	30	25,740	(27,350)	25,540	(27,140)	25,330	(26,920)

*Flows are for thermal/hydraulic design value (plant expected operating condition)

Table 4.15 Core region flow fractions

Region	Cycles 1,3,5...	Cycle 2	Cycles 4,6,8...
Fuel	0.65	0.66	0.66
Inner blanket	0.17	0.16	0.16
Radial blanket	0.12	0.12	0.12
Radial shield	0.0134	0.0134	0.0134
Control assemblies	0.0126	0.0126	0.0126
Bypass & leakage	0.0340	0.0340	0.0340
Total	1.0	1.0	1.0

Table 4.16 CRBR fuel, blanket, and control pin cladding design limits

Assembly (lifetime)	Steady-state (SS) operation	Anticipated + unlikely events	Extremely unlikely events
Fuel (80,000 MWd/MT)	<0.2% thermal creep strain with no plastic strain	(a) <0.3% thermal + plastic strain (including SS contribution) (b) CDF <1.0 (including SS contribution) (c) Cladding temperature <1,500°F for anticipated events (d) Cladding temperature <1,600°F for unlikely events	<Na boiling temperature <Cladding melting temperature
Inner blanket (2 cycles)	<0.2% thermal creep strain with no plastic strain	(a) <0.3% thermal + plastic strain (including SS contribution) (b) CDF <1.0 (including SS contribution) (c) Cladding temperature <1,500°F for anticipated events (d) Cladding temperature <1,600°F for unlikely events	<Na boiling temperature < Cladding melting temperature
Outer blanket (4-5 years)	<0.1% thermal creep strain with no plastic strain	(a) <0.2% thermal + plastic strain (including SS contribution) (b) CDF <1.0 (including SS contribution) (c) Cladding temperature <1,500°F for anticipated events (d) Cladding temperature <1,600°F for unlikely events	<Na boiling temperature <Cladding melting temperature
Control (328 full-power days)	Stress less than proportional elastic limit	(a) CDF <1.0 including SS contribution (b) See Table 4.2.37a of PSAR for anticipated events (c) Cladding temperature <1,600°F for unlikely events	<Na boiling temperature <Cladding melting temperature

NOTE: Cladding temperatures of 1,500°F and 1,600°F are guideline values only, which, if exceeded, require more detailed analysis to show strain and CDF limits are met.

Table 4.17 CRBR fuel assemblies rod temperature engineering uncertainty factors

Factor	Coolant	Film	Cladding	Gap	Fuel	Heat Flux
Direct						
Power level measurement and control system dead band	1.03 (1.0)					1.03
Inlet flow maldistribution	1.05					
Flow distribution calculational uncertainty (simulation bias)	1.03	1.022				
Cladding circumferential temperature variation		1.0* 1.7**	1.0**			
Statistical (3σ)						
Reactor ΔT variation	1.0(1.144)					
Wire wrap orientation	1.01					
Subchannel flow area	1.028	1.0				
Film heat transfer coefficient		1.12				
Pellet-cladding eccentricity		1.15	1.15			
Cladding thickness and conductivity			1.12			
Gap conductance				1.48***		
Fuel conductivity					1.10	
Coolant properties	1.01					
Flow distribution calculational uncertainty (calibration)	1.054	1.015				

*For fuel temperature calculations.
**For cladding midwall temperature calculations. Applies to the nominal temperature drop between cladding midwall and bulk coolant.
***Applies to beginning-of-life conditions.

NOTE: Some values of subfactors apply to both plant T&H and expected operating conditions except when two values are given; in this case, the values in parentheses apply to plant expected operating conditions, and the values not in parentheses apply to T&H operating conditions.

Table 4.18 CRBR fuel assemblies rod temperature nuclear uncertainty factors with and without control assembly influence

Factor	Coolant	Heat flux
Direct		
Physics modeling	1.02[1]	*1.02 (1.10)[1]
Control rod banking	1.02[2]	1.02[2]
Zero power plutonium reactor-7 flux tilt flux tilt	1.0[4]	1.0[4]
Statistical (3σ)		
Nuclear data	1.07	1.07
Criticality	1.01[3]	1.01[3]
Fissile fuel maldistribution	1.03	1.03

If assembly is influenced by adjacent control rod, replace with:

Factor		Coolant		Heat flux			
				Peak Power Position		"Top of Core"	
		Beginning of life	End of life	Beginning of life	End of life	Beginning of life	End of life
(1) Physics modeling	Adjacent	1.04	1.02	1.03	1.02	1.15	1.15
	Far side	1.01	1.02	0.95	1.02	1.30	1.15
(2) Control rod banking	Adjacent	1.04	1.02	1.04	1.02	1.01	1.02
	Far side	1.02	1.02	1.02	1.02	1.01	1.02
(3) Criticality	Adjacent	1.04	1.04	1.04	1.04	1.0	1.0
	Far side	1.01	1.01	1.01	1.01	1.03	1.03

(4) Zero power plutonium reactor 7 flux tilt - Assemblies 9, 10, 13, 14, 15, 16, 17, 23, 25, 37, 38, 41, 42, 43, 44, 45, 51, 53 (0.97 at beginning of life (BOL), 1.0 at end of life (EOL). Assemblies 8, 11, 19, 36, 39, 47, 65, 68, 101, 104 (0.99 at BOL, 1.0 at EOL). Assemblies 62, 98 (0.99 at BOL, 1.0 at EOL).

*Value not in parentheses applies at the peak power position (i.e., core midplane). Value in parentheses applies at the core lower/upper axial blanket interface except as superseded by Note (1).

Table 4.19 Three-sigma hot channel coolant temperature at thermal/hydraulic design value conditions

Factor	Hot channel factor	Increase in coolant ΔT (°F)	Total
<u>Direct</u>	<u>Coolant</u>		
Power level measurement and control system dead band	1.03 (1.0)*	4.5	4.5
Inlet flow maldistribution	1.05	7.5	7.5
Flow distribution calculational uncertainty (simulation bias)	1.03	4.5	4.5
Cladding circumferential temperature variation	-		
Physics modeling	1.02	3.0	3.0
Control rod banking	1.02	3.0	3.0
Zero power plutonium reactor flux tilt	1.0	0	
<u>Statistical (3σ)</u>			
Reactor ΔT variation	1.0 (1.144)*	0	
Wire wrap orientation	1.01	1.5	
Subchannel flow area	1.028	4.2	
Film heat transfer coefficent	-		
Pellet-cladding eccentricity	-		
Cladding thickness and conductivity	-		
Gap conductance	-		
Fuel conductivity	-		14.8**
Coolant properties	1.01	1.5	
Flow distribution calculational uncertainty (calibration)	1.054	8.1	
Nuclear data	1.07	10.5	
Criticality	1.01	1.5	
Fissile fuel maldistribution	1.03	4.5	
			37.3

*Values in parentheses are for plant expected operating conditions only.
**Square-root-of-sum-of-squares-of-statistical factors.

Table 4.20 Uncertainties applied in thermal/hydraulic analysis of in-vessel components

Component and parameter	Steady-state analysis (including anticipated transients)	Transient analysis (unlikely + extremely unlikely events)
Removable core component cladding temperatures	PEOC + direct + 2σ statistical hot channel factors	THDV + direct + 3σ statistical hot channel factors
Removable core component structural temperatures	THDV + direct hot channel factors	THDV + direct hot channel factors
Permanent core component structural temperatures	THDV + 20°F	THDV + 20°F
Mixed-mean outlet temperature from removable components	THDV + direct + 3σ statistical hot channel factors	THDV + direct + 3σ statistical hot channel factors

NOTE: PEOC = plant expected operating conditions; THDV = thermal/hydraulic design value.

Table 4.21 Out-of-pile core thermal/hydraulic development testing for fuel assemblies

Test title	Supporting information	Status
19 and 61-rod bundle heat transfer--sodium	Wire wrap bundle temperature distribution over wide operating range, including transients	Completed
217-rod low flow heat transfer--sodium	Low flow bundle temperature distribution	Completed
217-rod bundle mixing	Detailed bundle mixing	Completed
91-rod bundle mixing	Bundle swirl and mixing	Completed
Fuel bundle T&H	Flow split, ΔT, flow distribution, and mixing	In progress at MIT
11:1 scale wire wrap bundle air flow	Detailed axial and cross flow characterization and mixing	Completed
CRBR assembly flow and vibration	Verification of flow and vibration	Completed
Fast flux test facility assembly/bundle flow	Bundle pressure drop	Completed
Inlet/outlet nozzle and orifice flow	Cavitation and ΔP characterization	90% completed
Orifice cavitation proof test	Flow control orifice lifetime/ cavitation	In progress in EBR-II
Assembly outlet nozzle instrumentation	Correlate thermocouple outlet temperature measurements	Testing 90% complete

Table 4.22 Out-of-pile core thermal/hydraulic (T&H) development testing for blanket assemblies

Test title	Supporting information	Status
Full-scale 61-rod assembly heat transfer--sodium	Wire wrap bundle temperature distribution over wide operating range, including transients	95% completed
Blanket bundle T&H	Flow split, ΔP, flow distribution and mixing	In progress at MIT
5:1 scale wire wrap bundle air flow	Detailed axial and cross flow characterization	Completed
Assembly flow and vibration	Verification of flow and vibration characteristics	Completed
Full-scale bundle pressure drop--sodium and water	Bundle ΔP over wide flow range	Completed
Blanket flow orificing characterization	Pressure drop characterization	Planned
Assembly outlet nozzle instrumentation	Correlate thermocouple outlet temperature measurement	90% completed

Table 4.23 Out-of-pile core thermal/hydraulic development testing for primary control assemblies

Test title	Supporting information	Status
Hydraulic prototypic tests in water	ΔPs, bundle/bypass flow split, flotation	Completed
Scram tests in sodium	Scram insertion, effect of scram assisting and retarding forces	Completed
Orifice test in water	Characterization of orifice plates	Completed

Table 4.24 Out-of-pile core thermal/hydraulic development testing for secondary control assemblies

Test title	Supporting information	Status
Orifice tests in water	Characterization of orifice plates	Completed
Latch seal test in water	Pressure drop and velocities across latch seal	Completed
Static flow test in water	Detailed ΔPs in bundle, piston seal ring, flow split (up/down). Steady-state hydraulic scram assist force for parked position and full in	Completed
Bowed duct and guide tube in water	Effect of bowing on scram insertion	Completed
Prototype scram dynamics in sodium	Scram hynamics	50% complete
Extended limit in sodium	Confirmation of proper operation of driveline and scram under failed bellows conditions	Completed

Table 4.25 Reactor thermal/hydraulic (T&H) development testing inlet region

Test title	Supporting information	Status
Inlet plenum feature test--1/4 scale	Characterization of inlet plenum and lower internals T&H performance	Completed
Inlet plenum feature model particle mobility and bubble dispersion tests	Particle transport and bubble breakup characteristics	Completed
Inlet plenum feature test--1/21 scale	Visualization of inlet plenum flow patterns--determination of mixing and transport times	Completed
Piston ring leakage tests	Piston ring leakage rates	Completed
LIM orificing tests	Flow control to blanket assemblies, removable radial shields, and bypass	70% completed
LIM characterization tests	Flow distribution and pressure drop in lower inlet module	Completed
LIM orifice life test	Orifice lifetime characteristics	Completed
RRSA orifice tests in water	Characterization of orifice plates	Completed

Table 4.26 Reactor thermal/hydraulic development testing for outlet region

Test title	Supporting information	Status
Integral reactor flow model, outlet plenum feature flow and vibration test--Phase I testing	Plenum velocity patterns, mixing and ΔP characteristics, vibration, gas entrainment, and striping	Completed
Integral reactor flow model, outlet plenum feature flow and vibration test--Phase II testing	Hydraulic and vibration characteristics of upper internals	Hydraulic completed; vibration in fabrication
BCL outlet plenum stratification test	Flow distribution and temperature response to transient operation	Completed
ANL 1/10 scale outlet plenum tests	Temperature distribution and response at steady-state and transient operation	Completed
ANL 1/15 scale outlet plenum tests	Transient tests in water and sodium	Completed
ANL chimney vibroimpact tests	Full-scale flow-induced vibration of upper internals structure chimney	Completed
Fuel transfer and storage assembly	Heat transfer characteristics of stored fuel assembly	Completed

Table 4.27 Reactor thermal/hydraulic development testing striping tests

Test title	Supporting information	Status
Hanford Engineering Development Laboratory IRFM striping tests	Striping data on: chimney and instrument post, control rod shroud tube, upper internals structure and bypass, removable radial shield, blanket and fuel nozzles, core barrel, former rings, horizontal baffle, liner and suppressor plate, outlet nozzles, and so forth.	Completed
ANL striping tests	Striping data on: mixing tees, seven nozzle assembly with portion of upper internals	In progress
WARD striping tests	• Striping data on: seven assembly outlet nozzle feature test	Completed
	• Seven assembly outlet nozzles test	70% completed
	• Local interstitial flow striping test	Completed
	• Interstitial flow-water table tests	40% completed
	• Thermal striping tests in sodium--dunk and rotating cylinder	Completed