UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)

General Electric Company) Docket No. STN 50-447

Standard Plant)

AMENDMENT NO. 6 TO APPLICATION
FDA REVIEW OF 238 NUCLEAR ISLAND GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT (GESSAR II)

General Electric Company, applicant in the above captioned proceeding, hereby files Amendment No. 6 to the 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II).

Amendment No. 6 consists of two parts, a nonproprietary portion and a portion considered by the General Electric Company to be proprietary. The pages considered to be proprietary are so marked and are transmitted under separate cover.

Amendment No. 6 further amends GESSAR II by incorporating the General Electric Standard Application for Reactor Fuel (GESTAR II) into GESSAR II Sections 4.2, 4.3, 4.4, 5.2.2, 6.3.3, 9.1.2.3, and Chapter 15. The advantages of utilizing GESTAR II in this fashion are: (1) elmination of redundant reviews; (2) better utilization of NRC staff resources; and (3) areas subject to changing technology are maintained current. Amendment No. 6 also amends GESSAR II by clarifying portions of the text where obvious discrepancies exist.

Respectfully submitted,

General Electric Company

by: RVilla

Rudolph Villa, Manager
BWR Standardization

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

On this 31 day of August in the year 1982, before me, Karen S. Vogelhuber, Notary Public, personally appeared Rudolph Villa, personally proved to me on the basis of satisfactory evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

K.V.

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC-CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 21, 1984

Karen S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC-CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 21, 1984

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Respectfully submitted,

General Electric Company

by: s/R. Villa
R. Villa, Manager
BWR Standardization

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

On this 31 day of August in the year 1982, before me, Karen S. Vogelhuber, Notary Public, personally appeared Rudolph Villa, personally proved to me on the basis of satisfactory evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

S/K. S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA
Santa Clara County
My Commission Expires
December 21, 1984
175 Curtner Avenue
San Jose, CA 95125

JNF:sem/B08262
8/26/82

INSTRUCTIONS FOR FILING AMENDMENT NO. 6

Page 11.3-14 is reprinted one-sided and Pages 11.3-23 and 11.3-24 are reprinted back-to-back to correct page assembly mistakes in Amendment 4. Also, Page Nos. 4.4-42 and 4.4-43 should have deleted on Amendment 4. Included is a tab for Appendix 3I.

Remove and insert the pages listed below. Dashes (---) in the remove or insert column indicate no action required.

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SECTION 3.11

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CONTENTS (Continued)

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CONTENTS (Continued)

CONTENTS (Continued)

3.11.2.1.3.1.2 Temperature and Pressure Conditions Inside Containment - Main Steamline Break (MSLB) (Continued)

c.(1) NRC Staff Position 1.2.(3):

In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.

c.(2) GE Position:

See GE Position 3.11.2.1.3.1.1.c(2).

d.(1) NRC Staff Position 1.2.(4):

The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles. 3.39

d.(2) GE Position:

See GE Position 3.11.2.1.3.1.1.d(2).

e.(1) NRC Staff Position 1.2.(5):

Where qualification has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:

- Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in Subsection 1.1.(2).

3.11.2.1.3.1.2 Temperature and Pressure Conditions Inside Containment - Main Steamline Break (MSLB) (Continued)

- Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in Item 2 of Appendix B.

- If the calculated surface temperature exceeds the qualification temperature, the staff requires that (1) requalification testing be performed with appropriate margins, or (2) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For plants that are currently being reviewed or will be submitted for an operating license review within six months from issue date of this report, compliance with items (1) or (2) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated. 3

e.(2) GE Position:

Where qualification has been completed but only LOCA conditions were considered, it shall be demonstrated that the LOCA conditions exceed or are equivalent to the maximum calculated DBE conditions, or that the critical component of the product being qualified will not be exposed to conditions more severe than those for LOCA. The model used to calculate the peak MSLB temperature envelope shall be based on those assumptions defined in GE Position 3.11.2.1.3.1.1.b(2).

3.11.2.1.3.1.3 Effects of Chemical Spray

a.(1) NRC Staff Position 1.3:

The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Subsection 6.5.2 (NUREG-75/087), Paragraph II, Item (e) for caustic spray solution guidelines.

a.(2) GE Position:

The effects of spray, where applicable, shall be addressed for all product qualification. The chemical composition and concentration of the water spray used for qualification shall be equivalent to or more severe than that used in the containment spray system, taking into account the results of a single failure of the spray system. Water spray caused by activation of fire protection system is not addressed as part of a spray environment. 3.39

3.11.2.1.3.1.4 Radiation Conditions Inside and Outside Containment

a.(1) NRF Staff Position 1.4:

The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It

3.11.2.1.3.1.4 Radiation Conditions Inside and Outside Containment (Continued)

should be assumed that the DBA-related environmental conditions occur at the end of the equipment qualified life.

The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualification. Additional radiation margins identified in Subsection 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.

a.(2) GE Position:

The radiation environment for qualification of products important to safety shall be based on the normally expected radiation environment over the product qualified life, plus 3.39
that associated with the most severe DBA during or following which the equipment must remain functional. It shall be assumed that the DBA related environmental conditions occur at the end of the product qualified life.

Appropriate methods similar in nature and scope to that shown in NUREG-0588, Appendix D, for establishing radiation limits for BWR plants for qualification of products will be developed and justified. Additional radiation margins identified in Subsection 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.

b.(1) NRC Staff Position 1.4.(1):

The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the

3.11.2.1.3.1.4 Radiation Conditions Inside and Outside Containment (Continued)

atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr 85, for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable.

b.(2) GE Position:

See Table 3.11-10.

c.(1) NRC Staff Position 1.4.(2):

The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.

3.39

c.(2) GE Position:

The calculation of the radiation environment associated with a DBA shall take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.

d.(1) NRC Staff Position 1.4.(3):

The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.

3.11.2.1.3.1.4 Radiation Conditions Inside and Outside Containment (Continued)

d.(2) GE Position:

See Table 3.11-10.

e. (1) NRC Staff Position 1.4.(4):

Effects of engineered safety features (ESF) systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Subsection 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section.

e.(2) GE Position: 3.39

The effects of the ESF systems, such as containment spray and containment ventilation and filtration systems, which act to remove activity within the containment, shall be calculated on a basis consistent with the assumptions used to calculate the offsite dose. The activity removed from the containment will be added to the filler, sump or location where the activity ends. Appropriate decay will be considered. NUREG-75/087 Subsection 15.6.5 and its appropriate appendices shall be used as guidance.

f. (1) NRC Staff Position 1.4.(5):

Natural deposition (i.e., plateout) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plateout of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.

3.11.2.1.3.2.1 Selection of Methods (Continued)

b.(1) NRC Staff Position 2.1.(2):

The choice of the methods is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless: (1) testing of the component is impractical because of size limitations, and (2) partial type test data are provided to support the analytical assumptions and conclusions reached.

b.(2) GE Position:

See Table 3.11-10. 3.39

c.(1) NRC Staff Position 2.1.(3):

The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.

- Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.

- Any equipment (safety-related or nonsafety-related) that need not function in order to mitigate any accident but that must not fail in a manner detrimental to plant safety should be qualified by test to

3.11.2.1.3.2.1 Selection of Methods (Continued)

demonstrate its capability to withstand any accident environment for the time during which it must not fail.

- Equipment that need not function in order to mitigate any accident and whose failure in any mode, in any accident environment, is not detrimental to plant safety, need be qualified only for its nonaccident service environment. Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident and that its failure in any mode, in any accident environment, is not detrimental to plant safety.

c.(2) GE Position:

The environmental qualification of products exposed to DBE (LOCA or HELB) environments shall conform to the following positions. The responsible engineer establishes the bases associated with the time interval required for operability of this product. The operability and acceptance criteria shall be specified and the safety margins defined.

- Products that must function in order to mitigate an accident shall be qualified to demonstrate operability for the time required in the environmental conditions resulting from that accident. Type testing shall be the preferred method of qualification, subject to the limitations described in paragraph 3.11.2.1.3.2.1.b.(2) above.

3.11.2.1.3.2.1 Selection of Methods (Continued)

- Any product that is not required to function to mitigate an accident but that must not fail in a manner detrimental to plant safety shall be qualified to demonstrate its capability to withstand any accident environment for the time during which it must not fail. Type testing shall be the preferred method of qualification, subject to the limitations described in paragraph 3.11.2.1.3.2.1.b.(2) above.

- Any product that is not required to function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety is considered to be mild environment equipment and is not covered by this document. 3.39

d. (1) NRC Staff Position 2.1.(4):

For environmental qualification of equipment subject to events, other than a DBA, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data, may be found acceptable, subject to the applicability and detail of information provided.

d. (2) GE Position:

For environmental qualification of the product subject to events other than a LOCA or HELB, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, may be used, subject to the applicability and detail of information provided.

3.11.2.1.3.2.2 Qualification by Test

a.(1) NRC Staff Position 2.2.(1):

The failure criteria should be established prior to testing.

3.11.2.1.3.2.2 Qualification by Test (Continued)

a.(2) GE Position:

See Table 3.11-10.

b.(1) NRC Staff Position 2.2.(2):

Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.

b.(2) GE Position:

Test results shall demonstrate that the product can perform its design function for all service conditions postulated (with margin) during its installed life.

c.(1) NRC Staff Position 2.2.(3) 3.39

The items describea in Subsection 6.3 of IEEE Std. 323-1974, supplemented by Items (4) through (12) (NRC Staff Positions 2.2(4) through 2.2(12)) below, constitute acceptable guidelines for establishing test procedures.

c.(2) GE Position:

Section 4 of this document defines the acceptable guidelines for establishing test procedures. These procedures are consistent with Subsection 6.3 of IEEE Std. 323-1974.

d.(1) NRC Staff Position 2.2.(4):

When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelops the

3.11.2.1.3.2.2 Qualification by Test (Continued)

environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelops the conditions produced by the main steamline break and loss-of-coolant accidents).

d.(2) GE Position:

When establishing the simulated environmental profile for qualifying the product located inside containment, a single profile may be used that envelops the environmental conditions resulting from any design basis event (DBE) during any mode of plant operation for which the product is required to function.

e.(1) NRC Staff Position 2.2.(5):

Equipment should be located above flood level or protected against submergence by location in qualified watertight enclosures. Where equipment is located in watertight enclosure, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required. 3.39

e.(2) GE Position:

Products should be located above flood level or protected against submergence by location in watertight enclosures. Where a product is located in a watertight enclosure, qualification by test or analysis shall be used to demonstrate the adequacy of such protection. Products not located in watertight enclosures, that could be subject to submergence through which they must remain functional, shall be demonstrated to be qualified by test for the duration required.

3.11.2.1.3.2.2 Qualification by Test (Continued)

f.(1) NRC Staff Position 2.2.(6):

The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermcouple readings on or as close as practical to the surface of the component being qualified.

f.(2) GE Position:

The temperature to which the product is tested to demonstrate qualification shall be measured and recorded throughout the test. The test environment shall be shown to provide an adequate simulation of the postulated accident environment.

g.(1) NRC Staff Position 2.2.(7):

Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.

g.(2) GE Position:

Performance characteristics of the product shall be verified before, periodically during, and after testing throughout its range of required operability defined in the product performance specification.

h.(1) NRC Staff Position 2.2.(8):

Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.

h.(2) GE Position:

See Table 3.11-10.

3.11.2.1.3.2.4 Other Qualification Methods (Continued)

of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods.

b. GE Position:

Qualification by analysis or operating experience is defined in Subsection 3.11.2.2.4.

3.11.2.1.3.3 Margins 3.39

a.(1) NRC Staff Position 3.0.(1):

Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied to any margins (conservatism) applied during the derivation of the specified plant parameters.

a.(2) GE Position:

Quantified test margins shall be applied to the design environmental parameters, or it shall be shown that adequate margin is already included in the environmental requirements. In either case, the margins shall be justified as adequate and documented.

3.11.2.1.3.3 Margins (Continued)

b.(1) NRC Staff Position 3.(2):

In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Subsection 6.3.1.5, should be used as a guide. (Note exceptions stated in Subsection 1.4.)

b.(2) GE Position:

Qualification test margins shall be determined in accordance with the criteria presented in Subsection 3.11.2.2.4.

c.(1) NRC Staff Position 3.(3):

When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncertainties such as production errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).

c.(2) GE Position:

The qualification environmental profiles identified in Appendix C of NUREG-0588 may be used for in-containment equipment with only margin added for inaccuracies of the test equipment.

d.(1) NRC Staff Position 3.(4):

Some equipment may be required by the design to perform its safety function within only a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not

3.30

3.11.2.1.3.3 Margins (Continued)

be required to perform a safety function but must not fail within a short time period into the event and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least one hour in excess of the time assumed in the accident analysis. For all other equipment (e.g., postaccident monitoring, recombiners, etc.) the 10 percent time margin identified in Subsection 6.3.1.5 of IEEE Std. 323-1974 may be used.

d. (2) GE Position:

Some equipment may be required by the design to perform a safety function within only a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be 3.39
required to perform a safety function but must not fail within a short time period into the event and subsequent failures are also shown not to be detrimental to plant safety. For such equipment, qualification to the safety function time rather than event time is acceptable so long as such equipment is not subsequently applied and considered qualified for longer times unless it can be shown that the equipment's thermal time constant was less than 0.1 x the period tested and the longer period temperature is commensurately lower. For all other equipment, the 10% time margin will be used.

3.11.2.1.3.4 Aging

a. (1) NRC Staff Position 4.(1):

Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.

3.11.2.1.3.4 Aging (Continued)

a.(2) GE Position:

Aging effects on all products important to safety, regardless of their location in the plant, shall be considered and addressed in the qualification program.

b.(1) NRC Staff Position 4.(2):

The degrading influences discussed in Subsections 6.3.3, 6.3.4, and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging programs.

3.11.2.1.3.4 Aging (Continued)

b.(2) GE Position:

The degrading influences discussed in Subsection 3.11.2.2.4 and the electrical and mechanical stresses associated with cyclic operation of the product shall be considered and addressed as part of the aging programs.

c.(1) NRC Staff Position 4.(3):

Synergistic effects should be considered in the accelerated aging programs. An engineering evaluation shall be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), <u>Qualification Testing Evaluation Quarterly Reports</u>, for additional information.

c.(2) GE Position:

See Table 3.11-10.

3.39

3.11.2.1.3.4 Aging (Continued)

i.(2) GE Position:

The qualified life of the product and the basis for its selection shall be defined.

j.(1) NRC Staff Position 4.(10):

Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgment.

j.(2) GE Position:

The qualified life shall be established on the basis of the severity of the testing performed, the conservatisms employed 3.39
in the extrapolation of data, the operating history, or other methods that can be reasonably assumed, coupled with good engineering judgment.

3.11.2.1.3.5 Qualification Documentation

a.(1) NRC Staff Position 5.(1):

The staff endorses the requirements, stated in IEEE Std. 323-1974, that "the qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."

3.11.2.1.3.5 Qualification Documentation (Continued)

a.(2) GE Position:

Qualification documentation shall conform to the commitments made in Subsection 3.11.2.2.4.

b.(1) NRC Staff Position 5.(2):

The guidelines for documentation in IEEE Std. 323-1974, when fully implemented, are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program. 3.39

b.(2) GE Position:

Qualification documentation and files shall conform to the commitments made in Subsection 3.11.2.2.4.

3.11.2.2 Class 1E Product Environmental Qualification Basis

3.11.2.2.1 Scope

This subsection provides a compilation of requirements for the environmental qualification of Class 1E products* within the Nuclear Isand scope of responsibility. It is further restricted to those Class 1E products designed for applications which are exposed to environments resulting from design basis events (except where the only design basis event is a dynamic event).

*"Product" used herein is synonymous with "Class 1E Product".

3.11.2.2.1 Scope (Continued)

The manufacturers and users of Class 1E products are required to provide assurance that each product will meet or exceed its performance requirements throughout its installed life. This is accomplished through a disciplined program of quality assurance that includes but is not limited to design, qualification, production quality control, installation, maintenance, and periodic testing.

It is the primary role of qualification to assure that, for each type of Class 1E product, the design and the manufacturing processes are such that there is a high degree of confidence that the product will perform as required in the specified environment. Other steps in the quality assurance program invoke strict design and manufacturing control to assure that all products of the same type match that which was qualified and are suitabley applied, installed, maintained, and periodically tested.

The methods described in this subsection apply to work accomplished by the General Electric Nuclear Energy Business Operation (NEBO) as well as all its vendors and contractors.

3.11.2.2.2 Applicable Documents

If a conflict exists between the requirements contained in this subsection and those in a listed document, those in this subsection shall govern.

3.11.2.2.2.1 General Electric Documents

The following document forms a part of these requirements.

(1) Design Record Files

3.11.2.2.2.1 General Electric Documents (Continued)

(2) EMI Susceptibility Test Specification

3.11.2.2.2.2 Codes, Standards, and Regulations

The following codes, standards, and regulations, as interpreted by NEBO, form a part of these requirements to the limits specified in this subsection:

(1) NRC Regulatory Guides*

	No.	Title
(a)	1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (generally accepts IEEE 334-1971)
(b)	1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants (generally accepts IEEE 317-1976)
(c)	1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (generally accepts IEEE 382-1972)

*See Section 1.8 for revision and date.

Table 3.11-11

EXAMPLE OF A TEST SEQUENCE FOR TYPE TESTING

Pretest Inspection

Baseline Functional Test under
Normal Conditions

Operational Test under
Extremes of Functional Performance
Characteristics

Aging to End-of-Qualified
Life Conditions

Baseline Functional Test under
Normal Conditions

Operational Test under
Dynamic Conditions

Baseline Functional Test under
Normal Conditions

Operational Test under
Design Basis Event Conditions
Operational Test under Post-
Design Basis Event Conditions

Baseline Functional Test under
Normal Conditions

Post-Test Inspection

Test Report

Table 3.11-12

RECOMMENDED MARGINS

GE PROPRIETARY - provided under separate cover

3B.5.1.5 Inward Load Due to Negative Drywell Pressure (Continued)

corresponds to a change of the horizontal momentum of the water flowing through the vents.

The NRC has requested additional information on the vent backflow loads. The questions and responses are shown in Section 3BO.3 (responses to NRC question 3B.30). 3B.30

This same negative drywell condition can theoretically result in the flow of water over the weir wall into the drywell. The following evaluation of this negative drywell condition is applicable to the 238 Standard Plant but individual Mark III plants may use their plant-unique parameters for evaluation. Using the 238 Standard Plant predicted drywell depressurization time history (Figure 3B-51), a peak velocity of 30 ft/sec is calculated at the top of the weir wall. This velocity is decreased due to the effects of gravity with elevation and the spreading of the flow field so that the maximum elevation reached is 14 feet above the top of the weir wall (Figure 3B-52). Structures in the path of the water are designed for drag loads using the following equation: 3B.30

$$F = \frac{C_D A \rho V^2}{2g_c} \quad (3B.5\text{-}1)$$ 3B.30

where

F = Drag load force (lb/ft)

C_D = Drag coefficient

A = Structure projected area normal to V_n (ft^2)

ρ = Density of water (62.4 lbm/ft^3)

g_c = Newton's constant (32.2 $lbm/ft/lbf/sec^2$)

V = Velocity of fluid (ft/sec) (see Figure 3B-52 for 238 Standard Plant)

For circular cylinders, a conservative drag coefficient, C_D, of 1.2 can be assumed independent of Reynolds number. 3B.30

3B.5.1.5 Inward Load Due to Negative Drywell Pressure (Continued)

Drag forces on structures with sharp corners (e.g., rectangles and I beams) can be computed by considering forces on an equivalent cylinder of diameter, $D_{eq} = \sqrt{2}\, L_{max}$, where L_{max} is the maximum transverse dimension. L_{max} is defined as the diameter of a circumscribed cylinder about the cross section of the structure. For example, L_{max} equals $(a^2 + b^2)^{1/2}$ for a rectangular cross section of sides a and b. A C_D of 1.2 can be used in drag load calculation along with the equivalent diameter D_{eq}.

The impact load of the initial jet front is evaluated using the impact methodology presented below and the jet front velocity (Figure 3B-53) calculated from a transient analysis of a free jet.

1. Determine the impulse per unit of target area from the correlations given below. The impulse is given by an expression of the form:

$$I = \alpha V/32.2$$

where

I = impulse per unit area (lbf-sec/ft^2)
V = Plant-unique transient jet front velocity at structure location (Figure 3B-53 for 238 Standard Plant)

and α depends on the target geometry as follows:

Cylindrical target	$\alpha = 0.81\ D$
Flat target	$\alpha = 2.5\ W$

where

D = target diameter (in.), and
W = target width (in.).

3B.30

3B.5.3 Weir Wall Loads During a Small Break Accident (Continued)

weir wall other than hydrostatic pressure. Apart from that, the information in Subsection 3B.4.3 applies.

3B.5.4 Weir Wall Environment Envelope

The temperature and pressure for the drywell envelope data (Figure 3B-38) apply to the weir wall with the exception of the temperature of that part of the outside face which is below the elevation of the upper vents. This region will remain submerged and will be maintained at suppression pool temperature. The weir wall structure is totally within the drywell and effects of environmental conditions should be examined on this basis including the thermal cycling during chugging (Subsection 3B.4.6).

The first 6 hours of the environmental conditions (Figure 3B-38) are based on a small steam break. Faster shutdown by the operator can reduce the duration of the small break to 3 hours for a large break, the free volume inside the weir wall is flooded and environmental temperature conditions will correspond to the water temperature in this volume. This is less severe than the conditions of Figure 3B-38.

3B.5.5 Weir Annulus Multicell Effects

Chugging is conservatively considered to be synchronous for the Mark III load definitions. The typical pressure time history is shown on Figure 3B-46. Superimposed chugging spikes from adjacent vents, as confirmed by multicell tests[8], are minor and considered to be insignificant. In Figure 3B-54, these spikes are superimposed to indicate the typical magnitude.

3B.6 CONTAINMENT

The containment experiences dynamic loadings during all three classes of LOCA.

3B.6.1 Containment Loads During a Large Steamline Break (DBA)

Figure 3B-55 is a bar chart showing the loading conditions that the containment structure may experience during the DBA LOCA. Design loads for the various structures in the containment annulus are presented in Sections 3B.7 through 3B.12. Figures 3B-2 through 3B-6 show typical structures above the suppression pool.

3B.6.1.1 Compressive Wave Loading

Very rapid compression of the drywell air could, theoretically, result in a compressive wave in the weir annulus water. This wave could then travel down the weir annulus, through the vents, and across the pool to the containment wall. This phenomenon is not specifically included in the containment design conditions on the basis that the approximately 20-psi/sec pressure rate in the drywell is not sufficiently rapid to generate a compressive wave in the water. In addition, even if a 20-psi/sec wave were generated at the weir annulus surface, the very significant attenuation as the wave crosses the 18.5-ft-wide suppression pool would lead to insignificant containment wall loads. This phenomenon has never been observed in any GE pressure suppression test.

3B.6.1.2 Water Jet Loads

Examination of applicable PSTF data (Figure B3-18) indicates some evidence of a loading of the containment wall due to the water jet associated with the vent clearing process (i.e., less than 1 psid) as indicated by the small spike at 0.8 sec. Water jet loads are negligible when compared to the subsequent air bubble pressure

3B.6.1.2 Water Jet Loads (Continued)

discussed in Section 3B.6.1.3 and are not specifically included as a containment design load.

3B.6.1.3 Initial Bubble Pressure

The PSTF air test data for runs 3 and 4[9] have been examined for evidence of bubble pressure loading of the suppression pool wall opposite the vents. These tests were chosen because the drywell pressure at the time of vent clearing is comparable to that expected in a full-scale Mark III (i.e., approximately 20 psid) and because the vent air flow rates and associated pool dynamics would be more representative than the large-scale steam blowdown tests. The maximum bubble pressure load on the containment observed during PSTF testing was 10 psid (Figure 3B-18). Figure 3B-56 is a summary of all the peak containment wall pressure observed in PSTF tests during the bubble formation phase of the blowdown. The Mark III design load which is based on these tests is shown in Figure 3B-11.

The magnitude of the containment pressure increase following vent clearing is dependent upon the rate at which the drywell air bubble accelerates the suppression pool water. Circumferential variations in the air flow rate may occur due to drywell air/steam mixture variations but it results in negligible variations in the containment bubble pressure load (Attachment H).

The conservative asymmetric condition assumes that all air is vented on half of the drywell periphery and steam is vented on the other half.

The large-scale PSTF test data are the basis for specifying the maximum asymmetric load of 10 psi. Figure 3B-56 is a summary of all the peak containment wall pressures observed in PSTF tests during the bubble formation phase of the blowdown. Figure 3B-18 shows a typical transient. A maximum increase of 10 psid on the

3B.6.1.3 Initial Bubble Pressure (Continued)

containment wall was observed in the PSTF at the Mark III drywell peak calculated pressure of 36.5 psia; Figure 3B-56 shows the maximum increase close to zero. Thus, use of a 10-psid asymmetric pressure condition applied in a worst-case distribution as a bounding specification will be used for containment evaluation.

3B.6.1.4 Hydrostatic Pressure

In addition to the hydrostatic load due to the suppression pool water, the data presented in Attachment E is used to determine the hydrostatic pressure loads on the containment during an earthquake. During periods of horizontal accelerations, there will be an asymmetric distribution around the circumference of the containment. The maximum pool level above the pool bottom in the suppression pool is 22 feet and is 26 feet for the drywell and weir annulus.

3B.6.1.5 Local Containment Loads Resulting from the Structures at or Near the Pool Surface

Any structures in the containment annulus that are at or near the suppression pool surface experience upward loads during pool swell. If these structures are attached to the containment wall, then the upward loads are transmitted into the containment wall. Sections 3B.9 and 3B.10 discuss the types of loads that will be transmitted.

Localized loads on the containment wall resulting from the pressure losses associated with water flow past a body are depicted in Figure 3B-57. The data presented in this figure are based on drag type calculations and assumes that the affected structures have design features which preclude impact type loads from occurring.

3B.6.1.6 Containment Load Due to Pool Swell at the HCU Floor (Wetwell Pressurization)

This structure is approximately 20 feet above the pool surface and is 8 feet above the point where breakthrough begins. Froth will reach the HCU floor approximately 1/2 second after top vent clearing and will generate both impingement loads on the structures and a flow pressure differential as it passes through the restricted annulus area at this elevation.

The impingement will result in vertical loads on the containment wall from any structures attached to it and the flow pressure differential will result in an outward pressure loading on the containment wall at this location. The impingement loads will be 15 psi and the froth pressure drop across the HCU floor has been calculated to be 11 psi; the containment wall will see an 11-psi discontinuous pressure loading at this elevation. Figure 3B-58 shows details of the 11-psi pressure loading. The bases for both the impingement and flow pressure loading are discussed in Sections 3B.11 and 3B.12.

When evaluating the containment response to the pressure differential at the HCU floor, any additional loads transmitted to the containment via HCU floor supports (beam seats, etc.) must be assumed to occur simultaneously. These loads are based on the assumption that there is approximately 1500 ft^2 of vent area reasonably distributed around the annulus at this elevation. For plant configurations with HCU flow vent area other than 1500 ft^2 (see Figure 3B-59 for froth pressure drop). The question of circumferential variations in the pressure underneath the HCU floor is addressed in Section 3B.12 and Attachment K.

3B.6.1.7 Fallback Loads

No significant pressure loads are indicated from the data generated by the PSTF during the period when suppression pool water is

3B.6.1.7 Fallback Loads (Continued)

subsiding to its original level following pool swell. Figure 3B-18 shows that during the 2 to 5 seconds suppression pool fallback is occurring, the pool wall pressure probes show no evidence of pressures higher than the initial static pressure.

Structures within the containment annulus below the HCU floor will experience fallback induced drag loads as the water level subsides to its initial level. For design purposes, it is assumed that these structures will experience drag forces associated with water flowing at 35 ft/sec; typical drag coefficients are shown in Figure 3B-19. This is the terminal velocity for a 20-ft free fall and is a conservative bounding number.

3B.6.1.8 Post Pool-Swell Waves

Visual observations of PSTF tests indicate that following pool swell, the surface of the suppression pool is agitated with random wave action having peak to peak amplitudes of less than two feet. These waves do not generate significant containment loading conditions.

3B.6.1.9 Condensation Oscillation Loads

During the condensation phase of the blowdown, there have been some pressure oscillations measured on the containment wall in PSTF tests. Figures 3B-60 and 3B-61 show typical traces of the containment wall pressure fluctuations observed during the condensation phase of the $1/\sqrt{3}$-scale PSTF tests.

The forcing function to be used for design is described in Subsection 3B.4.1.5. The magnitude of the load on the containment wall is shown in Figures 3B-16 and 3B-17.

3B.6.1.10 Chugging

Examination of the PSTF data shows that attenuated vent system pressure fluctuations associated with the chugging phenomenon are transmitted across the suppression pool. Figures 3B-62 and 3B-63 show typical containment wall and basemat pressures from full-scale PSTF tests. Chugging loads on the containment are defined in Subsection 3B.4.1.9.2.

3B.6.1.11 Long-Term Transient

Following the blowdown, the Mark III containment system will experience a long term suppression pool temperature increase as a result of the continuing core decay heat. The operators will activate the RHR system to control the temperature increase but there will be a period of containment pressurization before the transient is terminated. Figure 3B-64 shows the envelope of containment atmospheric pressure and temperature for all postulated breaks. The figure defines only the containment atmospheric condition. Separate analyses are required to evaluate the transient structural response to these conditions. Peak design containment pressure is 15 psig and peak design containment temperature is 185°F.

The model used to simulate the long-term post-LOCA containment heatup transient is described in supplement 1 to Reference 5.

3B.6.1.12 Containment Environmental Envelope

Figure 3B-64 is a diagram showing the maximum design containment pressure and temperature envelope for any size of credible primary system rupture. The long-term containment pressure following a DBA is shown in Figure 3B-65.

3B.6.2 Containment Loads During an Intermediate Break Accident

Figure 3B-66 is the bar chart for the containment during an intermediate break that is of sufficient size to involve the ADS system. Since these breaks are typically quite small and because there is a two-minute timer delay on the ADS system, all the drywell air will have been purged to the containment prior to the time the ADS relief valves open. Thus, the containment will experience the loads from multiple relief valve actuation coupled with the 5-psi pressure increase produced by the drywell air purge and pool heatup. Since the former are pressure oscillations whose magnitude is not dependent upon the datum level, these loads are additive. Attachment A defines the loading magnitudes which are assumed for the SRV discharge.

The seismic-induced increase in suppression pool hydrostatic pressure as a result of horizontal accelerations is asymmetric. This loading sequence is discussed in more detail in Attachment E.

3B.6.3 Containment Loads During a Small Break Accident

No containment loads will be generated by a small break in the drywell that are any more severe than the loads associated with the intermediate or DBA break. Figure 3B-67 is the bar chart for this case.

There are unguarded reactor water cleanup unit (RWCU) lines in the containment that can release steam to the containment free space in the event of a rupture. The RWCU isolation valves and flow limiter for this system are designed to terminate the blowdown before significant containment pressurization can occur. Typically, a 2-psi pressure increase may occur.

Steam released by a pipe break in the containment may stratify and form a pocket of steam in the upper region of the containment. The steam temperature will be at approximately 220°F, whereas the

3B.6.3 Containment Loads During a Small Break Accident (Continued)

air temperature will be at approximately its initial prebreak temperature. This temperature stratification should be accounted for in the design.

Local temperatures of 330/250°F are possible in the event of reactor steam/liquid blowdowns to the containment.

3B.6.4 Safety/Relief Valve Loads

Relief valve operation can be initiated as a result of either a single failure, ADS operation, or a rise in reactor pressure to the valve setpoints. In addition, the containment can be exposed to SRV actuation loads any time the operator elects to open a valve(s) as during an isolated cooldown. The loads generated by SRV actuation are discussed in Attachment A.

3B.6.5 Suppression Pool Thermal Stratification

During the period of steam condensation in the suppression pool, the pool water in the immediate vicinity of the vents is heated. For the Mark III configuration, most of the condensing steam mass and energy are released to the pool through the top vents. By natural convection the hot water rises and the cold water is displaced towards the bottom of the pool. The vertical temperature gradient resulting from this effect is known as thermal stratification and is discussed in Attachment I. The momentary thermal stratification for a large break accident used in containment evaluation is shown in Figure 3B-40.

The NRC has requested additional information on suppression pool thermal stratification. The questions and responses are given in Section 3BO.3 (responses to NRC questions 3B.28 and 3B.29). 3B.28, 3B.29

3B.6.6 Containment Wall Multicell Effects

No multicell effects on the containment wall were observed for pool swell or condensation phenomena during multicell testing.[8,10]

3B.8 LOADS ON STRUCTURES IN THE SUPPRESSION POOL

There are certain structures within the suppression pool which will experience dynamic loads during both LOCA and/or SRV actuation.

3B.8.1 Design Basis Accident

Figure 3B-69 is the bar chart that defines the loads that structures in the suppression pool experience during the LOCA.

3B.8.1.1 Vent Clearing Jet Load

During the initial phase of the DBA, the drywell air space is pressurized and the water in the weir annulus vents is expelled to the pool and induces a flow field in the suppression pool. This induced flow field creates a dynamic load on structures submerged in the pool. However, this dynamic load is less (Attachment L) than the load induced by the LOCA air bubble which forms after the water is expelled. Since the air-bubble dynamic load is bounding, this load is conservatively used in place of the water jet load. The air-bubble load is discussed in Subsection 3B.8.1.2 and Subsection 3BL.2.3 of Attachment L.

3B.8.1.2 Drywell Bubble Pressure and Drag Loads Due to Pool Swell

During the initial phase of the DBA, pressurized drywell air is purged into the suppression pool through the submerged vents. After vent clearing, a single bubble is formed around each top vent. It is during the bubble growth period that unsteady fluid motion is created within the suppression pool. During this period, all submerged structures below the pool surface will be exposed to transient hydrodynamic loads.

The methodology and calculation procedures for determining submerged structures drag loads are discussed in Subsection 3BL.2.3

3B.8.1.2 Drywell Bubble Pressure and Drag Loads Due to Pool Swell (Continued)

of Attachment L. Structures in the suppression pool are designed conservatively for the LOCA drywell bubble pressure (Figure 3B-68) and acceleration drag (Attachment L). This applies to small submerged structures (e.g., pipes).

3B.8.1.3 Fallback Loads

There is no pressure increase in the suppression pool boundary during pool fallback as discussed in Subsection 3B.4.1.6. Structures within the containment suppression pool that are above the bottom vent elevation will experience drag loads as the water level subsides to its initial level. For design purposes, it is assumed that these structures will experience drag forces associated with water flowing at 35 ft/sec; that is the terminal velocity for a 20-ft free fall and is a conservative bounding number. Free fall height is limited by the HCU floor.

3B.8.1.4 Condensation Loads

Steam condensation begins after the vent is cleared of water and the drywell air has been carried over into the wetwell. Condensation oscillation phase is vibratory in nature and induces a bulk water motion and therefore creates drag forces on structures submerged in the pool. This condensation oscillation continues until pressure in the drywell decays.

The methodology and calculation procedures for determining condensation loads on submerged structures are discussed in Subsection 3BL.2.6 of Attachment L.

3B.8.1.5 Chugging

Following the condensation oscillation phase of the blowdown, the vent mass flux falls below a critical value and a random collapse of the steam bubbles occurs. This pressure-suppression phase is called chugging and causes a high-pressure wave (spike) on structures submerged in the pool.

The methodology and calculation procedures for determining chugging loads on submerged structures are discussed in Subsection 3BL.2.8 of Attachment L.

3B.8.1.6 Compressive Wave Loading

As discussed in Subsection 3B.6.1.1, the very rapid compression of the drywell air theoretically generates a compressive wave. But as pointed out in Subsections 3B.6.1.1 and 3B.6.1.2, there were no loads recorded on the containment wall in PSTF for this phenomenon. From this, it can be concluded that compression wave loads or structures in the suppression pool are significantly smaller than loads caused by the water jet for structures close to the drywell. For structures near the containment, neither compressive or jet loads are significant.

3B.8.1.7 Safety/Relief Valve Actuation

Loads on submerged structures due to SRV actuation are discussed in Attachment L.

3B.9 LOADS ON STRUCTURES AT THE POOL SURFACE

Some structures have their lower surfaces either right at the suppression pool surface or slightly submerged. This location means that these structures do not experience the high pool swell impact loads discussed in Section 3B.10. However, they experience pool-swell drag loads and LOCA-induced bubble loads. Relief valve loads must also be considered. These are:

(1) Pool swell drag loads produced by water flowing vertically past the structures at 40 ft/sec (Subsection 3B.8.1.2 and Attachment M).

(2) Pressure loads generated by formation of the vent-exit air bubble immediately following LOCA vent clearing. This type of load will result when the structure is expansive enough to restrict pool swell and cause the bubble pressure to be transmitted through the pool to the under side of the structures. For the GE reference design, the TIP and drywell personnel lock platforms and the sump tanks below are the only structures in this category. All are located on the drywell wall. The maximum upward floor pressure specified for this design is equal to the maximum drywell pressure 21.8 psid (Figure 3B-10). Similar structures located on the containment wall would be designed for a maximum upward floor pressure of 10.0 psid (Figure 3B-68). This is conservative because the bubble pressure can never exceed the drywell pressure and no credit is taken for the attenuation of pressure associated with the head of water above the bubble. These structures should be designed conservatively for the combined loads specified (i.e., drag loads and bubble pressure).

3B.9 LOADS ON STRUCTURES AT THE POOL SURFACE (Continued)

(3) Loads due to the SRV actuation (Attachment A). Only structures with surfaces in the suppression pool will experience the SRV bubble loads.

Pool fallback loads are as discussed in Subsection 3B.4.1.6.

3B.10 LOADS ON STRUCTURES BETWEEN THE POOL SURFACE AND THE HCU FLOORS

Equipment and platforms located in the containment annulus region between the pool surface and the HCU platform experience pool-swell-induced dynamic loads whose magnitude is dependent upon both location and the geometry of the structure. The pool swell phenomenon can be considered as occurring in two phases (i.e., bulk pool swell followed by froth pool swell). The pool-swell dynamic-loading conditions on a particular structure in the containment annulus are dependent upon the type of pool swell that the structure experiences. In addition to location, the size of the structure is also important. Large platforms or floors will completely stop the rising pool and thus incur larger loadings, whereas small pieces of equipment and structural items will only influence the flow of a limited amount of water in the immediate vicinity of the structure. The steam tunnel and HCU floors are the only structures that could be categorized as expansive. Section 3B.11 discusses these structures.

The remainder of this section deals with relatively small structures defined as approximately 20 inches wide. Figure 3B-70 is the loading bar chart for these structures. Structures at this elevation will be subjected to vertical loads only. Horizontal loading mechanisms are not identified and 1/3-scale impact tests verify this conclusion.

The NRC has requested additional information on pool swell and pool swell impact loads. The questions and responses are shown in Section 3BO.3 (responses to NRC questions 3B.1, 3B.2, 3B.3, 3B.4, 3B.5, 3B.6, 3B.9, 3B.32 and 3B.33).

3B.2,3B.3,3B.4,3B.5,
3B.6,3B.9,3B.32,3B.33

3B.10.1 Impact Loads

Figure 3B-72 shows the impact loading profile that is applicable to small structures which are exposed to bulk pool swell. The

3B.10.1 Impact Loads (Continued)

PSTF air test data shows that after the pool has risen approximately 1.6 times vent submergence (i.e., 12 feet), the ligament thickness has decreased to 2 feet or less and the impact loads are then significantly reduced. However, bulk pool-swell impact loading is applied uniformly to any structures within 18 feet of the pool surface (Figure 3B-72). For evaluating the time at which impact occurs at various elevations in the containment annulus, a water surface velocity of 40 ft/sec is assumed. Bulk pool swell would start one second after LOCA.

The basis for the loading specification is the PSTF air test impact data discussed in Reference 9. Specifically, Test Series 5706, run number 4, is used. These tests involved charging the reactor simulator with 1000 psia air and blowing down through a 4.25-inch orifice. Fully instrumented targets located over the pool provided the impact data.

Additional tests have been conducted which provide impact data for typical structures that experience bulk pool swell. Data from these tests (Series 5805) indicate that the specified design load is conservative.

Impact loads are not specified for gratings. The width of the grating surfaces (typically 1/4 inch) do not sustain an impact load. This has been verified in the 1/3-scale PSTF Test Series 5805. Figure 3B-73 is used for calculating grating drag loads.

The NRC has requested additional information on grating drag loads The questions and responses are shown in Section 3BO.3 (responses to NRC question 3B.8). 3B.8

For structures above the 19-foot elevation but below the HCU floors, the froth impingement data portion in Figure 3B-74 is used. Again, this impingement load is applied to all small structures with the time history shown. For structures between 18 and 19 feet above the suppression pool design loads and duration are linearly interpolated from the values shown in Figures 3B-71 and 3B-74.

3B10.1 Impact Loads (Continued)

Figure 3B-75 is a summary of the loading specifications for small structures in the containment annulus as a function of height above the pool.

The influence of seismic-induced submergence variations on the pool-swell transient and resulting impact loads has been considered. It has been concluded that the effect on the magnitude pool-swell impact load is not significant. This conclusion is based on a consideration of the influence of submergence on swell velocity and the significant load attenuation which will result from the pool surface distortions. The very significant margins between the specified loads and the expected loads (Attachment B) provides confidence that any local increase in swell velocities will not result in loads in excess of design values.

The conservatism in these load definitions is illustrated in Attachment B.

3B.10.2 Drag Loads

In addition to the impact loads, structures that experience bulk pool swell are also subject to drag loads as the pool water flows past them with velocities as high as 40 ft/sec. Figures 3B-73, 3B-76, and 3B-19 provide drag load information for geometrical shapes. Data are applied to all small structures in the containment annulus between the pool surface and the HCU floors.

3B.10.3 Fallback Loads

Fallback loads are discussed in Subsections 3B.4.1.6, 3B.6.1.7, and 3B.8.1.3.

3B.11 LOADS ON EXPANSIVE STRUCTURES AT THE HCU FLOOR ELEVATION

At the HCU floor elevation there are portions of the floor which are comprised of beams and grating and other portions that are solid expansive structures. The bottom of the steam tunnel is at approximately the same elevation (19 ft, 6 in.). The small structure portion (beams and grating) of the HCU floor is discussed in Section 3B.12.

The expansive structures at this elevation experience an impulsive loading followed by an 11-psi pressure differential. The impulsive load is due to the momentum of the froth which is decelerated by the expansive structure. The 11-psi pressure differential is based on an analysis of the transient pressure in the space between the pool surface and the HCU floor resulting from the froth flow through the 1500 ft^2 vent area at this elevation (Subsection 3B.6.1.6). Figure 3B-76 shows the loading sequences and Figure 3B-74 shows the loading history.

The NRC has requested additional information on wetwell pressurization. The questions and responses are shown in Section 3BO.3 (responses to NRC questions 3B.7 and 3B.35). 3B.7,3B.35

PSTF Test Series 5706 is the basis for the froth impingement load of 15 psi lasting for 100 msec[3]. Representative tests of the expected Mark III froth conditions at the HCU floor are the 5-foot submergence tests of Series 5801, 5802, 5803, and 5804. These tests confirmed the adequacy of the 15-psi impingement load.

The NRC has requested additional information on froth impingement. The questions and responses are shown in Section 3BO.3 (response to NRC question 3B.34). 3B.34

The 11-psi froth flow pressure differential lasting for 3 sec is based on an analysis of the transient pressure in the space

3B.11 LOADS ON EXPANSIVE STRUCTURES AT THE HCU FLOOR ELEVATION (Continued)

between the pool surface and the HCU floor. The value of 11 psi is from an analysis that assumes that the density of the flow through the annulus restriction is the homogeneous mixture of the top 9 ft of the suppression pool (i.e., 18.8 lbm/ft^3). Supplement 1 to Reference 5 descrives the analytical model used to simulate the HCU floor flow pressure differential and presents a comparison of model predictions with test data. This is a conservative density assumption confirmed to the PSTF 1/3-scale tests which show average densities of approximately 10 lbm/ft^3. Reference 11 indicates the HCU floor pressure differential is in the 3-to-5-psi range.

The potential for circumferential variations in the pressure transient in the wetwell region beneath the HCU floor has been examined and on the basis of bounding calculations it is concluded that the pressure variation will be less than 0.5 psid (Attachment K).

3B.12 LOADS ON SMALL STRUCTURES AT AND ABOVE THE HCU FLOOR ELEVATION

Structures at the HCU floor elevation experience froth pool swell which involves both impingement and drag type forces. Figure 3B-12 shows the loading sequences. Only structures in the line of sight of the pool will experience froth pool-swell loads.

PSTF air tests show that the structures experience a froth impingement load of 15 psi lasting for 100 milliseconds[3]. The impingement data is shown in Figure 3B-74. Structures are designed for this short-term dynamic impingement load; grating structures are not subjected to this impingement load[12].

The NRC has requested additional information on froth impingement. The questions and responses are shown in Section 3BO.3 (response to NRC question 3B.34).

Following the initial froth impingement (Subsection 3B.6.1.6) there is a period of froth flow through the annulus restriction at this elevation.

The froth-flow pressure-differential load (i.e., drag-type force) specification of Figure 3B-74 is based on an analysis of the transient pressure in the space between the pool surface and the HCU floor. The value of 11 psi is from an analysis that assumes that the density of the flow through the annulus restriction is the homogeneous mixture of the top 9 feet of the suppression pool water and the free air between the HCU floor and the pool (i.e., 18.8 lb_m/ft^3). This is a conservative density assumption confirmed by the PSTF 1/3-scale tests which show an average density of approximately 10 lb_m/ft^3. Representative tests of the expected Mark III froth conditions at the HCU floor at the 5-foot submergence tests of Series 5801, 5802, 5803, and 5804. Reference 11 indicates the HCU floor pressure differential during these tests was in the 3-to-5-psi range (drag load on HCU floor).

3B.12 LOADS ON SMALL STRUCTURES AT AND ABOVE THE HCU FLOOR ELEVATION (Continued)

Those small structures above the HCU floor that could be exposed to pool-swell froth may be exposed to a drag load. The drag load is determined for the geometric shape of the structure (Figure 3B-19) using a froth density of 18.8 lb_m/ft^3 as in the HCU floor ΔP calculation and the velocity of the froth at the elevation of the structure. The velocity used is 50 ft/sec at 19-1/2 feet above the suppression pool and is decelerated by the effects of gravity. The velocity of 50 ft/sec is a bound of the available data[13]. No pool swell is assumed for structures more than 30 feet above the suppression pool.

The potential for circumferential variations in the pressure transient in the wetwell region beneath the HCU floor have been examined and on the basis of bounding calculations it is concluded that the pressure variation will be less than 0.5 psid (Attachment K).

Since the air tests were performed, additional PSTF tests have been conducted with the specific objective of providing further data on the interaction of pool swell with the HCU floors. The test results are in Reference 11. Supplement 1 to Reference 5 describes the analytical model used to simulate the HCU floor flow pressure differential and presents a comparison of model predictions with test data. The model is shown to be conservative.

SECTION 4.1

CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

4. REACTOR

4.1 SUMMARY DESCRIPTION

The reactor assembly consists of the reactor vessel, its internal components of the core (shroud, steam separator and dryer assemblies) and jet pumps. Also included in the reactor assembly are the control rods, control rod drive (CRD) housings and the control rod drives. Figure 3.9-8 (Reactor Vessel Cutaway) shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in Subsection 1.3.1.1, "Nuclear Steam Supply System Design Characteristics". Loading conditions for reactor assembly components are specified in Subsection 3.9.5.2.

4.1.1 Reactor Vessel

The reactor vessel design and description are covered in Section 5.3.

4.1.2 Reactor Internal Components

The major reactor internal components are the core (fuel, channels, control blades and instrumentation), the core support structure (including the shroud, top guide and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray spargers and the jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, in-core instrumentation, shroud head and steam separator assembly, and steam dryers are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

Important features of the reactor core are:

(1) The bottom-entry cruciform control rods. Rods of this design were first introduced in the Dresden-1 reactor in April 1961 and have accumulated thousands of hours of service.

(2) Fixed in-core fission chambers provide continuous power range neutron flux monitoring. A guide tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range detectors are located in-core and are axially retractable. The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is discussed in Subsection 7.6.2.1.

(3) As shown by experience obtained at Dresden-1 and other plants, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

(4) The reusable channels provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods and protect the fuel during handling operations.

4.1.2.1 Reactor Core (Continued)

(5) The mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. The core is designed to be subcritical at any time in its operating history with any one control rod fully withdrawn.

(6) The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows adequate clearance below the pressure vessel between CRD mechanisms for ease of maintenance and removal.

(7) The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel.

4.1.2.1.1 Fuel Assembly Description

The fuel assembly description is provided in Section 4.2.

4.1.2.1.2 Assembly Support and Control Rod Location

A few peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral support and guidance at the top of each control rod guide tube.

The top guide, mounted on top of the shroud, provides lateral support and guidance for the top of each fuel assembly. The reactivity

4.1.2.1.2 Assembly Support and Control Rod Location (Continued)

of the core is controlled by cruciform control rods and their associated mechanical hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent drive enters the core from the bottom, and can accurately position its associated control rod during normal operation and yet exert approximately ten times the force of gravity to insert the control rod during the scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling and convenient drive maintenance.

4.1.2.2 Shroud

The information on the shroud is contained in Subsection 3.9.5.1.1.1.

4.1.2.3 Shroud Head and Steam Separators

The information on the shroud head and steam separators is contained in Subsection 3.9.5.1.1.3.

4.1.2.4 Steam Dryer Assembly

The information on the steam dryer assembly is contained in Subsection 3.9.5.1.2.2.

4.1.3 Reactivity Control Systems

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned to

4.1.3.1 Operation (Continued)

counterbalance steam voids in the top of the core and effect significant power flattening.

These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

A description of the control rods is given in Subsection 4.2.2.4.1.

4.1.3.3 Supplementary Reactivity Control

The initial and reload core control requirements are met by use of the combined effects of the movable control rods, supplementary burnable poison, and variation of reactor coolant flow. The supplementary burnable poison description is provided in Section 4.2.

4.1.4 Analysis Techniques

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components are listed as follows:

(1) MASS
(2) SNAP (MULTISHELL)
(3) GASP
(4) NOHEAT
(5) FINITE
(6) DYSEA
(7) SHELL 5
(8) HEATER
(9) FAP-71
(10) CREEP-PLAST
(11) ANSYS
(12) CLAPS
(13) ASIST

Detail description of these programs are given in the following sections.

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description

The program, proprietary of the General Electric Company, is an outgrowth of the PAPA (Plate and Panel Analysis) program originally developed by L. Beitch in the early 1960s. The program is based on the principle of the finite element method. Governing matrix equations are formed in terms of joint displacements using a "stiffness-influence-coefficient" concept originally proposed by L. Beitch (Reference 2). The program offers curved beam, plate

4.1.4.1.6.4 Extent of Application

The current version of DYSEA has been used in all dynamic and seismic analysis since its development. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

4.1.4.1.7 SHELL 5

4.1.4.1.7.1 Program Description

SHELL 5 is a finite shell element program used to analyze smoothly curved thin shell structures with any distribution of elastic material properties, boundary constraints, and mechanical thermal and displacement loading conditions. The basic element is triangular whose membrane displacement fields are linear polynomial functions, and whose bending displacement field is a cubic polynomial function (Reference 6). Five degrees of freedom (three displacements and two bending rotations) are obtained at each nodal point. Output displacements and stresses are in a local (tangent) surface coordinate system.

Due to the approximation of element membr ane displacements by linear functions, the in-plane rotation about the surface normal is neglected. Therefore, the only rotations considered are due to bending of the shell cross-section, and application of the method is not recommended for shell intersection (or discontinuous surface) problems where in-plane rotation can be significant.

4.1.4.1.7.2 Program Version and Computer

A copy of the source deck of SHELL 5 is maintained in GE/NEBG. SHELL 5 operates on the internal computers.

4.1.4.1.7.3 History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated (Reference 7) in 1969. The program has been in production status at Gulf General Atomic, General Electric, and at other major computer operating systems since 1970.

4.1.4.1.7.4 Extent of Application

SHELL 5 has been used at General Electric to analyze reactor shroud support and torus. Satisfactory results were obtained.

4.1.4.1.8 HEATER

4.1.4.1.8.1 Program Description

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full-scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the nuclear steam supply system (NSSS) are modeled in detail (Reference 8).

4.1.4.1.8.2 Program Version and Computer

This program was developed at GE/NEBG in FORTRAN IV for the Honeywell 6000 computer.

4.1.4.1.8.3 History of Use

The program was developed by various individuals in GE/NEBG beginning in 1970. The present version of the program has been in operation since January 1972.

4.1.4.1.8.4 Extent of Application

The program is used in the hydraulic design of the feedwater spargers for each BWR plant, in the evaluation of design modifications, and the evaluation of unusual operational conditions.

4.1.4.1.9 FAP-71 (Fatigue Analysis Program)

4.1.4.1.9.1 Program Description

The FAP-71 computer code, or Fatigue Analysis Program, is a stress analysis tool used to aid in performing ASME-III Nuclear Vessel Code structural design calculations. Specifically, FAP-71 is used in determining the primary plus secondary stress range and number of allowable fatigue cycles at points of interest. For structural locations at which the $3S_m$ (P+Q) ASME Code limit is exceeded, the program can perform either (or both) of two elastic-plastic fatigue life evaluations: (1) the method reported in ASME Paper 68-PVP-3, or (2) the present method documented in Paragraph NB-3228.3 of the 1971 Edition of the ASME Section III Nuclear Vessel Code. The program can accommodate up to 25 transient stress states of as many as 20 structural locations.

4.1.4.1.9.2 Program Version and Computer

The present version of FAP-71 was completed by L. Young of GE/NEBG in 1971 (Reference 9). The program currently is on the NEBG Honeywell 6000 computer.

4.1.4.1.9.3 History of Use

Since its completion in 1971, the program has been applied to several design analyses of GE BWR vessels.

4.1.4.1.9.4 Extent of Use

The program is used in conjunction with several shell analysis programs in determining the fatigue life of BWR mechanical components subject to thermal transients.

4.1.4.1.10 CREEP/PLAST

4.1.4.1.10.1 Program Description

A finite element program is used for the analysis of two-dimensional (plane and axisymmetric) problems under conditions of creep and plasticity. The creep formulation is based on the memory theory of creep in which the constitutive relations are cast in the form of hereditary integrals. The material creep properties are built into the program. Any other creep properties can be included if required.

The plasticity treatment is based on kinematic hardening and von Mises yield criterion. The hardening modulus can be constant or a function of strain.

4.1.4.1.10.2 Program Version

The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in Reference 11.

4.1.4.1.10.3 History of Use

This program was developed in Reference 11. It underwent extensive program testing before it was put on production status.

4.1.4.1.10.4 Extent of Application

The program is used at GE/NEBG in the channel cross-section mechanical analysis.

4.1.4.1.11 ANSYS

4.1.4.1.11.1 Program Description

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

(1) Structural analysis, including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

(2) One-dimensional fluid flow analysis.

(3) Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

(4) An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

(5) Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures.

4.1.4.1.11.1 Program Description (Continued)

(6) Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

4.1.4.1.11.2 Program Version

The program is maintained current by Swanson Analysis Systems, Inc. of Pittsburgh, Pennsylvania and is supplied to General Electric.

4.1.4.1.11.3 History of Use

The ANSYS program has been used for productive analysis since early 1970. Users now include the nuclear, pressure vessels and piping, mining, structures, bridge, chemical, and automotive industries, as well as many consulting firms.

4.1.4.1.11.4 Extent of Application

ANSYS is used extensively in GE/NEBG for elastic and elastic-plastic analysis of the reactor pressure vessel, core support structures, reactor internals and fuel.

4.1.4.1.12 CLAPS

4.1.4.1.12.1 Program Description

CLAPS is a general-purpose, two-dimensional finite element program used to perform linear and nonlinear structural mechanics analysis. The program solves plane stress, plane strain and axi-symmetric problems. It may be used to analyze for instantaneous

4.1.4.1.12.1 Program Description (Continued)

pressure, temperature and flux changes, rapid transients and steady-state, as well as conventional elastic and inelastic buckling analyses of structural components subjected to mechanical loading.

4.1.4.1.12.2 Program Version

The current CLAPS program is documented in Section 2 of Reference 12.

4.1.4.1.12.3 History of Use

The CLAPS model was approved by the NRC April 1975.

4.1.4.1.12.4 Extent of Application

CLAPS is used for stress analysis of fuel assembly components.

4.1.4.1.13 ASIST

4.1.4.1.13.1 Program Description

The ASIST program is a General Electric code which can be used to obtain load distribution, deflections, critical frequencies and mode shapes in the "in-plane" or "normal-to-plane" modes for planar structures of any orientation that: (1) are statically indeterminate; (2) can be represented by straight or curved beams; and (3) are under basically any loading, thermal gradient, or sinusoidal excitation. Deformations and resulting load distributions are compared considering all strain energies (i.e., bending, torsion, shear and direct). ASIST also considers the effects of the deflected shape on loads and provides deflections calculated

4.1.4.1.13.1 Program Description (Continued)

for the structure. In addition to this beam column (large deflection) capability, the buckling instability of planar structures can also be calculated.

4.1.4.1.13.2 Program Version

The current program version is documented in Reference 12.

4.1.4.1.13.3 History of Use

The initial version of the ASIST program was developed by the General Electric Jet Propulsion Division. The program and its predecessors have been in use in the General Electric Aircraft Engine group for more than 10 years. Its application in GE/NEBG has a history longer than 6 years.

4.1.4.1.13.4 Extent of Application

The ASIST program has been used to determine spring constants, stresses, deflections, critical frequencies and associated modes shapes for frames, shafts, rotors, and other jet engine components. It has been used extensively as a design and analysis tool for various components of nuclear fuel assemblies.

4.1.4.2 Fuel Rod Thermal Analysis

The fuel rod thermal analyses models are documented in Section 2 of Reference 12.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in Section 4 of Reference 10. Subsection 4.4.4.6 also provides a complete stability analysis for the reactor coolant system.

4.1.4.4 Nuclear Analysis

The analysis techniques are described and referenced in Section 3 of Reference 12.

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were carried out using a one-dimensional, discrete ordinates, Sn transport code with general anisotropic scattering.

This code is a modification of a widely used discrete ordinates code which will solve a wide variety of radiation transport problems. The program will solve both fixed source and multiplication problems. Slab, cylinder, and spherical geometry are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, P sub (L) matrices for anistropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core mid-plane were calculated using a two-dimensional, discrete ordinate code. The two-dimensional code is an extension of the one-dimensional code.

4.1.4.6 Thermal-Hydraulic Calculations

Description of the thermal-hydraulic models are provided in Section 4 of Reference 12.

4.1.5 References

1. (Deleted)

2. L. Beitch, "Shell Structures Solved Numerically by Using a Network of Partial Panels," AIAA Journal, Volume 5, No. 3, March 1967.

3. E. L. Wilson, "A Digital Computer Program For the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Technical Memo No. 23, Aerojet General, July 1965.

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5. J. E. McConnelee, "Finite-Users Manual", General Electric TIS Report DF 69SL206, March 1969.

6. R. W. Clough and C. P. Johnson, "A Finite Element Approximation for the Analysis of Thin Shells," International Journal Solid Structures, Vol. 4, 1968.

7. "A Computer Program for the Structural Analysis of Arbitrary Three-Dimensional Thin Shells," Report No. GA-9952, Gulf General Atomic.

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9. Young, L. J., "FAP-71 (Fatigue Analysis Program) Computer Code," GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.

10. L. A. Carmichael and G. J. Scatena, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," January 1977 (NEDO-21506).

4.1.5 References (Continued)

11. "BWR Fuel Channel Mechanical Design and Deflection," September 1976 (NEDE-21354-P (Proprietary) and NEDO-21354 (Nonproprietary).

12. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

SECTION 4.2

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SECTION 4.2

ILLUSTRATIONS

4.2 FUEL SYSTEM DESIGN

See Appendix A, Section A.4.2 of Reference 1.

4.2.1 Design Bases

See Appendix A, Subsection A.4.2.1 of Reference 1.

4.2.2 Description and Design Drawings

See Appendix A, Subsection A.4.2.2 of Reference 1.

4.2.2.1 Control Rods

The control rods perform the dual function of power shaping and reactivity control. A design drawing of the control blade is seen in Figure 4.2-1 and 2. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod displacement tends to counterbalance steam void effects at the top of the core and results in significant power flattening.

The control rod consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder. The control rods are 9.868 in. in total span and are separated uniformly throughout the core on a 12-in. pitch. Each control rod is surrounded by four fuel assemblies.

The main structural member of a control rod is made of Type-304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting, and center post are welded into a single skeletal structure.

4.2.2.1 Control Rods (Continued)

The U-shaped sheaths are resistance welded to the center post, handle and castings to form a rigid housing to contain the boron-carbide-filled absorber rods. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. The control rods are cooled by the core bypass flow. The U-shaped sheaths are perforated to allow the coolant to circulate freely about the absorber tubes. Operating experience has shown that control rods constructed as described above are not susceptible to dimensional distortions.

The boron-carbide (B_4C) powder in the absorber tubes is compacted to about 70% of its theoretical density. The boron-carbide contains a minimum of 76.5% by weight natural boron. The boron-10 (B-10) minimum content of the boron is 18% by weight. Absorber tubes are made of Type-304 stainless steel. Each absorber tube is 0.220 in. in outside diameter and has a 0.027 in. wall thickness. Absorber tubes are sealed by a plug welded into each end. The boron-carbide is longitudinally separated into individual compartments by stainless steel balls at approximately 17-in. intervals. The steel balls are held in place by a slight crimp of the tube. Should boron-carbide tend to compact further in service, the steel balls will distribute the resulting voids over the length of the absorber tube.

4.2.2.2 Velocity Limiter

The control rod velocity limiter (Figure 4.2-3) is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against high reactivity insertion rate by limiting the control rod velocity in the event of a control rod drop accident. It is a one-way device in that the control rod scram velocity is not significantly affected, but the control rod dropout velocity is reduced to a permissible limit.

4.2.2.2 Velocity Limiter (Continued)

The velocity limiter is in the form of two nearly mated, conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart and is at a 15-degree angle relative to the upper conical element, with the peripheral separation less than the central separation.

The hydraulic drag forces on a control rod are proportional to approximately the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke, the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout, but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction.

Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 3.11 ft/sec.

4.2.3 Design Evaluation

See Appendix A, Subsection A.4.2.3 of Reference 1.

4.2.4 Testing, Inspection and Surveillance Plans

See Appendix A, Subsection A.4.2.4 of Reference 1.

4.2.5 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

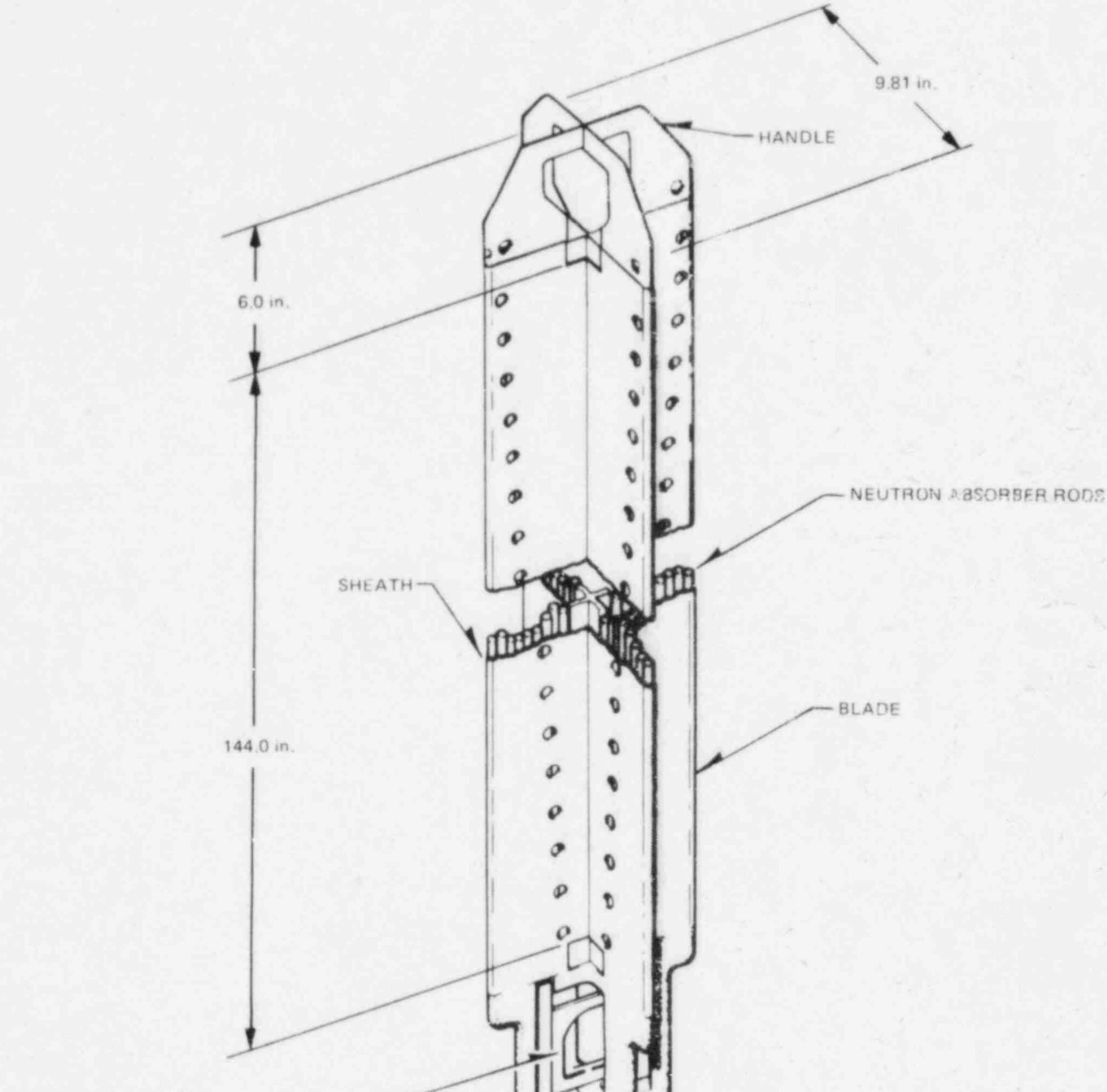

Figure 4.2-1. Control Rod Assembly

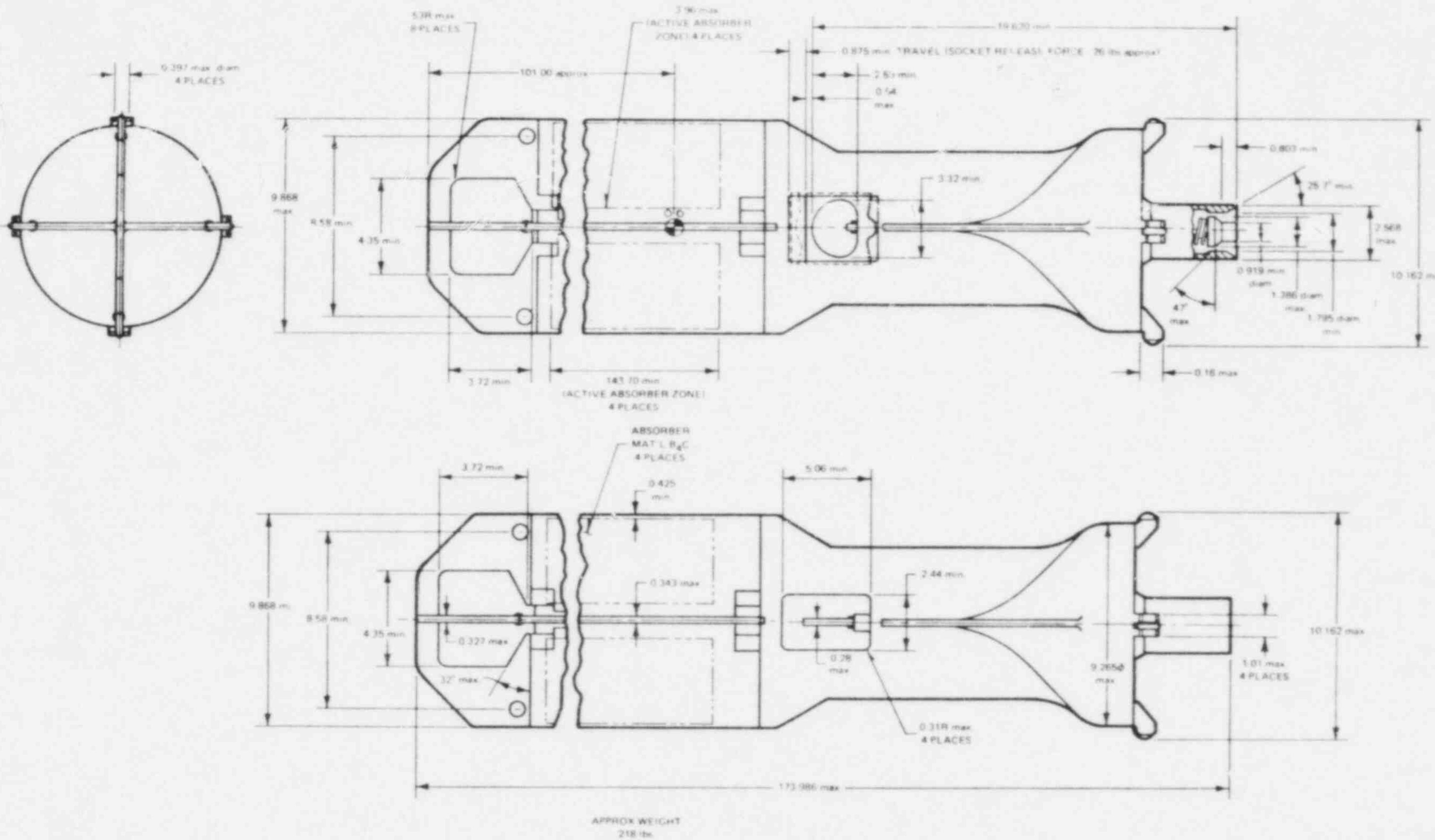

Figure 4.2-2. Control Rod Information Diagram

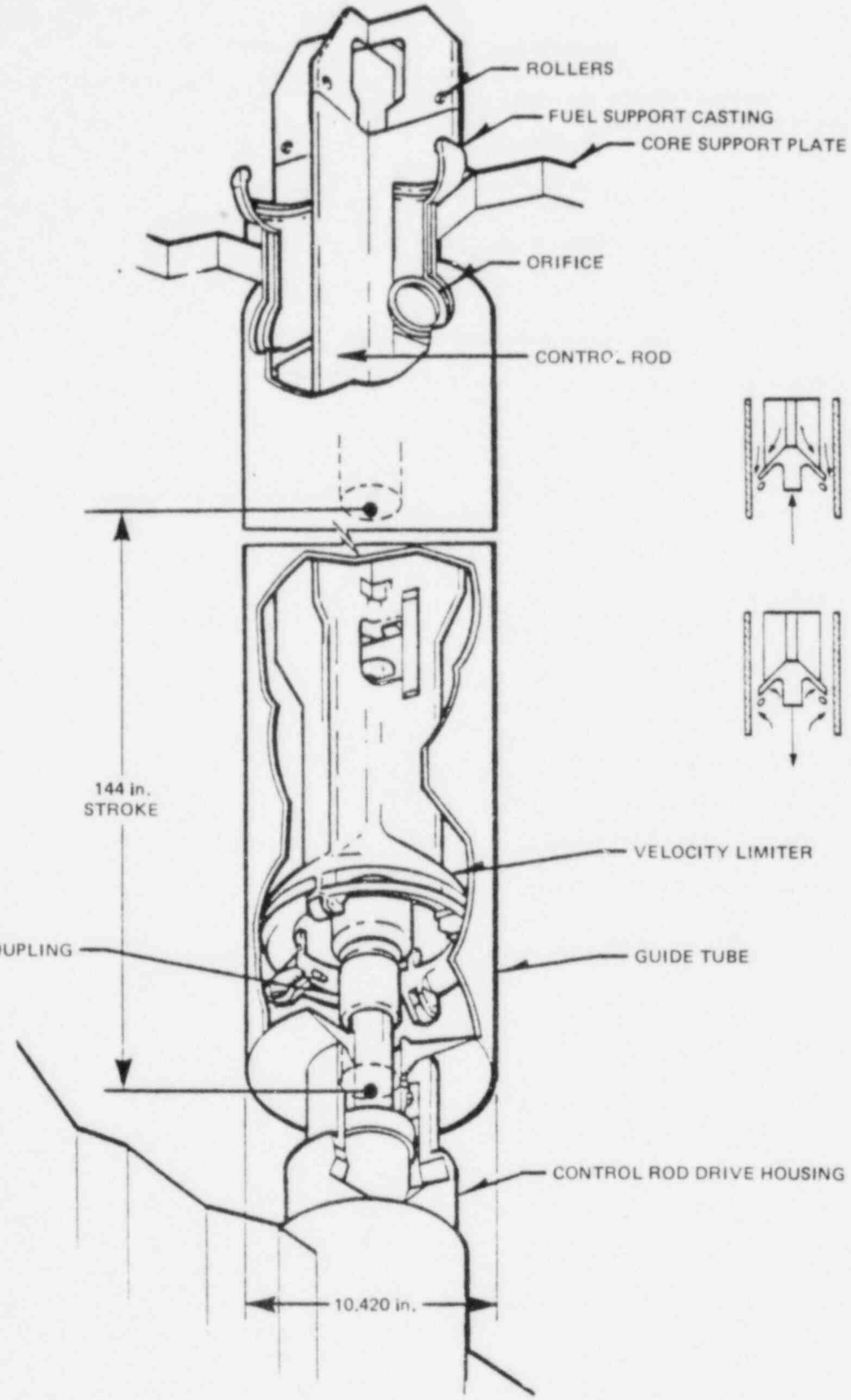

Figure 4.2-3. Control Rod Velocity Limiter

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ILLUSTRATIONS

4.3 NUCLEAR DESIGN

See Appendix A, Section A.4.3 of Reference 1.

4.3.1 Design Bases

See Appendix A, Subsection A.4.3.1 of Reference 1.

4.3.2 Description

See Appendix A, Subsection A.4.3.2 of Reference 1.

4.3.2.1 Nuclear Design Description

See Appendix A, Subsection A.4.3.2.1 of Reference 1. The reference core loading pattern for the initial core is to be provided by the applicant as shown in Table 4.3-1.

4.3.2.2 Power Distribution

See Appendix A, Subsection A.4.3.2.2 of Reference 1.

4.3.2.2.1 Power Distribution Calculations

See Appendix A, Subsection A.4.3.2.2.1 of Reference 1.

A full range of calculated power distributions along with the resultant exposure shapes and the corresponding control rod patterns are shown in Appendix 4A for a typical BWR/6.

4.3.2.2.2 Power Distribution Measurements

See Appendix A, Subsection A.4.3.2.2.2 of Reference 1.

4.3.2.2.3 Power Distribution Accuracy

See Appendix A, Subsection A.4.3.2.2.3 of Reference 1.

4.3.2.2.4 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct rearrangement of the core following refueling. Although a misplacement of a bundle in the core would be a very improbable event, calculations have been performed in order to determine the effects of such accidents on linear heat generation rate (LHGR) and critical power ratio (CPR). These results are presented in Chapter 15.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces perturbations in the power distribution. In addition, the in-core instrumentation system, together with the on-line computer, provides the operator with prompt information on power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as prescribed in Chapter 16 (Technical Specifications).

4.3.2.3 Reactivity Coefficients

See Appendix A, Subsection A.4.3.2.3 of Reference 1.

4.3.2.4 Control Requirements

See Appendix A, Subsection A.4.3.2.4 of Reference 1.

4.3.2.4.1 Shutdown Reactivity

To assure that the safety design basis for shutdown is satisfied, an additional design margin is adopted: k-effective is calculated to be less than or equal to 0.99 with the control rod highest worth fully withdrawn.

The cold shutdown margin for the reference core loading pattern is given in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Enrichment distributions for these rods are given in Section 2 of Reference 1.

Control rods are used during the cycle partly to compensate for burnup and partly to flatten the power distribution.

Reactivity balances are not used in describing BWR behavior because of the strong interdependence of the individual constituents of reactivity. Therefore, the design process does not produce components of a reactivity balance at the conditions of interest. Instead, it gives the k_{eff} (Table 4.3-2) representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

See Appendix A, Subsection A.4.3.2.5 of Reference 1.

4.3.2.6 Criticality of Reactor During Refueling

See Appendix A, Subsection A.4.3.2.5 of Reference 1.

4.3.2.7 Stability

See Appendix A, Subsection A.4.3.2.6 of Reference 1.

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel have been calculated using the one-dimensional discrete ordinates transport code described in Subsection 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region. The coolant water region between the fuel channel and the shroud was described containing saturated water at 550°F and 1050 psi. The material compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA 240, 304L, and ASME SA 533 grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A simple diagram showing the regions, dimensions, and weight fractions are shown in Figure 4.3-1.

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1 centimeter thick disc with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction.

4.3.2.8 Vessel Irradiations (Continued)

The neutron fluence is determined from the calculated flux by assuming that the plant is operated 90 percent of the time at 90 percent power level for 40 years or equivalent to 1 x 10^9 full power seconds. The calculated fluxes and fluence are shown in Table 4.3-3. The calculated neutron flux leaving the cylindrical core is shown in Table 4.3-4.

4.3.3 Analytical Methods

See Appendix A, Subsection A.4.3.3 of Reference 1.

4.3.4 Changes

See Appendix A, Subsection A.4.3.4 of Reference 1.

4.3.5 References

1. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

Table 4.3-1

REFERENCE CORE LOADING PATTERN

Fuel Designation	Number Loaded
(Provided by Applicant)	(Provided by Applicant)
Reference Core Loading Pattern:	(Provided by Applicant)

Table 4.3-2

CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C

Beginning of Cycle, K-effective	
Uncontrolled	(Provided by Applicant)
Fully Controlled	(Provided by Applicant)
Strongest Control Rod Out	(Provided by Applicant)
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle, Δk	(Provided by Applicant)

Table 4.3-3

CALCULATED NEUTRON FLUXES (USED TO EVALUATE VESSEL IRRADIATION)

Neutron Energy (MeV)	Average Flux In the Core (n/cm^2-sec)	Flux at the Core Boundary (n/cm^2-sec)	Flux at the Inside Surface Vessel (n/cm^2-sec)
>3.0	1.4×10^{13}	4.2×10^{12}	1.1×10^{9}
1.0 - 3.0	3.6×10^{13}	9.5×10^{12}	9.5×10^{9}
0.1 - 1.0	6.1×10^{13}	1.5×10^{12}	1.6×10^{9}

Maximum Fluence > 1.0 MeV at the vessel i.d. = $4.3 \times 10^{18} \frac{n}{cm^2}$ (2)

Notes:

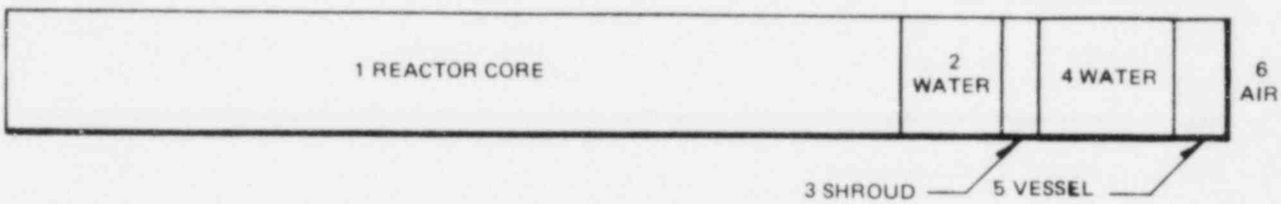
1. The calculated flux is a maximum in the axial direction but average over the azimuthal angle.

2. The maximum fluence is calculated using the flux and a capacity factor of 80% or 1×10^9 full power seconds. The fluence includes an azimuthal peaking factor and a factor to cover analytical uncertainties. The azimuthal peaking factor is derived from the results of a two-dimensional transport calculation. The two-dimensional analysis models the reactor bundle pattern in an r, θ geometry. Fluxes are calculated at the cylindrical core shroud surrounding the core. The peaking factor used was 1.4. In addition to the angular peaking factor, a safety factor of 2 was applied to ensure that the predicted values are conservative.

Table 4.3-4

CALCULATED NEUTRON FLUX AT CORE EQUIVALENT BOUNDARY

Group	Lower Energy Bound (eV)	Flux (n/cm^2-sec)
1	10.0×10^6	3.6×10^{10}
2	6.065×10^6	5.3×10^{11}
3	3.679×10^6	2.0×10^{12}
4	2.231×10^6	3.9×10^{12}
5	1.353×10^5	4.6×10^{12}
6	8.208×10^5	4.1×10^{12}
7	4.979×10^5	4.0×10^{12}
8	3.020×10^5	2.8×10^{12}
9	1.832×10^5	2.4×10^{12}
10	6.738×10^4	3.4×10^{12}
11	2.479×10^4	2.3×10^{12}
12	9.119×10^3	2.3×10^{12}
13	3.355×10^3	2.1×10^{12}
14	1.234×10^3	2.1×10^{12}
15	4.540×10^2	2.0×10^{12}
16	1.670×10^2	2.1×10^{12}
17	6.144×10^1	1.9×10^{12}
18	2.260×10^1	1.9×10^{12}
19	1.371×10^1	9.2×10^{12}
20	8.315	9.2×10^{12}
21	5.043	8.4×10^{12}
22	3.059	8.7×10^{11}
23	1.255	8.6×10^{11}
24	1.125	8.5×10^{11}
25	0.616	9.1×10^{11}
26	0.000	3.2×10^{13}



MATERIAL		RADIUS (inches)	MATERIAL	VOLUME AVERAGE DENSITY
NO.	NAME			
1	REACTOR CORE	92.58	WATER UO_2 304L STAINLESS STEEL ZIRCONIUM	0.31[illegible] g/cm^3 2.334 g/cm^3 0.056 g/cm^3 0.978 g/cm^3
2	WATER	99.9	WATER	0.74 g/cm^3
3	SHROUD	101.9	304L STAINLESS STEEL	FROM ASME SA 240
4	WATER	119.0	WATER	0.74 g/cm^3
5	VESSEL	125.0	CARBON STEEL	FROM ASME SA 533
6	AIR		AIR	1.3×10^{-3} g/cc

Figure 4.3-1. Model for One-Dimensional Transport Analysis of Vessel Fluence

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CONTENTS (Continued)

SECTION 4.4

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4.4 THERMAL-HYDRAULIC DESIGN

See Appendix A, Section A.4.4 of Reference 1.

4.4.1 Design Basis

See Appendix A, Subsection A.4.4.1 of Reference 1.

4.4.1.1 Safety Design Bases

See Appendix A, Subsection A.4.4.1.1 of Reference 1.

4.4.1.2 Power Generation Design Bases

See Appendix A, Subsection A.4.4.1.2 of Reference 1.

4.4.1.3 Requirements for Steady-State Conditions

For purposes of maintaining adequate thermal margin during normal steady-state operation, the MCPR must not be less than the required MCPR operating limit, and the MLHGR must be maintained below the design LHGR for the plant. This does not specify the operating power nor does it specify peaking factors. These parameters are determined subject to a number of constraints including the thermal limits given previously. The core and fuel design basis for steady-state operation (i.e., MCPR and LHGR limits) have been defined to provide margin between the steady-state operating conditions and any fuel damage condition to accommodate uncertainties and to assure thatno fuel damage results even during the worst anticipated transient condition at any time in life. The design steady-state MCPR operating limit and the peak LHGR is given in Table 4.4-1.

4.4.1.4 Requirements for Transient Conditions

See Appendix A, Subsection A.4.4.1.4 of Reference 1.

4.4.1.5 Summary of Design Bases

See Appendix A, Subsection A.4.4.1.5 of Reference 1.

4.4.2 Description of Thermal Hydraulic Design of Reactor Core

See Appendix A, Subsection A.4.4.2 of Reference 1.

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal and hydraulic standpoint is provided in Subsection 4.4.3.

A tabulation of thermal and hydraulic parameters of the core is given in Table 4.4-1, which gives comparison of this reactor with others of similar design.

4.4.2.2 Critical Power Ratio

See Appendix A, Subsection A.4.4.2.2 of Reference 1.

4.4.2.3 Linear Heat Generation Rate (LHGR)

See Appendix A, Subsection A.4.4.2.3 of Reference 1.

4.4.2.4 Void Fraction Distribution

The core average and maximum exit void fractions in the core at rated condition are given in Table 4.4-1. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the core are given in Table 4.4-2. The core average and maximum exit value is also provided. Similar distributions for steam quality are provided in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

See Appendix A, Subsection A.4.4.2.5 of Reference 1.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

See Appendix A, Subsection A.4.4.2.6 of Reference 1.

4.4.2.7 Correlation and Physical Data

See Appendix A, Subsection A.4.4.2.7 of Reference 1.

4.4.2.8 Thermal Effects of Operational Transients

See Appendix A, Subsection A.4.4.2.8 of Reference 1.

4.4.2.9 Uncertainties in Estimates

See Appendix A, Subsection A.4.4.2.9 of Reference 1.

4.4.2.10 Flux Tilt Considerations

See Appendix A, Subsection A.4.4.2.10 of Reference 1.

4.4.3 <u>Description of the Thermal and Hydraulic Design of the Reactor Coolant System</u>

See Appendix A, Subsection A.4.4.3 of Reference 1.

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The reactor coolant system is described in Section 5.4 and shown in isometric perspective in Figure 5.4-1. The piping sizes, fittings and valves are listed in Table 5.4-1.

4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the reactor coolant system is shown in Figure 5.1-1.

4.4.3.1.3 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown in Figure 5.1-2.

Table 4.4-5 provides the flow path length, height, liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel and recirculation loops of the reactor coolant systems.

Table 4.4-6 provides the lengths and sizes of all safety injection lines to the reactor coolant system.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in Figure 5.4-3. These curves are valid for all conditions with a normal operating range varying from approximately 20 percent to 115 percent of rated pump flow.

The pump characteristics, including considerations of NPSH requirements, are the same for the conditions of two-pump and one-pump operation as described in Subsection 5.4.1. Subsection 4.4.3.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump loads, bearing design flow starvation, and pump speed.

4.4.3.3 Power-Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

A BWR must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The power-flow map for the power range of operation is shown in Figure 4.4-1. The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. The boundaries on this map are as follows:

> Natural Circulation Line, A: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.
>
> 105% Steam Flow Rod Line or Rated Power (Whichever Is Less): The 105% steam flow rod line passes through 104.2% power at 100% flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern; however, rated power may not be exceeded. 105% steam flow rod line is based on a constant xenon concentration at 104.2% power and rated flow.
>
> Cavitation Protection Line: This line results from the recirculation pump, flow control valve and jet pump NPSH requirements.

4.4.3.3.1.1 Performance Characteristics

Other performance characteristics shown on the power-flow operating map are:

4.4.3.3.1.1 Performance Characteristics (Continued)

Constant Rod Lines: These lines show the change in power associated with flow changes, while maintaining constant control rod position.

Constant Position Lines for Flow Control Valve, B, C, D, and F: These lines show the change in flow associated with power changes while maintaining flow-control valves at a constant position.

4.4.3.3.2 Regions of the Power Flow Map

Region I - This region defines the system operational capability with the recirculation pumps and motors being driven by the low frequency motor-generator set at 25% speed. Flow is controlled by the flow control valve and power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to start up along curve C - FCV wide open at 25% speed.

Region II - This region shows the area where 25% pump speed and 100% pump speed operating regimes overlap. The switching sequence from the low frequency m-g set to 100% speed will be done in this region.

Region III - This is the low power area of the operating map where cavitation can be expected in the recirculation pumps, jet pumps, or flow control valves. Operation within this region is precluded by system interlocks which trip the main motor from the 100% speed power source to the 25% speed power source.

4.4.3.3.2 Regions of the Power Flow Map (Continued)

Region IV - This represents the normal operating zone of the map where power changes can be made, by either control rod movement or by core flow changes, through use of the flow control valves.

4.4.3.3.3 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions to the required values shown in Figure 4.4-1.

(1) Minimum Power Limits at Intermediate and High Core Flows: To prevent cavitation in the recirculation pumps, jet pumps, and flow control valves, the recirculation system is provided with an interlock to trip off the 100% speed power source and close the 25% speed power source if the difference between steamline temperature and recirculation pump inlet temperature is less than a preset value (9.8°F). This differential temperature is measured using high accuracy RTDs with a sensing error of less than 0.2°F at the two standard deviation (2σ) confidence level. This action is initiated electronically through a 15-sec time delay. The interlock is active while in both the automatic and manual operation modes.

(2) Minimum Power Limit at Low Core Flow: During low power, low loop flow operation, the temperature differential interlock may not provide sufficient cavitation protection to the flow control valves. Therefore, the system is provided with an interlock to trip off the 100% speed power source and close the 25% speed power source if the feedwater flow falls below a preset level (22% of rated) and the flow control valves are below

4.4.3.3.3 Design Features for Power-Flow Control (Continued)

a preset position (20% open). The feedwater flow rate and recirculation flow control valve position are measured by existing process control instruments. The speed change action is electronically initiated. This interlock is active during both automatic and manual modes of operation.

(3) Pump Bearing Limit: For pumps as large as the recirculation pumps, practical limits of pump bearing design require that minimum pump flow be limited to 20% of rated. To assure this minimum flow, the system is designed so that the minimum flow control valve position will allow this rate of flow.

(4) Valve Position: To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system is provided with an interlock to prevent starting the pumps, or to trip the pumps if the suction or discharge block valves are at less than 90% open position. This circuit is activated by a position limit switch and is active before the pump is started, during manual operation mode, and during automatic operation mode.

4.4.3.3.3.1 Flow Control

The principal modes of normal operation with valve flow control-low frequency motor generator (LFMG) set are summarized as follows: the recirculation pumps are started on the 100% speed power source in order to unseat the pump bearings. Suction and discharge block valves are full open and the flow control valve is in the minimum position. When the pump is near full speed, the main power source is tripped and the pump allowed to coast down to approximately 25%

4.4.3.3.1 Flow Control (Continued)

speed, where the LFMG set will power the pump and motor. The flow control valve is then opened to the maximum position, at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-1.

When reactor power is greater than approximately 20-28% of rated, the low feedwater flow interlock is cleared and the main recirculation pumps can be switched to the 100% speed power source. The flow control valve is closed to the minimum position before the speed change to prevent large increases in core power and potential flux scram. This operation occurs within Region II of the operating map. The system is then brought to the desired power-flow level within the normal operating area of the map (Region IV) by opening the flow control valves and by withdrawing control rods.

Control rod withdrawal with constant flow control valve position will result in power/flow changes along lines of constant c sub (v) (constant position). Flow control valve movement with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line.

4.4.3.4 Temperature-Power Operating Map (PWR)

Not applicable.

4.4.3.5 Load-Following Characteristics

See Appendix A, Subsection A.4.4.3.5 of Reference 1.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the reactor coolant system.

4.4.4 Evaluation

See Appendix A, Subsection A.4.4.4 of Reference 1.

4.4.5 Testing and Verification

See Appendix A, Subsection A.4.4.5 of Reference 1.

4.4.6 Instrumentation Requirements

See Appendix A, Subsection A.4.4.6 of Reference 1.

4.4.6.1 Loose Parts

To be supplied by Applicant.

4.4.7 References

1. "General Electric Standard Application for Reactor Fuel," (NEDE-24011, latest approved revision).

Table 4.4-1

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE

General Operating Conditions	(238-748)
Reference design thermal output (Mwt)	3579
Power level for engineered safety features (Mwt)	3730
Steam flow rate, at 420°F final feedwater temperature (millions lb/hr)	15.400
Core coolant flow rate (millions lb/hr)	104.0
Feedwater flow rate (millions lb/hr)	15.367
System pressure, nominal in steam dome (psia)	1040
System pressure, nominal core design (psia)	1055
Coolant saturation temperature at core design pressure (°F)	551
Average power density (kW/liter)	54.1
Maximum Linear Heat Generation Rate (kW/ft)	13.4
Average Linear Heat Generation Rate (kW/ft)	5.9
Core total heat transfer area (ft^2)	73,303

Table 4.4-1 (Continued)

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE

General Operating Conditions	(238-748)
Maximum heat flux (Btu/hr-ft^2)	361,600
Average heat flux (Btu/hr-ft^2)	159,500
Design operating minimum critical power ratio (MCPR)	1.20
Core inlet enthalpy at 420°F FFWT (Btu/lb)	527.7
Core inlet temperature, at 420°F FFWT (°F)	533
Core maximum exit voids within assemblies (%)	79.0
Core average void fraction, active coolant	0.414
Maximum fuel temperature (°F)	3435
Active coolant flow area per assembly (in.^{2})	15.164
Core average inlet velocity (ft/sec)	6.98
Maximum inlet velocity (ft/sec)	8.54
Total core pressure drop (psi)	26.4
Core support plate pressure drop (psi)	22.0
Average orifice pressure drop Central region (psi)	5.71

Table 4.4-1 (Continued)

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE

General Operating Conditions	(238-748)
Average orifice pressure drop Peripheral region (psi)	18.68
Maximum channel pressure loading (psi)	15.40
Average-power assembly channel pressure loading (bottom) (psi)	14.1
Shroud support ring and lower shroud pressure loading	25.7
Upper shroud pressure loading (psi)	3.7

]

Table 4.4-2

VOID DISTRIBUTION

Core Average Value - 0.414
Maximum Exit Value - 0.790
Active Fuel Length - 150 inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0	0
	2	0	0.008
	3	0.008	0.084
	4	0.042	0.204
	5	0.104	0.314
	6	0.178	0.402
	7	0.253	0.475
	8	0.323	0.532
	9	0.381	0.578
	10	0.429	0.614
	11	0.467	0.644
	12	0.498	0.668
	13	0.524	0.687
	14	0.545	0.703
	15	0.563	0.718
	16	0.579	0.730
	17	0.593	0.742
	18	0.606	0.753
	19	0.619	0.763
	20	0.631	0.773
	21	0.640	0.780
	22	0.648	0.785
	23	0.654	0.789
Top of Core	24	0.656	0.790

]

Table 4.4-3

FLOW QUALITY DISTRIBUTION

Core Average Value - 0.079
Maximum Exit Value - 0.332
Active Fuel Length - 150 inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0	0
	2	0	0
	3	0	0.004
	4	0.001	0.014
	5	0.004	0.031
	6	0.011	0.050
	7	0.020	0.072
	8	0.031	0.096
	9	0.043	0.118
	10	0.055	0.140
	11	0.066	0.161
	12	0.077	0.181
	13	0.087	0.199
	14	0.097	0.215
	15	0.106	0.231
	16	0.114	0.245
	17	0.122	0.260
	18	0.130	0.275
	19	0.138	0.289
	20	0.146	0.303
	21	0.153	0.315
	22	0.159	0.324
	23	0.163	0.330
Top of Core	24	0.165	0.332

Table 4.4-4

AXIAL POWER DISTRIBUTION USED TO GENERATE

VOID AND QUALITY DISTRIBUTIONS

	Node	Axial Power Factor
Bottom of Core	1	0.38
	2	0.69
	3	0.93
	4	1.10
	5	1.21
	6	1.30
	7	1.47
	8	1.51
	9	1.49
	10	1.44
	11	1.36
	12	1.28
	13	1.16
	14	1.06
	15	1.01
	16	0.97
	17	0.94
	18	0.97
	19	0.96
	20	0.91
	21	0.77
	22	0.59
	23	0.38
Top of Core	24	0.12

Table 4.4-5

REACTOR COOLANT SYSTEM GEOMETRIC DATA

	Flow Path Length (in.)	Height and Liquid Level (in.)	Elevation of Bottom of Each Volume* (in.)	Minimum Flow Areas (ft^2)
A. Lower Plenum	213.5	213.5 213.5	-170.5	84.0
B. Core	164.5	164.5 164.5	43.0	146.5 includes bypass
C. Upper Plenum and Separators	179.0	179.0 179.0	207.5	57.5
D. Dome (Above Normal Water Level	289.5	289.5 0	386.0	309.0
E. Downcomer Area	311.5	311.5 311.5	-27.5	66.0
F. Recirculation Loops and Jet Pumps	114.0 ft (one loop)	398.0 398.0	-392.0	132.5 in^2

* Reference Point is recirculation nozzle outlet centerline.

Table 4.4-6

LENGTHS OF SAFETY INJECTION LINES

Loop	Line	Nominal Diameter (in)	Pipe Schedule	Length (ft)
HPCS	HPCS 3	16	100	57
		12	100	48
	HPCS 4	12	80	154
LPCS	LPCS 2	14	40	92
		12	40	14
	LPCS 3	12	80	140
LPCI"A"	RHR 7	18	40	23
	RHR 12	18	40	4
	RHR 9	18	40	116
		14	40	45
	RHR 10	12	80	66
LPCI"B"	RHR 13	18	40	23
	RHR 18	18	40	4
	RHR 15	18	40	104
		14	40	112
	RHR 16	12	80	61
LPCI"C"	RHR 21	18	40	96
	RHR 22	14	80	216
		12	80	58

NOTE Lengths given are to the nearest foot, and are measured from the appropriate pump outlet nozzle to the RPV nozzle.

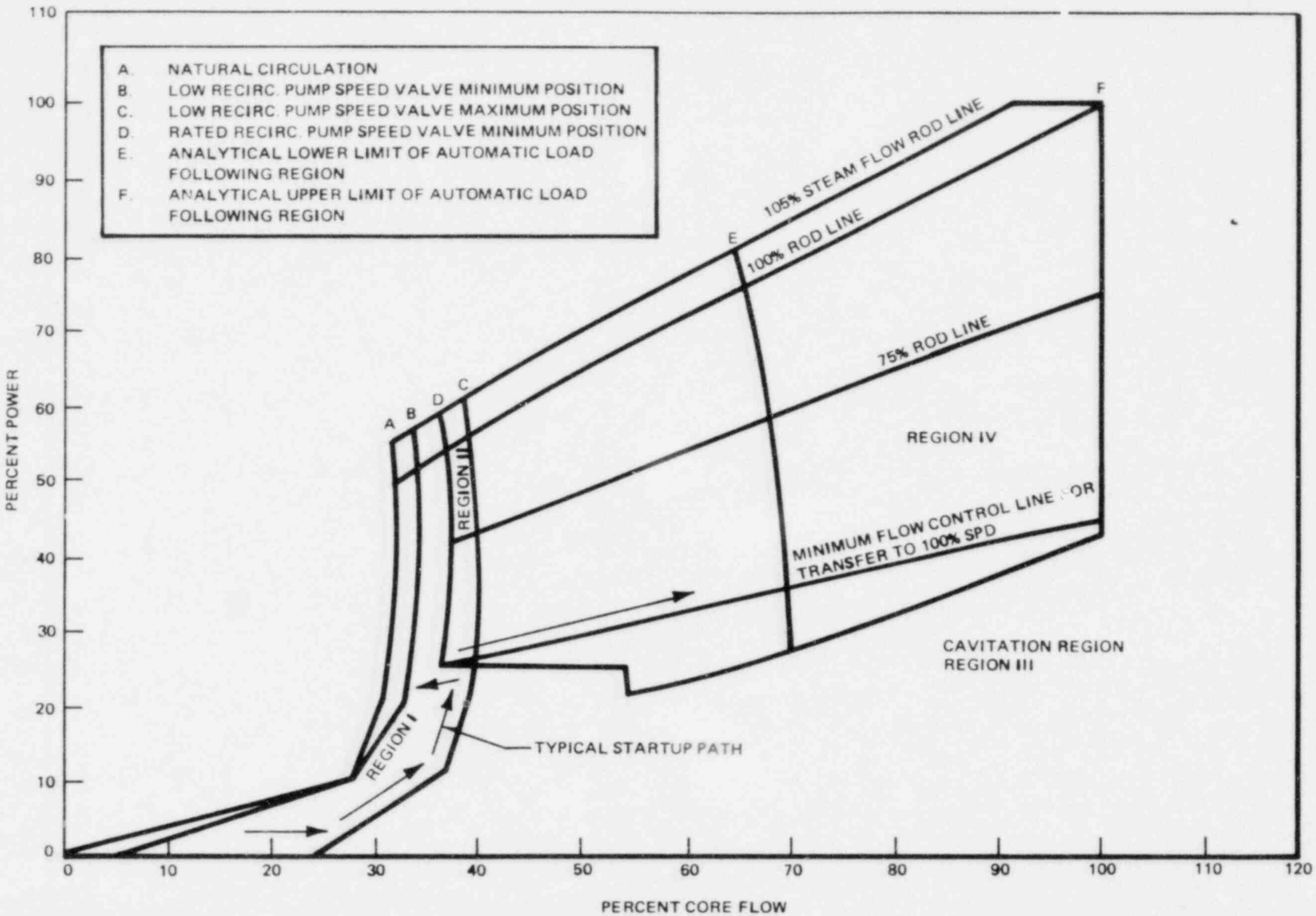

Figure 4.4-1. Power-Flow Operating Map

4.6.1.1.2.2.1 Drive Piston

The drive piston is mounted at the lower end of the index tube. The function of the index tube is similar to that of a piston rod in a conventional hydraulic cylinder. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder). Because the type of inner seal used is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction.

A pair of nonmetallic bushings prevents metal-to-metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented, step-cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents piston contact with the cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 in.^{2} versus 4.1 in.^{2} for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and assures a higher force for insertion than for withdrawal.

4.6.1.1.2.2.2 Index Tube

The index tube is a long hollow shaft made of nitrided stainless steel. Circumferential locking grooves, spaced every 6 in. along the outer surface, transmit the weight of the control rod to the collet assembly

4.6.1.1.2.2.3 Collet Assembly

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. This assembly prevents the index tube from accidentally moving downward. The assembly consists of the collet fingers, a return spring, a guide cap, a collet housing (part of the cylinder, tube, and flange) and the collet piston.

Locking is accomplished by fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a force of approximately 150 lb supplied by a spring. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive-in signal. A pressure, approximately 180 psi above reactor vessel pressure, must then be applied to the collet piston to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so they do not engage a locking groove.

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube.

If reactor water is used during a scram to supplement accumulator pressure, it is drawn through a filter on the guide cap.

APPENDIX 4A

CONTENTS

APPENDIX 4A

ILLUSTRATIONS (Continued)

APPENDIX 4A

CONTROL ROD PATTERNS AND ASSOCIATED POWER DISTRIBUTION FOR TYPICAL BWR

4A.1 INTRODUCTION

This appendix contains a typical simulation of an equilibrium cycle. This cycle was analyzed using the three-dimensional BWR simulator[1]. Qualification of this model is documented in Reference 2. The control rod patterns used are just one example of a set of control rod patterns which could be used to provide the radial and axial power shaping (Subsection 4.3.2.5) needed to meet the Technical Specifications.

The basic control rod strategy for this case consists of alternating between four types of control rod patterns: A2-B2-A1-B1. The definition of these rod pattern types, commonly referred to as control rod sequences, is given in Reference 3. By changing sequences regularly [typically every 1 GWd/st (1.1023 GWd/Te)], the locations of the deeply inserted rods are continually being moved. This precludes any bundle from being significantly controlled over a long period of exposure. In turn, control rod history is reduced and the bundle average exposure is more evenly distributed to obtain a better power distribution and better MCPR performance.

4A.2 POWER DISTRIBUTION STRATEGY

A basic operating principle used to minimize power peaking throughout an operating cycle has been developed and is applied to boiling water reactors. The principle is described in Reference 1. The main concept is that "for any given set of end-of-cycle conditions, the power peaking factor is maintained at the minimum value when the power shape does not change during the operating cycle".

4A.3 RESULTS OF CORE SIMULATION STUDIES

The following table itemizes the exposure step and its related figure numbers:

Incremental Exposure (GWd/st)	Sequence*	Figure Numbers
6.69	All-rods-out - Haling EOC	4A-1a through 4A-1d
0.2	A-2	4A-2a through 4A-2e
1.0	B-2	4A-3a through 4A-3e
2.0	A-1	4A-4a through 4A-4e
3.0	B-1	4A-5a through 4A-5e
4.0	A-2	4A-6a through 4A-6e
5.0	B-2	4A-7a through 4A-7e
6.0	A-1	4A-8a through 4A-8e
6.6	All rods out	4A-9a through 4A-9e

The detailed data presented demonstrates that this design can be operated throughout this cycle with adequate margins to allow for operating flexibility. The variation of the maximum linear heat generation rate (MLHGR) with cycle exposure is presented in Figure 4A-10. Significant margin exists relative to the MLHGR safety limit. Maximum average planar linear heat generation rates (MAPLHGR) are not calculated for this design since calculations show the peak clad temperature (PCT) to be less than the 2200°F limit when the maximum single rod is at the 13.4 kW/ft limit. Adherence to the MLHGR limit will always assure meeting the MAPLHGR limit. The variation of the MCPR with cycle exposure is shown in Figure 4A-11. Similarly, a large margin is indicated with respect to the expected MCPR operating limit.

4A.4 REFERENCES

1. J. A. Woolley, "Three-Dimensional BWR Core Simulator", January 1977 (NEDO-20953A).

2. C. R. Parkos, "BWR Simulator Methods Verification", January 1977 (NEDO-20946A).

3. C. F. Paone, "Banked Position Withdrawal Sequence", September 1976 (NEDO-21231).

I / J	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1										0.3003	0.3751	0.4104	0.4224	0.4272	0.4216
2								0.3264	0.4470	0.6476	0.7892	0.8983	0.8501	0.9179	0.8340
3							0.3973	0.7105	0.8897	0.8405	1.0449	0.9271	1.0938	0.9306	0.9846
4						0.4135	0.7566	0.9627	0.9058	1.1126	0.9967	1.1841	1.0292	1.1840	0.9919
5					0.4228	0.7741	0.9921	0.9291	1.0382	1.0492	1.1179	1.0873	1.1582	1.0837	1.1220
6				0.4169	0.7790	1.0108	0.9478	1.1650	1.0415	1.2384	1.0224	1.2853	1.1098	1.2855	1.0328
7			0.4015	0.7645	1.0059	1.0239	1.0607	1.0728	1.1411	1.1122	1.1962	1.1506	1.2385	1.1460	1.1783
8		0.3299	0.7175	0.9717	0.9383	1.1657	0.9693	1.2345	1.0273	1.2973	1.0679	1.3422	1.1495	1.3314	1.0479
9		0.4507	0.8980	0.9138	1.0451	1.0396	1.1325	1.0255	1.1956	1.1518	1.2314	1.1653	1.2452	1.1481	1.1739
10	0.3038	0.6522	0.8489	1.1252	1.0482	1.2477	1.1135	1.2993	1.1351	1.3368	1.0769	1.3257	1.0722	1.3017	1.0298
11	0.3789	0.7970	1.0573	1.0498	1.1363	1.0354	1.2034	1.0719	1.2427	1.0854	1.2336	1.0652	1.2242	1.1085	1.1411
12	0.4152	0.9065	0.9373	1.199[illegible]	[illegible]	1.3049	1.1610	1.3530	1.1803	1.3502	1.1549	1.3026	1.0385	1.2438	0.9801
13	0.4254	0.8553	1.1027	1.0377	1.1670	1.1156	1.2403	1.1534	1.2547	1.0836	1.2228	1.0462	1.1834	0.9901	1.0766
14	0.4291	0.9217	0.9336	1.1891	1.0875	1.2876	1.1448	1.3304	1.1540	1.3135	1.1235	1.2648	1.0663	1.1812	0.9185
15	0.4221	0.8348	0.9860	0.9923	1.1190	1.0310	1.1779	1.0487	1.1826	1.0378	1.1473	0.9920	1.0890	0.9220	0.9463

Figure 4A-9d. Integrated Power per Bundle at 6.6 GWd/st Cycle Exposure

I / J	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1										25451.2	24479.5	24674.7	24815.8	24845.5	24938.7
2								25421.8	25008.9	3986.5	13835.1	5564.2	14365.2	5949.9	14615.4
3							24730.8	13488.2	5562.5	21248.2	6521.4	21941.6	7072.6	22318.8	15111.2
4						25454.8	13930.5	6198.4	20959.7	7230.6	21849.8	7636.5	22205.5	7879.8	22964.1
5					24960.7	14147.2	6456.7	22129.9	16057.2	20076.8	15368.1	21107.3	14923.6	21578.1	15781.5
6				25454.2	14110.3	6583.4	21864.7	7771.8	22141.2	8220.4	28390.7	8231.0	23279.3	8138.1	26609.4
7			24720.0	13907.1	6488.4	15959.0	16142.2	20022.9	15651.3	21540.0	14774.7	21506.6	14110.2	21493.9	15664.3
8		25320.8	13411.0	6124.1	21760.8	7660.9	28696.2	8005.6	28590.3	8100.6	28381.7	7802.9	23113.6	7646.6	27504.0
9		24979.5	5451.4	20828.6	15889.5	22298.0	15536.4	28565.3	14787.8	21388.5	14386.4	21633.2	14024.6	21715.0	16261.0
10	25284.8	3896.0	21117.1	7026.1	20653.9	7961.1	21486.9	8007.8	22964.5	7551.7	28395.6	7773.1	28031.4	7956.0	27843.9
11	24417.0	13724.6	6360.5	21505.3	14817.1	27894.4	14736.8	28240.4	14039.5	28136.9	13873.8	28543.2	12763.1	22294.5	16229.8
12	24443.1	5465.6	21756.3	7407.2	20950.7	7695.7	21294.4	7692.4	21323.0	7469.9	21729.8	7865.1	28432.7	8212.6	28683.6
13	24781.0	14323.9	6972.3	22076.4	14833.1	23267.4	14308.8	23218.2	13994.0	28016.8	14038.6	28243.7	12814.2	28443.5	16382.4
14	24827.9	5904.2	22304.3	7835.0	21517.7	8182.1	21705.3	7910.7	21717.8	7847.4	21874.4	7979.9	22272.3	8074.4	28690.5
15	24946.7	14596.9	15061.0	22982.1	16055.7	26829.3	15714.7	27506.3	15814.8	27365.3	16151.4	28011.3	16107.2	28452.3	21915.2

Figure 4A-9e. Average Bundle Exposure at 6.6 GWd/st Cycle Exposure

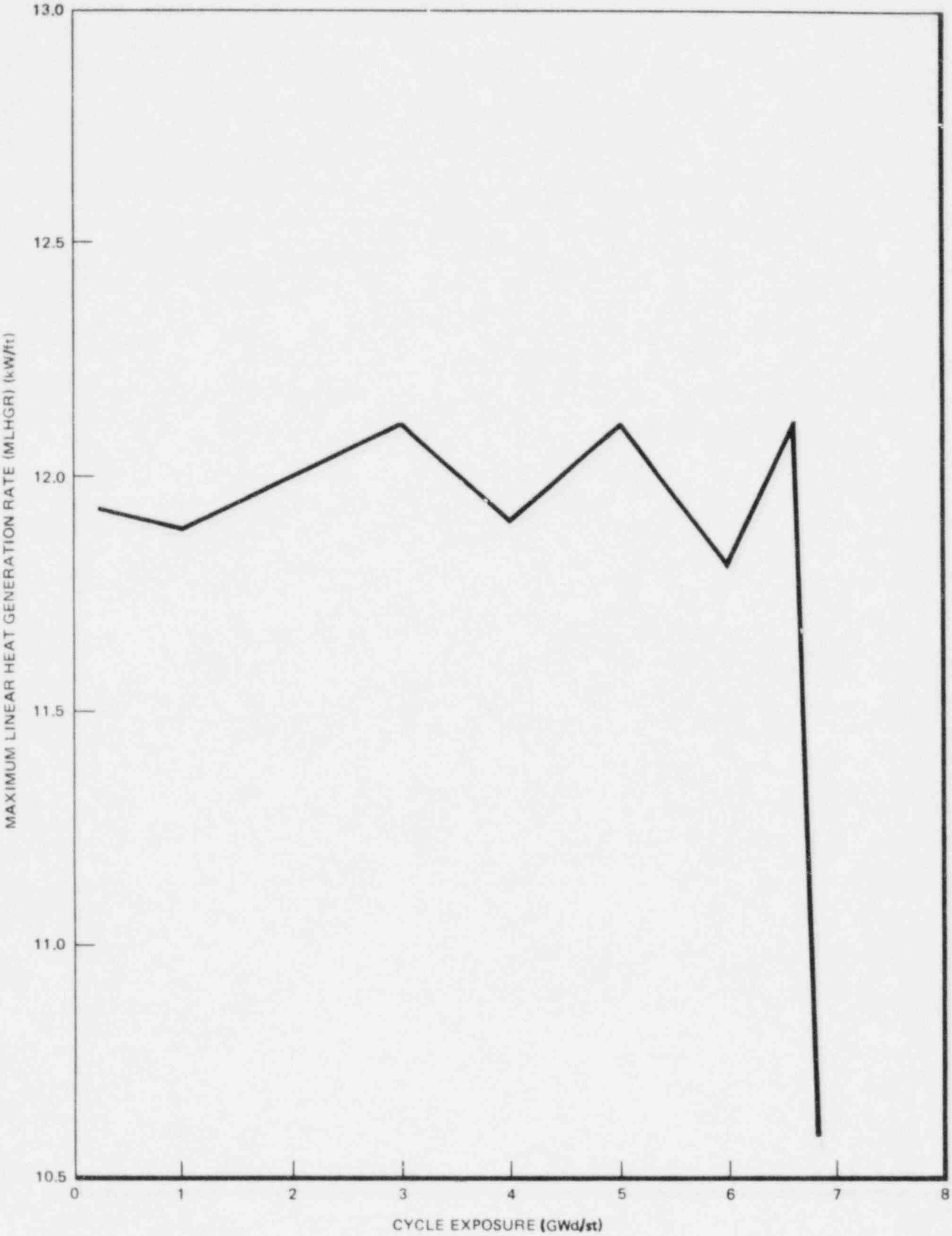

Figure 4A-10. Maximum Linear Heat Generation Rate as a Function of Cycle Exposure

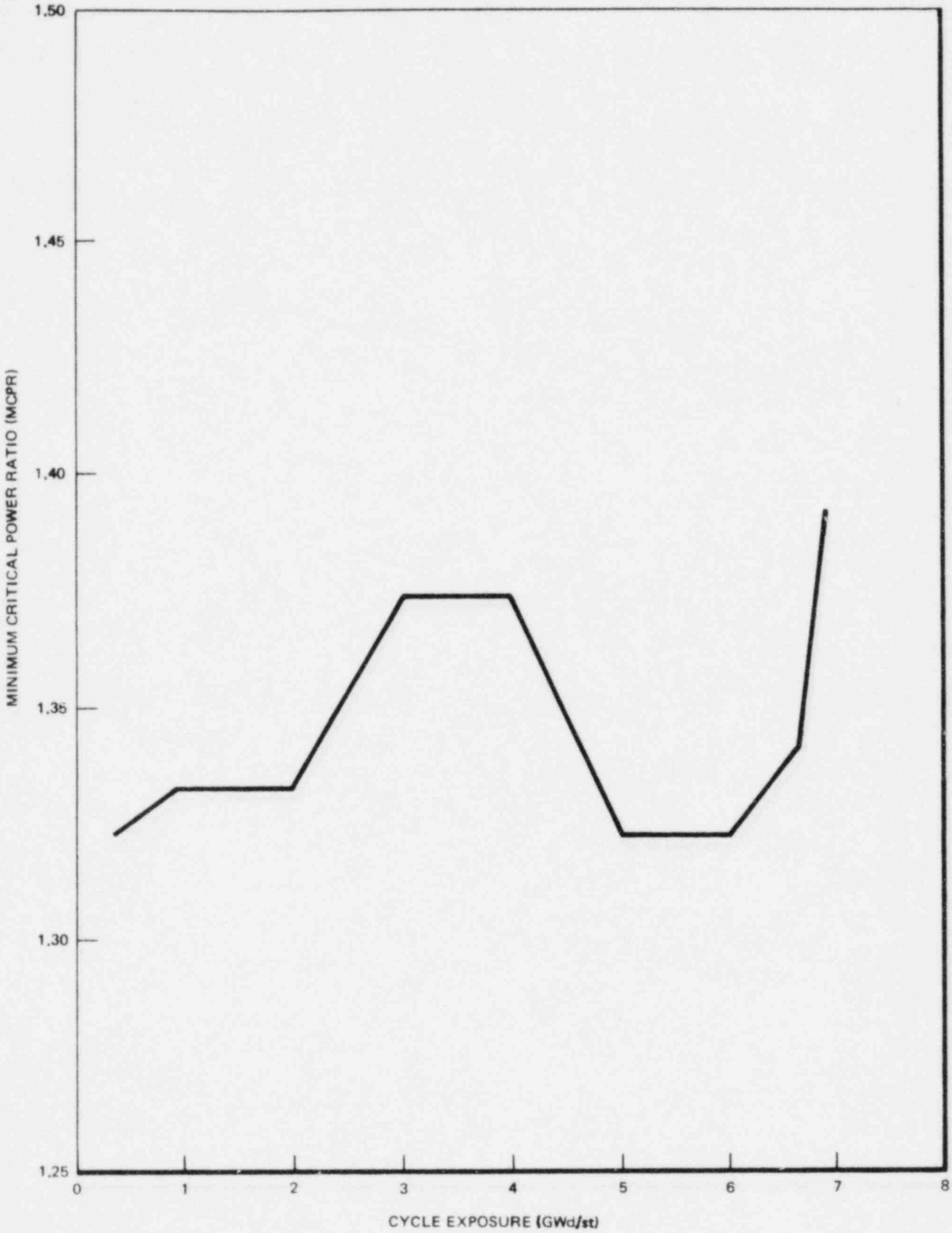

Figure 4A-11. Minimum Critical Power Ratio as a Function of Cycle Exposure

SECTION 5.2

CONTENTS

CONTENTS (Continued)

SECTION 5.2

TABLES

SECTION 5.2

ILLUSTRATIONS

5.2.2.1.1 Safety Design Bases (Continued)

(3) permit verification of its operability; and

(4) withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions.

5.2.2.1.2 Power Generation Design Bases

The nuclear pressure-relief system safety/relief valves have been designed to meet the following power generation bases:

(1) discharge to the containment suppression pool, and

(2) correctly reclose following operation so that maximum operational continuity is obtained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions as discussed in Subsection 5.2.3 of Reference 3.

5.2.2.1.3 Discussion (Continued)

The SRV setpoints are listed in Table 5.2-2 and satisfy the ASME Code specifications for safety valves because all valves open at less than the nuclear system design pressure of 1250 psig.

The automatic depressurization capability of the nuclear system pressure relief system is evaluated in Section 6.3, Emergency Core Cooling Systems, and in Section 7.3, Engineered Safety Feature Systems.

The following criteria are used in selection of relief valves:

(1) must meet requirements of ASME Code, Section III;

(2) must qualify for 100% of nameplate capacity credit for the overpressure protection function; and

(3) must meet other performance requirements such as response time, etc., as necessary to provide relief functions.

The SRV discharge piping is designed, installed, and tested in accordance with ASME Code, Section III.

5.2.2.1.4 Safety/Relief Valve Capacity

SRV capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components, up to and including applicable addenda. The essential ASME requirements which are met by this analysis are as follows.

5.2.2.1.4 Safety/Relief Valve Capacity (Continued)

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of two sources: a direct or a flux trip signal. The direct scram trip signal is derived from position switches mounted on the main steamline isolation valves, the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Credit is taken for 50% of the total installed safety/relief valve capacity operating by the power-operated mode as permitted by ASME III. Credit is also taken for the remaining safety/relief valve capacity which opens by the spring mode of operation direct from inlet pressure.

The rated capacity of the pressure-relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.10 x 1250 psig = 1375 psig) for events defined in Section 15.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

Table 5.2-3 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

See Appendix A, Subsection A.5.2.2.2.1 of Reference 3.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions.

5.2.2.2.2.1 Operating Conditions

(1) operating power = 3729 MWt (104.2% of nuclear boiler rated power);

(2) vessel dome pressure $\leq$ 1045 psig; and

(3) steamflow = 16.71×10^6 lb/hr (105% of nuclear boiler rated steamflow).

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

See Appendix A, Subsection A.5.2.2.2.2.2 of Reference 1.

5.2.2.2.2.3 Safety/Relief Valve Transient Analysis Specification

(1) Simulated valve groups:

power-actuated relief mode - 4 groups
spring-action safety mode - 5 groups

(2) opening pressure setpoint (maximum safety limit):

power-actuated relief mode -	group 1	1125 psig
	group 2	1135 psig
	group 3	1145 psig
	group 4	1155 psig

spring-action safety mode -	group 1	1175 psig
	group 2	1185 psig
	group 3	1195 psig
	group 4	1205 psig
	group 5	1215 psig

(3) reclosure pressure setpoint (% of opening setpoint) both modes:

maximum safety limit (used in analysis)	98
minimum operational limit	89

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically the assumed

5.2.2.2.2.3 Safety/Relief Valve Transient Analysis Specification (Continued)

setpoints in the analysis are at least 1 to 2% above the actual nominal setpoints. Conservative safety/relief valve response characteristics are also assumed, therefore, the analysis conservatively bounds all safety/relief valve operating conditions.

5.2.2.2.2.4 Safety/Relief Valve Capacity

Sizing of the SRV capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients.

The method used to determine total valve capacity is as follows.

Whenever system pressure increases to the relief pressure setpoint of a group of valves having the same setpoint, half of those valves are assumed to operate in the relief mode, opened by the pneumatic power actuation. When the system pressure increases to the valve spring set pressure of a group of valves, those valves not already considered open are assumed to begin opening and to reach full open at 103% of the valve spring set pressure.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from an MSIV closure with flux scram transient as documented in Subsection S.2.3 of Reference 3. Results of this analysis is given in Figure 5.2-3.

5.2.2.2.3.2 Low-Low Set Relief Function

In order to assure that no more than one relief valve reopens following a reactor isolation event, two relief valves are provided with lower opening and closing setpoints and four relief valves are provided with lower closing setpoints. These setpoints override the normal setpoints following the initial opening of the relief valves and act to hold open these valves longer, thus preventing reopening of more than one valve subsequently. This system logic is referred to as the low-low set relief logic and functions to ensure that the containment design basis of one safety/relief valve operating on subsequent actuations is met.

5.2.2.2.3.2 Low-Low Set Relief Function (Continued)

The low-low set relief function is armed from the same pressure sensors which initiate opening of the SRV set at any of the normal relief setpoints. Thus, the low-low set valves will not actuate during normal plant operation even though the reopening setpoint of one of the valves is in the normal-operating pressure range. This arming method results in the low-low set SRVs opening initially during an overpressure transient at the normal relief opening setpoint.

The lowest setpoint low-low set valve will cycle to remove decay heat. Since this valve will have a larger differential between its opening and closing set pressures than assumed for the normal relief function, the number of single safety/relief valve actuations during isolation events will be reduced. Table 5.2-2 shows the opening and closing setpoints for the low-low set safety/relief valves.

The assumptions used in the calculation of the pressure transient after the initial opening of the relief valves are:

(1) the transient event is a 3-second closure of all MSIVs with position scram,

(2) nominal relief valve setpoints are used,

(3) the maximum expected relief capacity is used,

(4) relief valve opening response time (Figure 5.2-7) is used,

(5) the normal closing setpoint of the relief valves is 100 psi below the opening setpoint, and

(6) ANS + 20% decay heat at infinite exposure is used.

(4) continuous monitoring of floor drain sump level and a source of water for calibration and testing is provided.

These satisfy Position C.8 requirements.

Limiting unidentified leakage to 5 gpm and identified to 25 gpm satisfies Position C.9.

For commitment, and revision number, see Section 1.8.

5.2.6 References

1. J. M. Skarpelos and J. W. Bagg, Chloride Control in BWR Coolants, GE, June 1973 (NEDO-10899).

2. M. B. Re nolds, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws, GE, April 1968 (GEAP-5620).

3. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved version).

Table 5.2-4

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Figure 5.2-1
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Figure 5.2-[illegible]
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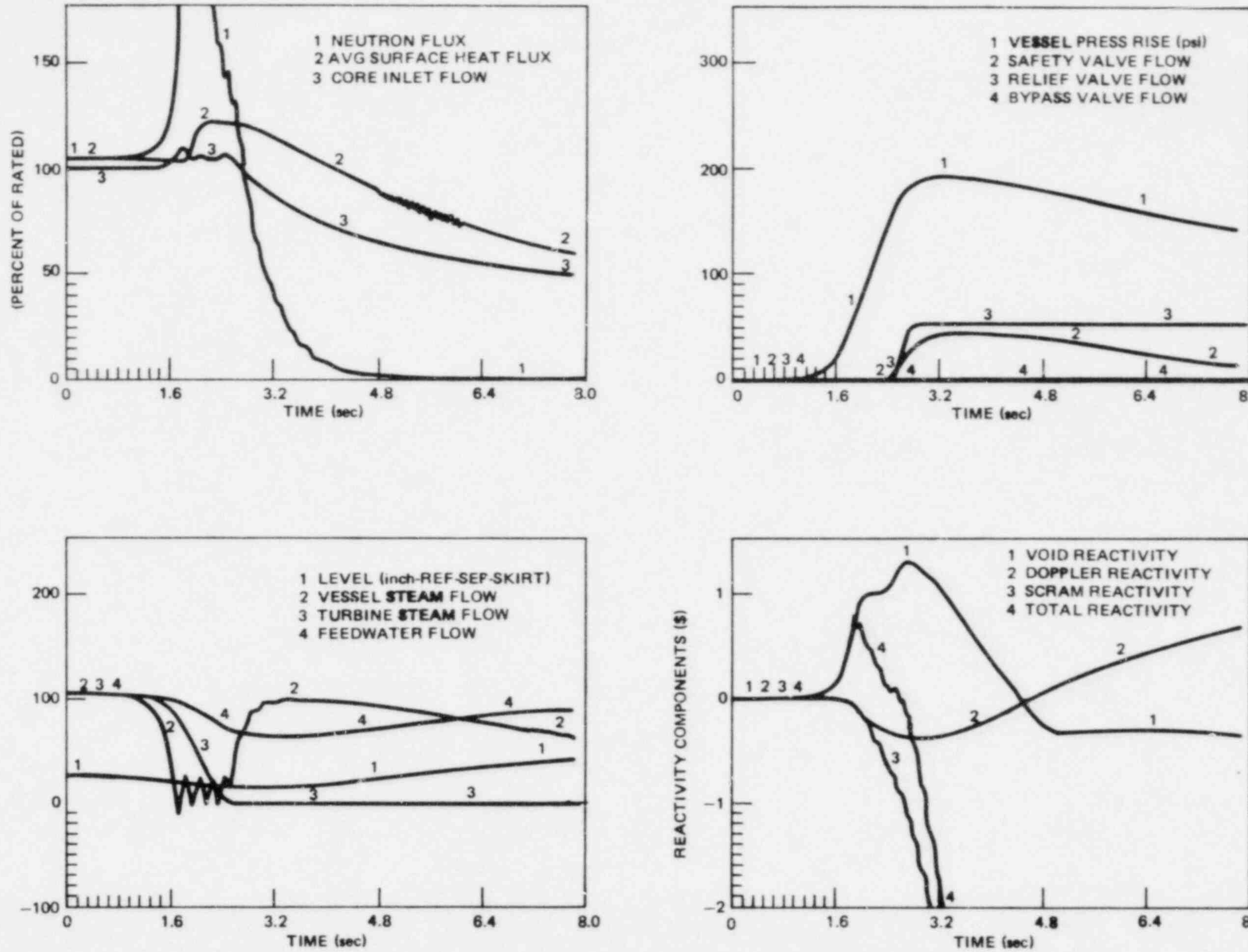

Figure 5.2-3. MSIV Closure with Flux Scram and Installed Safety/Relief Valves Capacity

Figure 5.2-4
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6.3 EMERGENCY CORE COOLING SYSTEMS

6.3.1 Design Bases and Summary Description

Section 6.3.1 provides the design bases for the Emergency Core Cooling System (ECCS) and a summary description of the several systems as an introduction to the more detailed design descriptions provided in Subsection 6.3.2 and the performance analysis provided in Subsection 6.3.3.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of coolant accidents (LOCA) caused by ruptures in primary system piping. The functional requirements (e.g., coolant delivery rates) specified in detail in Table 6.3-1 are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10CFR50, paragraph 50.46, (Acceptance Criteria for Emergency Core Cooling System for Light Water-Cooled Nuclear Power Reactors). These requirements are summarized in Subsection 6.3.3.2. In addition, the ECCS is designed to meet the following requirements:

(1) Protection is provided for any primary system line break up to and including the double-ended break of the largest line.

(2) Two independent phenomenological cooling methods (flooding and spraying) are provided to cool the core.

6.3.1.1.1 Performance and Functional Requirements (Continued)

(3) One high-pressure cooling system is provided which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 1 in. nominal diameter.

(4) No operator action is required until 10 min after an accident to allow for operator assessment and decision.

(5) The ECCS is designed to satisfy all criteria specified in Section 6.3 for any normal mode of reactor operation.

(6) A sufficient water source and the necessary piping, pumps and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a LOCA.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

(1) The ECCS must conform to all licensing requirements and good design practices of isolation, separation and common mode failure considerations.

(2) In order to meet the above requirements, the ECCS network shall have built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment shall make up the ECCS:

1 High Pressure Core Spray (HPCS)
1 Low Pressure Core Spray (LPCS)
3 Low Pressure Coolant Injection (LPCI) Loops
1 Automatic Depressurization System (ADS)

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6.3.1.1.1 Performance and Functional Requirements (Continued)

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1 High Pressure Core Spray (HPCS)
1 Low Pressure Core Spray (LPCS)
3 Low Pressure Coolant Injection (LPCI) Loops
1 Automatic Depressurization System (ADS)

6.3.2.8 Manual Actions (Continued)

as indicating the operation of the ECCS. ECCS flow indication is the primary parameter available to assess proper operation of the system. Other indications, such as position of valves, status of circuit breakers, and essential power bus voltage, are available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Section 7.3. Other available instrumentation is listed in the P&IDs for the individual systems. Much of the monitoring instrumentation available to the operator is discussed in more detail in Chapter 5 and Section 6.2.

6.3.3 ECCS Performance Evaluation

The performance of the ECCS is determined through application of the 10CFR50 Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. Analytical models are documented in Subsection S.2.5.2 of Reference 4.

The ECCS analyses results provided in this subsection constitute the lead plant analysis for BWR/6 plants. MAPLHGR results are for a fuel enrichment of approximately 3 wt% U-235. Analytical results specified in Reference 5 for non-lead plant BWR/6 analyses are provided by applicant.

The accidents, as listed in Chapter 15, for which ECCS operation is required are:

Subsection	Title
15.2.8	Feedwater Piping Break
15.6.4	Spectrum of BWR Steam System Piping Failures Outside of Containment
15.6.5	Loss-of-Coolant Accidents

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates (MAPLHGR) calculated in this performance analysis provide the basis for Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10CFR50.46 are listed, and, for each criterion, applicable parts of Subsection 6.3.3 (where conformance is demonstrated) are indicated. A detailed description of the methods used to show compliance are shown in Subsection S.2.5.2 of Reference 4.

Criterion 1: Peak Cladding Temperature

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 is shown in Subsections 6.3.3.7.3 (Break Spectrum), 6.3.3.7.4 (Design Basis Accident), 6.3.3.7.5 (Transition Break), 6.3.3.7.6 (Small Break), and specifically in Table 6.3-4 (MAPLHGR, maximum local oxidation, and peak cladding temperature versus exposure).

Criterion 2: Maximum Cladding Oxidation

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformation to Criterion 2 is shown in Figure 6.3-8 (break spectrum plot), Table 6.3-4 (local oxidation versus exposure) and Table 6.3-5 (break spectrum summary).

6.3.3.4 System Performance During the Accident (Continued)

Key ECCS actuation setpoints and time delays for all the ECCS systems are provided in Table 6.3-1. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. The delay time due to valve motion in the case of the high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

The flow delivery rates analyzed in Subsection 6.3.3 can be determined from the head-flow curves in Figures 6.3-3, 6.3-4 and 6.3-5 and the pressure versus time plots discussed in Subsection 6.3.3.7. Simplified piping and instrumentation and process diagrams for the ECCS are referenced in Subsection 6.3.2. The operational sequence of ECCS for the DBA is shown in Table 6.3-2.

Operator action is not required, except as a monitoring function, during the short-term cooling period following the LOCA. During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for ECCS

See Appendix A, Subsection A.6.3.3.5 of Reference 4.

6.3.3.6 Limits on ECCS System Parameters

See Appendix A, Subsection A.6.3.3.6 of Reference 4.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

See Appendix A, Subsection A.6.3.3.7.1 of Reference 4.

6.3.3.7.2 Accident Description

See Appendix A, Subsection A.6.3.3.7.2 of Reference 4.

6.3.3.7.3 Break Spectrum Calculations

The analysis results presented in this section were obtained from a typical LOCA analysis which is representative of this plant size and product line. A plant specific LOCA analysis will be submitted as an FSAR amendment later in the SAR review cycle.

A complete spectrum of postulated break sizes and locations is considered in the evaluation of ECCS performance. For ease of reference, a summary of all figures and tables presented in Subsection 6.3.3 is shown in Table 6.3-6.

A summary of the results of the break spectrum calculations is shown in tabular form in Table 6.3-5 and graphically in Figure 6.3-8. Conformance to the acceptance criteria (PCT $^{\circ}$ = 2200°F, local oxidation = 17% and core-wide metal-water reaction = 1%) is demonstrated. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Recirculation Line Break Calculations

Important variables from the analyses of the DBA are shown in Figures 6.3-10 thru 6.3-19. These variables are:

(1) core average pressure as a function of time;

(2) core flow as a function of time;

6.3.3.7.4 Large Recirculation Line Break Calculations (Continued)

(3) core inlet enthalpy as a function of time;

(4) minimum critical power ratio as a function of time;

(5) water level as a function of time;

(6) pressure as a function of time from SAFE/REFLOOD;

(7) fuel rod convective heat transfer coefficient as a function of time;

(8) peak cladding temperature as a function of time from

(9) average fuel temperature as a function of time; and

(10) PCT rod internal pressure as a function of time.

The maximum average planar linear heat generation rate (MAPLHGR), maximum local oxidation and peak cladding temperature as a function of exposure from the analysis of the DBA are shown in Table 6.3-4.

Important variables in two other large break calculations (break size = 0.80 x DBA break size, 0.60 x DBA break size) are shown in Figures 6.3-20 through 6.3-35.

6.3.3.7.5 Transition Recirculation Line Break Calculations

Important variables from the analysis of the transition (1.0 ft^2) break are shown in Figures 6.3-36 through 6.3-47. These variables are:

(1) core average pressure (large break methods) as a function of time;

(2) core flow (large break methods) as a function of time;

(3) core inlet enthalpy (large break methods) as a function of time;

(4) minimum critical power ratio (large break methods) as a function of time;

(5) water level (large break methods) as a function of time;

(6) pressure (large break methods) as a function of time;

(7) fuel rod convective heat transfer coefficient (large break methods) as a function of time;

(8) peak cladding temperature (large break methods) as a function of time;

(9) water level (small break methods) as a function of time;

(10) pressure (small breaks methods) as a function of time;

6.3.3.7.5 Transition Recirculation Line Break Calculations (Continued)

(11) fuel rod convective heat transfer coefficients (small break methods) as a function of time; and

(12) peaking cladding temperature (small break methods) as a function of time.

6.3.3.7.6 Small Recirculation Line Break Calculations

Important variables from the analysis of the small break yielding the highest cladding temperature are shown in Figures 6.3-48 through 6.3-51. These variables are:

(1) water level as a function of time;

(2) pressure as a function of time;

(3) convective heat transfer coefficients as a function of time; and

(4) peak cladding temperature as a function of time.

The same variables resulting from the analysis of a less limiting small break are shown in Figures 6.4-52 through 6.3-55.

6.3.3.7.7 Calculations for Other Break Locations

Reactor water level and vessel pressure and peak cladding temperature and fuel rod convective heat transfer coefficients are shown in Figures 6.3-56 through 6.3-59 for the core spray line break, Figures 6.3-60 through 6.3-63 for the feedwater line break, and in Figures 6.3-64 and 6.3-65 for the main steamline break inside the containment.

6.3.3.7.7 Calculations for Other Break Locations (Continued)

An analysis was also done for the main steamline break outside the containment. Reactor water level and vessel pressure and peak cladding temperature and fuel rod convective heat transfer coefficients are shown in Figures 6.3-68 through 6.3-71.

6.3.3.7.8 Improved Decay Heat Correlation

Section I.A.4 of 10CFR50, Appendix K, requires use of the 1971 ANS Standards Subcommittee proposed decay heat standard for ECCS licensing evaluations. The current method for applying the 1971 standards in BWR LOCA calculations is outlined in GE's approved ECCS evaluation model (Reference 6.3-2). In 1979, the American National Standards Institute approved and the ANS published a much improved decay heat standard (Reference 6.3-3). A detailed technical basis for an improved GE BWR decay heat correlation based on the 1979 standard is outlined in Appendix 6A. Use of the improved correlation in the currently approved GE LOCA models will provide increased ECCS criteria margins.

Application of the correlation described in Appendix 6A is optional. To use it in place of the current method, a utility must provide the NRC with a request for exemption from Section I.A.4 of 10CFR50, Appendix K. The utility must reference Appendix 6A as the technical justification for the exemption.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Section 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10CFR50.46 acceptance criteria, given operation at or below the MAPLHGRs in Table 6.3-4.

6.3.4.2.4 LPCI Testing

Each LPCI loop can be tested during reactor operation. The test conditions are tabulated in Figures 6.3-4a, b and c. During plant operation, this test does not inject cold water into the reactor because the injection line check valve is held closed by vessel pressure, which is higher than the pump pressure. The injection line portion is tested with reactor water when the reactor is shut down and when a closed system loop is created. This prevents unnecessary thermal stresses.

To test an LPCI pump at rated flow, the test line valve to the suppression pool is opened, the pump suction valve from the suppression pool is opened (this valve is normally open) and the pumps are started using the remote/manual switches in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the LPCI System returns to the operating mode. The valves in the test bypass lines are closed automatically to assure that the LPCI pump discharge is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCS, LPCS, LPCI and ADS is discussed in Subsection 7.3.1 and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The HPCS, LPCS, LPCI and ADS can be manually initiated from the control room.

6.3.5 Instrumentation Requirements (Continued)

The HPCS, LPCS and LPCI are automatically initiated on low reactor water level or high drywell pressure. (See Table 6.3-8 for specific initiation levels for each system) The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one LPCI or LPCS pump is operating. The HPCS, LPCS and LPCI automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The LPCS and LPCI system injection into the RPV begin when reactor pressure decreases to system discharge shutoff pressure.

HPCS injection begins as soon as the HPCS pump is up to speed and the injection valve is open, since the HPCS is capable of injecting water into the RPV over a pressure range from 1177 psid* to 200 psid[3].

6.3.6 References

1. H.M. Hirsch, "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems", January 1973 (NEGO-10739).

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix", (NEDO-20566), submitted August 1974, and "General Electric Refill Reflood Calculation" (Supplement to Safe Code Description), transmitted to USNRC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

3. "Decay Heat Power in Light Water Reactors", ANSI/ANS 5.1-1979, Approved by American National Standards Institute, August 29, 1979.

4. "General Electric Standard Application for Reactor Fuel-United States Supplement", NEDE-20411-P-A-US (latest approved revision).

5. Letter MFN-255-77, Darrell G. Elsenhut (NRC) to E.D. Fuller (GE)", Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants", June 30, 1977.

[3]psid - differential pressure between RPV and pump suction source.

SECTION 9.1

CONTENTS

CONTENTS (Continued)

9.2

CONTENTS (Continued)

CONTENTS (Continued)

9.1.1.3.2 Structural Design (Continued)

(6) The nominal center-to-center spacing for the fuel assemblies or bundles between rows is 6.56 inches. The maximum spacing between racks is 2.0 inches.

(7) Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion.

(8) The racks are designed to withstand, while maintaining the nuclear safety design basis, the impact force generated by the vertical free-fall drop of a fuel assembly from a height of 6 ft.

(9) The rack is designed to withstand a pullup force of 4000 lb and a horizontal force of 1000 lb. There are no readily definable horizontal forces in excess of 1000 lb and, in the event a fuel assembly should jam, the maximum lifting force of the fuel-handling platform grapple (assumes limit switches fail) is 3000 lb.

(10) The new fuel storage racks require no periodic special testing or inspection for nuclear safety purposes.

9.1.1.3.3 Protection Features of the New Fuel Storage Facilities

The new fuel storage vault is housed in the Fuel Building (Subsection 3.8.4). The vault and Fuel Building are Seismic Category I structures. The Fuel Building provides protection from severe natural phenomena such as tornadoes, tornado missiles, floods and high winds. Fire protection features are described in Subsection 9.5.1 and Appendix 9A.

The storage rack structure is designed to withstand the impact resulting from a falling weight. Tests using a simulated fuel bundle of the correct weight and size have been conducted to

9.1.1.3.3 Protection Features of the New Fuel Storage Facilities (Continued)

verify that the rack casting can withstand the impact from a bundle dropped from a maximum allowable height above the array. Procedural fuel-handling requirements and equipment design dictate that no more than one bundle at a time can be handled over the storage racks and at a maximum height of 6 ft above the upper rack. Therefore, the racks cannot be displaced in a manner causing critical spacing as a result of impact from a falling object.

The five-ton general-purpose building crane can traverse the full length of the fuel building. A corridor is provided along the shield building side (not over) of the pools and vault; roof hatches are provided in the vicinity of the FPPCU equipment. This permits removal of major equipment by way of the hatch, thus eliminating the need to move these components along or over the pools and vault. The shipping cask cannot be lifted or moved above the new fuel vault because of inadequate clearance.

Should it become necessary to move major loads along or over the pools, administrative controls will require that the load be moved over the empty portion of the spent fuel pool and to avoid the area of the new fuel storage vault.

New fuel is carried to the new fuel vault and placed in the storage rack using the fuel-handling platform. During positioning of new fuel into the new fuel racks, the grapple is always above the upper fuel rack casting, and the grapple interfaces only with the fuel bundle bail and could not engage the fuel rack. Thus, the transfer devices used for new fuel handling to the new fuel vault cannot impose uplift loads on the rack castings.

9.1.2.1.1.2 Safety Design Bases - Nuclear (High Density) (Continued)

taken for neutron leakage, the values reported as effective neutron multiplication factors are, in reality, infinite neutron multiplication factors.

(c) The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculational procedure to assure that the specific k_{eff} limit is met.

9.1.2.1.2 Storage Design Bases

(1) The fuel storage racks provided in the spent fuel storage pool provide storage for 326% of one full core fuel load.

(2) The fuel storage racks provided in the containment pool provide storage for 68% of one full core fuel load.

(3) The spent fuel racks are designed and arranged so that fuel assemblies and bundles can be handled efficiently during refueling operations.

9.1.2.2 Facilities Description

(1) The spent fuel storage racks provide storage in the containment and spent fuel pools for spent fuel received from the reactor vessel during the refueling operation. The spent fuel storage racks are top entry racks designed to maintain the spent fuel while precluding the possibility of criticality under normal and abnormal conditions. The upper tieplate of the fuel element rests against the rack to provide lateral support. The lower tieplate sits

9.1.2.2 Facilities Description (Continued)

in the bottom of the rack, which supports the weight of the fuel.

(2) The rack arrangement is designed to prevent accidental insertion of fuel assemblies or bundles between adjacent modules. The storage rack is designed to provide accessibility to the fuel bail for grappling purposes. Nominal fuel spacing from center to center is 6.56 inches by 6.56 inches.

(3) The location of the spent fuel pool and the containment pool within the complex is shown in Section 1.2.

9.1.2.3 Safety Evaluation

9.1.2.3.1 Criticality Control

See Appendix A, Subsection A.9.1.2.3.1 of Reference 1, for GE fuel racks.

9.1.2.3.2 Structural Design and Material Compatibility Requirements

(1) The spent fuel pool contains 12 racks, four each of 13x13 racks and eight each of 13x17 racks, which provides storage for a maximum of 2444 fuel assemblies or bundles.

(2) The containment pool contains three 13x13 racks, which provides storages for a maximum 507 fuel assemblies or bundles.

(3) The fuel storage racks are designed to be supported above the pool floor by a support structure. The support structure allows sufficient pool water flow for natural convection cooling of the stored fuel. Since the modules are freestanding (i.e., no supports above the base), the support structure also provides the required dynamic stability.

(4) The racks include individual solid tube storage compartments, which provide lateral restraints over the entire length of the fuel assembly or bundle.

(5) The weight of the fuel assembly or bundle is supported axially by the rack fuel support.

9.1.2.3.2 Structural Design (Continued)

(6) The racks are fabricated from stainless steel. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications at the time of equipment order.

(7) The nominal center-to-center spacing for the fuel assemblies or bundles between rows is 6.56 inches. The maximum spacing between racks is 2.0 inches.

(8) Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion.

(9) The racks are designed to withstand, while maintaining the nuclear safety design basis, the impact force generated by the vertical free-fall drop of a fuel assembly from a height of 6 ft.

(10) The rack is designed to withstand a pullup force of 4000 lb and a horizontal force of 1000 lb. There are no readily definable horizontal forces in excess of 1000 lb and in the event a fuel assembly should jam, the maximum lifting force of the fuel-handling platform grapple (assumes limit switches fail) is 3000 lb.

(11) The fuel storage racks are designed to handle irradiated fuel assemblies. The expected radiation levels are well below the design levels.

The fuel storage facilities will be designed to Seismic Category I requirements to prevent earthquake damage to the stored fuel.

From the foregoing analyses, it is concluded that the spent fuel storage arrangement and design meet the safety design bases.

9.1.2.3.3 Protective Features of Spent Fuel Storage Facilities (Continued)

The FPCCU system described in Subsection 9.1.3 provides adequate and continuous cooling for the spent fuel.

From the foregoing analyses, it is concluded that the spent fuel storage arrangement and design meet the safety design bases and satisfy the intent of Regulatory Guide 1.13.

9.1.2.4 Testing Inspection

The spent fuel storage racks require no periodic special testing or inspection for nuclear safety purposes.

9.1.2.5 Summary of Radiological Considerations

By adequate design and careful operational procedures, the safety design bases of the spent fuel storage arrangement are satisfied. Thus, the exposure of plant personnel to radiation is maintained well below published guideline values. Further details of radiological considerations, including those for the spent fuel storage arrangement, are presented in Chapter 12.

9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

9.1.3.1.1 Safety Design Bases

The Fuel Pool Cooling and Cleanup (FPCCU) System shall be designed to remove the decay heat from the fuel assemblies, maintain pool water level and remove radioactive materials from the pool and thus minimize the release of radioactive elements stored in the containment upper pool and the pools in the fuel building.

9.1.3.1.2 Power Generation Design Bases

The FPCCU System shall:

(1) minimize corrosion product buildup and shall control water clarity, so that the fuel assemblies can be efficiently handled underwater;

(2) minimize fission product concentration in the water which could be released from the pool to the refueling building environment;

(3) monitor fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy;

(4) maintain the pool water temperature below 125°F under normal operating conditions. The temperature limits of 125°F is set to establish an acceptable environment for personnel working in the vicinity of the fuel pool. A maximum normal heat load from spent fuel stored in the pool is the sum of the decay heat released by the average spent fuel batch discharged from the 18-month equilibrium fuel cycle at the earliest refueling time, plus the heat being released by the batch discharged at the previous refueling. The heat sources are based on full power operation for four years prior to removal of fuel assemblies from the reactor and a batch size of 37% of the core. Also, the system shall remove the heat transferred from the drywell to the containment pool through the drywell head. The RHR system will be used to supplement the FPCCU System under the maximum load condition as defined in Subsection 9.1.3.3.

(5) maximize water purity for visual purposes.

9.1.3.2 System Description (Continued)

The filter-demineralizer vessel is constructed of phenolic resin-coated carbon steel. A post-strainer in the effluent stream of the filter-demineralizer limits the migration of filter material. The filter-holding element can withstand a differential pressure greater than the developed pump head for the system.

The filter-demineralizer units are located separately in shielded cells with enough clearance to permit removing filter elements from the vessels.

Each cell contains only the filter-demineralizer and piping. All valves (inlet, outlet, recycle, vent, drain, etc.) are located on the outside of one shielding wall of the room, together with necessary piping and headers, instrument elements and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The filter-demineralizers are controlled from a local panel. A differential pressure and conductivity instrument provided for each filter-demineralizer unit indicates when backwash is required. Suitable alarms, differential pressure indicators and flow indicators monitor the condition of the filter-demineralizers.

System instrumentation is provided for both automatic and remote-manual operations. A low-low level switch stops the circulating pumps when the fuel pool drain tank reserve capacity is reduced to the volume that can be pumped in approximately one minute with one pump at rated capacity (1100 gpm). Level switches are also located in both containment and fuel storage pools. Whenever the water levels are too high or too low, an alarm and indicator light are activated. Temperature elements are provided to

9.1.3.2 System Description (Continued)

record pool temperature in the main control room. (Spent Fuel Pool Cooling and Cleanup System Instrumentation and Controls are described in Section 7.6.)

The circulating pumps are controlled from the control room and a local panel. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm is indicated in the control room and on the local panel. The circulating pump motors are powered from their corresponding unit shutdown board. These boards receive power from the diesel-generators if normal power is not available. Circulating pump motor loads are considered nonessential loads and will be operated as required under accident conditions.

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for normal building occupancy. Radioactive particulates removed from the fuel pool are collected in filter-demineralizer units which are located in shielded cells. For these reasons, the exposure of plant personnel to radiation from the FPCCU System is minimal. Further details of radiological considerations for this system are described in Chapter 12.

The circulation patterns within the containment upper pool and spent fuel storage pool are established by placing the diffusers and skimmers so that particles dislodged during refueling operations are swept away from the work area and out of the pools.

The return lines to the pool are prevented from siphoning the pool in the event of a pipe rupture by redundant vacuum breakers at the high point of the lines.

9.1.3.2 System Description (Continued)

Heat from pool evaporation is handled by the building ventilation system. Makeup water is provided through a remote-operated valve.

Irradiated fuel shall not be stored in the upper containment storage pool during reactor operation.

9.1.3.3 Safety Evaluation

The maximum possible heat load is the decay heat of the full core load of fuel at the end of the fuel cycle plus the remaining decay heat of the spent fuel discharged at previous refuelings. The temperature of the fuel pool water may be permitted to rise to approximately 150°F under these conditions. During shutdown conditions, if it appears that the fuel pool temperature will exceed 125°F, the operator connects the FPCCU System to the RHR System. Combining the capacities enables the two systems to keep the water temperature below 125°F. The RHR System will be used only to supplement the fuel pool cooling when the reactor is shut down. The reactor will not be started up whenever portions of the RHR systems are needed to cool the fuel pool. The connecting piping from the fuel storage pool to the RHR system is designed Seismic Category I and is completely independent of the fuel pool system piping. These connections may also be utilized during emergency conditions to assure cooling of the spent fuel regardless of the availability of the fuel pool cooling system. The volume of water in the storage pool is such that there is enough heat absorption capability to allow sufficient time for switching over to the RHR system for emergency cooling.

The 150°F temperature limit is set to assure that the fuel building environment does not exceed equipment environmental limits.

9.1.3.3 Safety Evaluation (Continued)

The spent fuel storage pool is designed so that no single failure of structures or equipment will cause inability to: (1) maintain irradiated fuel submerged in water; (2) re-establish normal fuel pool water level; or (3) remove decay heat from the pool. In order to limit the possibility of pool leakage around pool penetrations, the pool is lined with stainless steel. In addition to providing a high degree of integrity, the lining is designed to withstand abuse that might occur when equipment is moved about. No inlets, outlets or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with syphon breakers, checkvalves, or other suitable devices to prevent inadvertent pool drainage. Interconnected drainage paths are provided behind the liner welds. These paths are designed to: (1) prevent pressure buildup behind the liner plate; (2) prevent the uncontrolled loss of contaminated pool water to other relatively cleaner locations within the containment or fuel-handling area; and (3) provide expedient liner leak detection and measurement. These drainage paths are formed by welding channels behind the liner weld joints and are designed to permit free gravity drainage or pumping to the equipment drain tank.

A makeup water system and pool water level instrumentation are provided to replace evaporative and leakage losses. Makeup water during normal operation will be supplied from condensate. Redundant loops of the essential service water system (which are both Seismic Category I) can be used as a source of makeup water in case of failure of the normal makeup water system. The cooling portion of the fuel pool system is designed to Seismic Category I up to and including the isolation valves for the filter demineralizer. Also, a Seismic Category I bypass is provided around the filter demineralizer. This will assure continued performance of the heat removal function if the filter-demineralizer portion is

9.1.4.1 Design Bases (Continued)

These platforms are Safety Class 2 and Seismic Category I from a structural standpoint in accordance with 10CFR50, Appendices A and B. Allowable stress due to safe shutdown earthquake (SSE) loading is 120% of yield or 70% of ultimate, whichever is least. A dynamic analysis is performed on the structures using the response spectrum method with load contributions resulting from each of three directions acting simultaneously being combined by the RMS procedure. Working loads of the platform structures are in accordance with the AISC Manual of Steel Construction. All parts of the hoist systems are designed to have a safety factor of five, based on the ultimate strength of the material. A redundant load path is incorporated in the fuel hoists so that no single component failure could result in a fuel bundle drop. Maximum deflection limitations are imposed on the main structures to maintain relative stiffness of the platform. Welding of the platforms is in accordance with AWS D14-1 or ASME Boiler and Pressure Vessel Code Section IX. Gears and bearings meet AGMA Gear Classification Manual and ANSI B3.5. Materials used in construction of load bearing members are to ASTM specifications. For personnel safety, OSHA Part 1910-179 is applied. Electrical equipment and controls meet ANSI CI, National Electric Code, and NEMA Publication No. IC1, MG1.

The auxiliary fuel grapple and the main telescoping fuel grapples have redundant lifting features and an indicator which confirms positive grapple engagement.

The fuel grapple is used for lifting and transporting fuel bundles. It is designed as a telescoping grapple that can extend to the proper work level and, in its fully retracted state, still maintain adequate water shielding over fuel.

In addition to redundant electrical interlocks to preclude the possibility of raising radioactive material out of the water, the

9.1.4.1 Design Bases (Continued)

cables on the auxiliary hoists incorporate an adjustable, removal stop that will jam the hoist cable against some part of the platform structure to prevent hoisting when the free end of the cable is at a preset distance below water level.

Provision of a separate cask loading pool, capable of being isolated from the fuel storage pool, will eliminate the potential accident of dropping the cask and rupturing the fuel storage pool. Furthermore, limitation of the travel of the crane handling the cask will preclude transporting the cask over any fuel storage pool. (See Chapter 15 for accident considerations.)

9.1.4.2 System Description

Table 9.1-1 is a listing of typical tools and servicing equipment supplied with the nuclear system. The following paragraphs describe the use of some of the major tools and servicing equipment and address safety aspects of the design where applicable.

9.1.4.2.1 Spent Fuel Cask

(Applicant to supply)

9.1.4.2.2 Overhead Bridge Cranes

9.1.4.2.2.1 Containment Polar Crane

(Applicant to supply) 9.2

9.1.4.2.2.2 Cask Crane

(Applicant to supply)

9.1.4.2.3.7 Jib Crane

The jib crane (Figure 9.1-9) consists of a motor-driven swing boom monorail and a motor-driven trolley with an electric hoist. The jib crane is mounted along the edge of the fuel building fuel storage pool to be used during refueling operations. Use of the jib crane leaves the refueling platform or fuel-handling platform free to perform general fuel shuffling operations and still permit uninterrupted fuel preparation in the work area. The hoist has two full-capacity brakes and in-series adjustable up-travel limit switches. Upon hoisting, the first of two independently adjustable limit switches automatically stop the hoist cable terminal approximately 8 ft below the jib crane base. Continued hoisting is possible by depressing a momentary contact (up-travel override pushbutton on the pendant) together with the normal hoisting push button. The second independently adjustable limit switch automatically interrupts hoist power at the maximum safe uptravel limit. When the jib crane is used in the handling of hazardous radioactive materials that must be kept below a specific water level, a fixed mechanical stop is installed on the hoist cable to prevent further hoisting when that level is reached. The jib crane is normally located adjacent to the fuel storage pool and connected to the service outlet provided.

9.1.4.2.3.8 Fuel-Handling Platform

Refer to Subsection 9.1.4.2.7 for a description of the fuel-handling platform.

9.1.4.2.3.9 Channel-Handling Boom

A channel-handling boom (Figure 9.1-10) with a spring-loaded balance reel is used to assist the operator in supporting a portion of the weight of the channel as it is removed from the fuel assembly. The boom is set between the fuel preparation machines. With the channel-handling tool attached to the reel, the channel may be conveniently moved between the fuel preparation machines.

9.1.4.2.3.10 Fuel Transfer System

The Inclined Fuel Transfer System (Figure 9.1-11) is used to transfer fuel, control rods, defective fuel storage containers and other small items between the containment and the fuel building pools by means of a carriage traveling in a transfer tube (a 23-in. I.D. stainless steel pipe). In the containment upper pool, the transfer tube connects to pool penetration and to a sheave box. Connected to the sheave box is a 24-in. flap valve, a vent pipe, cable enclosures and a fill valve. In the fuel building pool, the transfer tube connects to a 24-in. gate valve. A bellows connects the building penetration to the valve and transfer tube to prevent water entrapment between the tube and penetration. A 4-in. Weldolet located on the transfer tube approximately 2 ft above the fuel building pool water level and a motor-operated valve are provided for connections to a drain pipe for water level control in the transfer tube. A containment isolation assembly containing a blind flange and a bellows, which connects the containment isolation assembly to the containment penetration, is provided to make containment isolation. A hand-operated 24-in. gate valve is provided to isolate the reactor building pool water from the transfer tube so that the blind flange can be installed. A hydraulically actuated upender is provided in each pool for rotating part of the carriage (tilt tube) to the vertical position for loading and unloading and to the inclined position for transfer. The carriage consists of the tilt tube and a follower connected with a pivot pin which allows upending of the tilt tube while maintaining the follower in the inclined position. The carriage has rollers and wheels which ride on tracks within the transfer tube and upenders to assure low friction, correct carriage orientation and smooth transition across valves and between other components. The tilt tube is designed to accept two different inserts - a fuel bundle insert with a two-bundle capacity and a control rod insert for control rods, defective fuel storage container, and other small items.

9.1.4.2.4 Servicing Aids

General area underwater lights are provided with a suitable reflector for illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional illumination. Drop lights are used for illumination where needed.

A radiation-hardened designed portable underwater closed circuit television camera is provided. The camera may be lowered into the reactor vessel and/or fuel storage pool to assist in the inspection and/or maintenance of these areas. The camera lens is capable of pitching ninety degrees, which allows infinite scanning of three hundred and sixty degrees, solid angle.

A general-purpose, plastic viewing aid is provided to float on the water surface to provide better visibility. The sides of the viewing aid are brightly colored to allow the operator to observe it in the event of filling with water and sinking. A portable, submersible-type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for offsite burial. Fuel pool tool accessories are also provided to meet servicing requirements. A fuel sampler is provided. This is to be used to detect defective fuel assemblies during open vessel periods while the fuel is in the core. The fuel sampler head isolates individual fuel assemblies by sealing the top of the fuel channel and pumping water from the bottom of the fuel assembly, through the fuel channel, to a sampling station, and returning the water to the primary coolant system. After a "soaking" period, a water sample is obtained and is radiochemically analyzed.

9.1.4.2.5 Reactor Vessel Servicing Equipment

The essentiality and safety classifications, the quality group, and the seismic category for this equipment are listed in Table 9.1-3. Following is a description of the equipment designs in reference to that table.

9.1.4.2.5.1 Reactor Vessel Service Tools

These tools are used when the reactor is shut down and the reactor vessel head is being removed or reinstalled. Tools in this group are:

Stud Handling Tool

Stud Wrench

Nut Runner

Stud Thread Protector

Thread Protector Mandrel

Bushing Wrench

Seal Surface Protector

Stud Elongation Measuring Rod

Dial Indicator Elongation Measuring Device

Head Guide Cap

These tools are designed for a 40-yr life in the specified environment. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used. When carbon steel is used, it is either hard chrome plated, 9.1.4.2.5.1 Reactor Vessel Service Tools (Continued)

parkerized, or coated with an approved paint per Regulatory Guide 1.54.

9.1.4.2.5.2 Steamline Plug

The steamline plugs are used during reactor refueling or servicing; they are inserted in the steam outlet nozzles from inside of the reactor vessel to prevent a flow of water from the reactor well into the main steamline during servicing of safety relief valves, main isolation valves, or other components of the main steamlines, while the reactor water level is at the refueling level. The steamline plug design provides two seals of different types. Each one is independently capable of holding full head pressure. The equipment is constructed of noncorrosive materials. All calculated safety factors are 5 or better. The plug body is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.3 Shroud Head Stud Wrench

This is a hand-held tool for tightening and loosening the shroud head studs. It is designed for a 40-yr life and is made of aluminum for easy handling and to resist corrosion.

9.1.4.2.5.4 Head Holding Pedestal

Three pedestals are provided for mounting on the refueling floor for supporting the reactor vessel head and strongback/carousel during periods of reactor service. The pedestals have studs which engage three evenly spaced stud holes in the head flange. The flange surface rests on replaceable wear pads made of aluminum.

9.1.4.2.5.4 Head Holding Pedestal (Continued)

When resting on the pedestals, the head flange is approximately 3 ft above the floor to allow access to the seal surface for inspection and O-ring replacement.

The pedestal structure is a carbon steel weldment coated with an approved paint. It has a base with bolt holes for mounting it to the concrete floor.

A seismic analysis was made to determine the seismic forces imposed onto the pedestals, floor anchors, using the floor response spectrum method. The structure is designed to withstand these calculated forces and meet the requirements of AISC.

9.1.4.2.5.5 Head Stud Rack

The head stud rack is used for transporting and storage of eight reactor pressure vessel studs. It is suspended from the Reactor Building polar crane hook when lifting studs from the reactor well to the operating floor.

The rack is made of aluminum to resist corrosion and is designed for a safety factor of 5 with respect to the ultimate strength of the material.

The structure is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.6 Dryer and Separator Strongback

The Dryer and Separator Strongback is a lifting device used for transporting the steam dryer or the shroud head with the steam separators between the reactor vessel and the storage pools. The

9.1.4.2.5.7 Head Strongback/Carousel (Continued)

allowance and a safety factor of 5 in reference to the ultimate strength of the material used. After completion of welding and before painting, the lifting assembly is proof load tested and all load-affected welds and lift pins are magnetic-particle inspected.

The steel structure is designed in accordance with "The Manual of Steel Construction" by AISC. Aluminum structures are designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

The strongback is tested in accordance with American National Standard for overhead hoists ANSI B30.16-1973, Paragraph 16-1.2.2.2 and such that one hook pin and one main beam of the structure is capable of carrying the total load, and so that no single component failure will cause the load to drop or swing uncontrollably out of an essentially level attitude. The ASME Boiler and Pressure Vessel Code, Section IX, Welder Qualification is applied to all welder structures.

Regulatory Guide 1.54

General compliance or alternate assessment for Regulatory Guide 1.54, which provides design criteria for protective coatings, may be found in Subsection 6.1.2.

9.1.4.2.6 In-Vessel Servicing Equipment

The instrument handling tool is attached to the refueling platform auxiliary hoist and is used for removing and installing neutron source holders. Each in-core instrumentation guide tube is sealed by an O-ring on the flange, and, in the event that the seal needs replacing, an in-core guide tube sealing tool is provided. The

9.1.4.2.6 In-Vessel Servicing Equipment (Continued)

tool is inserted into an empty guide tube and sits on the beveled guide tube entry in the vessel. When the drain on the Water Seal Cap is opened, hydrostatic pressure seats the tool. The flange can then be removed for seal replacement.

The auxiliary hoist on the refueling platform is used with appropriate grapples to handle control rods, flux monitor dry tubes, sources and other internals of the reactor. Interlocks on both the grapple hoists and auxiliary hoist are provided for safety purposes; the refueling interlocks are described and evaluated in Section 7.6.

9.1.4.2.7 Refueling Equipment

Fuel movement and reactor servicing operations are performed from platforms which span the refueling, servicing and storage cavities. The containment is supplied with a refueling platform for fuel movement and servicing, an auxiliary platform for servicing operations from the refueling floor level and a vessel platform for reactor servicing from the vessel flange level. The fuel building is supplied with a fuel-handling platform for fuel movement and servicing.

9.1.4.2.7.1 Refueling Platform

The refueling platform is a gantry crane, which is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools on bedded tracks in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for core, storage rack or upender placement. Control of the platform is from an operator station on the main

9.1.4.2.8 Storage Equipment (Continued)

which are designed for the defective fuel. These may be used to isolate leaking or defective fuel while in the fuel pool and during shipping.

9.1.4.2.9 Under-Reactor Vessel Servicing Equipment

The primary functions of the under-reactor vessel servicing equipment are to: (1) remove and install control rod drives; (2) service thermal sleeves; and (3) install and remove the neutron detectors. Table 9.1-4 lists the equipment and tools required for servicing. Of the equipment listed, the equipment-handling platform and the CRD handling equipment are powered pneumatically.

The CRD handling equipment is designed for the removal and installation of the control rod drives from their housings. This equipment is used in conjunction with the equipment-handling platform.

The equipment-handling platform provides a working surface for equipment and personnel performing work in the under-vessel area. It is a polar platform capable of rotating 360°.

9.1.4.2.9 Under-Reactor Vessel Servicing Equipment (Continued)

The water seal cap is designed to prevent leakage of primary coolant from in-core detector housings during detector replacement. It is designed to industrial codes, manufactured from noncorrosive material.

The thermal sleeve installation tool locks, unlocks and lowers the thermal sleeve from the CRD guide tube.

The in-core flange seal test plug is used to determine the pressure integrity of the in-core flange O-ring seal. It is constructed of noncorrosive material.

The key bender is designed to install and remove the antirotation key that is used on the thermal sleeve.

9.1.4.2.10 Description of Fuel Transfer

The Fuel-Handling and Transfer System provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. Subsection 9.1.4.2.9 described the equipment and methods utilized in fuel handling. The following subsections describe the integrated fuel transfer system which ensures that the design bases of the fuel-handling system and the requirements of Regulatory Guide 1.13 are satisfied.

9.1.4.2.10.1 Arrival of Fuel on Site

The new fuel is delivered to the plant on flatbed truck or railcar. The new fuel is delivered to the receiving stations in the Fuel Building through the rail and truck entry door. There, the incoming new fuel is unloaded, inspected and prepared for use as described in Subsection 9.1.4.2.10.2.1.

9.1.4.2.10.2.3.1 Vessel Head Removal (Continued)

Next, the head, strongback and carousel are transported by the Reactor Building crane to the head holding pedestals on the refueling floor. The head holding pedestals keep the vessel head elevated to facilitate inspection and O-ring replacement.

The six studs in line with the fuel transfer canal are removed from the vessel and placed in the rack provided for them. The loaded rack is transported to the refueling floor for storage. Removal of these studs provides a path for fuel movement.

9.1.4.2.10.2.3.2 Dryer Removal

The reactor well is filled with water from the main condenser and the upper pool gates removed and stowed. The dryer-separator strongback is lowered by the reactor building crane and attached to the dryer lifting lugs. The dryer is lifted from the reactor vessel and transported underwater to its storage location in the dryer storage pool adjacent to the reactor well.

9.1.4.2.10.2.3.3 Separator Removal

In preparation for the separator removal, the steamline plugs are installed in the four main steam nozzles. The separator is then unbolted from the shroud using the four shroud head bolt wrenches furnished for this purpose. When the unbolting is accomplished, the dryer separator strongback is lowered into the vessel and attached to the separator lifting lugs. The separator is lifted from the reactor vessel and transported underwater to the storage location in the separator pool adjacent to the reactor well.

9.1.4.2.10.2.3.4 Fuel Bundle Sampling

During reactor operation the core offgas radiation level is monitored. If a rise in offgas activity has been noted, the

9.1.4.2.10.2.3.4 Fuel Bundle Sampling (Continued)

reactor core may be sampled during shutdown to locate any leaking fuel assemblies. The fuel sampler isolates up to a 16-bundle array in the core. This stops water circulation through the bundles and allows fission products to concentrate if a bundle is defective. After 10 minutes, a water sample is taken for fission product analysis. If a defective bundle is found, it is transferred to the Fuel Building storage pool and stored in a special defective fuel storage container to minimize background activity in the storage pool.

9.1.4.2.10.2.4 Refueling and Reactor Servicing

The gate isolating the containment pool from the reactor well is removed, thereby interconnecting the containment pool, the reactor well and the fuel transfer area. The refueling of the reactor can now begin.

9.1.4.2.10.2.4.1 Refueling

During an annual equilibrium outage, approximately 25% of the fuel is removed from the reactor vessel, 25% of the fuel is shuffled in the core (generally from peripheral to center locations) and 25% new fuel is installed. The actual fuel handling is done with the refueling platform. It is used as the principal means of transporting fuel assemblies between the reactor well and the containment pool; it also serves as a hoist and transport device. It provides an operator with work surface for almost all the other servicing operations. The platform travels on track extending along each side of the reactor well and pool and supports the refueling grapple and auxiliary hoists. The platform design permits travel over safety railings placed around the pools. The grapple is suspended from a trolley that can traverse the width of the platform. Platform movement is controlled from an operator station on the trolley. Railing should be provided to keep all

9.1.4.5.2 Fuel Support Grapple (Continued)

and indicator lights. This system provides the operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

9.1.4.5.3 Inclined Fuel Transfer Tube

The instrumentation sensors for this system provide the inputs to a programmable controller that automatically sequences the opening and closing of valves, the inclination and vertical upending of the fuel carriage, water levels, and the carriage traversing speeds.

The microprocessor control and proximity-type sensors also provide monitor and status conditions of the fuel transfer operation on each of the two operator's consoles, one located in the Fuel Building and the other on the RPV refueling floor. Monitor indicators and interlocks are provided in the reactor control room to indicate whenever personnel have accessed radiation hazardous areas along the transfer tube's route.

9.1.4.5.4 Other

Refer to Table 9.1-1 for additional refueling and servicing equipment not requiring instrumentation.

9.1.4.5.5 Radiation Monitoring

The radiation monitoring equipment for the refueling and servicing equipment is evaluated in Subsection 7.6.1.

9.1.5 References

1. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

11.3.2.18 Maintainability of Gaseous Radwaste System (Continued)

(4) shielding of nonradioactive auxiliary subsystems from the radioactive process stream.

Design features which reduce leakage and releases of radioactive material include the following:

(1) extremely stringent leak rate requirements placed upon all equipment, piping, and instruments and enforced by requiring as-installed helium leak tests of the entire process system;

(2) use of welded joints wherever practicable;

(3) specification of valve types with extremely low leak rate characteristics (i.e., bellows seal, double stem seal, or equal);

(4) routing of drains through steam traps to the main condenser; and

(5) specification of stringent seat-leak characteristics for valves and lines discharging to the environment via other systems.

11.3.2.19 Quality Classification and Construction and Testing Requirements

Equipment and piping will be designed and constructed in accordance with the requirements of the applicable codes in Table 11.3-6 and will comply with the welding and material requirements and the system construction and testing requirements as follows.

11.3.3.5.3 Charcoal Performance

The ability of the charcoal to delay the noble gases can be continuously evaluated by comparing activity measured and recorded by the process activity monitors at the exit of the cooler/condenser and at the exit of the charcoal adsorbers.

Experience with boiling water reactors has shown that the calibration of the offgas and vent effluent monitors changes with isotopic content. Isotopic content can change depending on the presence or absence of fuel cladding leaks in the reactor and the nature of the leaks. Because of this possible variation, the monitors are calibrated against grab samples periodically and whenever the radiation monitor after the offgas condenser shows significant variation in noble gas activity indicating a significant change in plant operations.

Grab sample points are located upstream and downstream of the first charcoal bed and downstream of the last charcoal bed and can be used for periodic sampling if the monitoring equipment indicates degradation of system delay performance.

11.3.3.5.4 Post Filter

On installation, replacement, and at periodic intervals during operation these particulate filters will be tested using a DOP (dioctylphthalate) aerosol to determine whether an installed filter meets the minimum in-place efficiency of 99.99% retention.

11.3.4 Radioactive Releases

11.3.4.1 Release Points

Airborne radioactive releases to the environs are from one or more monitored roof vent locations or points. The plant vent-release point is above the containment building dome which is the tallest

11.3.4.1 Release Points (Continued)

structure in the Nuclear Island. The plant vent releases ventilation air from the containment, fuel, and portions of the auxiliary buildings. The standby gas treatment system has its own vent with the release point by the containment building parapet. The radwaste building has two release points, one for ventilation air and the other for vapor from the detergent waste evaporator. The turbine building release point is above its roof and releases building ventilation air and discharges from the mechanical vacuum pump. Process offgas is also released via the turbine building vent.

The Applicant will provide all release points of gaseous waste to the environment on process flow diagrams, general arrangement drawings, or site plot plan. These release points will include height of release, inside dimensions of release point exit, effluent temperature, and effluent exit velocity. 11.5

11.3.4.2 Gland Seal Offgas System

Activity releases from the turbine gland sealing system are presented in Section 12.3. This system is provided with separate clean steam made from demineralized condensate. The effect of clean steam utilization is negligible activity releases from the turbine gland sealing system.

11.3.4.3 Mechanical Vacuum Pump

Activity releases from the mechanical vacuum pump are presented in Section 12.3. The mechanical vacuum pump will be operated for short periods of time during startups. Past BWR experience indicates that the mechanical vacuum pump is continuously utilized at a few plants during a refueling/maintenance outage but not others. The effluent from the mechanical vacuum pump is routed to the turbine building vent for discharge to the environs.

Table 14.1-1

PREOPERATIONAL TESTS

Subsection	Test Title	Page
14.2.12.1.1	Feedwater Control System	14.2-27
14.2.12.1.2	Reactor Feedwater System	*
14.2.12.1.3	Reactor Feedwater Pump Driver Control System	*
14.2.12.1.4	Reactor Water Cleanup System	14.2-28
14.2.12.1.5	Standby Liquid Control System	14.2-29
14.2.12.1.6	Nuclear Boiler System	14.2-30
14.2.12.1.7	Residual Heat Removal System	14.2-31
14.2.12.1.8	Reactor Core Isolation Cooling System	14.2-32
14.2.12.1.9	Reactor Recirculation System and Control	14.2-34
14.2.12.1.10	Rod Control and Information System	14.2-35
14.2.12.1.11	Control Rod Drive Hydraulic System	14.2-36
14.2.12.1.12	Fuel Handling and Vessel Servicing Equipment	14.2-37
14.2.12.1.13	Low Pressure Core Spray System	14.2-38
14.2.12.1.14	High Pressure Core Spray System	14.2-39
14.2.12.1.15	Fuel Pool Cooling and Cleanup System	14.2-41
14.2.12.1.16	Leak Detection System	14.2-42
14.2.12.1.17	Liquid and Solid Radwaste Systems	14.2-43
14.2.12.1.18	Reactor Protection System	14.2-47
14.2.12.1.19	Neutron Monitoring System	14.2-48

*Applicant will supply.

Table 14.1-1

PREOPERATIONAL TESTS (Continued)

*Applicant will supply.

Table 14.1-1

PREOPERATIONAL TESTS (Continued)

*Applicant will supply.

Table 14.1-1

PREOPERATIONAL TESTS (Continued)

Subsection	Test Title	Page
14.2.12.1.62	Main Turbine Electro-Hydraulic Control System	*
14.2.12.1.63	Condensate System	*
14.2.12.1.64	Condensate Polishing Demineralizer System	*
14.2.12.1.65	Condensate Storage System	*

*Applicant will supply.

Table 14.1-3

STARTUP TEST PROGRAM

STI No.	Test Title	Cold Test or Open RPV	Heatup	Test Conditions[1] 1	2	3	4	5	6	7	Warranty
1	Chemical and radiochemical	X	X	X		X		X	X		
2	Radiation measurements	X	X	X		X			X		
3	Fuel loading	X									
4	Full core shutdown margin	X									
5	Control rod drive system	X	X	X[2]					X		
6	SRM perf. and control rod seq.	X	X	X							
7	Water level measurements		X	X	X	X	X	X	X		
8	IRM performance	X	X	X							
9	LRPM calibration			X		X			X		
10	APRM calibration		X	X	X	X		X	X		X
11	NSSS process computer	X	X	X[3]					X		
12	RCIC System		X		X						
13	Selected process temperatures		X	X	X		X				
14	System expansion		X	X		X					
15	Core power distribution			X		X			X		
16	Core performance		X	X	X	X	X	X	X		X
17	Core power-void mode response				X		X		X		
18	Pressure regulator										
	Setpoint changes			X,BP	X	X,no BP	X	X	X,A		
	Backup regulator			X,BP	X	X,no BP	X	X,no BP	X,A		
19	Feedwater system										
	FW pump trip								M[14]		
	Water level setpoint change				X	X	X	X	X,A		
	Heater loss								X[12]		
20	Turbine valve surveillance				X[4]			X[5],SP	X[7,8],SP		
21	Main steam isolation valves		X	X[3],SP							
	one valve				X[4],SP			X[5,8],SP			
	full isolation								X[2,6,10,13],SP		
22	Relief valves										
	flow demonstration				X[4,13]						
	operational		X[13]		X[4]						
23	Turbine trip and generator							X[2],SD	X[2],SD*		
	load rejection				L,SP						
24	Shutdown from outside control room			X							
25	Recirculation Flow Control System		X	X	M[4],A[4]	M,A[14]		M[5],A[5]			
26	Recirculation System										
	trip one pump					X[13]			X[13]		
	trip two pumps					X[13]					
	system performance			X		X	X		X		
	non-cavit. verif.			X							
27	Loss of turbine/generator and offsite power				X[2,13],SP						

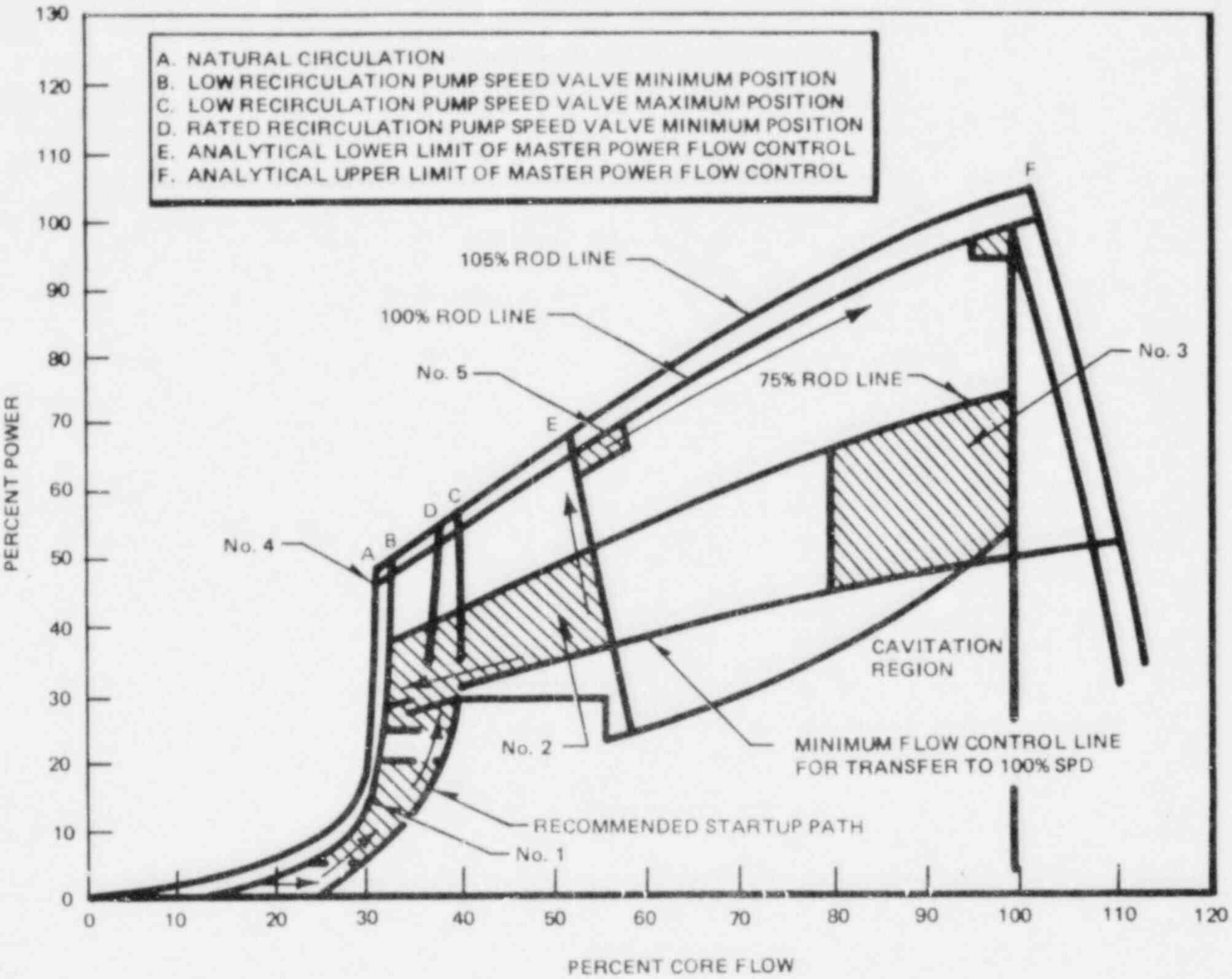
Table 14.1-3

START-UP TEST PROGRAM (continued)

STI No.	Test Title	Cold Test or Open RPV	Heatup	1	2	3	4	5	6	7	Warranty
				Test Conditions							
28	Drywell piping vibration		X	X		X			X		
29	RPV internals vibration		X	X	X	X	X	X	X		
30	Recirc. System flow calibration	X			X[4]	X		X,X[5]	X		
31	Reactor Water Cleanup System		X								
32	Residual Heat Removal System		X	X							
33	Drywell atmosphere cooling		X		X				X		
34	Cooling Water System		X						X		
35	Offgas System		X		X	X		X	X		
36	Suppression Pool Makeup System		X								
37	Inclined fuel transfer		X								

1. See Figure 14.1-1 for test conditions region map.
2. Perform Test 5, timing of 4 slowest control rods in conjunction with these scrams.
3. Between test conditions 1 and 3.
4. Between test conditions 2 and 3.
5. Between test conditions 5 and 6.
6. Before 100% turbine trip.
7. Future maximum power test point.
8. Determine maximum power without scram.
9. Perform at 100% core flow, 50% ± 2.5% power
10. Anywhere > 75% power.
11. 70 - 80% power.
12. 80 - 90% power.
13. Do STI 28 in conjunction with this test.
14. Demonstrate recirculation system runback feature.

L = Local Flow Control Mode
M = Master Manual Flow Control Mode
X = Local or Master Manual Flow Control Mode
A = Automatic Flow Control Mode
SP = Scram Possibility
SD = Scram Definite
BP = Bypass Valve Response
* = Do either Stop Valve or Control Valve Trip



TEST CONDITION (TC)	TEST CONDITION REGION DEFINITION
1	BEFORE MAIN GENERATOR SYNCHRONIZATION AND RECIRCULATION PUMPS OPERATING ON LOW FREQUENCY POWER SUPPLY
2	BETWEEN 50% AND 75% CONTROL ROD LINES, AT OR BELOW THE ANALYTICAL LOWER LIMIT OF MASTER FLOW CONTROL MODE
3	FROM 50% TO 75% CONTROL ROD LINES AND CORE FLOW BETWEEN 80% AND MAXIMUM ALLOWABLE
4	NATURAL CIRCULATION AND WITHIN 5% OF THE INTERSECTION WITH 100% ROD LINE
5	MID-POWER RANGE WITHIN 5% OF THE 100% CONTROL ROD LINE AND 0 TO +5% CORE FLOW OF THE MINIMUM FLOW LINE, FOR MASTER FLOW CONTROL IN MANUAL MODE, AND FOR AUTOMATIC POWER CONTROL IN AUTO MODE
6	WITHIN 0 TO -5% OF RATED THERMAL POWER, AND WITHIN 5% OF RATED CORE FLOW RATE
7	AT 105% OF RATED THERMAL POWER AND 100% CORE FLOW

Figure 14.1-1. Approximate Power Flow Map Showing Startup Test Conditions

CONTENTS (Continued)

CONTENTS (Continued)

14.2.7 Conformance of Test Programs to Regulatory Guides (Applicant will Confirm)

14.2.7.1 Conformance with Regulatory Guide 1.68

The test and startup program shall conform to the requirements of Regulatory Guide 1.68, Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors, except where specifically noted below. This regulatory guide will be reviewed by the Applicant for applicability of individual items in the guide to the specific facility and its systems. The applicability to this plant determines the nature and scope of testing to be performed. Actual exceptions to the testing required by this guide have been specifically addressed and are discussed in Subsection 14.2.7.2. Areas where the guide does not apply are not considered to be exceptions.

14.2.7.2 Exceptions to Regulatory Guide 1.68

The exceptions to Regulatory Guide 1.68 follow with an explanation of the justification for the exception:

(1) Section C.2.b: Operational limitations for the protection of public health and safety are included in the Technical Specification for the plant. Startup instructions contain notes of caution which supplement the Technical Specification. The Technical Specification should be the instrument for describing operational (including testing) limitations. Therefore, the identification of "safety precautions" in test procedures should be limited to those items which, if not observed, could lead to reduction of system safety performance below expected levels and not the minor procedural and test details which would not cause such a reduction.

14.2.7.2 Exceptions to Regulatory Guide 1.68 (Continued)

(2) Section C.2.c: The generic simulation tests appearing in safety analysis reports should appear by reference in preoperational and initial startup test programs where onsite full simulation tests are not possible. The guide wording would change to "...less than full simulation should be provided or referenced for test where full..."

(3) Appendix A, Section C.2.h: The comparison of critical control rod pattern with predicted patterns (Appendix A, Section C.2.d) provides required knowledge of effective over-all rod worth. Individual control rod calibrations cannot be performed in a meaningful manner in a large multi-rodded BWR. Therefore, this part of the guide is not applicable to boiling water reactors.

(4) Appendix A, Section C.2.i: The functional requirement of the reactor head cooling system design is required at operating pressures less than or equal to 135 psig. Therefore, for this paragraph to applicable, "($\leq$135 psig)" should be part of last sentence.

(5) Appendix A, Section D.2.a: A 50% power test is safer than the 25% test, since the water level and core power are less severe.

(6) Appendix A, Section D.2.b: Friction tests are performed on four drives at rated pressure.

(7) Appendix A, Section D.2.f: It is necessary to make more than two calibrations and, therefore, not appropriate to limit the test to 50% and 100% power levels.

14.2.7.3 Conformance with or Exception to Regulatory Guides Other Than RG 1.68

(1) Regulatory Guide 1.9, Selection of Diesel Generator Set Capacity for Standby Power Supplies

Refer to Subsection 14.2.12.1.55 for a description of the emergency diesel generator system preoperational test.

(2) Regulatory Guide 1.20: Vibration Measurements on Reactor Internals

Refer to Subsection 14.2.12.3.29 for a description of the vibration measurements preoperational test.

(3) Regulatory Guide 1.30: Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electric Equipment

Applicant will supply.

(4) Regulatory Guide 1.33: Quality Assurance Program Requirements (Operations)

Applicant will supply.

(5) Regulatory Guide 1.41: Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Load Group Assignments

Applicant will supply.

14.2.7.3 Conformance with or Exception to Regulatory Guides Other Than RG 1.68 (Continued)

(6) Regulatory Guide 1.52: Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Applicant will supply.

(7) Regulatory Guide 1.58: Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel

General Electric startup operations personnel qualifications meet the requirements of this guide as follows.

a. General Electric personnel are selected and trained according to the criteria of ANSI N18.1-1971 (NRC Regulatory Guide 1.8) with the exception of NRC Licensing.

b. The Operations Manager meets the equivalent of ANSI N18.1, Paragraph 4.2.2, Operations Manager. The Operations Manager is normally present for preoperational testing and will therefore be qualified at the time that preoperational testing is begun.

c. The Operations Superintendent meets the equivalent of ANSI N18.1, Paragraph 4.3.1, Supervisors Requiring AEC Licenses. The Operating Superintendent will normally be present for preoperational testing and therefore will be qualified at the time preoperational testing is begun.

14.2.12.1.17.2 Solid Radwaste System

(1) Purpose

(a) Demonstrate the reliable operation of the solid radwaste system and verify component interconnections.

(b) Verify that design flow rates through the system can be achieved.

(2) Prerequisites

The following general conditions should be considered.

(a) Construction tests are completed and approved.

(b) All instrumentation calibration sheets are completed and approved.

(c) Section 1 of the liquid radwaste handling system preoperational test has been satisfactorily completed.

(d) The following support systems and/or equipment should be available:

1. demineralized water;
2. instrument air;
3. electrical power;
4. service air; and
5. service water.

14.2.12.1.17.2 Solid Radwaste System (Continued)

The following safety precautions should be observed:

(a) Verify that all safety and construction tags have been removed.

(b) Verify visually that system components, piping, and pipe hangers do not suffer excessive vibration or movements.

(c) Monitor tank levels to ensure that no tanks overflow and that intended flowpaths are correctly lined up.

(d) Under no circumstances should actual radioactive materials be used for test.

(3) Test Methods and Acceptance Criteria

(a) Operations will be conducted to verify operation of the spent resin transfer and handling portions of the system.

(b) Operations will be conducted to verify operation of the sludge handling portions of the system.

(c) The solid radwaste processing and drumming will be demonstrated during Steps (a) and (b). Additionally, concentrated waste processing will be demonstrated.

(d) Operations will be conducted to verify operation of the crane(s).

14.2.12.1.19 Neutron Monitoring System Preoperational Test (Continued)

range monitors (IRM) and voltage preamplifiers, and average power range monitors (APRM) will have been calibrated per vendor instructions.

(3) General Test Methods and Acceptance Criteria

NMS capability is verified by the integrated operation of the following:

(a) all SRM detectors, their respective insert and retract mechanisms, and cables;

(b) SRM channel including pulse preamp, remote meter and recorder, trip logic, logic bypass and related lamps and annunciators, control system interlocks, refueling instrument trips, and power supply;

(c) all IRM detectors and their respective insert and retract mechanisms and cables;

(d) IRM channels including voltage preamps, remote recorders, RC&IS interlocks, RPS trips, annunciators and lamps, and power supplies;

(e) all local power range monitor (LPRM) detectors, and their respective cables, and power supplies;

(f) all APRM channels including trips, trip bypasses, annunciators and lamps, remote recorders, RC&IS interlocks, RPS trips, and power supplies; and

14.2.12.1.19 Neutron Monitoring System Preoperational Test (Continued)

(g) recirculation flow bias signal including flow unit, flow transmitters, and related annunciators, interlocks, and power supplies.

14.2.12.1.20 Traversing In-Core Probe System Preoperational Test

(1) Purpose

Verify the operation of the traversing in-core probe (TIP) system including the TIP detector, controls and interlocks, containment secure lamp, and containment isolation circuits.

(2) Prerequisites

The construction tests have been completed and the SCG has reviewed and approved the test procedure and the initiation of testing. Additionally, the TIP detector and dummy detector, ball valve time delay, core top and bottom limits, clutch, X-Y recorder, and purge system will have been shown to be operational.

(3) General Test Methods and Acceptance Criteria

With the exception of the shear valve, which is not tested, TIP system capability is verified by the integrated operation of the following:

(a) indexer cross-calibration interlock;

(b) shear valve control monitor lamp; and

14.2.12.1.29 Upper Pool Storage System Preoperational Test (Continued)

(2) Prerequisites

The construction tests have been completed and the SCG has reviewed and approved the test procedure and the initiation of testing.

(3) General Test Methods and Acceptance Criteria

Verification of the capability to transfer upper pool fluid to the lower pool at the flow desired.

14.2.12.1.30 Plant Process Sampling System (Radwaste) Preoperational Test

Applicant will supply.

14.2.12.1.31 Reactor Vessel Flow-Induced Vibration Preoperational Test

(1) Purpose

The reactor vessel flow-induced vibration test contains the engineering requirements for the preoperational vibration inspection and flow excitation of reactor internals. These requirements are intended to fulfill provisions of NRC Regulatory Guide 1.20 with respect to the vibration assessment of reactor intervals.

(2) Prerequisites

(a) Recirculation system preoperational testing shall be completed sufficiently to allow safe operation of the recirculation pumps at rated volumetric flow for extended operation.

14.2.12.1.31 Reactor Vessel Flow-Induced Vibration Preoperational Test (Continued)

(b) Capability to maintain the reactor water temperature equal to or greater than 150°F shall be provided throughout the duration of the flow test.

(c) Installation of the steam separator shall be accomplished, prior to RPV heatup, to ensure that a minimum of 50°F delta temperature is achieved following shroud head bolt torquing.

(d) Capability to maintain the reactor vessel pressure equal to or greater than 100 psi above the saturation pressure of the RPV water shall be provided throughout the duration of the flow test.

(e) Capability to bypass the recirculation system cavitation interlock protection shall be provided.

(f) Reactor assembly must include all core support structures and components, jet pumps, spargers, shroud head, and reactor vessel head.

(g) Reactor assembly shall not include any temporary hardware devices, such as blade guides.

(h) Reactor control rod blades must be removed or positioned fully withdrawn in their guide tubes with drive motion valved out of service and hydraulic accumulators vented to atmosphere.

(i) In-core instruments, neutron sources, and fuel shall not be installed throughout the duration of the test.

14.2.12.1.31 Reactor Vessel Flow-Induced Vibration Preoperational Test (Continued)

4. A minimum accumulation of 35 hours of rated volumetric core flow with two recirculation pumps of balanced flows is required.

5. A minimum accumulation of 14 hours of single-loop operation to each individual recirculation loop with the drive flow the same as that required for balanced loop flow at rated volumetric core flow is required.

14.2.12.1.32 Air Positive Seal (APS) System Preoperational Test

(1) Purpose

Verify the ability of the APS system to supply the design quantities of air at the design pressure for sealing and preventing release of fission products from primary containment to the environment.

(2) Prerequisites

The construction tests have been completed and the Startup Coordinating Group has reviewed and approved the test procedure, schedule, staffing, and plant condition. The following support systems must be available:

(a) electrical power;
(b) condensate;
(c) emergency service water (ESW); and
(d) normal and clean radioactive waste (CRW) drains.

14.2.12.1.32 Air Positive Seal (APS) System Preoperational Test (Continued)

(3) Test Methods and Acceptance Criteria

(a) System component checkout shall be made.

(b) Any temporary instruments needed for safe testing and adequate records shall be installed.

(c) Set ESW and condensate flow rates to post LOCA conditions. Operate compressors to show that cooling requirements are satisfied.

(d) Record time for each compressor to fill the receiver while the receiver leakage is 23 scfm.

(e) Conduct system tests by tripping the automatic controls. Test all safety interlocks.

(f) System flow rates and pressures shall meet design requirements. Interlocks and automatic features shall function according to design. All components shall be successfully tested.

14.2.12.1.33 Cask Decontamination System Preoperational Test

(1) Purpose

Verify the operation of the cask decontamination system in two modes: automatic spray or hand cleaning. Test interfaces with other systems.

14.2.12.1.33 Cask Decontamination System Preoperational Test (Continued)

(2) Prerequisites

The construction tests have been completed and the SCG has reviewed and approved the test procedure schedule, staffing, and plant condition. The following systems must be operational and available.

(a) condensate;
(b) radwaste;
(c) service air;
(d) floor drains;

(3) Test Methods and Acceptance Criteria

(a) Start the decontamination high-pressure pump, vent the discharge pipe, the relief valve will now open. When the decontamination holding tank reaches low level, the low-suction pressure alarm should sound. Shut down the pump.

(b) Refill the holding tank with condensate and open the spray valves and vent valve P48-FF032. Start the decontamination high-pressure pump and booster pump. Close vent valve P48-FF032 and test the automatic spray system.

(c) Test the hand-held lance. As a safety measure, have a standby operator present during this test.

(d) Repeat tests b and c using condensate directly from its supply header.

(e) Repeat tests b and c using demineralized water.

14.2.12.1.33 Cask Decontamination System Preoperational Test (Continued)

(f) Test the transfer pump by pumping water to the dirty radwaste system.

(g) The system is acceptable if all design flows and pressures are met, the automatic spray traverses the full length of track, and all instruments function according to design.

14.2.12.1.34 Reactor Island Chilled Water System Preoperational Test

(1) Purpose

Verify the capability of the Reactor Island nonessential chilled water system to supply design quantities of chilled water at design temperature to air conditioning cooling coils located in the Reactor Building, Auxiliary Building, Fuel Building, and Radwaste Building. The automatic features of the system which are essential shall be tested.

(2) Prerequisites

(a) The construction tests have been completed and the SCG has reviewed and approved the preoperational test procedures, schedule, staffing, and plant condition.

(b) The following systems shall be available.

1. demineralized water;
2. essential service water;
3. service air;
4. instrument air;

14.2.12.1.44 Nuclear Island Process Radiation Monitoring System Preoperational Test (Continued)

(e) Check each circuit by manual action to simulate a low radiation alarm. Observe the resulting action of the process status light and the out-of-service annunciator.

(f) For each alarm actuation in Paras (d) and (e), check the operation of the performance monitoring system.

(g) The system is acceptable when all actions conform to specifications.

14.2.12.1.45 Suppression Pool Makeup System (SPMS) Preoperational Test

(1) Purpose

Verify the ability of the SPMS to supply the design quantities of makeup water from the upper containment pool to the suppression pool after a LOCA.

(2) Prerequisites

(a) The construction tests have been completed and the SCG has reviewed and approved the test procedure, schedule, staffing, and plant condition.

(b) The following systems must be operational and available:

1. electrical power,
2. instrument air,
3. condensate, and
4. DRW drain.

14.2.12.1.45 Suppression Pool Makeup System (SPMS) Preoperational Test (Continued)

(3) Test Methods and Acceptance Criteria

(a) Any temporary instruments and equipment needed for safe and adequate testing are to be installed.

(b) System components are to be checked out as follows:

1. manually open and close all valves, then leave in operating mode;

2. check and calibrate instruments;

3. check automatic circuits;

4. with valve mode switch in OFF position and reactor mode switch in NOT IN REFUEL position, test each valve via its radiation monitoring system (RMS) for OPEN and CLOSE (position lights must correctly indicate the valve position); and

5. Repeat step 4 with reactor mode switch in REFUEL position (inoperative valves indicate proper interlock).

(c) Set the suppression pool water level low enough to prevent added water from flowing over the weirwall. Then simulate a LOCA signal (one division at a time) and dump the upper pool manually. The dump time should be close to, but not longer than, 8.8 minutes. Adjust the dump time, if necessary, by changing the size of the restriction orifice.

14.2.12.1.48 CO_2 Fire Protection System Preoperational Test (Continued)

(b) The following systems must be operational and available:

1. electrical power;
2. carbon dioxide storage;
3. Diesel Generator Rooms and Switchgear Room HVAC; and
4. Diesel generator starting air.

(3) Test Methods and Acceptance Criteria

(a) Actuate the automatic CO_2 flooding system by providing two heat sources at cross zones. The Control Room will note high temperature within Diesel Generator Room and CO_2 flow into affected space. Verify time delay of 30 seconds between alarm and CO_2 flow.

(b) With the CO_2 supply block valve closed, activate the manual CO_2 discharge mode by the local keylock while the starting air compressors are in operation. The compressors will be deenergized and cannot be restarted with the trip signal present.

(c) Verify isolation of the Diesel Generator Room. Check that all H & V fans have stopped, and all louvers and dampers close on CO_2 system actuation.

(d) Check alarm circuits for visual and audible alarms, sound levels and time delay.

(e) Determine the CO_2 flow rates and concentration.

(f) The system is acceptable when all valves, instruments, circuits and alarms function according to design specifications.

14.2.12.1.49 Suppression Pool Cleanup Preoperational Test

(1) Purpose

Verify the system integrity of the Suppression Pool Cleanup (SPCU) System. The demineralizer's effectiveness in removing radioactive ions is not tested as this capability lies beyond the scope of a preoperational test.

(2) Prerequisites

(a) The construction tests have been completed and the SCG has reviewed and approved the test procedure, schedule, staffing, and plant condition.

(b) The following systems must be operational and available:

1. normal and DRW drains;
2. electrical power;
3. service and instrument air; and
4. spent resin storage tank.

(3) Test Methods and Acceptance Criteria

(a) Install any temporary instruments and equipment needed for safe and adequate testing.

(b) System component checkout shall be made including calibration of instruments.

(c) Maintain the suppression pool at the normal water level.

14.2.12.3.5.3 Description (Continued)

binding caused by thermal expansion of the core components. A list of all control rod drive tests to be performed during startup testing follows.

CONTROL ROD DRIVE SYSTEM TESTS

Test Description Indication	Accumulator Pressure	Preop Test	Reactor Pressure with Core Loaded psig (kg/cm^2) 0	600 (42.2)	800 (56.2)	Rated
Normal times insert/ withdraw		all	all			4*
Coupling		all	all**			
Friction			all			4*
Scram	Normal	all	all	4*	4*	all
Scram	Minimum		4*			
Scram	Zero					4*
Scram (scram discharge volume high level)	Normal		4 (full core scram)			
Scram		normal				4***

NOTE

Single CRD scrams should be performed with the charging valve closed (do not ride the charging pump head).

*Value refers to the four slowest CRDs as determined from the normal accumulator pressure scram test at ambient reactor pressure. Throughout the procedure, the four slowest CRDs imply the four slowest compatible with rod worth minimizer and CRD sequence requirements.

**Establish initially that this check is normal operating procedure.

***Scram times of the four slowest CRDs will be determined at 25%, 60%, and 100% of rated power during planned reactor scrams.

14.2.12.3.5.4 Criteria

Level 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 inches per second (9.14 cm/sec) indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

The mean scram time of all operable CRDs must not exceed the following times. (Scram time is measured from the time the pilot scram valve solenoids are denergized.)

Percent Inserted	Rod Position	Scram Time (seconds) Vessel Dome Pressure $\geq$950 psig (66.9 kg/cm^2)	Scram Time (seconds) Vessel Dome Pressure <950 psig (66.9 kg/cm^2)
4.5	46	0.358	0.454
25.4	36	1.096	1.260
46.2	26	1.860	1.885
87.9	06	3.419	4.838

The mean scram time of the three fastest CRDs in a two-by-two array must not exceed the following times. (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

Percent Inserted	Rod Position	Scram Time (seconds) Vessel Dome Pressure $\geq$950 psig (66.9 kb/cm^2)	Scram Time (seconds) Vessel Dome Pressure <950 psig (66.9 kb/cm^2)
4.5	46	0.379	0.482
25.4	36	1.169	1.335
46.2	26	1.971	1.998
87.9	06	3.624	5.128

14.2.12.3.5.4 Criteria (Continued)

Level 2

Each CRD must have a normal insert or withdraw speed of 3.0 ± 0.6 inches per second (7.62 ±1.52 cm/sec) indicated by a full 12-foot stroke in 40 to 60 seconds.

With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid (1 kg/cm^2) for a continuous drive in, a settling test must be performed; in which case, the differential settling pressure should not be less than 30 psid (2.1 kg/cm^2) nor should it vary by more than 10 psid (0.7 kg/cm^2) over a full stroke.

14.2.12.3.6 Startup Test Number 6 - SRM Performance and Control Rod Sequence

14.2.12.3.6.1 Purpose

Demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

14.2.12.3.6.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedure and the initiation of testing. The control rod drive system must be operational.

14.2.12.3.6.3 Description

The operational neutron sources will be installed and source range monitor count-rate data will be taken during rod withdrawals to

14.2.12.3.6.3 Description (Continued)

critical and compared with stated criteria on signal and signal count-to-noise count ratio.

A withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated-power configuration. Critical rod patterns will be recorded periodically as the reactor is heated to rated temperature.

Movement of rods in a prescribed sequence is monitored by the rod worth minimizer which will prevent out of sequence withdrawal. Also not more than two rods may be inserted out of sequence.

As the withdrawal of each rod group is completed during the power ascension, the electrical power, steam flow, control valve position, and APRM response will be recorded.

14.2.12.3.6.4 Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRMs or fuel loading chambers.

There must be a minimum count rate of 3 counts/second on the required operable SRMs or fuel loading chambers.

The IRMs must be on scale before the SRMs exceed the rod block set point.

14.2.12.3.7 Startup Test Number 7 - Water Level Measurement

14.2.12.3.7.1 Purpose

(1) Check the calibration of the various level indicators,

(2) Measure the reference leg temperature and recalibrate the affected wide-range level instruments if the measured temperature is different than the value assumed during the initial calibration

(3) Collect plant data which can be used to investigate the effects of core flow, carryunder, and subcooling on indicated wide-range level systems.

14.2.12.3.7.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedure and the initiation of testing. All system instrumentation is installed and calibrated. All system controls and interlocks have been checked.

14.2.12.3.7.3 Description

To monitor the reactor vessel water level, four level instrument systems are provided. These are:

(1) shutdown range level system;
(2) narrow range level system;
(3) wide range level system; and
(4) fuel zone level system.

These systems are used respectively as follows:

(1) shutdown range level system - water level measurement in cold, shutdown conditions;

14.2.12.3.7.3 Description (Continued)

(2) narrow range level system - feedwater flow and water level control functions;

(3) wide range level system - safety functions; and

(4) full zone level system - safety functions.

The test is divided into three parts. The first part will be done at rated temperature and pressure and under steady-state conditions and will verify that the reference leg temperature of the wide range level instrument is the value assumed during initial calibration. If not, the instruments will be recalibrated using the measured value. The second part of the test consists of reading all of the level indicators to verify that they are working properly. The Level 2 criteria will determine whether recalibration is necessary. There should be reasonable agreement between indications at hot standby. The third part of the test will collect data at various operating conditions to help define the effect of core flow velocity, subcooling, and carryunder on indicated wide range level.

14.2.12.3.7.4 Criteria

Level 2

The narrow range level system readings should agree with each other within ±1.5 inches of the average reading.

The wide range level system indicators should agree with each other within ±6 inches of the average reading.

14.2.12.3.12.3 Description (Continued)

vessel. The CST injections consist of controlled and quick starts at reactor pressures ranging from 150 psig (10.5 kg/cm^2) to rated with corresponding pump discharge pressures throttled between 250 psig (17.5 kg/cm^2) and 100 psi above rated pressure. During this part of the testing, proper operation of the system will be verified and adjustments made as required to meet this criteria. The reactor vessel injection will consist of a cold quick start of the system with all flow routed to the reactor vessel at $\geq$25% power.

14.2.12.3.12.4 Criteria

Level 1

The time from actuating signal to required flow must be less than 30 seconds at any reactor pressure between 150 psig (10.5 kg/cm^2) and rated.

With pump discharge at any pressure between 150 psig (10.5 kg/cm^2) and 100 psi above rated pressure, 5% of rated feedwater flow is required. (The 100 psi is a conservatively high valve for line losses. The measured valve may be used if available.)]

The RCIC turbine must not trip off during startup.

If any Level 1 criteria are not met, the reactor will only be allowed to operate up to a restricted power level defined by Figure 14.2-1 oi the startup test instructions.

Level 2

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

14.2.12.3.12.4 Criteria (Continued)

The differential pressure switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at 300% of the maximum required steady state flow.

For small speed or flow changes in either manual or automatic mode, the decay ratio of each recorded RCIC system variable must be less than 0.25 in order to demonstrate acceptable stability.

The margins to avoid the overspeed trip shall be at least 10% of the trip value.

14.2.12.3.13 Startup Test Number 13 - Selected Process Temperatures

14.2.12.3.13.1 Purpose

Verify that the setting of the low-speed recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region provides assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

14.2.12.3.13.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedures and initiation of testing. System and test instrumentation have been calibrated.

14.2.12.3.13.3 Description

During initial heatup while at hot standby conditions, the bottom drain line temperature and applicable reactor parameters are monitored. The recirculation pumps shall be operated at low frequency m-g set speed and the flow control valves set at minimum

14.2.12.3.15.4 Criteria (Continued)

NOTE

A minimum of two and up to six data sets may be used to meet the above criteria. If the 7.8% total TIP uncertainty criteria cannot be met by the six sets of data, testing may continue provided the MCPR limit is adapted to reflect the TIP uncertainty.

Additional data sets may be obtained in order to improve the TIP uncertainty by increasing the TIP data base and the MCPR limit adjusted, accordingly. If the 7.8% total TIP uncertainty becomes satisfied, the MCPR limit can be returned to its original value.

Level 2

In the TIP reproducibility test, the TIP traces shall be reproducible in the nonboiling region within ±3.5% relative error, or ±0.15 inch (3.8 mm), the absolute error at each axial position, whichever is greater.

14.2.12.3.16 Startup Test Number 16 - Core Performance

14.2.12.3.16.1 Purpose

(1) Evaluate the core thermal power
(2) Evaluate the following core performance parameters:

  (a) maximum linear heat generation rate (MLHGR);
  (b) minimum critical power ratio (MCPR); and
  (c) maximum average planar linear heat generation rate (MAPLHGR).

14.2.12.3.16.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedures and initiation of testing. System instrumentation has been installed and calibrated and test instrumentation calibrated.

14.2.12.3.16.3 Description

The core performance evaluation is employed to determine the principal thermal and hydraulic parameters associated with core behavior. These parameters are:

(1) core flow rate;
(2) core thermal power level;
(3) MLHGR; and
(4) MAPLHGR.

These core performance parameters will be evaluated by manual calculation techniques described in Startup Test Instruction 19 or may be obtained from the process computer.

If the process computer is used as a primary means to obtain these parameters, it must be proven that it agrees with BUCLE within 2% on all thermal parameters (see Startup Test Number 11), or the results must be corrected to do so. If the BUCLE and process computer results do not agree within 2% for any thermal parameter, the parameter calculated by the process computer will be corrected by a multiplication factor to bring it within the 2% criteria.

14.2.12.3.16.4 Criteria

Level 1

The maximum linear heat generation rate (MLHGR) of any rod during steady-state conditions shall not exceed the limit specified by the Plant Technical Specifications.

14.2.12.3.18.1 Purpose (Continued)

(2) Demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the setpoints at an appropriate value.

(3) Demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

14.2.12.3.18.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

14.2.12.3.18.3 Description

The pressure setpoint will be decreased rapidly and then increased rapidly by about 10 psi (0.7 kg/cm^2) and the response of the system will be measured in each case. It is desirable to accomplish the setpoint change in less than 1 second. At specified test conditions the load limit setpoint will be set so that the transient is handled by control valves, bypass valves, and both. The backup regulator will be tested by simulating a failure of the operating pressure regulator so that the backup regulator takes over control. The response of the system will be measured and evaluated and regulator settings will be optimized. At certain conditions, the results of the backup regulator test will be included with the test report on the core power-void mode test. Because the near step transient occurs downstream of the log filter, this disturbance yields valuable stability data in the midfrequency range (i.e., 0-13.0 Hz).

14.2.12.3.18.4 Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to pressure regulator changes.

Level 2

In all tests the decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to pressure regulator changes when the plant is operating above the lower limit setting of the master flow controller and $\leq$0.50 when below core flow.

Pressure control deadband, delay, etc., shall be small enough that steady-state limit cycles, if any, shall produce turbine steam flow variations no larger than ±0.5% of rated flow as measured by the gross generated electrical power.

Optimum gain values for the pressure control loop shall be determined to give the fastest return from the transient condition to the steady-state condition within the limits of the above criteria.

During the simulated failure of the controlling pressure regulator, if the setpoint of the backup pressure regulator is optimumly set, the backup regulator shall control the transient that the peak neutron flux and/or peak vessel pressure so remain below the scram settings by 7.5% and 10 psi respectively.

Following a ±10 psi (0.7 kg/cm^2) pressure setpoint adjustment, the time between the setpoint change and the occurrence of the pressure peak shall be $\leq$ 10 seconds in recirculation manual mode.

14.2.12.3.21.3 Description

At 5% power or greater, both slow and fast single valve closure will be performed. A test of the simultaneous full closure of all MSIVs will be performed at about 100% of rated thermal power. Correct performance of the RCIC and relief valves will be shown. Reactor process variables will be monitored to determine the transient behavior of the system during and following the main steamline isolation. The maximum power conditions at which individual valve full closure tests can be performed without a reactor scram are to be established, and one individual valve full-closure test will be performed on the 100% power load line to check ability to perform surveillance tests on this load line.

The MSIV closure times will be determined from the MSL isolation data by multiplying 1.1 times the time increment between closure initiation and activation of the 90% closed light.

14.2.12.3.21.4 Criteria

Level 1

(1) MSIV closure time, exclusive of electrical delay shall be no faster than 3.0 seconds (average of the fastest valve in each steamline) and no slower than 5.0 seconds including electrical delay (each valve, not averaged).

(2) The positive change in vessel dome pressure occurring within the first 30 seconds after a closure of all MSIV valves must not exceed the Level 2 criteria by more than 25 psi. The positive change is simulated heat flux and must not exceed the Level 2 criteria by more than 2% of the rated value.

14.2.12.3.21.4 Criteria (Continued)

Level 2

(1) During full closure of individual valves, peak vessel pressure must be 10 psi (0.7 kg/cm^2) below scram, peak neutron flux must be 7.5% below scram, and steam flow in individual lines must be 10% below the isolation trip setting.

(2) The RCIC system shall adequately take over water level protection. The relief valves must reclose properly (without leakage) following the pressure transient.

(3) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of the full isolation from full power shall not exceed the predicted values.

(Predicted values shall be referenced to actual test conditions of initial power level and dome pressure and will use beginning-of-life (BOL) nuclear data. Worst-case design or technical specification values of all hardware performance shall be used in the prediction with the exception of control rod insertion time and the delay from beginning of turbine control valve or stop valve motion to the generation of the scram signal. The predicted pressure and heat flux will be corrected for the actual measured values of these two parameters.)

14.2.12.3.23.3 Description

The turbine stop valves will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, and reactor protection system (RPS). Additionally, the peak values and change rates of reactor steam pressure and heat flux will be determined. The effect of recirculation pump overspeed, if any, will be checked during the generator load rejection. The ability to ride through a load rejection within bypass capacity without a scram will be demonstrated.

14.2.12.3.23.4 Criteria

Level 1

(1) Bypass valve (BPV) quick opening should begin by 0.1 second after start of stop valve closure, and bypass flow should be $\geq$ 80% of BPV capacity within another 0.2 second (i.e., within 0.3 second of start of stop valve closure).

(2) Feedwater system settings must prevent flooding of the steamline following threse transients.

(3) The two pump-drive flow-coastdown transients during the first three seconds must be equal to, or faster than, that specified in Startup Test Number 26.

14.2.12.3.23.4 Criteria (Continued)

(4) The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi. The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

(5) Pressure and heat flux must be within 25 psi and 2% of prediction.

Level 2

(1) There shall be no MSIV closure during the first three minutes of the transient, and operator action shall not be required during that period to avoid the MSIV trip.

(2) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

(Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use beginning-of-life (BOL) nuclear data. Worst-case design or technical specification values of all hardware performance shall be used in the prediction with the exception of control rod insertion time and the delay from beginning of turbine control valve or stop valve motion to the generation of the scram signal. The predicted pressure and heat flux will be corrected for the actual measured values of these two parameters.)

14.2.12.3.27 Startup Test Number 27 - Loss of Turbine Generator and Offsite Power

14.2.12.3.27.1 Purpose

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

14.2.12.3.27.2 Prerequisites

The preoperational tests have been completed, the SCG has reviewed and approved the test procedures and initiation of testing. All controls and interlocks are checked and instrumentation calibrated.

14.2.12.3.27.3 Description

The turbine stop valves will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, and Reactor Protection System (RPS). Additionally, the peak values and change rates of reactor steam pressure and heat flux will be determined. The effect of recirculation pump overspeed, if any, will be checked during the generator load rejection. The ability to ride through a load rejection within bypass capacity without a scram will be demonstrated.

14.2.12.3.27.4 Criteria

Level 1

(1) Bypass valve quick opening should begin by 0.1 second after start of stop valve closure, and bypass flow should be $\geq$80% of BPV capacity within another 0.2 second (i.e., within 0.3 second of start of stop valve closure).

14.2.12.3.27.4 Criteria (Continued)

(2) Feedwater system settings must prevent flooding of the steamline following these transients.

(3) The two-pump drive-flow coastdown transient during the first three seconds must be equal to or faster than that specified in Startup Test Number 26.

(4) The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi. The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

(5) Pressure and heat flux must be within 25 psi and 2% of prediction.

Level 2

(1) There shall be no MSIV closure during the first three minutes of the transient and operation action shall not be required during that period to avoid the MSIV trip.

(2) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

SECTION 15.0

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SECTION 15.0

15.0 GENERAL

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events.

General Electric has developed a unique systematic approach to plant safety consistent with the General Electric boiling water reactor technology base. The key to the General Electric approach to plant safety is the Nuclear Safety Operational Analysis. A generic nuclear safety operational analysis has been developed for each of the recent General Electric boiling water reactor product lines. It has then been modified to be compatible with the specific plant configuration being evaluated. Key inputs into the nuclear safety operational analysis are derived from the applicable regulations and through industry codes and standards. The generic nuclear safety operational analysis for BWR/6 plants is given in Appendix 15A.

General Electric evaluates the entire spectrum of events in the nuclear safety operational analysis in order to establish the most limiting or design basis events in a meaningful manner. It is the design basis events that are quantified in this chapter.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), off-design abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and, finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system).

15.0.1 Nuclear Safety Operational Analysis

See Appendix A, Subsection A.15.0.1 of Reference 1.

15.0.2 Event Analytical Objective

See Appendix A, Subsection A.15.0.2 of Reference 1.

15.0.3 Analytical Categories

See Appendix A, Subsection A.15.0.3 of Reference 1.

15.0.4 Event Evaluation

15.0.4.1 Identification of Causes and Frequency Classification

See Appendix A, Subsection A.15.0.4.1 of Reference 1.

15.0.4.2 Identified Unacceptable Results

15.0.4.2.1 Unacceptable Results for Incidents of Moderate Frequency (Anticipated (Expected) Operational Transients)

See Appendix A, Subsection A.15.0.4.2.1 of Reference 1.

15.0.4.2.2 Unacceptable Results for Infrequent Incidents (Abnormal (Unexpected) Operational Transients)

See Appendix A, Subsection A.15.0.4.2.2 of Reference 1.

15.0.4.2.3 Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents)

See Appendix A, Subsection A.15.0.4.2.3 of Reference 1.

15.0.4.3 Sequence of Events and Systems Operations

See Appendix A, Subsection A.15.0.4.3, of Reference 1.

15.0.4.4 Analysis Basis

See Appendix A, Subsection A.15.0.4.4, of Reference 1.

15.0.4.4.1 Evaluation Models

See Appendix A, Subsection A.15.0.4.4.1, of Reference 1.

15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed within this section have values for input parameters and initial conditions as specified in Table 15.0-1. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

The normal maximum allowable reactor operating condition is the 100%/100% power/flow condition. The maximum P/F measurement

15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events (Continued)

uncertainty is usually $\sim$2%. However, the Chapter 15 analyses are based on 105% steam flow condition, which corresponds to $\sim$104.2% power level. The transient results at this condition are more severe than that at rated condition. Thus, the Chapter 15 analyses bounds all the operating condition.

The dynamic parameters assumed in Chapter 15 are much more conservative than on bounding the normal operating values. For example, the Doppler coefficient is expected to be varied from -0.14¢/°F to -0.24¢/°F, while the analysis used -0.13¢/°F. The void coefficient normally varied from -5¢/% to -10¢/% during the cycle, while conservative values of -14¢/% and -4¢/% were used in the analysis. The scram reactivity in Figure 15.0-2 presents a conservative lower bound on the minimum scram reactivity and also defines the minimum scram characteristic for permissible operation. 15

The analytical values for some system characteristics, like SRV delay/stroke time, recirculation pump trip coastdown time constant, etc., bound the design specification for that system. These values will be checked during the startup test.

All setpoints for the protection system assumed in the analyses are conservative, which includes instrument uncertainty, calibration error and instrument drift. The nominal and allowable values for these setpoints, shown in the technical specifications assure that the setpoints will not exceed what are assumed in the analyses.

In conclusion, the values used in Chapter 15 analyses are conservative values, which include uncertainties, and bound the operating band. Therefore Chapter 15 analysis will cover the whole operating conditions and cycle points.

15.0.4.4.3 Initial Power/Flow Operating Constraints

The analyses basis for most of the transient safety analyses is the thermal power at rated core flow (100%) corresponding to 105% nuclear boiler rated (NBR) steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (104.2% rod line A-D), the lower bound is the zero power line H-J, the right bound is the rated flow line A-H, and the left bound is the natural circulation line D-J.

The power/flow map (A-D-J-H-A) represents the acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. See Subsection 4.4.3.3 for power/flow map operating instructions.

The upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the GETAB operating limit.

Certain localized events are evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event.

15.0.4.4.3 Initial Power/Flow Operating Constraints (Continued)

Reactor operation up to the APRM rod block line, which is above the power levels corresponding to the design flow control line except at low drive flows, is assumed for ECCS analysis.

General Compliance or Alternate Approach Assessment for Regulatory Guide 1.49

For commitment and revision number, see Section 1.8.

Regulatory Guide 1.49 requires that the proposed licensed power level be restricted to a reactor core power level of 3800 MWt thermal or less, and that analyses and evaluations in support of the application should be made at 1.02 times the proposed licensed power level.

The rated thermal power for the standard 238 size reactor is 3579 MWt. The safety analyses and evaluations were made for a 104.2% power level of 3729 MWt. Both of these are in compliance with the subject Guide Requirements.

15.0.4.5 Evaluation of Results

The results of the standard plant equilibrium core analyses are provided for each event. The results of the transient analyses are given in Table 15.0-2. Based on these transient results, the limiting events have been identified. Reasons why the other events are not limiting are given in the event documentation. The limiting events are as follows:

1. Limiting Pressurization Events: Pressure Controller Downscale Failure, Generator Load Rejection without Bypass, and Turbine Trip Without Bypass,

15.0.4.5 Evaluation of Results (Continued)

2. Limiting Decrease in Core Coolant Temperature Event: Loss of Feedwater Heating (manual control), and

3. Limiting Temperature Decrease/Pressurization Event Feedwater Controller Failure (Maximum Demand).

The Load Rejection and Turbine Trip without Bypass Events are categorized as infrequent events but are included in this list as they are not limiting events. Results reported in Table 15.0-2 for pressurization events were calculated using ODYN Option A. The resulting initial core MCPR operating limit is 1.18.

Results of the transient analyses for individual plant reference core loading patterns will differ from the standard plant results. However, the relative results between events will not change. Therefore, only the results of the identified limiting events need to be provided by the Applicant. These results will be provided in the format given in Tables 15.0-4 and 15.0-5.

15.0.4.5.1 Effect of Single Failures and Operator Errors

The effect of a single equipment failure or malfunction or operator error is provided in Appendix 15A.

15.0.4.5.2 Analysis Uncertainties

See Appendix A, Subsection A.15.0.4.5.2 of Reference 1.

15.0.4.5.3 Barrier Performance

See Appendix A, Subsection A.15.0.4.5.3 of Reference 1.

15.0.4.5.4 Radiological Consequences

See Appendix A, Subsection A.15.0.4.5.4 of Reference 1.

15.0.5 References

1. "General Electric Standard Application for Reactor Fuel - United States Supplement,"(NFDE-24011-P-A, latest approved revision).

Table 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

1.	Thermal Power Level (MWt)	
	Warranted Value	3579
	Analysis Value	3729
2.	Steam Flow (lb/hr)	
	Warranted Value	15.40×10^6
	Analysis Value	16.71×10^6
3.	Core Flow (lb/hr)	104.0×10^6
4.	Feedwater Flow Rate (lb/sec)	
	Warranted Value	4269
	Analysis Value	4483
5.	Feedwater Temperature (°F)	425
6.	Vessel Dome Pressure (psig)	1045
7.	Vessel Core Pressure (psig)	1056
8.	Turbine Bypass Capacity (% NBR)	35
9.	Core Coolant Inlet Enthalpy (But/lb)	528.9
10.	Turbine Inlet Pressure (psig)	960
11.	Fuel Lattice	P8 x 8R
12.	Core Average Gap Conductance (Btu/sec-ft2-°F)	0.1892
13.	Core Leakage Flow (%)	12.9
14.	Required MCPR Operating Limit	
	First Core	1.18
	Reload Core	1.19
15.	MCPR Safety Limit	
	First Core	1.06
	Reload Core	1.07
16.	Doppler Coefficient (-)¢/°F	
	Analysis Data (REDY only)*	0.132

*For transients simulated on the ODYN computer model, this input is calculated by ODYN.

Table 15.0-1 (Continued)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

17.	Void Coefficient (-)¢/% Rated Voids	
	Analysis Data for Power Increase Events (REDY only)*	14.0
	Analysis Data for Power Decrease Events (REDY only)*	4.0
18.	Core Average Rated Void Fraction (%) (REDY only)*	42.54
19.	Scram Reactivity, $\$\Delta\kappa$ Analysis Data (REDY only)*	Subsection S.2.2, Reference 1
20.	Control Rod Drive Position versus time	Subsection S.2.2, Reference 1
21.	Nuclear characteristics used in ODYN simulations	EOEC**
22.	Jet Pump Ratio (M)	2.257
23.	Safety/Relief Valve Capacity (% NBR) at 1210 psig	110.8
	Manufacturer	***
	Quantity Installed	19
24.	Relief Function Delay (sec)	0.4
25.	Relief Function Response Time Constant (sec)	0.1
26.	Safety Function Delay (sec)	0.0
27.	Safety Function Response Time Constant (sec)	0.2
28.	Set Points for Safety/Relief Valves	
	Safety Function (psig)	1175,1185,1195,1205,1215
	Relief Function (psig)	1125,1135,1145,1155
29.	Number of Valve Groupings Simulated	
	Safety Function (No.)	5
	Relief Function (No.)	4

*For transients simulated on the ODYN model, this input is calculated by ODYN.
**EOEC = End of Equilibrium Cycle.
***Applicant to Supply

Table 15.0-1 (Continued)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

30.	S/R Valve Reclosure Setpoint - Both Modes (% of setpoint)	
	- Maximum Safety Limit (used in analysis)	98
	- Minimum Operational Limit	89
31.	High Flux Trip (% NBR)	
	Analysis setpoint (122 x 1.042)	127.2
32.	High Pressure Scram Setpoint (psig)	1095
33.	Vessel Level Trips (ft above bottom of separator skirt bottom)	
	Level 8 - (L8)(ft)	5.89
	Level 4 - (L4)(ft)	4.04
	Level 3 - (L3)(ft)	2.165
	Level 2 - (L2)(ft)	(-)1.739
34.	APRM Simulated Thermal Power Trip Scram % NBR	
	Analysis Setpoint (114 x 1.042)	118.8
	Time Constant (sec)	7
35.	Recirculation Pump Trip Delay (sec)	0.14
36.	Recirculation Pump Trip Inertia Time Constant for Analysis (sec)***	5
37.	Total Steamline Volume (ft^3)	3850
38.	Set pressure of Recirculation pump trip (psig) (Nominal)	1135

*For transients simulated on the ODYN model, this input is calculated by ODYN.
**EOEC = End of Equilibrium Cycle.
***The inertia time constant is defined by the expression:

$$t = \frac{2 \pi J_o n}{g T_o}$$

, where t = inertia time constant (sec);

J_o = pump motor inertia (lb-ft);

n = rated pump speed (rps);

g = gravitational constant (ft/sec^2); and

T_o = pump shaft torque (lb-ft).

Table 15.0-2

RESULTS SUMMARY OF APPLICABLE TRANSIENT EVENTS

Sub-section I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR -	Frequency Category*	Duration of Blowdown: No. of Valves First Blow-down	Duration of Blowdown: Duration of Blow-down (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of Feedwater Heater, Auto Flow Control	111.5	1045	1087	1034	105.8	**	a	0	0
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	124.2	1060	1102	1047	113.7	0.12	a	0	0
15.1.2	15.1-3	Feedwater Cntl Failure, Max Demand	124.3	1163	1193	1159	105	0.10	a	19	5
15.1.3	1.51-4	Pressure Controller Fail - Open	104.2	1138	1161	1136	100	**	a	10	5
15.1.4		Inadvertent Opening of Safety or Relief Valve			See Text						
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temp			See Text						
15.2		INCREASE IN REACTOR PRESSURE			See Text						
15.2.1	15.2-1	Pressure Controller Downscale Failure	156.8	1187	1221	1181	102.6	0.09	a	19	7
15.2.2	15.2-2	Generator Load Rejection, Bypass-On,	128.2	1160	1189	1157	100	**	a	19	5
15.2.2	15.2-3	Generator Load Rejection, Bypass-Off,	198.7	1203	1233	1202	102.7	0.08	b	19	7
15.2.3	15.2-4	Turbine Trip, Bypass-On	114.5	1158	1188	1155	100	**	a	19	5
15.2.3	15.2-5	Turbine Trip, Bypass-Off	179.4	1202	1231	1201	101.3	0.05	b	19	7
15.2.4	15.2-6	Inadvertent MSIV Closure	105.3	1177	1207	1174	100	**	a	19	5
15.2.5	15.2-7	Loss of Condenser Vacuum	113.7	1157	1186	1153	100	**	a	19	5
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104.2	1100	1112	1098	100	**	a	1	5
15.2.6	15.2-9	Loss of All Grid Connections	105.3	1159	1184	1156	100	**	a	19	7
15.2.7	15.2-10	Loss of All Feedwater Flow	104.2	1045	1086	1034	100	**	a	0	0
15.2.8		Feedwater Piping Break See Table 15.0-2, event 15.6.6									
15.2.9		Failure of RHR Shutdown Cooling				See Text					

*Frequency definition is discussed in Subsection 15.0.4.1.
**See Subsection 15.0.4.5.
[a]Moderate frequency
[b]Infrequent

Table 15.0-2 (Continued)

RESULTS SUMMARY OF APPLICABLE TRANSIENT EVENTS (Continued)

Subsection I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial	ΔCPR -	Frequency Category*	Duration of Blowdown: No. of Valves First Blow-down	Duration of Blowdown: Duration of Blow-down (sec)
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104.3	1046	1087	1035	100	**	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104.2	1141	1155	1139	100	**	a	10	5
15.3.2	15.3-3	Fast Closure of One Main Recirc. Valve	104.2	1135	1149	1133	100	**	a	10	5
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves	104.2	1142	1153	1139	100	**	a	10	5
15.3.3	15.3-5	Seizure of One Recirculation Pump	104.2	1139	1153	1137	100	**	c	10	5
15.3.4		Recirc. Pump Shaft Break	See Subsection 15.3.3								
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling				See Text			b		
15.4.1.2		RWE - Startup				See Text			b		
15.4.2		RWE - At Power				See Text			a		
15.4.3		Control Rod Misoperation	See Subsections 15.4.1 and 15.4.2								
15.4.4	15.4-1	Abnormal Startup of Idle Recirculation Loop	109.1	988	1002	983	148.7	***	a	0	0
15.4.5	15.4-2	Fast Opening of One Main Recirc. Valve	235.3	978	997	974	135	***	a	0	0
15.4.5	15.4-3	Fast Opening of Both Main Recirc. Valves	162.2	974	990	971	123.4	***	a	0	0
15.4.7		Misplaced Bundle Accident				See Text		0.10	b		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104.2	1045	1087	1034	100	**	a	0	0
15.5.1		BWR Transients	See appropriate Events in Sections 15.1 and 15.2								

*Frequency definition is discussed in Subsection 15.0.3.1.
**See Subsection 15.0.3.3.1.
***Transients initiated from low power.
[a]Moderate frequency
[b]Infrequent
[c]Unexpected

Table 15.0-3

SUMMARY OF ACCIDENTS

Subsection I.D.	Title	Failed Fuel Rods GE Calculated Value	Failed Fuel Rods NRC Worst-Case Assumption
15.3.3	Seizure of One Recirculation Pump	None	
15.3.4	Recirculation Pump Shaft Break	None	
15.4.9	Rod Drop Accident	<770	770
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100%
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Gas Treatment System Failure	N/A	N/A
15.7.3	Liquid Radwaste Tank Failure	N/A	N/A
15.7.4	Fuel-Handling Accident	<125	125
15.7.5	Cask Drop Accident	None	None
15.8	ATWS	SPECIAL EVENT STILL UNDER NEGOTIATION	

Table 15.0-4

CORE-WIDE TRANSIENT ANALYSIS RESULTS

Transient	Flux (% NBR)	Q/A (% NBR)	Delta CPR (Nominal ODYN)	Figure
Pressure Controller Failure	*	*	*	*
Load Rjection w/o Bypass	*	*	*	*
Turbine Trip w/o Bypass	*	*	*	*
Loss of Feedwater Heating	*	*	*	*
Feedwater Controller Failure	*	*	*	*

Table 15.0-5

INITIAL CORE MCPR VALUES

Non-Pressurization Events	MCPR	
Loss of Feedwater Heating	*	
Fuel Loading Error	*	
Rod Withdrawal Error	*	
Pressurization Events	**Option A MCPR**	**Option B MCPR**
Pressure Controller Failure	*	*
LR or TT w/o Bypass	*	*
Feedwater Controller Failure	*	*

*To be provided by Applicant

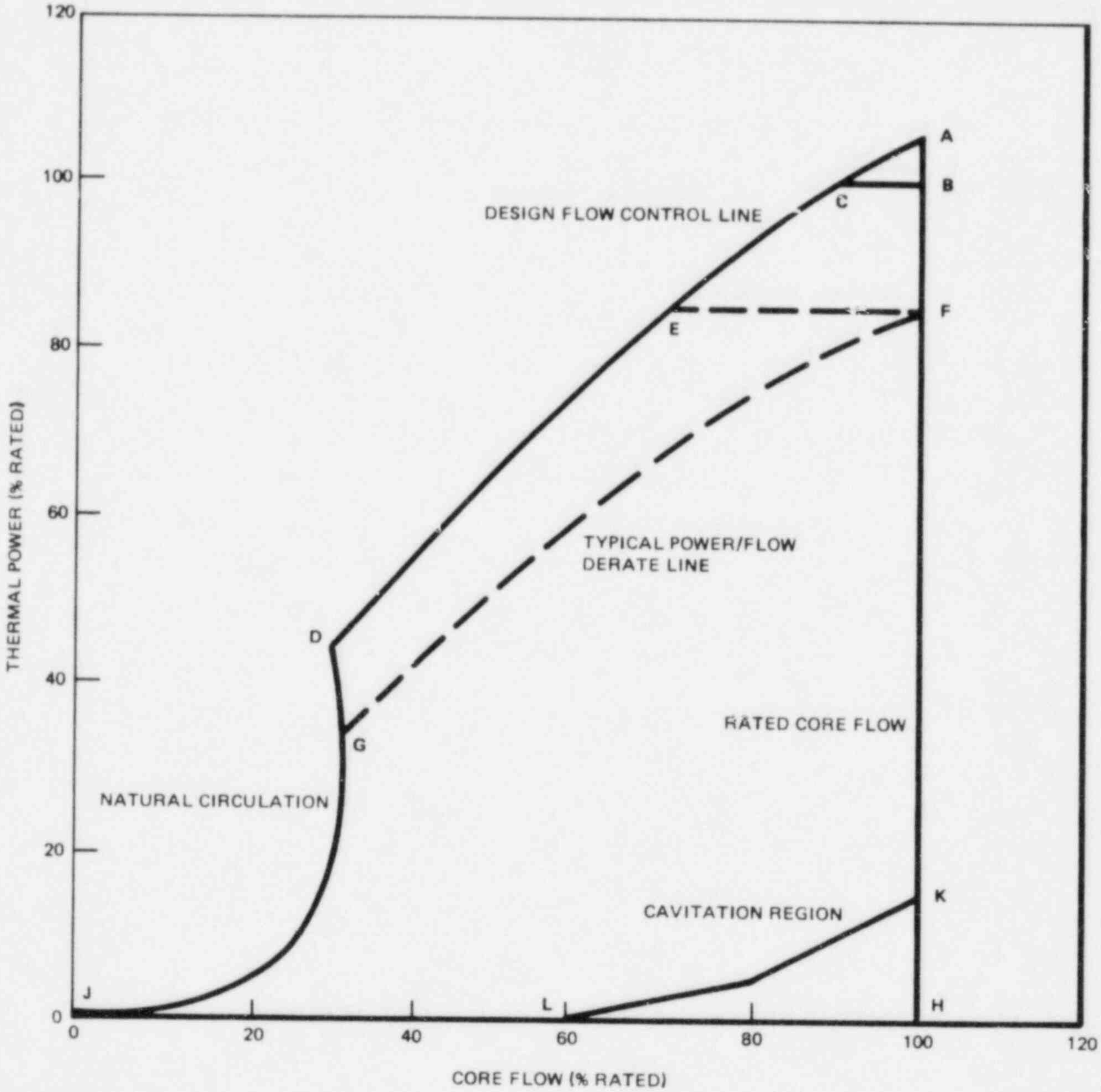

Figure 15.0-1. Typical Power/Flow Map

SECTION 15.1

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CONTENTS (Continued)

CONTENTS (Continued)

15.1.1.3 Core and System Performance

15.1.1.3.1 Input Parameters and Initial Conditions

The transient is simulated by programming a change in feedwater enthalpy corresponding to a 100°F loss in feedwater heating.

15.1.1.3.2 Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. In order to maintain the initial steam flow with the reduced inlet temperature, reactor thermal power increases above the initial value and settles at about 110% NBR (106% of initial power), below the flow-referenced APRM simulated thermal power scram setting and core flow is reduced to approximately 80% of rated flow. The MCPR reached in the automatic control mode is greater than for the more limiting manual flow control mode.

The increased core inlet subcooling aids thermal margins, and smaller power increase makes this event less severe than the manual flow control case given below. Nuclear system pressure does not change and, consequently, the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case. This transient is illustrated in Figure 15.1-1.

In the manual mode, no compensation is provided by core flow, and thus the power increase simulated is greater than in the automatic mode. A scram on high APRM simulated thermal power may occur. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 114% of its initial value and average fuel temperature increases 128°F. The increased core inlet subcooling aids core thermal margins and minimum MCPR remains above the safety limit. Therefore, the design basis is

15.1.1.3.2 Results (Continued)

satisfied. The transient responses of the key plant variables for this mode of operation are shown in Figure 15.1-2.

After the reactor scrams, water level drops to the low level trip point (L3) for recirculation pump trip (not shown in Table 15.1-2).

This transient is less severe from lower initial power levels for two main reasons: (1) lower initial power levels will have CPR values greater than the limiting initial CPR value assumed, and (2) the magnitude of the power rise decreases with lower initial power conditions. Therefore, transients from lower power levels will be less severe.

The Applicant will provide reanalysis of this event for the specific core configuration.

15.1.1.4 Barrier Performance

As noted above and shown in Figures 15.1-1 and 15.1-2, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Since this event does not result in any additional fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.1.2 Feedwater Controller Failure - Maximum Demand

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. This event is a combination coolant temperature decreased pressurization transient. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

15.1.2.1.2 Frequency Classification

This event is considered to be an incident of moderate frequency.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

With excess feedwater flow, the water level rises to the high-level reference point, at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 15.1-3 lists the sequence of events for Figure 15.1-3. The figure shows the changes in important variables during this transient.

15.1.2.2.1.1 Identification of Operator Actions

The operator should:

(1) observe that high feedwater pump trip has terminated the failure event;

15.1.2.2.1.1 Identification of Operator Actions (Continued)

(2) switch the feedwater controller from auto to manual control in order to try to regain a correct output signal; and

(3) identify causes of the failure and report all key plant parameters during the event.

15.1.2.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high level scram and tripping of the main turbine and feedwater pumps, recirculation pump trip (RPT), and low water level initiation of the reactor core isolation cooling (RCIC) system and the high pressure core spray (HPCS) system to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.3 Core and System Performance

The simulated feedwater controller transient is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 sec. Scram occurs simultaneously and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above the safety limit. The turbine bypass system opens to limit peak pressure in the steamline near the S/R valves to 1159 psig and the pressure at the bottom of the vessel to about 1193 psig.

The level will gradually drop to the Low Level reference point (Level 2), activating the RCIC/HPCS systems for long-term level control.

15.1.2.3 Core and System Performance (Continued)

The applicant will provide reanalysis of this event for the specific core configuration.

15.1.2.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.3 Pressure Regulator Failure - Open

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow demand to 130% NB rated in the analysis.

15.1.3.1.1 Identification of Causes (Continued)

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine admission valves can be fully opened and the turbine bypass valves can be partially opened until the maximum steam flow demand is satisfied.

15.1.3.1.2 Frequency Classification

This transient distribance is categorized as an incident of moderate frequency.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

Table 15.1-4 lists the sequence of events for Figure 15.1-4.

15.1.3.2.1.1 Identification of Operator Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer operation to the backup controller in time to prevent the full transient. If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (825 psig) in the run mode, the following sequence of operator actions is expected during the course of the event. Once isolation occurs, the pressure will increase to a point where the relief valves open. The operator should:

(1) monitor that all rods are in;

(2) monitor reactor water level and pressure;

(3) observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries;

15.1.3.2.1.1 Identification of Operator Actions (Continued)

(4) observe that the reactor pressure relief valves open at their setpoint;

(5) observe that RCIC and HPCS initiate on low-water level;

(6) secure both HPCS and RCIC when reactor pressure and level are under control;

(7) monitor reactor water level and continue cooldown per the normal procedure; and

(8) complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.1.3.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems, except as otherwise noted.

Initiation of HPCS and RCIC system functions will occur when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized.

If these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.3 Core and System Performance

15.1.3.3.1 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator output to a high value, which causes the turbine admission valves to open fully and the turbine bypass valves to open partially. Since the controlling and backup regulator outputs are gated by a high value gate, the effect of such a failure in the backup regulator would be exactly the same. A regulator failure with 130% steam flow was simulated as a worst case, since 115% is the normal maximum flow limit. A reactor scram and trip of the main and feedwater turbines occur on high water level.

A 5-sec isolation valve closure instead of a 3-sec closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

15.1.3.3.2 Results

Figure 15.1-4 shows graphically how the high water level turbine trip and the isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. In this simulation, the depressurization rate is large enough such that water level swells to the sensed level trip setpoint (L8), initiating reactor scram and main turbine and feedwater turbine trips. Position switches on the turbine stop valves initiate recirculation pump trip (RPT). After the turbine trip, the failed pressure regulator now signals the bypass to open to full bypass flow

15.1.3.3.2 Results (Continued)

of 35% NBR steam flow. After the pressurization resulting from the turbine stop valve closure, pressure again drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint when main steamline isolation finally terminates the depressurization. The turbine trip and isolation limit the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. No significant reduction in fuel thermal margins occur; therefore, this event does not have to be analyzed for specific core configurations.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required since the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. Peak pressure in the bottom of the vessel reaches 1161 psig, which is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure reaches 1138 psig, just slightly above the setpoint of the second pressure relief group. Minimum vessel dome pressure of 790 psig occurs at about 30 sec.

15.1.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.4 Inadvertent Safety/Relief Valve Opening

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. Opening and closing circuitry at the individual valve level (as opposed to groups of valves) is subject to a single failure. It is therefore simply postulated that a failure occurs and the event is analysed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident but, due to a lack of a comprehensive data basis, it is being analyzed as an incident of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-5 lists the sequence of events for this event.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must reclose the valve as soon as possible and check that reactor and T-G output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.3 Core and System Performance

The opening of a S/R valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected and this event does not have to be reanalyzed for specific core configurations.

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequences of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is

15.1.4.5 Radiological Consequences (Continued)

contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside of Containment in a PWR

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions, no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-6.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation since the nuclear system pressure is too high to permit operation of the shutdown cooling (RHRs).

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the ope ator in the same manner normally used to control power in the source or intermediate power ranges.

15.1.6.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise would be terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here and this event does not have to be analyzed for specific core configurations.

15.1.6.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Since this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

SECTION 15.2

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CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

15.2.1.2.1.3.2 Pressure Regulation Downscale Failure

The operator should:

(1) monitor that all rods are in;

(2) monitor reactor water level and pressure;

(3) observe turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries);

(4) observe that the reactor pressure relief valves open at their setpoint;

(5) monitor reactor water level and continue cooldown per the normal procedure; and

(6) complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.2.1.2.2 Systems Operation

15.2.1.2.2.1 One Pressure Regulator Failure - Closed

Normal plant instrumentation and control are assumed to function. This event requires no protection system or safeguard systems operation.

15.2.1.2.2.2 Pressure Regulation Downscale Failure

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems. Specifically, this transient takes credit for high

15.2.1.2.2.2 Pressure Regulation Downscale Failure (Continued)

neutron flux scram to shut down the reactor. High system pressure is limited by the pressure relief valve system operation.

15.2.1.3 Core And System Performance

15.2.1.3.1 One Pressure Regulator Failure - Closed

Qualitative evaluation provided only.

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, in less than approximately 2 sec, due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip setpoints; therefore, this event does not have to be analyzed for specific core configurations.

15.2.1.3.2 Pressure Regulation Downscale Failure

A pressure regulation downscale failure is simulated at 105% NBR steam flow condition in Figure 15.2-1.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux increase is limited to 157% NBR by the reactor scram. Peak fuel surface heat flux does not exceed 102.6% of its initial value. MCPR for this transient is still above the safety MCPR limit. Therefore, the design basis is satisfied. The applicant will provide reanalysis of this event for the specific core configuration.

15.2.1.4 Barrier Performance

15.2.1.4.1 One Pressure Regulator Failure - Closed

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed (Table 15.0-1); therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2 Pressure Regulation Downscale Failure

Peak pressure at the S/R valves reaches 1181 psig. The peak nuclear system pressure reaches 1221 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.1.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input. for this event is much less than those consequences identified in Subsection 15.2.4.5 (for a Type 2 event). Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than 40% NB rated. In addition, recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy the single-failure criterion and credit is taken for these protection features.

The pressure relief system, which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

Same as Subsection 15.2.2.2.2.1, except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.3 Core and System Performance

15.2.2.3.1 Input Parameters and Initial Conditions

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable speed change takes place.

The closure chacteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 sec.

Auxiliary power is independent of any T-G overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies.

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Events caused by low water level trips such as initiation of HPCS and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure

15.2.2.3.1 Input Parameters and Initial Conditions (Continued)

effects have occurred, and are expected to be less severe than those already experienced by the system.

15.2.2.3.2 Results

15.2.2.3.2.1 Generator Load Rejection with Bypass

Figure 15.2-2 shows the results of the generator trip from 105% rated steam flow conditions. Peak neutron flow rises 24% above NB rated conditions.

The average surface heat flux shows no increase from its initial value, and MCPR does not significantly decrease below its initial value. Therefore, this event does not have to be reanalyzed for a specific core configuration.

15.2.2.3.2.2 Generator Load Rejection with Failure of Bypass

Figure 15.2-3 shows that, for the case of bypass failure, peak neutron flux reaches about 199% of rated, and average surface heat flux reaches 102.7% of its initial value. Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria, and the MCPR limit is permitted to fall below the safety limit for the incidents of moderate frequency. However, the MCPR for this event, with a value of 1.14, is well above the safety limit. The Applicant will provide reanalysis of this event for the specific core configuration.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal operating range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the S/R valves reaches 1202 psig. The peak nuclear system pressure reaches 1233 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5. Therefore, the radiological exposures noted in Subsection 15.2.4.5 for Type 2 exposure cover these consequences of this event.

15.2.3.3 Core and System Performance

15.2.3.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40% of NBR power level.

15.2.3.3.2 Input Parameters and Initial Conditions (Continued)

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.2 Results

15.2.3.3.2.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105% NBR steam flow conditions in Figure 15.2-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 114.5% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value. Therefore, this event d[illegible] have to be reanlyzed for a specific core configuration.

15.2.3.3.2.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105% NB rated steam flow conditions in Figure 15.2-5.

Peak neutron flux reaches 180% of its rated value, and average surface heat flux reaches 101% of its initial value. Therefore, this transient is less severe than the generator load rejection with failure of bypass transient described in Subsection 15.2.2.3.3.2.

15.2.3.3.2.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 40% of rated power, the turbine stop valve closure and turbine control valve closure scrams and Recirculation pump trip

15.2.3.3.2.3 Turbine Trip with Bypass Valve Failure, Low Power (Continued)

(RPT) are automatically bypassed. At these lower power levels, turbine first-stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve setpoints and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values. Therefore, this event does not have to be reanalyzed for a specific core configuration.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1188 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1158 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The S/R valves open and close sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. Peak nuclear system pressure reaches 1231 psig at the vessel bottom; therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1202 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in Subsection 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for a Type 2 event. Therefore, the radiological exposures noted in Section 15.2.4.5 cover the consequences of this event.

15.2.4.2.1.1 Identification of Operator Actions (Continued)

(7) when the reactor pressure has decayed sufficiently for RHR operation, put it into service per procedure,

(8) before resetting the MSLIV isolation, determine the cause of valve closure,

(9) observe turbine coastdown and break vacuum before the loss of sealing steam (check T-G auxiliaries for proper operation),

(10) not reset and open MSLIVs unless conditions warrant and be sure the pressure regulator setpoint is above vessel pressure, and

(11) survey maintenance requirements and complete the scram report

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steamline Isolation Valves

MSLIV closures initiate a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the protection system.

The pressure relief system which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.2.2 Closure of One Main Steamline Isolation Valve

A closure of a single MSLIV at any given time will not initiate a reactor scram. This is because the valve position scram trip logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.3 Core and System Performance

15.2.4.3.1 Input Parameters and Initial Conditions

The main steam isolation valves close in 3 to 5 sec. The worst case (the 3-sec closure time) is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90% open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow. Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pump and initiate the HPCS and RCIC systems.

15.2.4.3.2 Results

15.2.4.3.2.1 Closure of All Main Steamline Isolation Valves

Figure 15.2-6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steamlines while the reactor is operating at 105% of NBR steam flow. Neutron flux increases slightly, and fuel surface heat flux shows no increase.

Water level decreases sufficiently to cause a recirculation system trip on the Level 3 (L3) trip at 1.9 sec and initiation of the HPCS and RCIC system on the Level 2 (L2) trip at some time greater than 10 sec. However, there is a delay up to 30 sec before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.4.3.3.2 Closure of One Main Steamline Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires

15.2.4.3.2.2 Closure of One Main Steamline Isolation Valve (Continued)

an initial power reduction to approximately 75 to 80% of design conditions in order to avoid high flux scram, high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3-sec closure of one main steam isolation valve during 105% rated power conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIVs at full power. No quantitative analysis is furnished for this event. However, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV setpoints. Therefore, this event does not have to be reanalyzed for specific core configurations.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down (such as operating state C, as defined in Appendix 15A) will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in Subsection 15.2.4.3.2.1.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steamline Isolation Valves

The nuclear system relief valves begin to open at approximately 2.7 sec after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1207 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline is 1174 psig.

15.2.4.4.2 Closure of One Main Steamline Isolation Valve

No significant effect is imposed on the RCPB, since, if closure of the valve occurs at an unacceptably high operating power level, a flux of pressure scram will result. The main turbine bypass system will continue to regulate system pressure via the other three "live" steamlines.

15.2.4.5 Radiological Consequences

15.2.4.5.1 General Observations

The radiological impact of many transients and accidents involves the consequences: (1) which do not lead to fuel rod damage as a direct result of the event itself; (2) additionally, many events

15.2.4.5.1 General Observations (Continued)

do not lead to the depressurization of the primary system but only the venting of sensible heat and energy via fluids at coolant loop activity through relief valves to the suppression pool, (3) in the case of previously defective fuel rods, a depressurization transient will result in considerably more fission product carryover to the suppression pool than hot-standby transients; and (4) the time duration of the transient varies from several minutes to more than four hours.

The above observations lead to the realization that radiological aspects can involved a broad spectrum of results. For example:

(1) Transients where appropriate operator action (seconds) results in quick return (minutes) to planned operation, little radiological impact results.

(2) Where major RCPB equipment failure requires immediate plant shutdown and its attendant depressurization under controlled shutdown time tables (4 hours), the radiological impact is greater.

In order to envelope the potential radiological impact, a worst case like example No. 2 is described below. However, it should be noted that most transients are like example No. 1 and the radiological envelope conservatively overpredicts the actual radiological impact by a factor greater than 100.

15.2.4.5.2 Depressurization - Shutdown Evaluation

15.2.4.5.2.1 Fission Product Release from Fuel

While no fuel rods are damaged as a consequence of this event, fission product activity associated with normal coolant activity

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-13 lists the sequence of events for Figure 15.2-7.

15.2.5.2.1.1 Identification of Operator Actions

The operator should:

(1) verify auto transfer of buses supplied by generator to incoming power - if automatic transfer has not occurred, manual transfer must be made,

(2) monitor and maintain reactor water level at required level,

(3) check turbine for proper operation of all auxiliaries during coastdown,

(4) depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes,

(5) put the mode switch in the STARTUP position before the reactor pressure decays to <850 psig,

(6) secure the RCIC operation if auto-initiation occurred due to low water level,

15.2.5.2.1.1 Identification of Operator Actions (Continued)

(7) monitor control rod drive positions and insert both the IRMs and SRMs,

(8) investigate the cause of the trip, make repairs as necessary, and complete the scram report, and

(9) cooldown the reactor per standard procedure if a restart is not intended.

15.2.5.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

Tripping functions incurred by sensing main turbine condenser vacuum pressure are designated in Table 15.2-14.

15.2.5.3 Core and System Performance

15.2.5.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40% NBR power level.

The analysis presented here is a hypothetical case with a conservative 2 in. Hg/sec vacuum decay rate. Thus, the bypass system is available for several seconds, since the bypass is signaled to close at a vacuum level of about 10 in. Hg less than the stop valve closure.

15.2.5.3.2 Results

Under this hypothetical 2 in. Hg/sec vacuum decay condition, the turbine bypass valve and main steamline isolation valve closure would follow main turbine and feedwater turbine trips about 5 sec after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steamline isolation valve closure tends to be minimal, since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steamline flow. Figure 15.2-7

15.2.5.3.3 Results (Continued)

shows the transient expected for this event. It is assumed that the plant is initially operating at 105% of NBR steam flow conditions. Peak neutron flux reaches 114% of NBR power, while average fuel surface heat flux shows no increase. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1186 psig at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1157 psig. A comparison of these values to those for Turbine Trip at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steamline isolation.

15.2.5.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event

15.2.5.5 Radiological Consequences (Continued)

is much less than those consequences identified in Subsection 15.2.4.5; therefore, the radiological exposures noted in Subsection 15.2.4.5 for Type 2 events cover these consequences of this event.

15.2.6 Loss of Offsite AC Power

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Auxiliary Power Transformer

Causes for interruption or loss of the auxiliary power transformer can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high current operation, as well as oper-tor error which trips the transformer breakers.

15.2.6.1.1.2 Loss of All Grid Connections

Loss of all grid connections can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabil-ities will cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

15.2.6.2.2.1 Loss of Auxiliary Power Transformer (Continued)

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the auxiliary transformer) provide the following simulation sequence (assuming a solid state reactor trip system):

(1) all electrical pumps are tripped at a reference time, t=0, with normal coastdown times for the recirculation pumps.

(2) Within 8 sec, the loss of main condenser circulating water pumps causes condenser vacuum to drop to the main turbine and feedwater turbine trip setting, causing stop valve closure and scram when the stop valves are less than 90% open, assuming 0.5 in. Hg/sec vacuum decay rate. However, scram, main turbine and feedwater turbine tripping may occur earlier than this time, if water level reaches the high water level (Level 8) setpoint before 8 sec.

(3) At approximately 28 sec, the loss of condenser vacuum is expected to reach the MSIV and bypass valves closure setpoint and main steamline isolation setpoint.

Operation of the HPCS and RCIC system functions are not simulated in this analysis. Their operation occurs at some time beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

15.2.6.2.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.2.2.1 with the following additional concern:

The loss of all grid connections is another feasible, although improbable, way to lose all auxiliary power. This event would add a generator load rejection to the above sequence at time, t=0. The load rejection immediately forces the turbine control valves closed, causes a scram and initiates recirculation pump trip (RPT) (already tripped at reference time t=0).

15.2.6.3 Core and System Performance

15.2.6.3.1 Loss of Auxiliary Power Transformer

Figure 15.2-8 shows graphically the simulated transient. The initial portion of the transient is similar to the recirculation pump trip transient. At 4 sec turbine trip, scram, and feedwater turbines trip on high water level. Main steamline isolation valves and turbine bypass valves close at 28 sec on their condenser vacuum setpoint.

Sensed level drops to the RCIC and HPCS initiation setpoint at approximately 27 sec after loss of auxiliary power. The RHRS, in the steam condensing mode, is initiated to dissipate the heat.

There is no significant increase in fuel temperature or decrease in the operating MCPR value, fuel thermal margins are not threatened and the design basis is satisfied. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.6.3.2 Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in Subsection 15.2.2. Figure 15.2-9 shows graphically the simulated event and does not have to be reanalyzed for specific core configurations.

15.2.7.2.1.1 Identification of Operator Actions (Continued)

(3) verify that the recirculation pumps trip on reactor low level,

(4) secure HPCS when reactor level and pressure are under control,

(5) continue operation of RCIC until decay heat diminishes to a point where the RHR system can be put into service,

(6) monitor turbine coastdown, break vacuum as necessary, and

(7) complete scram report and survey maintenance requirements.

15.2.7.2.2 Systems Operation

Loss of feedwater flow results in a proportional reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. Reactor protection system responds within 1 sec after this trip to scram the reactor. The low level (L3) scram trip function meets the single-failure criterion.

Containment isolation, if water level reaches (L1), would also initiate a main steamline isolation valve position scram trip signal as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time.

15.2.7.3 Core and System Performance

The results of this transient simulation are shown in Figure 15.2-10. Feedwater flow terminates at approximately 5 sec. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 5 sec. Water level continues to drop until, first, the recirculation flow is runback at level 4 (L4) and then the vessel level (L3) scram trip setpoint is reached, whereupon the reactor is shut down and the recirculation pumps are tripped to low frequency speed. Vessel water level continues to drop to the L2 trip. At this time, the recirculation pumps are tripped, and the HPCS and RCIC operation is initiated. MCPR remains considerably above the safety limit, since increases in heat flux are not experienced. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.7.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

The consequences of this event do not result in any fuel failure. Therefore, no analysis of the radiological consequences is required.

15.2.8 Feedwater Line Break

(Refer to Subsection 15.6.6)

15.2.9 Failure of RHR Shutdown Cooling

Normally, in evaluating component failure considerations associated with the RHRS-Shutdown Cooling mode operation, active pumps

15.2.9 Failure of RHR Shutdown Cooling (Continued)

or instrumentation (all of which are redundant for safety system portions of the RHRS aspects) would be assumed to be the likely failed equipment. For purposes of worst-case analysis, the single recirculation loop suction valve to the redundant RHRS loops is assumed to fail. This failure would, of course, still leave two complete RHRS loops for LPCI, pool, and containment cooling minus the normal RHRS-Shutdown Cooling loop connection. Although the valve could be manually manipulated open, it is assumed failed indefinitely. If it is now assumed that the single active failure criterion is applied, the plant operator has one complete RHRS loop available with the further selective worst-case assumption that the other RHRS loop is lost.

Recent analytical evaluations of this event have required additional worst-case assumptions. These included:

(1) loss of all offsite AC power;

(2) utilization of safety shutdown equipment only; and

(3) operator involvement only after 10 min after coincident assumptions.

These accident-type assumptions certainly would change the initial incident (malfunction of RHRS suction valve) from a moderate frequency incident to a classification in the design basis accident status. However, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 105% NBR steam flow when a long-term loss of offsite power occurs, causing multiple SRV actuation (Subsection 15.2.6) and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power, an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. The operator then establishes a shutdown cooling path for the vessel through the ADS valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event. However, for the following reasons, it could be considered an infrequent incident:

(1) no RHR valves have failed in the shutdown cooling mode in BWR total operating experience, and

(2) the set of conditions evaluated is for multiple failure as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-18.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are identical to those described in Subsection 15.2.6 (Loss of Offsite Power Event with Isolation/Scram). The operator should do the following:

(1) at approximately 10 min into the transient, initiate suppression pool cooling (again for purposes of this analysis, it is assumed that only one RHR heat exchanger is available);

(2) initiate RPV shutdown depressurization by manual actuation of 3 ADS valves;

(3) after the RPV is depressurized to approximately 100 psig, the operator should attempt to open one of the two RHR shutdown cooling suction valves (these attempts are assumed unsuccessful); and

(4) at 100 psig RPV pressure, the operator establishes a closed cooling path as described in the notes for Figure 15.2-11.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation, credit is taken for the plant and reactor protection systems and/or the ESF utilization.

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. During this time, MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-min time period assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action.

15.2.9.3.2 Mathematical Model

In evaluating this event, the important parameters to consider are reactor depressurization rate and suppression pool temperature. Models used for this evaluation are described in References 3 and 4.

15.2.9.3.3 Input Parameters and Initial Conditions

Table 15.2-19 shows the input parameters and initial conditions used in evalution of this event. The data in Table 15.2-19 is utilized with the following clarifications: 15.2

a. The suppression pool mass of 8.696 x 10^6 includes the suppression pool water to LWL and the volume to completely fill the RPV and the steam lines with water at 100°F.

b. The three values of flow rates given are runout flow rates. The actual flow rates are functions of pressure and are so input into the computer codes used to analyze the transient. The RHR system would not be operated in the LPCI mode for pressure less than 100 psig, but would be operated in pool cooling mode. 15.2

15.2.9.3.4 Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using other, normal shutdown cooling equipment. In cases where both of the RHRS shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-12). An evaluation has been performed assuming the worst single failure that could disable the RHRS shutdown cooling valves.

The analysis demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The evaluation assures 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 safety function can be accomplished, assuming a worst-case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (Reference 5 and Figure 15.2-11).

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. Even if it is additionally postulated that all of the ADS or relief valves discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell.

15.2.9.3.4 Results (Continued)

The systems have suitable redundancy in components such that, for onsite electrical power operation (assuming offsite power is not available) and for offsite electrical power operation (assuming onsite power is also not available), the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 psia and 125°F) conditions.

15.2.9.3.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCS systems, together with the nuclear boiler pressure relief system.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event (loss of offsite power), which results in reactor isolation and subsequent relief valve actuation and suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure and temperature are reduced to and maintained at saturated conditions at approximately 100 psig. The reactor vessel is depressurized by manually opening selected SRVs. Reactor vessel makeup water is automatically provided via the RCIC/HPCS systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

15.2.9.3.4.1 Full Power to Approximately 100 psig (Continued)

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC/HPCS and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

The results of this analysis are applicable to all BWR/6 core configurations.

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

(1) the vessel is at 100 psig and saturated conditions;

(2) a worst-case single failure is assumed to occur (i.e., loss of a division of emergency power); and

(3) there is no offsite power available.

In the event that the RHR's shutdown suction line is not available because of single failure, the first action to be taken will be to maintain the 100 psig level while personnel gain access and effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling suction line cannot be repaired, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

The RHR shutdown cooling line valves are in two divisions (Division 1 = the outboard valve, and Division 2 = the inboard valve) to satisfy containment isolation criteria. For evaluation

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown (Continued)

purposes, the worst-case failure is assumed to be the loss of a division of emergency power, since this also prevents actuation of one shutdown cooling line valve. Engineered safety feature equipment available for accomplishing the shutdown cooling function includes (for the selected path):

ADS (DC Division 1 and DC Division 2)

RHR Loop (A) (Division 1)

HPCS (Division 3)

RCIC (DC Division 1)

LPCS (Division 1)

Since availability or failure of Division 3 equipment does not affect the normal shutdown mode, normal shutdown cooling is easily available through equipment powered from only Divisions 1 and 2. It should be noted that, conversely, the HPCS system is always available for coolant injections if either of the other two divisions fails. For failure of Divisions 1 or 2, the following systems are assumed functional:

(1) Division 1 Fails, Divisions 2 and 3 Functional:

Failed Systems	Functional Systems
RHR Loop (A)	HPCS
LPCS	ADS
	RHR Loops B and C
	RCIC

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown (Continued)

Assuming the single failure is a failure of Division 1 emergency power, the safety function is accomplished by establishing one of the cooling loops described in Activity C1 of Figure 15.2-11.

(2) Division 2 Fails, Divisions 1 and 3 Functional:

Failed Systems	Functional Systems
RHR Loops B and C	HPCS
	ADS
	RHR Loop A
	RCIC
	LPCS

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-11. Figures 15.2-13, 15.2-14, 15.2-15 and 15.2-16 show RHR loops A, B and C (simplified).

Using the above assumptions and following the depressurization rate shown in Figure 15.2-17, the suppression pool temperature is shown in Figure 15.2-18.

The results of this analysis are applicable to all BWR/6 core configurations.

15.2.9.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient, in excess of the criteria for which the fuel, pressure vessel or containment are designed. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described below.

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15.3.1.2.1.3.2 Trip of Two Recirculation Pumps

The operator should ascertain that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation, monitoring reactor water level and pressure control after shutdown. When both reactor pressure and level are under control, the operator should secure both HPCS and RCIC as necessary. The operator should also determine the cause of the trip prior to returning the system to normal.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of One Recirculation Pump

Tripping a single recirculation pump requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.2.2 Trip of Two Recirculation Pumps

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems.

Specifically, this transient takes credit for vessel level (L8) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine stop valves. High system pressure is limited by the pressure relief valve system operation.

15.3.1.3 Core and System Performance

15.3.1.3.1 Input Parameters and Initial Conditions

Pump motors and pump rotors are simulated with minimum specified rotating inertias.

15.3.1.3.2 Results

15.3.1.3.2.1 Trip of One Recirculation Pump

Figure 15.3-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 5.7 sec. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 131% of normal diffuser flow and 54% of rated core flow. MCPR remains significantly above the safety limit; thus, the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.1.3.2.2 Trip of Two Recirculation Pumps

Figure 15.3-2 graphically shows this transient with minimum specified rotating inertia. MCPR remains unchanged. No scram is initiated directly by pump trip. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip, thereby shutting down the main turbine and feed pump turbines, and scramming. Subsequent events, such as main steamline isolation and initiation of RCIC and HPCS systems occurring late in this event, have no significant effect on the results. Therefore, this does not have to be reanalyzed for specific core configurations.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Recirculation Pump

Figure 15.3-1 results indicate a basic reduction in system pressures from the initial conditions. Therefore, the RCPB barrier is not threatened.

15.3.1.4.2 Trip of Two Recirculation Pumps

The results shown in Figure 15.3-2 indicate that peak pressures stay well below the 1375 psig limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for a Type 2 event. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.2 <u>Recirculation Flow Control Failure - Decreasing Flow</u>

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

Master controller malfunctions can cause a decrease in core coolant flow. A downscale failure of either the master power controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator has a velocity limiter which limits the maximum valve stroking rate to 11%/sec. A postulated failure of the input demand signal, which is utilized in both loops, can decrease core flow at the maximum valve stroking rate established by the loop limiter.

Failure within either loop's controller can result in a maximum valve stroking rate as limited by the capability of the valve hydraulics.

15.3.2.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Fast Closure of One Main Recirculation Valve

Table 15.3-3 lists the sequence of events for Figure 15.3-3.

15.3.2.2.1.2 Fast Closure of Two Main Recirculation Valves

Table 15.3-4 lists the sequence of events for Figure 15.3-4.

15.3.2.2.1.3 Identification of Operator Actions

15.3.2.2.1.3.1 Fast Closure of One Main Recirculation Valve

As soon as possible, the operator should verify that no operating limits are being exceeded. The operator should determine the cause of failure prior to returning the system to normal.

15.3.2.2.1.3.2 Fast Closure of Two Main Recirculation Valves

As soon as possible, the operator must verify that no operating limits are being exceeded. If they are, corrective actions must be initiated. Also, the operator must determine the cause of the trip prior to returning the system to normal.

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Fast Closure of One Main Recirculation Valve

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

15.3.2.2.2.2 Fast Closure of Two Main Recirculation Valves

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

15.3.2.3.1.1 Fast Closure of One Main Recirculation Valve

Failure within either loop controller can result in a maximum stroking rate of 60%/sec as limited by the valve hydraulics.

15.3.2.3.1.2 Fast Closure of Two Main Recirculation Valves

A downscale failure of either the master power controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator circuitry has a velocity limiter which limits maximum valve stroking rate to 11%/per sec. Recirculation loop flow is allowed to decrease to approximately 25% of rated before high water level

15.3.2.3.1.2 Fast Closure of Two Main Recirculation Valves (Continued)

(L8) causes trip of the recirculation pumps due to stop valve closure. This is the flow expected when the flow control valves are maintained at a minimum open position.

15.3.2.3.2 Results

15.3.2.3.2.1 Fast Closure of One Recirculation Valve

Figure 15.3-3 illustrates the maximum valve stroking rate which is limited by hydraulic means. Even though a turbine trip on high water level occurs, the MCPR remains significantly above the safety limit. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.2.3.2.2 Fast Closure of Two Recirculation Valves

Figure 15.3-4 illustrates the expected transient which is similar to a two-pump trip. This analysis is very similar to the two-pump trip described in Subsection 15.3.1. Design of limiter operation is intended to render this transient to be less severe than the two-pump trip. MCPR remains significantly greater than the safety limit; therefore, no fuel damage occurs. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5.

15.3.3.2.1.1 Identification of Operator Actions

The operator should ascertain that the reactor scrams from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump, and he should monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

Operation of safe shutdown features, though not included in this simulation, is expected to be utilized in order to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2 (see Appendix 15A for further details).

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The nonlinear dynamic model described in Subsection S.2.2 of Reference 1 is used to simulate this event.

15.3.3.3.2 Input Parameters and Initial Conditions

For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 105% NBR steamflow. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value (i.e., the least negative value in Table 15.0-2).

15.3.3.3.3 Results

Results for this event are documented in Subsection S.2.5.5 of Reference 1. Based on these results, this event does not have to be reanalyzed for specific core configurations.

15.3.3.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.4 Recirculation Pump Shaft Break

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered as a DBA event. It has been evaluated as a very mild accident in relation to other DBAs such as the LOCA. The analysis has been conducted with consideration to a single- or two-loop operation. (Refer to Chapter 5 for specific mechanical considerations and Chapter 7 for electrical aspects.)

15.3.4.1 Identification of Causes and Frequency Classification (Continued)

This postulated event is bounded by the more limiting case of recirculation pump seizure. Quantitative results for this more limiting case are presented in Subsection 15.3.3.

15.3.4.1.1 Identification of Causes

The case of recirculation pump shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the break of the pump shaft.

15.3.4.1.2 Frequency Classification

This event is considered a limiting fault but results in effects which can easily satisfy an event of greater probability (i.e., infrequent incident classification).

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one recirculation pump (Subsection 15.3.4.1.1.) will cause the core flow to decrease rapidly resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (Level 8), scram, main turbine trip and feedwater pump trip will be initiated. Subsequently, the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by HPCS and RCIC flow.

15.3.4.2.1.1 Identification of Operator Actions

The operator should ascertain that the reactor scrams resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump; and he should monitor reactor water level and pressure control after shutdown.

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (L8) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the pressure relief system operation.

Operation HPCS and RCIC systems is expected in order to maintain adequate water level control.

15.3.4.3 Core and System Performance

If this extremely unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a reactor scram and trip of the main and feedwater turbines. Since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, there is no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

The severity of this pump shaft break event is bounded by the pump seizure event (Subsection 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump seizure event, the loop flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event and this event does not have to be reanalyzed for specific core configurations.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.5 References

1. "General Electric Standard Applications for Reactor Fuel-United States Supplement,"(NEDE-24011-P-A, latest approved revision).

SECTION 15.4

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15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Rod Withdrawal Error - Low Power

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes, alone, is considered low enough to warrant its being categorized as an infrequent incident, since there is no postulated set of circumstances which results in an inadvertent rod withdrawal error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal or Withdrawal

During refueling operations, safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Insertion With Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel

15.4.1.1.2.2 Fuel Insertion With Control Rod Withdrawn (Continued)

is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

15.4.1.1.2.3 Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block by the refueling interlocks. Since the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

Finally, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles.

15.4.1.1.2.5 Identification of Operator Actions

No operator actions are required to preclude this event, since the protection system design as discussed above will prevent its occurrence.

15.4.1.1.3 Core and System Performance

Since the possibility of inadvertent criticality during refueling is precluded, the core and system performances were not analyzed. The withdrawal of the highest worth control rod during refueling will not result in criticality. This is verified experimentally by performing shutdown margin checks (see Subsection 4.3.2 for a

15.4.1.1.3 Core and System Performance (Continued)

description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Since no fuel damage can occur, no radioactive material will be released from the fuel. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since there is not a postulated set of circumstances for which this event could occur.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event, since no radioactive material is released from the fuel.

15.4.1.2 Continuous Control Rod Withdrawal Error During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of the initial causes of error of this event, alone, is considered low enough to warrant its being categorized as an infrequent incident. The probability of further single failures postulated for this event is even lower because it is contingent upon the simultaneous failure of two redundant inputs to the rod control and information system (RCIS), concurrent with a high worth rod, out-of-sequence rod selection, plus operator nonacknowledgment of continuous alarm annunciations prior to safety system actuations.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Continuous control rod withdrawal errors during reactor startup are precluded by the RCIS. The RCIS prevents the withdrawal of an out-of-sequence control rod in the 100%-75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the low power setpoint. Since only in-sequence control rods can be withdrawn in the 100%-75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the low power setpoint, there is no basis for the continuous control rod withdrawal error in the startup and low power range. (See Subsection 15.4.2 for description of continuous control rod withdrawal above the low power setpoint. The bank position mode of the RCIS is described in Reference 1.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event, since the plant design as discussed above prevents its occurrence.

15.4.1.2.3 Core and System Performance

The performance of the RCIS prevents erroneous selection and withdrawal of an out-of-sequence control rod. This, the core and system performance is not affected by such an operator error. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.4.1.2.4 Barrier Performance

As evaluation of the barrier performance was not made for this event, since there is no postulated set of circumstances for which this error could occur.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

The Rod Withdrawal Error (RWE) transient results from a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously until the Rod Withdrawal Limiter (RWL) function of the Rod Control and Information System (RCIS) blocks further withdrawal.

15.4.2.1.2 Frequency Classification

The frequency of occurrence for the RWE is assumed to be moderate, since definite data do not exist. The frequency of occurrence diminishes as the reactor approaches full power by virtue of the reduced number of control rod movements. A statistical approach, using appropriate conservative acceptance criteria, shows that consequences of the majority of RWEs would be very mild and hardly noticeable.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for this transient is presented in Table 15.4-1.

15.4.2.2.2 System Operations

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod or gang of control rods continuously until the RWL inhibits further withdrawal. The RWL utilizes rod position indications of the selected rod as input.

During the course of this event, normal operation of plant instrumentation and conrols is assumed, although no credit is taken for this except as described above. No operation of any engineered safety feature (ESF) is required during this event.

15.4.2.3 Core and System Performance

15.4.2.3.1 Input Parameters and Initial Conditions]

The reactor core is assumed to be on MCPR and MLHGR technical specification limits prior to RWE initiation. A statistical analysis of the rod withdrawal error results (Appendix 15B) initiated from a wide range of operating conditions (exposure, power, flow, rod patterns, xenon conditions, etc) has been performed, establishing allowable rod withdrawal increments applicable to all BWR/6 plants. These rod withdrawal increments were determined such that the design basis ΔMCPR (minimum critical power ratio) for rod withdrawal errors initiated from the technical specification operating limit and mitigated by the RWL system withdrawal restrictions, provides a 95% probability at the 95% confidence level that any randomly occurring RWE will not result in a larger ΔMCPR. MCPR was verified to be the limiting thermal performance parameter and therefore was used to establish the allowable withdrawal increments. The 1% plastic strain limit on the clad was always a less limiting parameter.

15.4.2.3.2 Resu'_s

The calculated r_ults demonstrate that, should a rod or gang be withdrawn a distance equal to the allowable rod withdrawal increment, there exists a 95% probability at the 95% confidence level that the resultant ΔMCPR will not be greater than the design basis ΔMCPR. Furthermore, the peak LHGR will be substantially less than that calculated to yield 1% plastic strain in the fuel clad.

These results of the generic analyses in Appendix 15B show that a control rod or gang can be withdrawn in increments of 12 in. at power levels ranging from 70-100% of rated, and 24 in. at power levels ranging from 20-70% (Table 15.4-2). See Subsection 15.4.1.2 for RWE's below 20% reactor power. The 20% and 70% reactor core power levels correspond to the Low Power Set Point (LPSP) and High Power Set Point (HPSP) of the RWL. Results of either the generic or plant specific analysis will be provided by the Applicant.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power for RWEs initiated from rated conditions is less than 4% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event, since no radioactive material is released from the fuel.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluation cited in Subsections 15.4.1 and 15.4.2 and does not have to be reanalyzed for specific core configurations.

15.4.4 Abnormal Startup of Idle Recirculation Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4-3 lists the sequence of events for Figure 15.4-1.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) adjust rod pattern, as necessary, for new power level following idle loop start;

(2) determine that the idle recirculation pump suction and discharge block values are open and that the flow control valve in the idle loop is at minimum position and, if not, place them in this configuration;

(3) readjust flow of the running loop downward to less than half of the rated flow;

(4) determine that the temperature difference between the two loops is no more than 50°F;

(5) start the idle loop pump and adjust flow to match the adjacent loop flow (monitor reactor power); and

(6) readjust power, as necessary, to satisfy plant requirements per standard procedure.

NOTE: The time to do the above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF action occurs as a result of the transient.

15.4.4.3 Core and System Performance

15.4.4.3.1 Input Parameters and Initial Conditions

One recirculation loop is idle and filled with cold water (100°F). (Normal procedure when starting an idle loop with one pump already running requires that the indicated idle loop temperature be no more than 50°F lower than the indicated active loop temperature.)

The active recirculation loop is operating with the flow control valve position that produces about 70% of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 33% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

The idle recirculation pump suction and discharge block valves are open and the recirculation flow control valve is closed to its minimum open position. (Normal procedure requires leaving an idle loop in this condition to maintain the loop temperature within the required limits for restart.)

15.4.4.3.2 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15.4-1. Shortly after the pump begins to move, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply. The motor approaches synchronous speed in approximately 3 sec because of the assumed minimum pump and motor inertia.

A short-duration neutron flux peak is produced as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at 80% of rated

15.4.4.3.2 Results (Continued)

before decreasing after the cold water washed out of the loop at about 18 sec. No damage occurs to the fuel barrier and MCPR remains significantly above the safety limit as the reactor settles out at its new steady-state condition. Therefore, this event does not have to reanalyzed for specific core configurations.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient (Figure 15.4-1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master controller of neutron flux controller can cause an increase in the core coolant flow rate. Failure within a loop's flow controller can also cause an increase in core coolant flow rate.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

15.4.5.2.1.1 Fast Opening of One Recirculation Valve

Table 15.4-4 lists the sequence of events for Figure 15.4-2.

15.4.5.2.1.2 Fast Opening of Two Recirculation Valves

Table 15.4-5 lists the sequence of events for Figure 15.4-3.

15.4.5.2.1.3 Identification of Operator Actions

Initial action by the operator should include:

(1) transfer flow control to manual and reduce flow to minimum, and

(2) identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

(1) observe that all rods are in;

(2) check the reactor water level and maintain above low level (L2) trip to prevent MSLIVs from isolating;

(3) switch the reactor mode switch to the STARTUP position;

15.4.5.2.1.3 Identification of Operator Actions (Continued)

(4) continue to maintain vacuum and turbine seals;

(5) transfer the recirculation flow controller to the manual position and reduce setpoint to zero;

(6) Survey maintenance requirements and complete the scram report;

(7) monitor the turbine coastdown and auxiliary systems; and

(8) establish a restart of the reactor per the normal procedure

NOTE: Time required from first trouble alarm to restart would be approximately 1 hr.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls and the reactor protection system. Operation of engineered safeguards is not expected.

15.4.5.3 Core and System Performance

15.4.5.3.1 Input Parameters and Initial Conditions]

In each of these transient events, the most severe transient results when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is 54% NBR power and 33% core flow. The maximum stroking rate of the recirculation loop valves for a master controller failure driving two loops is limited by individual loop controls to 11%/sec

15.4.5.3.1 Input Parameters and Initial Conditions (Continued)

Maximum stroking rate of a single recirculation loop value for a loop controller failure is limited by hydraulics to 30%/sec.

15.4.5.3.2 Results

15.4.5.3.2.1 Fast Opening of One Recirculation Valve

Figure 15.4-2 shows the analysis of a failure where one recirculation loop main valve is opened at its maximum stroking rate of 30%/sec. Table 15.4-4 provides the sequence of events of this failure.

The rapid increase in core flow causes a sharp rise in neutron flux, initiating a reactor scram at approximately 1.3 sec. The peak neutron flux reached was 235% of NBR value, while the accompanying average fuel surface heat flux reaches 73% of NBR at approximately 2.2 sec. MCPR remains considerably above the safety limit and average fuel temperature increases only 108°F. Reactor pressure is discussed in Subsection 15.4.5.4.

15.4.5.3.2.2 Fast Opening of Two Recirculation Valves

Figure 15.4-2 illustrates the failure where both recirculation loop main valves are opened at a maximum stroking rate of 11%/sec. Table 15.4-5 shows the sequence of events for this failure. It is very similar to the above transient. Flux scram occurs at approximately 1.6 sec, peaking at 162% of NB rated, while the average surface heat flux reaches 67% of NB rated at approximately 2.3 sec. MCPR remains considerably above the safety limit and average fuel temperature increases 80°F. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Opening of One Recirculation Valve

This transient results in a very slight increase in reactor vessel pressure (Figure 15.4-2) and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Opening of Two Recirculation Valves

This transient results in a very slight increase in reactor vessel pressure (Figure 15.4-3) and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs. This is a PWR event.

15.4.7 Misplaced Bundle Accident

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the equilibrium core loading. First, a bundle must be misloaded into a wrong location in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to also be put in an

15.4.7.1.1 Identification of Causes (Continued)

incorrect location or discharged. Third, the misplaced bundles would have to be overlooked during the core verification process performed following core loading.

15.4.7.1.2 Frequency Classification

This unlikely event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequency incident based on the following data:

Expected Frequency: 0.002 events/operating cycle

The above number is based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.7.3 Core and System Performance

This event is discussed in Subsection S.2.5.4 of Reference 2.

An analysis was performed to quantify the worst fuel bundle loading error for the GESSAR II equilibrium cycle. A summary of the results of that analysis is presented in Table 15.4-8. As can be seen, MCPR remains well above the MCPR safety limit, and MLHGR does not exceed the 1% plastic strain limit for the clad. Therefore, no violation of fuel limits occurs as a result of this event. Because this event is dependent upon the specific core configuration, the applicant will provide the results of this event.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since it is very mild and highly localized event. No perceptable change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.8 <u>Spectrum of Rod Ejection Assemblies</u>

Not applicable to BWRs. This is a PWR event.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the CRD system assemblies. The CRD housing support assemblies are described in Chapter 4.

15.4.9 Control Rod Drop Accident (CRDA)

This limiting fault is fully described in Subsection S.2.5.1 of Reference 2 for the GESSAR II banked position withdrawal sequence.

15.4.9.1 Evaluation of Results

The radiological evaluations are based on the assumed failure of 770 fuel rods. The number of rods which exceed the damage threshold is less than 770 for all plant operating conditions or core exposure, provided the peak enthalpy is less than the 280 cal/gm design limit.

The results of the compliance-check calculation (Table 15.4-11) indicate that the maximum incremental rod worth is well below the worth required to cause a CRDA which would result in 280 cal/gm peak fuel enthalpy. The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 770 rods for the radiological evaluation is conservative.

15.4.9.2 Barrier Performance

An evaluation of the barrier performance was not made for this accident, since this is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.3 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CRF100 guidelines. This analysis is referred to as the "Design Basis Analysis".

15.4.9.3 Radiological Consequences (Continued)

(2) The second analysis is based on assumptions considered to provide a realistic yet conservative estimate of ratiological consequences. This analysis is referred to as the "Realistic Analysis".

A schematic of the leakage path is shown in Figure 15.4-4.

15.4.9.3.1 Design Basis Analysis

The specific models, assumptions and the program used for computer evaluation are described in Reference 3. Specific parametric values used in the evaluation are presented in Table 15.4-12.

15.4.9.3.1.1 Fission Product Release from Fuel

The failure of 770 fuel rods is used for this analysis. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting (taken as 2842°C) is estimated to be 0.0077.

Fuel reaching melt conditions is assumed to release 100% of the noble gas inventory and 50% of the iodine inventory. The remaining fuel in the damaged rods is assumed to release 10% of both the noble gas and iodine inventories.

A maximum equilibrium inventory of fission products in the core is based on 1000 days of continuous operation at 3651 MWt. No delay time is assumed, but it is assumed that the failed rods have been operated at power level 1.5 times that of the average power level of the core.

15.4.9.3.1.2 Fission Product Transport to the Environment]

The transport pathway is shown in Figure 15.4-4 and consists of carryover with steam to the turbine condenser prior to MSLIV closure, and leakage from the condenser to the environment. No credit is taken for the turbine building.

Of the activity released from the fuel, 100% of the noble gases and 10% of the iodines are assumed to be carried to the condenser before MSLIV closure is complete.

Of the activity reaching the condenser, 100% of the noble gases and 10% of the iodines (due to partitioning and plateout) remain airborne. The activity airborne in the condenser is assumed to leak directly to the environment a rate of 1.0% per day. Radioactive decay is accounted for during residence in the condenser; however, it is neglected after release to the environment.

The activity airborne in the condenser is presented in Table 15.4-13. The cumulative release of activity to the environment is presented in Table 15.4-14.

15.4.9.3.1.3 Results]

The calculated exposures from the design basis analysis are presented in Table 15.4-15 and are well within the guidelines of 10CFR100.

15.4.9.3.2 Realistic Analysis]

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in

15.4.9.3.2 Realistic Analysis (Continued)

Reference 4. Specific values of parameters used in the evaluation are presented in Table 15.4-12.

15.4.9.3.2.1 Fission Product Release from Fuel

The following assumptions are used in calculating the fission product activity released from the fuel:

(1) The reactor has been operating at design power for 1000 days until 30 min prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30-min time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 5).

(3) The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the fission products already present in the fuel.

15.4.9.3.2.2 Fission Product Transport to the Environment]

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

(1) The recirculation flow rate is 25% of rated, and the steam flow to the condenser is 5% of rated. The 25% recirculation flow and 5% steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5% steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steamlines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steamline isolation valves to achieve full closure.

(2) The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 sec after detection of high radiation in the main steamlines and to be fully closed at 5 sec from the receipt of the closure signal. The signal originates from the main steamline radiation monitors. The total amount of fission product activity transported to the condenser before the steamlines are isolated is, therefore, governed by the 5.5-sec isolation time and the conditions in (1) above.

(3) All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

15.4.9.3.2.2 Fission Product Transport to the Environment (Continued)

(4) The mass ratio of the halogen concentration in steam, to that of the water, is assumed to be 2%.

(5) Fission product plate-out is neglected in the reactor vessel, main steamlines, turbine and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100% of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. The partition factor assumed applicable is 100, while the ratio of air volume to water volume is taken as 3. Based on the above conditions, the activity airborne in the condenser is presented in Table 15.4-16.

The following assumptions and conditions are used to evaluate the activity released to the environment:

(1) The leak rate out of the condenser is 0.5% of the combined condenser and turbine free volume (2.47E5 ft^3) per day.

(2) The activity released from the condenser becomes a r-borne in the turbine building. The turbine building ventilation rate is 1327% per day.

Based on the above assumptions, the integrated fission product release to the environment is presented in Table 15.4-17.

15.4.9.3.2.3

The calculated off-site exposures for the realistic analysis are presented in Table 15.4-18 and demonstrate the wide margin of conservatism in the design basis analysis.

15.4.10 References

1. C. J. Paone, "Bank Position Withdrawal Sequence", September 1976 (NEDO-21231).

2. "General Electric Standard Application for Reactor Fuel-United States Supplement," NEDE-24011-P-A-US, (Latest approved revision).

3. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143).

4. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

5. N. R. Horton, W. A. Williams, K. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors", March 1976 (APED-5756).

Table 15.4-7

INPUT PARAMETERS AND INITIAL CONDITIONS FOR THE FUEL BUNDLE LOADING ERROR

(1)	Power (% rated)	100
(2)	Flow (% rated)	100
(3)	MCPR operating limit*	1.20
(4)	MLHGR operating limit (kW/ft)*	13.4
(5)	Core Exposure	End of Cycle

*These are above the current operating limits. Since these limits do not go into the calculation of the MCPR associated with a mislocated bundle, differences in the safety operating limits will not effect these results.

Table 15.4-8

RESULTS OF MISPLACED BUNDLE ANALYSIS
EQUILIBRIUM CYCLF

(1)	MCPR Safety Limit	1.07
(2)	MCPR with misplaced bundle	1.14
(3)	LHGR 1% plastic strain limit	>20 kW/ft
(4)	LHGR with misplaced bundle*	14.9

*Does not include any densification penalty.

Table 15.4-12 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data:		
A. Site Boundary and LPZ distances (m)	*	*
B. X/Q's for time intervals of:		
(1) 0-1 hr - SB/LPZ	2.0E-3/1.0E-2	2.0E-3/1.0E-3
(2) 1-8 hr - SB/LPZ	3.8E-4	3.8E-4
(3) 8-16 hr - SB/LPZ	1.0E-4	1.0E-4
(4) 16 hr-3 days - LPZ	3.4E-5	3.4E-5
(5) 3-26 day - LPZ	7.5E-6	7.5E-6
IV. Dose Data:		
A. Method of dose calculation	Reference 3	Reference 4
B. Dose conversion assumptions	Reference 3	Reference 4
C. Peak activity concentrations in condenser	Table 15.4-13	Table 15.4-16
D. Doses	Table 15.4-15	Table 15.4-18

*Applicant to Supply

Table 15.4-13

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.1E 03	2.1E 03	2.0E 03	1.5E 03	1.2E 02
I132	3.6E 05	3.1E 03	2.7E 03	2.0E 03	1.1E 03	3.2E 02	9.4E 01	2.5E 00	7.5E-10	0.
I133	3.3E 03	3.3E 03	3.2E 03	3.1E 03	2.9E 03	2.6E 03	2.2E 03	1.5E 03	1.3E 02	9.5E-08
I134	5.6E 03	3.8E 03	2.6E 03	1.2E 03	2.4E 02	1.0E 01	4.2E-01	3.1E-05	0.	0.
I135	4.7E 03	4.5E 03	4.2E 03	3.8E 03	3.1E 03	2.0E 03	1.3E 03	3.7E 02	1.8E-01	0.
Total I	1.9E 04	1.7E 04	1.5E 04	1.2E 04	9.5E 03	7.0E 03	5.7E 03	3.9E 03	1.6E 03	1.2E 02
Kr83m	2.5E 04	2.1E 04	1.8E 04	1.2E 04	5.7E 03	1.3E 03	2.8E 02	3.2E 00	5.8E-12	0.
Kr85m	6.1E 04	5.6E 04	5.2E 04	4.5E 04	3.3E 04	1.8E 04	9.5E 03	1.5E 03	2.0E-02	0.
Kr85	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.5E 03	1.5E 03	1.2E 03
Kr87	1.2E 05	9.5E 04	7.3E 04	4.2E 04	1.4E 04	1.6E 03	1.8E 02	2.5E 01	0.	01
Kr88	1.8E 05	1.6E 05	1.4E 05	1.1E 05	6.6E 04	2.4E 04	9.1E 03	4.6E 02	7,8E-06	0.
Kr89	1.8E 05	3.1E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe131m	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe133m	6.1E 04	6.1E 04	6.0E 04	6.0E 04	5.8E 04	5.5E 04	5.2E 04	4.4E 04	1.7E 04	4.1E 00
Xe133	3.6E 05	3.5E 05	3.5E 05	3.5E 05	3.5E 05	3.4E 05	3.3E 05	3.1E 05	2.0E 05	5.1E 03
Xe135m	9.7E 04	2.6E 04	6.7E 03	4.4E 02	1.9E 00	3.6E-05	6.9E-10	9.	0.	0.
Xe135	6.5E 04	6.2E 04	6.0E 04	5.6E 04	4.8E 04	3.5E 04	2.6E 04	1.0E 04	4.3E 01	0.
Xe137	3.9E 05	2.1E 03	9.2E 00	1.8E-04	7.0E-14	0.	0.	0.	0.	0.
Xe138	4.3E 05	1.0E 05	2.4E 04	1.3E 03	3.6E 00	2.9E-05	2.4E-10	0.	0.	01
Total NG	2.0E 06	9.4E 05	7.9E 05	6.8E 05	5.7E 05	4.8E 05	4.3E 05	3.7E 05	2.2E 05	6.5E 03

Table 15.4-14

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO ENVIRONMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	1.5E-02	4.6E-01	9.1E-01	1.8E 00	3.6E 00	7.2E 00	1.1E 01	2.1E 01	7.3E 01	2.1E 02
I132	2.5E-02	7.0E-01	1.3E 00	2.3E 00	3.5E 00	4.5E 00	4.8E 00	4.9E 00	4.9E 00	4.9E 00
I133	2.3E-02	6.9E-01	1.4E 00	2.7E 00	5.2E 00	9.8E 00	1.4E 01	2.3E 01	4.0E 01	4.1E 01
I134	3.9E-02	0.8E-01	1.6E 00	2.4E 00	2.9E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00
I135	3.3E-02	9.5E-01	1.9E 00	3.5E 00	6.4E 00	1.1E 01	1.3E 01	1.7E 01	1.9E 01	1.9E 01
Total I	1.4E-01	3.8E 00	7.1E 00	1.3E 01	2.2E 01	3.5E 01	4.6E 01	6.9E 01	1.4E 02	2.8E 02
Kr83m	1.8E-01	4.9E 00	8.9E 00	1.5E 01	2.2E 01	2.7E 01	2.8E 01	2.8E 01	2.8E 01	2.8E 01
Kr85m	4.2E-01	1.2E 01	2.4E 01	4.4E 01	2.7E 00	1.2E 00	1.4E 02	1.6E 02	1.6E 02	1.6E 02
Kr85	1.1E-02	3.3E-01	6.5E-01	1.3E 00	2.6E 00	5.2E 00	7,8E 00	1.6E 01	6.1E 01	4.0E 02
Kr87	8.7E-01	2.3E 01	4.0E 01	6.4E 01	8.5E 01	9.4E 01	9.5E 01	9.5E 01	9.5E 01	9.5E 01
Kr88	1.2E 00	3.5E 01	6.6E 01	1.2E 02	1.9E 02	2.6E 02	2.8E 02	3.0E 02	3.0E 02	3.0E 02
Kr89	1.4E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00
Xe131m	1.1E-02	3.2E-01	6.3E-01	1.3E 00	2.5E 00	5.0E 00	7.5E 00	1.5E 01	5.4E 01	2.0E 02
Xe133m	4.3E-01	1.3E 01	2.5E 01	5.0E 01	9.9E 01	1.9E 02	2.8E 02	5.2E 02	1.4E 03	1.9E 03
Xe133	2.5E 00	7.4E 01	1.5E 02	2.9E 02	5.9E 02	1.2E 03	1.7E 03	3.3E 03	1.1E 04	2.5E 04
Xe135m	6.9E -01	1.2E 01	1.4E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01
Xe135	4.5E-01	1.3E 01	2.6E 01	5.0E 01	9.3E 01	1.6E 02	2.1E 02	3.0E 02	3.5E 02	3.5E 02
Xe137	3.0E 00	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01
Xe138	3.0E 00	4.9E 01	6.0E 01	6.3E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01
Total NG	1.4E 01	2.6E 02	4.4E 02	7.4E 02	1.3E 03	2.1E 03	2.9E 03	4.9E 03	1.3E 04	2.8E 04

TABLE 15.4-15

CONTROL ROD DROP ACCIDENT
(DESIGN BASIS ANALYSIS)
Radiological Effects

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	0.22	2.55
Low Population Zone	0.16	4.08

SECTION 15.5

CONTENTS

TABLES

ILLUSTRATIONS

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent HPCS Startup

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCS system is postulated for this analysis (i.e., operator error).

15.5.1.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.5.1.2 Sequence of Events and Systems Operation

15.5.1.2.1 Sequence of Events

Table 15.5-1 lists the sequence of events for Figure 15.5-1.

15.5.1.2.1.1 Identification of Operator Actions

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCS has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

15.5.1.2.2 System Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant

15.5.1.2.2 System Operation (Continued)

instrumentation and controls--specifically, the pressure regulator and the vessel level control which respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this transient event.

The system is assumed to be in the manual flow control mode of operation.

15.5.1.3 Core and System Performance

15.5.1.3.1 Input Parameter and Initial Conditions

The water temperature of the HPCS system was assumed to be 40°F with an enthalpy of 11 Btu/lb.

Inadvertent startup of the HPCS system was chosen to be analyzed, since it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.3.2 Results

Figure 15.5-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the upper core plenum. Within 3 sec, the full HPCS flow is established at approximately 5.1% of the rated feedwater flow rate. This flow is nearly 102% of the HPCS flow at rated pressure. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the upper plenum causes a reduction in steam flow, which results in some depressurization as the pressure regulator responds to the event. In the automatic flow control mode, following a momentary decrease, neutron power settles out at a level slightly above operating level. In manual mode the flux level settles out slightly below operating level. In either case, pressure and thermal variations are relatively small and no significant consequences are experienced. MCPR remains well above the safety limit and, therefore, fuel thermal margins are maintained. Therefore, this event does not have to reanalyzed for specific core configurations.

15.5.1.3.3.1 Consideration of Uncertainties (Continued)

worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15.5-1 indicates a slight pressure reduction from initial conditions; therefore, no further evaluation is required as RCPB pressure margins are maintained.

15.5.1.5 Radiological Consequences

Since no activity is released during this event, a detailed evaluation is not required.

15.5.2 Chemical Volume Control System Malfunction (or Operator Error)

This section is not applicable to BWR. This is of PWR interest.

15.5.3 BWR Transients Which Increase Reactor Coolant Inventory

These events are discussed and considered in Sections 15.1 and 15.2.

SECTION 15.6

CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the MSLIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and reactor protection system action and ESF action is given in Sections 6.3, 7.3 and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

15.6.4.3 Core and System Performance

Quantitative results (including math models, input parameters, and consideration of uncertainties) for this event are given in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

Refer to Subsection 16.5.6.3.2. 15.2

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to Section 6.3 for ECCS analysis.

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3 (Secondary Containment Functional Design).

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSLIVs:

(1) the reactor is operating at the power level associated with maximum mass release;

(2) nuclear system pressure is 1040 psia and remains constant during closure;

(3) an instantaneous circumferential break of the main steamline occurs;

(4) isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 5.5 sec;

15.6.5.2.2 Systems Operations (Continued)

flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any Reactor and Plant Protection System are discussed in Sections 6.2, 6.3, 7.3, 7.6 and 8.3, and Appendix 15A.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide conservative assessment of the expected consequences of this very improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.3, 7.3, 7.6, 8.3 and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-1. For the LOCA analysis, the 15.2

reactor is initialized at conditions corresponding to 105% of rated steam flow (see Table 6.3-1). At this condition the total power is greater than the 10CFR50 Appendix K requirement of 102% of rated power. These conditions maximize the system inventory loss during the transient by producing a conservatively high system pressure and break flow. This additional inventory loss will produce a longer period of core uncovery resulting in a conservatively high peak cladding temperature calculation for the LOCA analysis. 15.2

15.6.5.3.3 Results

Results of this event are given in detail in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological input is minimized and within limits. Continued operator control and surveillance is examined and guaranteed.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of the largest single primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Sections 3.8.2.3, 3.6, and 6.2 for details and results of the analyses).

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet

15.6.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown from Pipes", ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143).

3. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

4. F. J. Prutschy, G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup", August 1972 (NEDO-10585).

5. D. G. Weiss and V. O. Nguyen, "Control Room Accident Exposure Evaluation - CRDS Program", January 1979 (NEDO-23909)

SECTION 15.6

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15.7.1.1.2.2 Identification of Operator Actions

Gross failure of this system may require manual isolation of this system from the main condenser. This isolation results in high condenser pressure and a reactor scram. The operator should monitor the turbine-generator auxiliaries and break vacuum as soon as possible. The operator should notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for reentry. The time needed for these actions is about 2 min.

15.7.1.1.2.3 Systems Operation

In analyzing the postulated offgas system failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for functioning of normally operating plant instruments and controls and other systems only in assuming the following:

(1) capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by Area Radiation Monitoring System, in an alarmed loss of flow in the Offgas System, and in an alarmed increase in activity at the vent release;

(2) capability to isolate the system and shutdown the reactor; and

(3) operational indicator and annunciators in the main control room.

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation, necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Subsection 15.2.5.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the offgas system pressure boundary. No credit is taken for performance of secondary barriers, except to the extent inherent in the assumed equipment release fractions discussed in Subsection 15.7.1.1.5.

15.7.1.1.5 Radiological Consequences

15.7.1.1.5.1 General

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

15.7.1.1.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of direct release of fission products to the environment from the failed component through the building ventilation system. The release of activity to the environment is presented in Table 15.7-4.

15.7.1.1.5.2.3 Results

The calculated exposures for the Design Basis Analysis are presented in Table 15.7-5 and are well within the guidelines of 10CFR100. These results apply to all BWR/6 core configurations for plants with a Gaseous Waste Management System similar to that described in Section 11.3.

15.7.1.1.5.3 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-2.

15.7.1.1.5.3.1 Fission Product Release

15.7.1.1.5.3.1.1 Initial Conditions

The activity in the offgas system is based on the following normal operating conditions:

(1) 30 SCFM air inleakage, and
(2) 100,000 μCi/sec Noble Gas after 30-min delay.

The activity stored in the various equipment pieces before the postulated failure is given in Table 12A-1 (Appendix 12A).

15.7.1.1.5.3.1.2 Assumptions

The only credible failure that could result in loss of carbon from the vessels is the failure of the concrete structure surrounding the vessel. A circumferential failure of the vessel could result from concrete falling on the vessel in either of two ways:

(1) Pending Load - the vessel being supported in the center and loaded on each end. This could result in a tear around 50% of the circumference.

(2) Shearing Load - the vessel being supported and loaded near the same point from above.

In either case, no more than 10-15% of the carbon would be displaced from the vessel. Iodine is strongly bonded to the charcoal and would not be expected to be removed by exposure to the air. However, the conservative assumption is made that 1% of the iodine activity contained in the absorber tanks is released to the vault containing the offgas equipment.

Measurements made at KRB indicate that offgas is about 30% richer in Kr than air. Therefore, if this carbon is exposed to air, it will eventually reach equilibrium with the noble gases in the air. However, the first few inches of carbon will blanket the underlying carbon from the air. A 10% loss of noble gas activity from a failed vessel is conservative because of the small fraction of carbon exposed to the air.

Prefilters: Because of the design features of the prefilter vessel (approximately 24 in. diameter, 4 ft. height, 350 psig design pressure, 1/2 in. wall thickness and collapsible filter media), a failure mechanism cannot be postulated that will result in emission of filter media or daughter products from this vessel. However, to illustrate the consequences of a radioactivity loss from this vessel, 1% release of particulate activity is assumed.

15.7.1.1.5.3.1.2 Assumptions (Continued)

Holdup Pipe: Pipe rupture and depressurization of the pipe is considered. Normally, the pipe will operate at less than 16 psia and depressurize to 14.7 psia. The possible loss of solid daughters and noble gases and iodines is conservatively taken as 20%. The model used assumes retention and washout of 60% of the particulate daughters for the calculation of the holdup pipe inventory.

Piping: It is assumed that the seismic event causing the pipe failure is accompanied by a reactor isolation, stopping steam flow to the steam jet air ejectors. Therefore, the resulting release from failed piping is not significant compared to those failures previously considered.

15.7.1.1.5.3.2 Fission Product Transport to the Environment

The release of activity to the environment is presented in Table 15.7-6.

15.7.1.1.5.3.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-7. These results apply to all BWR/G core configurations for plants with a Gaseous Waste Management System similar to that described in Section 11.3.

15.7.1.2 Malfunction of Main Turbine Gland Sealing System

(Applicant to supply.)

15.7.1.3 Failure of Main Turbine Steam Air Ejector Lines

(Applicant to supply.)

15.7.2 Liquid Radioactive System Failure

15.7.2.1 Identification of Causes and Frequency Classification

15.7.2.1.1 Identification of Cause

The event which could cause a failure in the liquid radwaste system is a liquid radwaste tank rupture by a seismic occurrence.

Although the system consists of Non-Seismic Category I Equipment, the liquid radwaste tanks are constructed in accordance with sound engineering principles. Therefore, simultaneous failure of all of the tanks is unlikely. However, for purposes of this analysis, a simultaneous failure releasing the contained liquid activity of all tanks is assumed.

Radwaste equipment/component failure could occur in either some combination of storage facilities, such as storage tankage, or in processing equipment such as pumps, valves, etc. Failure of storage tanks would be indicated by a rapid loss of level where no process has been initiated. Tank water contents on the building floor would initiate sump pump activity and area radiation alarms, due to spreading of the contents to the drains and other areas. Sump pump run times would indicate relatively large volumes are involved. 15.5

Failure of processing equipment would be shown to the operator at the time that equipment was required to perform. Processing indications, including levels and flows, would fail to respond to the initiation indicating that a component or combination of components has not operated. The component failure would be identified by observing all variables involved in the process. Those operating and responding would determine process viability and pinpoint the failed component.

15.7.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.2.5 Radiological Consequences

15.7.2.5.1 General

Two radiological analyses are provided for this accident:

(1) The "design basis analysis" is based upon conservative assumptions considered to be acceptable to the NRC for the purpose of determining design adequacy to meet 10CFR100 guidelines.

(2) The conservative "realistic analysis" is considered to provide a realistic estimate of radiological consequences.

15.7.2.5.2 Design Basis Analysis

The liquid radwaste tank failure analysis is evaluated in accordance with the following parameters:

(1) simultaneous rupture of the liquid radwaste tanks and release of all liquid contents;

(2) 10% of total iodine inventory released becomes airborne for release to environs;

(3) release takes place over 2-hr period;

(4) Atmospheric dispersion is 5 percentile probable χ/Q; and

(5) χ/Q at the site boundary is 2.0E-3 sec/m^3.

15.7.2.5.2.1 Fission Product Release

The activity contained as I-131, 132, 133, 134, and 135 in the major radwaste tanks liquid is shown in Table 15.7-8. Activity content is based upon the design basis source term of 100,000 μCi/sec. Tank volumes are presented in Section 11.2. The computational methods for determining the values in Table 15.7-8 are given in Reference 1a. 15.6

15.7.2.5.2.2 Fission Product Transport to the Environment

It is conservatively assumed that the activity listed in Table 15.7-9 is released from the building at ground level.

15.7.2.5.2.3 Results

The resultant thyroid inhalation exposures from the iodine activity released to the enviornment are listed in Table 15.7-10. Since very little noble gas activity is released, the whole body dose is negligible. It should be noted that the assumption of release to the environment of 10% of the iodine activity contained in the radwaste tanks, using 5% probable χ/Q will undoubtedly result in an overestimate of real exposure by a factor to 10 to 100. However, exposures are well within the guidelines of

15.7.2.5.2.3 Results (Continued)

10CFR100. These results apply to all BWR/6 core configurations for plants with a Liquid Waste Management System similar to that described in Subsection 11.2.

15.7.2.5.3 Realistic Analysis

Parameters used in the design basis analysis would be pertinent to the realistic analysis with the following exception:

(1) 1% of total iodine inventory released becomes airborne for release to environs, and

(2) only the concentrated waste tank (greatest iodine content) is ruptured and releases all liquid contents.

15.7.2.5.3.1 Fission Product Release

The activity contained in the concentrated waste tank is as shown in the appropriate row and column in Table 15.7-8.

15.7.2.5.3.2 Fission Product Transport to the Environment

It is conservatively assumed that the activity listed in Table 15.7-11 is released from the building at ground level.

15.7.2.5.3.3 Results

Radiological effects from a realistic basis reduces the dose over the design basis effect by a factor of about 20 due to less iodine released from a single tank. Results from the design basis (Table 15.7-10) are already substantially under 10CFR100 guidelines. These results apply to all BWR/6 core configurations for plants with a Liquid Waste Management System similar to that described in Section 11.2. The computational methods for determining the values in Table 15.7-10 are given in Reference 1b. 6

15.7.3.4 Design Basis Accident

The design basis accident is based on a conservative assessment of this accident. Two release pathways are analyzed. An airborne release is analyzed in which 10% of the iodine inventory is assumed to be released to the environment, and a surface water release is analyzed in which 90% of the concentrated waste tank is assumed to be released directly to the surface water. The specific models, assumptions and programs used for computer evaluation are described in References 1 and 2. Specific values of parameters used in the evaluation are presented in Table 15.7-12.

15.7.3.4.1 Fission Product Release

The fission product release is identified in Subsection 15.7.3.5.1 and is based on an offgas release rate of 100,000 μCi/sec at 30 minutes.

15.7.3.4.2 Fission Product Release to the Environment

Tables 15.7-13 and 15.7-14 present the information on activity released to the environment.

15.7.3.4.3 Results

Table 15.7-15 provides the airborne radiological effects from this event. It should be noted that the referenced computer program which is used to evaluate the radiological consequences of this event is based on the assumption that the activity in the aquatic life is at equilib ium levels. Since this assumption will result in an over-estimate of the actual consequence, the radiological doses in Table 15.7-16 are considered to be very conservative.

he results apply to all BWR/6 core configurations for plants with a Liquid Waste Management System similar to that described in Section 11.2.

15.7.3.5 Realistic Analysis

The realistic analysis is based on a realistic (but still conservative) assessment of this accident. The specific models, assumptions and the program used for computer evaluation are also described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-12.

15.7.3.5.1 Fission Product Release

The fission produce release is based on an offgas release rate of 100,000 μCi/sec at 30-min decay.

15.7.3.5.2 Fission Product Transport to the Environment

Table 15.7-17 presents the information on activity released to the environment.

15.7.3.5.3 Results

It should be noted that the referenced computer program which is used to evaluate the radiological consequences of this event is based on the assumption that the activity in the aquatic life is at equilibrium levels. Since this assumption will result in an overestimate of the actual consequence, the radiological doses in Table 15.7-18 are considered to be very conservative. These results apply to all BWR/6 core configurations for plants with a Liquid Waste Management System similar to that described in Section 11.2.

15.7.4.2.2 Identification of Operator Actions (Continued)

(3) the fuel-handling foreman should make the operations shift engineer aware of the a :ident;

(4) the shift engineer should immediately determine if the normal ventilation system has isolated and the standby gas treatment is in operation;

(5) the shift engineer should initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the reactor building;

(6) the plant superintendent or delegate should determine if the standby gas treatment system is performing as designed;

(7) the duty shift engineer should post the appropriate radiological control signs at the entrance of the reactor building; and

(8) before entry to the refueling building is made, a careful study of conditions, radiation levels, etc., will be performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts.

To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

The fuel assembly is expected to impact on the spent fuel racks at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based on 1% uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled

15.7.4.5 Radiological Consequences (Continued)

power condition is assumed because it is not expected that fuel handling can begin within 24 hr following initiation of reactor shutdown. Figure 15.7-1 indicates the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

The Design Basis Analysis is based on Regulatory Guide 1.25. The specific models, assumptions and the program used for computer evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.7-19.

15.7.4.5.1.1 Fission Product Release from Fuel

Per the conditions in Regulatory Guide 1.25, the following conditions are assumed applicable for this event:

(1) Power Level - 3651 MWt for 3 years

(2) Plenum Activity - 10% of the radioactivity for iodine and noble gases except Kr-85 and 30% for Kr-85.

(3) Fission Product Peaking Factor - 1.5 for those rods damaged.

(4) Activity Released to Fuel Building - 10% of the noble gas activity and 0.1% for the iodine activity.

Based on the above conditions, the activity released to the fuel building is presented in Table 15.7-20.

15.7.4.5.1.2 Fission Product Transport to the Environment

Also, per the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the fuel building (Table 15.7-20) is released to the environment over a 2-hr period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-21.

15.7.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-22 and are well within the guidelines of 10CFR100. These results are applicable to all BWR/6 core configurations.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-19.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission release estimates for the fuel-handling accident are based on the following assumptions:

(1) The reactor fuel has an average irradiation time of 1000 days at NBR up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hr decay period allows time to shut down the reactor, depressurize the nuclear system, remove the reactor vessel head and remove the reactor vessel upper internals.

15.7.4.5.2.1 Fission Product Release from Fuel (Continued)

It is not expected that these operations could be accomplished in less than 24 hr and probably will require at least 48 hrs.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments (Reference 4).

(3) Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products is assumed to be released.

(4) It is assumed that 101 fuel rods fail. This is considered to be conservative because it is expected that much less than 101 rods would be damaged.

15.7.4.5.2.2 Fission Product Transport to the Environment

The following assumptions and conditions are assumed in calculating the release of activity to the environments.

(1) The fission product activity released to the refueling building will be in proportion to the removal efficiency of the water in the fuel pool. Because water has a negligible effect on removal of the noble gases, the gases are assumed to be instantaneously released from the pool to the building.

(2) The iodine activity airborne is in proportion to the partition factor and the ratio of the volume of air (V_a) to the volume of water (V_w) for which the respective

15.7.4.5.2.2 Fission Product Transport to the Environment (Applicability to be confirmed by Applicant) (Continued)

values are applicable. It is assumed that a partition factor of 100 and a V_a/V_w of 3 is applicable for this event. It should be noted that the volume assumed for V_a is not equal to the total volume of air in the refueling building, but is nevertheless considered to be a conservative estimate of the volume of air which may form an equilibrium condition with the activity in the fuel storage pool.

(3) The ventilation rate from the refueling building to the environment via the SGTS is 9 air changes per day. Based on these assumptions, the activity airborne in the refueling building is shown in Table 15.7-23.

Due to isolation of the refueling building and initiation of the SGTS, the release rate to the environment is 9 air changes per day. Considering an SGTS efficiency for iodine of 99.9%, the integrated activity discharged to the environment is presented in Table 15.7-24.

15.7.4.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-25 and are well below the guidelines set forth in 10CFR100. These results are applicable to all BWR/6 core configurations.

15.7.5 Spent Fuel Cask Drop Accident

15.7.5.1 Identification of Cause

Due to the redundant nature of the crane, the cask drop accident is not believed to be a credible accident. However, the accident

SECTION 15.8

CONTENTS

APPENDIX 15B

BWR/6 GENERIC ROD WITHDRAWAL ERROR ANALYSIS

15B.3 RWL SYSTEM OVERVIEW

15B.3.1 RWL and RBM System Comparison

The RWL System proposed for BWR/6 performs the same function as the RBM System on earlier BWR product lines (i.e., blocking an inadvertent rod withdrawal such that the RWE design criteria are not violated). However, the hardware and operations of the two systems are significantly different. A brief discussion of these differences follows.

The sequence of events prior to rod block for the RBM System is described in Table 15B-1. Basically, the system consists of two redundant RBM channels, each receiving input signals from up to eight LPRMs surrounding the selected control rod. Each RBM channel signal is the average of the input LPRM signals. Prior to rod withdrawal, each RBM channel reading is normalized to an assigned Average Power Range Monitor (APRM) channel reading. After normalization and subsequent rod withdrawal, a rod block occurs if either RBM channel signal reaches a preset rod block trip setpoint.

The dual-channel RWL System does not require direct core response feedback from the LPRMs during a rod withdrawal. Instead, the distance a rod is withdrawn is monitored by position indicator switches situated along the control rod drive (CRD) mechanism. The relationship between withdrawal distance and the margin to fuel safety limits is analytically determined. The allowable rod withdrawal distance as a function of core power is set such that there is a high degree of confidence the $\Delta MCPR_{DB}$ and $\Delta LHGR_{DB}$ are not violated. When an operator attempts to withdraw a rod further than the prespecified withdrawal distance, the rod is blocked. The time sequence of events for a BWR/6 RWE transient is given in Table 15B-2.

15B.3.2 RWL System Operational Description

The RWL System blocks rod withdrawals at prespecified, power-dependent increments. This system is operational between the Low Power Setpoint (LPSP) (20^{+18}_{-0}% of rated power) and rated power. [Below the LPSP, rod pattern restrictions are enforced by the Banked Position Withdrawal Sequence (BPWS)[2]]. A High Power Setpoint (HPSP) is established at $\leq$70% core power. Between the LPSP and the HPSP, rod withdrawals are limited to 2 ft, while between the HPSP and rated power, withdrawals are limited to 1 ft. These withdrawal restrictions were established by the generic BWR/6 RWE analysis discussed in this appendix.

15B.3.3 Why Replace the RBM System?

The concept of ganged control rods was introduced with BWR/6. A gang consists of a maximum of four control rods that can be selected and withdrawn simultaneously. This improves plant startup times and fuel performance, since symmetrical radial power shapes can be maintained during power changes.

The advantages of ganged rod withdrawals complicate a RBM System approach. Instead of monitoring the neutron flux increase around a single rod, it would be necessary to monitor the response around up to four rods. This would require the analysis of up to 64 LPRM input signals. The proper treatment of all combinations of instrument failure and response would require a more complex hardware system and supporting analysis than on pre-BWR/6 plants.

Since the RWL System blocks strictly on incremental distance withdrawn, all concerns relative to LPRM instrument response are eliminated. In addition, the required hardware logic is simplified.

The two systems are equivalent from an analytical standpoint, since setpoints are based on calculated ΔMCPRs in both cases. Although

15B.4.3.2.1.1 Rod Pattern Development (Continued)

with equilibrium xenon distributions consistent with the power/flow operating state. The only constraints were: (1) MCPR $\geq$ operating limit MCPR at rated conditions (1.18); (2) MLHGR $\geq$ 13.4 kW/ft; and (3) the calculated neutron multiplication factor equal to the projected critical k_{eff} ± 0.005. Once these three constraints were satisfied, no further attempt was made to flatten the radial power shape to increase margins to thermal limits. Thus, at low powers the rod search module is essentially unconstrained and the rod patterns were not optimized to achieve favorable IMCPR and ΔMCPR performance.

Projected through-the-cycle rod patterns (1000 MWd/t intervals) at rated conditions were the basis for the history-dependent exposure distributions, as opposed to worst-case distributions. These exposure distributions had been optimized to meet thermal margins and reactivity requirements. The rod patterns were consistent with current BWR operating philosophy.

The rod pattern search module was initialized at random exposure points and power/flow conditions. A nonoptimized initial rod pattern was input to the rod pattern search module, and the margin to constraints was checked to determine if any rod pattern adjustments were required. The final, nonoptimized rod pattern represented expected short-term MCPR capability, assuming the core accumulated the greatest portion of its exposure with optimized rod patterns. This is consistent with normal operations wherein many rod pattern adjustments are made at low powers as the core is maneuvered up to rated power, with the major exposure accumulation at steady-state, near rated conditions.

15B.4.3.2.1.2 IMCPR Database

The output of Step 1 is a database consisting of sampled values of IMCPR at various power and flow conditions (Figure 15B-2 and

15B.4.3.2.1.2 IMCPR Database (Continued)

Attachment A). A mixture of core sizes, cycles, cycle exposures and rod sequences are included (Table 15B-3). A total of 71 datapoints are included in the database. Biases introduced by the various parameters are discussed in Subsection 15B.4.3.2.1.3. To establish that the statistically determined value of IMCPR as a function of power and flow is converged (i.e., the database is of sufficient size), the statistical model (Subsection 15B.4.3.2.2) was updated periodically to incorporate additional data. Figure 15B-3 shows the fluctuations in the nominal value of IMCPR at several power and flow conditions as a function of the number of datapoints. No core parameter or calculational uncertainties or biases were considered in these sampled IMCPR values. As the number of datapoints increased, the nominal IMCPR approached constant values. As more data were added, these values tended to oscillate. Once this oscillation was obtained at all power and flow conditions ($\gtrsim$55 datapoints in Figure 15B-3), the database was considered complete.

15B.4.3.2.1.3 IMCPR Database Biases

Biases associated with the variable core parameters (i.e., core size, core average exposure, fuel cycle, rod sequence and core average enrichment) were qualitatively evaluated using crossplots of residuals from the statistical fit of the IMCPR database (Subsection 15B.4.3.2.2) versus the subject parameter (Attachment B). The fit residual is the difference between the observed value and the corresponding estimate of the mean divided by the corresponding estimate of the standard deviation.

Crossplots provide information on how well the assumed relationship fits the data and how variables in the model affect a dependent variable. Lack of fit (i.e., random scatter) in a crossplot indicates that no biases exist in the database. Note that all crossplots of Attachment B, except Figure 15BB-1, are random scatter

15B.4.3.3.4 Optimum Withdrawal Increments (Continued)

Below 20% power, the RWL System is not functional. Instead, rod movements are restricted by the Rod Pattern Control System (RPCS) enforcement of the Banked Position Withdrawal Sequence (BPWS). The BPWS requires rod groups to be banked at 1, 2, 3 and 12 ft out. Thus, the maximum withdrawal distance under BPWS constraints is 9 ft. This corresponds to a 99% probability/50% confidence that the safety limit MCPR will not be violated for a random RWE initiated at ≤38% power using Section 15BF.4 methodology. Thus, the BPWS assures an acceptable RWE response.

15B.4.3.3.5 IMCPR Technical Specification as a Function of Core Power

With the RWL withdrawal increments fixed and the ΔMCPR response known, Equation 15B.4-7 was used to calculate the required IMCPR technical specification to protect against the RWE transient. The results of this calculation are given in Table 15B-6 and graphically presented in Figure 15B-15. Comparison of this curve to Figure 15B-6 indicates that BWR/6 should have minimum difficulty in establishing rod patterns which satisfy the $IMCPR_{TS}$.

15B.4.3.3.6 Rod Movement Restriction Technical Specification

The input power signal to the RWL System originates from the first-stage turbine pressure. When operating with steam bypass, this signal gives a biased power indication. This can result in greater withdrawal distances being allowed than the design and licensing basis support. For example, for a core operating at 50% power with 35% bypass, the input power signal corresponds to ∿15% power. Since this power is below the low power setpoint (LPSP) (typically 20% of rated power), the RPCS is functional instead of the RWL System. The RPCS enforces the BPWS, which allows a maximum 9-ft withdrawal. Thus, instead of rod withdrawals limited to 2 ft at 50% power, the potential exists for a 9-ft withdrawal.

15B.4.3.3.6 Rod Movement Restriction Technical Specification (Continued)

If a 9-ft withdrawal did occur, however, the probability of violating the safetly limit MCPR is small. At 30% power/36% flow, $IMCPR_{95/50}$ is 2.41 (Figure 15B-6). From Equations 15B.4-3 and 15B.4-4, $(\Delta MCPR/IMCPR)_{95/50}$ is 0.48. Thus, the final MCPR following a 9-ft withdrawal at 30% power is 1.25 (2.41 - 0.48 * 2.41), which is greater than the safety limit MCPR. Above 30% power, the operator is typically in a power-shaping mode. In this case, the rod pattern will most likely not conform to BPWS requirements. As a result, when operating above 30% power with sufficient bypass to result in a RPCS input power signal less than the LPSP, violation of BPWS constraints will result in both insert and withdrawal blocks on all rods.

To ensure that the above situation does not occur, the following technical specification is required:

> "Do not withdraw control rods when operating above the LPSP with steam bypass."

A detailed evaluation of this power signal bias and its impact is provided in Attachment F.

15B.4.3.3.7 Comparison of Generic and Deterministic RWE Analyses

Plant-specific deterministic RWE analyses are typically performed only at rated conditions. Several plant-specific analyses have been completed in support of early BWR/6 FSARs prior to the completion of the generic analysis. The allowable withdrawal distances from these plant-specific analyses (Table 15B-7) are in general agreement with the 1-ft withdrawal distance from the generic analysis. Thus, at rated conditions, the overall conservatism in the generic analysis is equivalent to the conservatism inherent in the deterministic analysis.

19.1.3 Chapter 3 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	210.1	3.1	Subsections 3.6.1.1.4, 3.6.2.2.1, 3.6.2.3.1 and 3.6.2.3.2.2	4
	210.2	3.2	Tables 3.9-11 and 3.9-12	
	220.1	3.3	Section 3.7.2.6	
	220.2	3.4	Subsection 19.3.3.3	
	220.3	3.5	Subsection 19.3.3.5	
	220.4	3.6	Table 3.8-3 and Subsection 19.3.3.6	
	220.5	3.7	Subsection 3.8.2.5	
	220.6	3.8	Subsection 19.3.3.8	
	220.7	3.9	Subsection 19.3.3.9	
	220.8	3.10	Subsection 19.3.3.10	
	241.2	3.11	Subsection 19.3.3.11	4
	241.3	3.12	Response in 9/82*	
	241.4	3.13	Response in 9/82*	
	241.5	3.14	Response in 9/82*	
	241.6	3.15	Response in 9/82*	
	241.7	3.16	Response in 9/82*	
	241.8	3.17	Response in 9/82*	
	241.9	3.18	Response in 9/82*	
	241.10	3.19	Subsection 19.3.3.19	4
	241.11	3.20	Response in 9/82*	
	241.12	3.21	Response in 9/82*	
	241.13	3.22	Response in 9/82*	
	241.14	3.23	Response in 9/82*	
	241.15	3.24	Subsection 3A.1.2	4
	241.16	3.25	Subsection 3A.1.2	4
	241.17	3.26	Response in 9/82*	
	241.18	3.27	Response in 9/82*	
	241.19	3.28	Response in 9/82*	
	241.20	3.29	Response in 9/82*	
	241.21	3.30	Response in 9/82*	
	241.22	3.31	Subsection 19.3.3.31	5
	241.23	3.32	Response in 9/82*	
	241.24	3.33	Response in 9/82*	
	241.25	3.34	Subsection 19.3.3.34	5
	241.26	3.35	Subsection 3A.5.2	5
Note 2	251.1	3.36	Subsection 3.5.1.3	4

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19.1.3 Chapter 3 - Question/Response Index (Continued)

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	270.1	3.37	Tables 3.11-2 through 3.11-9	5
↑	270.2	3.38	Subsection 3.11.4	5
	270.3	3.39	Subsection 3.11.2.1.3	5
↓	270.4	3.40	Subsection 3.11.2.1.1	5
Note 2	371.1	3.41	Table 3.10-1	4

**Environmental Qualification

Chapter - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

3. See Section 3BO.1 for Appendix 3B Question/Response Index.