ATTACHMENT 1

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: ACCIDENT EVALUATION

Listing of Item Numbers by
Technical Specification Problem Sheet (TSPS) Number

TSPS No.	Item Nos.*
275	1.B.01

*Item number format: 1.A.02

- 02 — Item number within category
- A — Category designator
- 1 — Attachment number

A. TYPOGRAPHICAL ERRORS, EDITORIAL CHANGES, AND CLARIFICATIONS

No technical specification changes in this category are included with this attachment.

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

The following change is proposed to render the technical specification consistent with the as-built plant. In all such cases, the as-built plant is consistent with the safety analyses and the licensing basis.

In that this proposed change is inherently consistent with the safety analyses and the licensing basis, it is concluded that the proposed change does not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed change does not involve a significant hazards consideration.

A description of this change including justification for the change is provided below:

1. (TSPS 275), Reactor Vessel Water Level, Technical Specifications 3/4.8.1.2, 3/4.9.11.1, 3/4.9.11.2, 3/4.9.8, and Bases 3/4.9.11

   The proposed change affects the specified minimum water level which must be maintained above the reactor pressure vessel flange during OPERATIONAL CONDITIONS 4,5 and "*". Based on the as-built elevation of the reactor vessel flange, the actual depth of water above the flange is at least 22 feet 8 inches, instead of 23 feet. This change reflects the as-built elevation of the flange and does not impact overall pool volume or depth. This change has an insignificant impact on the iodine scrubbing function and heat sink capability of the large water volume. The proposed change also conservatively maintains at least 8.5 feet of water over the top of the active fuel for shielding purposes during irradiated fuel assembly transfer as specified in FSAR Section 12.3.2.2.2.c. This change will not adversely impact plant safety because it is consistent with the FSAR, the safety analysis and the as-built plant. (Page 3/4 8-9, 3/4 9-10, 3/4 9-16, 3/4 9-17, and Bases 3/4 9-2)

C. ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES

No technical specification changes in this category are included with this attachment.

D. REGULATORY REQUIREMENTS/REQUESTS/RECOMMENDATIONS

No technical specification changes in this category are included with this attachment.

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

(AFFECTED PAGES ARE PROVIDED IN THE ORDER OF ASCENDING PAGE NUMBERS.)

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator 11 and/or 12, and diesel generator 13 when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. A day tank containing a minimum of 220 gallons of fuel.
   2. A fuel storage system containing a minimum of:
      a) 48,000 gallons of fuel each for diesel generators 11 and 12.
      b) 39,000 gallons of fuel for diesel generator 13.
   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With all offsite circuits inoperable and/or with diesel generators 11 and/or 12 of the above required A.C. electrical power sources inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than ~~23 feet~~ 22 FEET 8 INCHES above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator 13 of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator 13 to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

*When handling irradiated fuel in the primary or secondary containment.

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

22 FEET 8 INCHES

3.9.8 At least ~~23 feet~~ of water shall be maintained over the top of the reactor pressure vessel flange. 275

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode train of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger train.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to ~~23 feet~~ 22 FEET 8 INCHES above the top of the reactor pressure vessel flange. 275

ACTION:

a. With no RHR shutdown cooling mode train OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode train in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode train of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode trains of the residual heat removal (RHR) system shall be OPERABLE and at least one train shall be in operation,* with each train consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger train.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than ~~23 feet~~ 22 FEET 8 INCHES above the top of the reactor pressure vessel flange. 275

ACTION:

a. With less than the above required shutdown cooling mode trains of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode train.

b. With no RHR shutdown cooling mode train in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode train of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

3/4.9.7 CRANE TRAVEL - SPENT FUEL AND UPPER CONTAINMENT FUEL STORAGE POOLS

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pools ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL AND UPPER CONTAINMENT FUEL STORAGE POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE and in operation or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

22 FEET 8 INCHES

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than ~~23 feet~~ of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and ~~23 feet~~ of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

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3/4.9.12 HORIZONTAL FUEL TRANSFER SYSTEM

The purpose of the horizontal fuel transfer system specification is to control personnel access to those potentially high radiation areas immediated adjacent to the system and to assure safe operation of the system.

ATTACHMENT 2

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: CONTAINMENT/SYSTEMS

Listing of Item Numbers by
Technical Specification Problem Sheet (TSPS) Number

TSPS No.	Item Nos.*	TSPS No.	Item Nos.*
012	2.C.01	172	2.A.08
019	2.A.04	233	2.C.07
020	2.B.01	235	2.D.01
031	2.D.03	240	2.B.03
067	2.D.02	266	2.A.09
069	2.D.04	269	2.A.01
127	2.C.02	276	2.A.02
144	2.A.07	283	2.A.06
164	2.C.04	294	2.C.03
167	2.C.05	306	2.B.02, 2.A.03
168	2.C.10	320	2.C.08
169	2.C.01	379	2.C.09
170	2.C.06	818	2.B.04
171	2.A.05		

*Item number format: 1.A.02

1 — Attachment number
A — Category designator
02 — Item number within category

A. TYPOGRAPHICAL ERRORS, EDITORIAL CHANGES, AND CLARIFICATIONS

These proposed changes correct obvious typographical errors, implement editorial changes such as correction of spelling errors, punctuation errors, and grammatical errors or provide clarification of the basic meaning or intent of the subject technical specifications.

MP&L has determined that the proposed changes do not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration.

A description of these changes including necessary justification for the changes is provided below:

TYPOGRAPHICAL ERRORS

Typographical errors are being corrected by this submittal as listed below. Correction of these typographical errors is purely an administrative change. (See attached revised technical specification pages for exact changes proposed.)

	TSPS No.	TS Page No.
1.	269	3/4 6-17
2.	276	3/4 6-9
3.	306	3/4 6-29

EDITORIAL CHANGES

Proposed editorial changes to the technical specifications are discussed below:

4. (TSPS 019), Drywell Purge System, Technical Specification 3.6.7.3

   The proposed change to delete the word "continued" adjacent to "Surveillance Requirements", corrects an obvious editorial error. (Page 3/4 6-58)

5. (TSPS 171), E61-F056 and E61-F057 Valve Description, Technical Specification Table 3.6.4-1

   The Specification currently labels valves E61-F056 and E61-F057 as "Purge Rad. Detector" isolation valves. The correct description for these valves is "Purge Filter Train Isolation." The proposed change is purely administrative to correct this labeling error. (Page 3/4 6-32)

6. (TSPS 283), Improper Building Name, Technical Specification 5.2.3

   The subject specification states that secondary containment consists of the Reactor Building and the Enclosure Building. In the Grand Gulf design the building surrounding the primary containment is designated the Auxiliary Building rather than Reactor Building. Therefore, Specification 5.2.3 should be revised to state that secondary containment consists of the Auxiliary Building and the Enclosure Building. The proposed change is editorial in that it provides the plant specific designation for the building rather than general terminology. (Page 5-1)

CLARIFICATIONS

Clarifications to the technical specifications to improve understanding and readability are discussed below:

7. (TSPS 144), Primary Containment Integrity Leak Rate Testing Requirements, Technical Specification 3/4.6.1.1

   Surveillance Requirement 4.6.1.1.a, as presently written, states the equipment hatch seals are required to be leak-tested following the closure of any penetration that is subject to Type B testing, (Appendix J, 10CFR50) even if the equipment hatch had not been opened. The words "equipment hatch" should be deleted, thereby clarifying that the intent of the surveillance is to require only those penetrations that have been opened to have their seals leak-tested after the penetration is closed. The proposed change is considered administrative in that its purpose is to clarify the intent of the Surveillance Requirement. (Page 3/4 6-1)

8. (TSPS 172), Drywell Bypass Leakage, Bases 3/4.6.2.2

   This proposed change expands the bases for drywell bypass leakage to:

   a. Provide an explanation of the $A/\sqrt{k}$ term and an appropriate FSAR reference.

   b. Provide an equivalent flow in scfm for the design drywell leakage rate (allowable drywell leakage capability) at 3 psid.

   c. Provide an equivalent flow in scfm for the integrated drywell leakage value at 3 psid.

These additions provide clarification and are consistent with the safety analysis. (Page B 3/4 6-3)

9. (TSPS 266), Reopening of Isolation Valves - MSIVs Excluded, Technical Specification 3/4.6.4

The proposed change to the footnote of the subject technical specification will exclude the Main Steam Isolation Valves (MSIVs) from the provision that permits isolation valves to be reopened under administrative controls, thus rendering this specification consistent with the requirements of Technical Specification 3/4.4.7. This proposed change involves no safety significance as it represents a clarification of the intent of the Technical Specification. (Page 3/4 6-27)

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

The following changes are proposed to render the technical specifications consistent with the as-built plant. In all such cases, the as-built plant is consistent with the safety analyses and the licensing basis.

In that these proposed changes are inherently consistent with the safety analyses and the licensing basis, it is concluded that the proposed changes do not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration.

A description of these changes including justification for the changes is provided below:

1. (TSPS 020), Containment Isolation Valve Testing, Technical Specification Table 3.6.4-1

Revisions to Technical Specification Table 3.6.4-1 are proposed to require pneumatic testing of several valves that previously required hydrostatic testing. MP&L committed to adopt this more stringent method and to apply for these changes in a letter from L. F. Dale to H. R. Denton dated September 12, 1983 (AECM-83/0540). The proposed changes are listed below, by page number:

a. Page 3/4 6-29 - Delete footnote (c) notation for valves E12-F008-A, E12-F009-B, and E12-F023-A.

b. Page 3/4 6-30 - Delete footnote (c) notation for valves E12-F042A-A, E12-F028A-A, E12-F037A-A, E12-F042B-B, E12-F028B-B, E12-F037B-B. Delete footnote (d) notation for valves E12-F021-B and E21-F012-A.

c. Page 3/4 6-34 - Delete footnote (c) notation for valves E12-F027A-A, E12-F027B, E12-F042C-B, E22-F004-C, and E21-F005-A. Delete footnote (d) notation for valves E12-F064C-B and E21-F011-A.

d. Page 3/4 6-37 - Delete footnote (c) notation for valves E12-F308, E51-F066, E12-F344, E12-F044A, E12-F025A, E12-F107A, E12-F025B, E12-F044B, and E12-F107B.

e. Page 3/4 6-38 - Delete footnote (c) notation for valves E12-F234, E12-F041C-B, E22-F005, E22-F218, E22-F201, E21-F006, E21-F200, and E21-F207. Delete footnote (e) notation for valves E12-F280, E12-F281, E21-F217, and E21-F218.

f. Page 3/4 6-39 - Change footnote (e) notation to (c) for valves E51-F251 and E51-F252.

g. Page 3/4 6-40 - Delete footnote (d) notation for valves E12-F036 and E12-F005.

h. Page 3/4 6-42 - Delete footnote (c) notation for valves E12-F002, E12-F342, E12-F061, E12-F056C, E12-F311, E12-F304, E22-F021, E21-F013, E21-F222, and E21-F221.

An evaluation has been performed which determined that eleven penetrations, which are currently tested hydrostatically, are located above the minimum suppression pool drawdown level and could therefore be exposed to containment atmosphere if a worst case single failure is considered. Standard Review Plan 6.2.6, "Containment Leakage Testing", states that hydrostatic testing of isolation valves is permissible if the line is not a potential containment atmosphere leak path; otherwise, pneumatic testing should be performed. Since the evaluation determined that it would be possible for the penetrations to become uncovered due to suppression pool drawdown, this change is proposed to require pneumatic testing of the isolation valves associated with these penetrations by deleting the hydrostatic testing footnotes. The valves associated with four of the affected penetrations have not yet been pneumatically tested; these valves will be pneumatically tested on their next scheduled test date. A list of the penetrations and subject valves is presented below:

Penetration No.	Valve No.
18	E12-F023 E12-F344 E12-F342 E12-F061 E51-F066
24	E12-F280 E12-F281
32	E21-F217 E21-F218
76	E12-F005

It should be noted that a design change is required in order to be able to perform pneumatic testing on four of the affected valves, namely valves E12-F280, E12-F281, E21-F217, and E21-F218. It should also be noted that although penetration 27 had previously been identified as requiring pneumatic testing of its isolation valves subsequent investigation has determined that this penetration is below the minimum suppression pool drawdown level; therefore, exposure to containment atmosphere will not occur and pneumatic testing for its valves is not required.

In addition, a recent design change has added a valve and capped tee downstream of containment isolation valves E51-F251 and E51-F252 to enable direct hydrostatic testing of these valves rather than testing during system functional tests. The system configuration was such that direct observation of leakage from these valves could not be accomplished during system functional tests. Table 3.6.4-1, page 3/4 6-39, should be revised to reflect this design change by deleting footnote (e) and adding footnote (c).

The proposed changes, which require pneumatic testing of valves previously tested hydrostatically, represent additional limitations not presently included in the Technical Specifications. Pneumatic testing provides additional assurance that the affected isolation valves will provide an adequate seal and is a more stringent surveillance requirement than the previously required hydrostatic testing, as documented in a letter from A. Schwencer (NRC) to J. P. McGaughy (MP&L) dated September 23, 1983.

The proposed changes represent no adverse impact to the safe operation of GGNS as they represent more stringent requirements and render the subject technical specification consistent with the as-built design. (Pages 3/4 6-29, 3/4 6-30, 3/4 6-34, 3/4 6-37, 3/4 6-38, 3/4 6-39, 3/4 6-40, and 3/4 6-42)

2. (TSPS 306), Containment and Drywell Isolation Valves, Technical Specification Tables 3.6.4-1 and 3.6.6.2-1

   Changes are proposed to revise the maximum isolation times of 61 valves in Section 1 of Table 3.6.4-1. In a previous Technical Specification change submittal (letter No. AECM-83/0492 from L. F. Dale (MP&L) to H. R. Denton (NRC), dated August 29, 1983), MP&L identified valves which have analytical closure time requirements. Subsequently, MP&L has identified that the reactor water cleanup (RWCU) valves in Section 1 of the table should also be included in the list of valves that have analytical closure time requirements. The revisions to the RWCU (G33) valves' closure times reflect General Electric's analyses of radiological dose and containment overpressurization resulting from a break in the RWCU system. These analyses have shown that a maximum isolation time of 35 seconds for these valves will ensure an acceptable response for a RWCU pipe break either inside or outside of containment. Additionally, it should be noted that although containment spray valves E12-F028A and B do have an analytical closure limit, the limit specified is based upon the maximum analytical opening time, not the maximum analytical closure time as stated in previous submittals.

In order to assure identification of all valves that have analytical closure times, MP&L initiated a review to be conducted by both the NSSS vendor and the Architect/Engineer. This review identified all valve closure times that are based on analytical values. The valves in Table 3.6.4-1 that have been determined to have analytical closure time are listed below along with the analytical isolation times.

Valve Number	Maximum Isolation Times (Seconds)
B21-F028A,B,C,D	5
B21-F022A,B,C,D	5
E12-F008	40
E12-F009	40
E12-F024A,B	90
M41-F011	4
M41-F012	4
M41-F034	4
M41-F035	4
M41-F015	4
M41-F013	4
M41-F016	4
M41-F017	4
E51-F063	20
E51-F064	20
G33-F028	35
G33-F034	35
G33-F039	35
G33-F040	35
G33-F001	35
G33-F004	35
G33-F252	35
G33-F053	35
G33-F054	35
G33-F250	35
G33-F251	35
G33-F253	35

The remaining revisions to the maximum isolation times are for valves that have no analytical closure times. For valves without analytical closure times, the margin of safety and the consequences of an accident will not be affected as long as valve closure is assured. MP&L has determined that the maximum isolation time for these valves should be based on the results of ASME Section XI testing. This is consistent with ASME Section XI, which states that the owner shall set the maximum stroke time for valves.

An additional proposed change would move valves E12-F042A and B from Section 1.a to Section 2.a of the table. These valves are LPCI injection valves which open on the signals listed as group 5 isolation signals in Table 3.3.2-1. While these valves do receive a closure signal from a containment spray initiation signal, their inclusion in section 2.a of the table is appropriate since they do not receive an automatic containment isolation signal. Note that a discussion of the four sections of Table 3.6.4-1 has been added to Bases Section 3/4.6.4 as resolution to TSPS 170.

The final proposed change would revise an incorrect penetration number and add information designating the divisional power supply associated with the valves in Table 3.6.4-1 and Table 3.6.6.2-1. The designators A, B, and C, and (A), (B), and (C), correspond to electrical divisions 1, 2, and 3 for motor-operated valves and the solenoids for air-operated valves, respectively. No designators were added to the Main Steam Isolation Valves; these valves receive power from both Division 1 and 2. These changes do not affect any specification requirement but are enhancements that will provide additional information.

The proposed changes do not affect safety, even though the closure times are increased for most of the affected valves, because all of the proposed times are less than or equal to the analytical closing values, if applicable. A time history of the closure times for these valves (observed during surveillance testing) will be maintained and reviewed to detect degradation of valve performance, thereby decreasing the probability of an unexpected valve failure.

(Pages 3/4 6-29, 3/4 6-30, 3/4 6-31, 3/4 6-32, 3/4 6-33, 3/4 6-34, 3/4 6-35, 3/4 6-36, 3/4 6-37, 3/4 6-38, 3/4 6-44, 3/4 6-48, 3/4 6-49, 3/4 6-50, 3/4 6-51 and 3/4 6-52)

3. (TSPS 240), Containment and Drywell Hydrogen Recombiners, Technical Specification 3/4.6.7.1

   The proposed changes to the subject Specification will delete all references to drywell hydrogen recombiner systems because the GGNS plant design incorporates hydrogen recombiners only in the containment, not in the drywell. The proposed changes involve no safety significance as they represent a clarification of plant design and do not affect the technical content of the specifications. (Page 3/4 6-56)

4. (TSPS 818), Secondary Containment Isolation, Technical Specifications 1.37, 3/4.6.6.1, and Bases 3/4.6.6

   The proposed changes to Definition 1.37, Surveillance Requirement 4.6.6.1, and Bases 3/4.6.6 add rupture discs and blind flanges, as appropriate, to the list of equipment necessary for SECONDARY CONTAINMENT INTEGRITY. These revisions are required because the GGNS design utilizes these components as one method of ensuring secondary containment integrity. The proposed changes involve no safety significance as they represent clarifications that more accurately reflect plant design and the intent of the existing Specifications. (Pages 1-7, 3/4 6-46, and B 3/4 6-6)

C. ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES

The following proposed changes are enhancements which are consistent with the safety analyses and the licensing basis and which provide clarification, render areas consistent with the philosophy and intent of the technical specifications, or provide additional plant operational margin.

Since these proposed changes are included in the current licensing bases and are bounded by existing safety analyses, the proposed changes do not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration.

A description of these changes including justification for the changes is provided below:

1. (TSPS 012 and 169), Containment Spray and Suppression Pool Cooling, Technical Specifications 3.6.3.2, 3.6.3.3 and Bases 3/4.6.3

   The following technical specification changes are proposed:

   a) Change "SSW heat exchanger" to "RHR heat exchanger," in Technical Specification 3.6.3.2.b and 3.6.3.3.b,

   b) Add the containment spray spargers to Technical Specification 3.6.3.2.b,

   c) Revise "ACTION b" of Technical Specification 3.6.3.2 to be consistent with "ACTION b" of Technical Specification 3.6.3.3,

   d) Change the time in "ACTION a" of Technical Specification 3.6.3.3 from 7 days to 72 hours,

   e) Add new Surveillance Requirement 4.6.3.2.d, and

   f) Revise Bases 3/4.6.3 to reflect the added Surveillance Requirement.

   The change from SSW to RHR heat exchanger corrects an error in terminology and is made to reflect correct nomenclature. The addition of the containment spray spargers to Specification 3.6.3.2.b reflects the system design and is proposed to ensure system OPERABILITY. The changes to the ACTION statements are enhancements to achieve consistency between the containment spray and suppression pool cooling modes of RHR operation. The addition of the requirement to perform an air or smoke flow test to Surveillance Requirement 4.6.3.2 and the revision to Bases 3/4.6.3 constitutes an additional requirement not presently included in the technical specifications. The design of the containment spray system is such that nozzle obstruction should not occur unless caused by maintenance activities; therefore, the surveillance frequency should not be time dependent but should instead be coordinated with the completion of applicable maintenance activities. The containment spray nozzles were initially air-tested during the preoperational test phase and no maintenance has been performed on the system since that time which could cause nozzle blockage. (Pages 3/4 6-24, 3/4 6-25 and B 3/4 6-4)

2. (TSPS 127), Drywell to Containment Differential Pressure, Technical Specification Bases 3/4.6.2.5

   The change in the minimum differential pressure from -0.1 to -0.26 in Specification 3.6.2.5 and the corresponding correction to Bases 3/4.6.2.5 were previously submitted as Item 22 of AECM-83/0565. The 0.26 psid value resulted from an analysis performed to resolve one of the Humphrey concerns, (overflow of the weir wall). An additional change to the Bases, which states that the 2.0 psid limit for positive drywell to containment pressure will "not allow clearing of the top vent" is proposed to reflect the results of the drywell/containment analyses. The proposed change is considered an enhancement that will render the specifications consistent with safety analyses. (Page B 3/4 6-3)

3. (TSPS 294), Containment Leakage Rates, Technical Specification 3/4.6.1.2

   The proposed changes to the subject technical specification clarify the leakage requirements for containment isolation valves and penetrations and their applicable testing methods so as to ensure compliance with 10 CFR 50. As presently written, the specification requires only ECCS and RCIC containment isolation valves (which are in hydrostatically tested lines that penetrate the primary containment) to be included in the group of valves that must have a combined leakage rate of less than or equal to 1 gpm times the total number of valves in the group. Other containment isolation valves should also be included in the group of valves that are subject to containment leak rate requirements; therefore the terms "ECCS and RCIC" should be deleted from the appropriate sections so as to broaden the scope of this specification. In addition, an editorial change to Specification 3.6.1.2.b and Action b removed a portion of the text and placed it in a footnote, thus rendering these sentences more readable. The proposed changes involve no safety significance as they represent a clarification of the technical specifications addressing containment leakage. (Pages 3/4 6-2, 3/4 6-3, 3/4 6-4)

4. (TSPS 164), Primary Containment Integrity Requirements, Definition 1.30 and Technical Specification 3/4.6.1.1

   The proposed changes to Definition 1.30 and Surveillance Requirement 4.6.1.1 will render the definition of containment air lock and suppression pool OPERABILITY consistent with the intent of Specifications 3.6.1.3 and 3.6.3.1. Specifications 3.6.1.3 and 3.6.3.1 contain exceptions to the basic OPERABILITY requirements that permit continued operation under the conditions specified in the ACTIONS. Since PRIMARY CONTAINMENT INTEGRITY is maintained if the conditions of these ACTIONS are satisfied, the term OPERABLE in Definition 1.30 and Surveillance Requirement 4.6.1.1 should be deleted and replaced with a statement that requires the containment air locks and suppression pool to be in compliance with requirements of Technical Specifications 3.6.1.3 and 3.6.3.1, respectively.

   These changes are considered enhancements in that they clarify the intent of those specifications related to primary containment integrity. (Pages 1-6 and 3/4 6-1)

5. (TSPS 167), Drywell Integrity, Definition 1.10 and Technical Specification 3/4.6.2.1

   The proposed changes to Definition 1.10 and Surveillance Requirement 4.6.2.1 will render the definition of drywell air lock and suppression pool OPERABILITY consistent with the intent of Specifications 3.6.2.3 and 3.6.3.1. Specifications 3.6.2.3 and 3.6.3.1 contain exceptions to the basic OPERABILITY requirements that permit continued operation under the conditions specified in the ACTIONS. Since DRYWELL INTEGRITY is maintained if the conditions of these ACTIONS are satisfied, the term OPERABLE in Definition 1.10 and Surveillance Requirement 4.6.2.1 should be deleted and replaced with a statement that requires the drywell air locks and suppression pool to be in compliance with the requirements of Technical Specifications 3.6.2.3 and 3.6.3.1, respectively. These changes are considered enhancements in that they clarify the intent of those specifications related to DRYWELL INTEGRITY. (Pages 1-2 and 3/4 6-13)

6. (TSPS 170), Additional Bases Information Concerning Table 3.6.4-1, Technical Specification Bases 3/4.6.4

   Technical Specification 3.6.4 addresses the containment and drywell isolation valves which are listed by section in Table 3.6.4-1. The proposed technical specification change to add a description of the different valve categories listed in Table 3.6.4-1 is made to the Bases to clarify the arrangement of the table. This proposed clarification is an enhancement and involves no safety significance because it is consistent with the information presently contained in Table 3.6.4-1. (Page B 3/4 6-5)

7. (TSPS 233), RHR Flows for Containment Spray Mode, Technical Specification Bases 3/4.6.3

The proposed change to Bases Section 3/4.6.3 is to add a statement confirming that Surveillance Requirements 4.5.1.b and 4.6.3.2.b provide adequate assurance that the Containment Spray System will be OPERABLE when required. Sufficient flow through the RHR heat exchangers will ensure sufficient flow to the containment spray nozzles, since the minimum acceptable flow and total developed head values, stated in the surveillance requirement, account for inherent system losses. This change is an enhancement that clarifies the Bases for the Containment Spray System specification. (Page B 3/4 6-4)

8. (TSPS 320), Depressurization Systems Bases, Technical Specification Bases 3/4.6.3

This change clarifies that the pressure used in the reactor blowdown analysis was 1060 psia instead of 1089 psig. The 1060 psia is consistent with the numerical value listed in FSAR Table 15.0-3 "Input Parameters and Initial Conditions for Transients Used in ODYN Code." This change is an enhancement which is consistent with the safety analysis that is referenced above. (Page B 3/4 6-4)

9. (TSPS 379), Air Lock Seal Decay Test, Technical Specification Bases 3/4.6.1.3 and 3/4.6.2.3

This proposed change to the Bases provides clarification that a shorter time period for the leak-detection test may be used if the test is conducted in accordance with ANSI N45.4-1972.

The methodology used for assuring that the leakage rate can be accurately determined with a test period less than 48 hours is described below. This methodology was developed and used in previous tests. Further refinements may be made to the methodology as experience dictates.

Three test instruments are used, a pressure gauge, a barometer, and a thermometer. Considering accuracy, readability and repeatability, for the pressure gauge there is an uncertainty of ±0.07 psi; for the barometer, the uncertainty is ±0.02 psi; and for temperature variations, the uncertainty is ±0.005 psi. The algebraic sum of the uncertainties is 0.095 psi. A conservative ±0.1 psi uncertainty is used.

Although the pressure decay is exponential, linear interpolation will be used, which is more conservative; e.g., for a 12 hour test, the allowable pressure loss for a 30 day exponential decay is 0.588 psi; for a linear interpolation, the pressure loss cannot exceed 0.500 psi.

Using the linear interpolation of the allowable 30 day pressure loss, a plot of the actual pressure decay minus 0.1 psi is plotted on the same chart. When the plot of the actual pressure decay crosses the plot of the 30 day linear interpolation, the test will be considered to have been successfully completed.

This method is conservative and even with zero pressure loss, a minimum of 2.4 hours is required before it may be concluded that the results are acceptable based on the ±0.1 psi uncertainty that is used.

The proposed change to the Bases of Technical Specification 3/4.6.1.3 and 3/4.6.2.3 provides an operational enhancement that is consistent with the philosophy and intent of the technical specifications.

Approval of this change by the NRC will constitute prior acceptance and an acknowledgement that leakage rates can be accurately determined during test periods that are less than 48 hours. (Page B 3/4 6-1 and B 3/4 6-3)

10. (TSPS 168), Emergency Core Cooling Systems - Suppression Pool, Depressurization Systems - Suppression Pool, and Depressurization Systems Bases, Technical Specifications 3/4.5.3, 3/4.6.3 and Bases 3/4.6.3

    The proposed changes to the subject technical specifications are as follows:

    a. Revise the suppression pool low and high water levels (depths) to 18' 4-1/12" and 18' 9-3/4", respectively, to make them consistent with FSAR TABLE 6.2-50, "SUPPRESSION POOL GEOMETRY - 251 PLANT."

    b. Delete the reference to OPERATIONAL CONDITION 1 or 2 in LCO 3.6.3.1.b, ACTION 3.6.3.1.b, and Surveillance Requirement 4.6.3.1.b.

    c. Expand Bases 3/4.6.3, "DEPRESSURIZATION SYSTEMS," to provide bases for:

        1. Suppression pool volumes
        2. Suppression pool levels (depths)
        3. Suppression pool temperatures

    The changes to the suppression pool water levels are considered enhancements made to be consistent with the volumes used for the safety analyses. The deletions of the reference to OPERATIONAL CONDITION 1 or 2 are considered enhancements which make the Limiting Conditions for Operation, ACTION statements, and Surveillance Requirements consistent with the Applicability Statement. The changes to Bases 3/4.6.3 are enhancements which provide substantial clarification and are consistent with the safety analyses. (Pages 3/4 5-8, 3/4 5-9, 3/4 6-20, 3/4 6-21, B 3/4 6-4)

D. REGULATORY REQUIREMENTS/REQUESTS/RECOMMENDATIONS

The following changes are proposed to render the technical specifications consistent with recent changes in NRC policy and the Code of Federal Regulations, as well as to implement changes or enhancements recently requested or recommended by NRC reviewers.

These proposed changes are required to render the technical specifications consistent with recent NRC guidance, and it has been concluded based on a review of each item that the proposed changes do not:

- Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- Involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration.

A description of these changes including justification for the changes is provided below:

1. (TSPS 235), Containment Air Lock, Technical Specification 4.6.1.3

   Technical Specification Surveillance Requirement 4.6.1.3.a should be revised to add a footnote to both "72 hours" time limits, which will specify that the provisions of Technical Specification 4.0.2 are not applicable. This change clarifies that no extension of the 72 hour time limits to demonstrate OPERABILITY of the containment air lock(s) is allowed and ensures that the Technical Specifications are in compliance with 10CFR50, Appendix J. (Page 3/4 6-6)

2. (TSPS 067), Type A Test Accuracy Determination, Technical Specification 3/4.6.1.2

   Surveillance Requirement 4.6.1.2.c.1 should be revised so as to provide clarification concerning the method to be used for verification of the accuracy of Type A containment leakage rate testing (Appendix J, 10CFR50). In addition, Surveillance Requirement 4.6.1.2.c.3 should be revised to indicate that the required quantity of gas injected into, or bled from, the containment during the supplemental test must be between 0.75 L_a and 1.25 L_a. These changes are considered to be enhancements and are in compliance with 10 CFR 50 Appendix J. (Page 3/4 6-3)

3. (TSPS 031), Drywell Air Lock, Technical Specification 3/4.6.2.3

Surveillance Requirement 4.6.2.3.a is changed to require that air lock OPERABILITY be demonstrated within 72 hours, instead of 8 hours, after closing. Surveillance Requirement 4.6.2.3.b is changed to add a requirement that airlock OPERABILITY be demonstrated following any maintenance that could affect the sealing capability. The change to Surveillance Requirement 4.6.2.3.a makes the Technical Specification consistent with 10 CFR 50 Appendix J, paragraph D.2.iii and industry standards. The change to Surveillance Requirement 4.6.2.3.b represents an additional requirement not included in 10CFR50 Appendix J that will ensure that the airlock is OPERABLE prior to establishing DRYWELL INTEGRITY and following maintenance that could affect air lock sealing capability. (Page 3/4 6-16)

4. (TSPS 069), Containment and Drywell Hydrogen Ignition System, Technical Specification 3/4.6.7.2

This proposed change replaces present Specification 3/4.6.7.2 with an expanded Specification that adds requirements which ensure OPERABILITY of the H_2 ignition system. Changes to the Limiting Condition for Operation (LCO) require at least two igniter assemblies in each enclosed area in the Containment to be OPERABLE as well as all igniter assemblies adjacent to any inoperable igniter assembly in each open area in the Containment and Drywell. Proposed changes to the ACTION Statements are provided to coincide with changes to the LCO's. New ACTION "a" requires that with less than two igniter assemblies OPERABLE in any enclosed area in the Containment, at least two must be restored to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours. New ACTION "b" requires that with any adjacent igniter assemblies within open areas of the Containment or Drywell inoperable, the igniter assemblies must be restored so that all igniter assemblies adjacent to an inoperable igniter assembly are OPERABLE within 7 days or be in at least HOT SHUTDOWN within the next 12 hours. The current ACTION Statement is retained as new ACTION Statement "c." Proposed changes to Surveillance Requirement 4.6.7.2 are made to demonstrate OPERABILITY of the H_2 igniters required OPERABLE by the LCO. New Surveillance 4.6.7.2.a.1 and 4.6.7.2.a.2 require energizing the supply breakers at least once per 92 days and verifying a visible glow from each normally accessible igniter assembly in the Containment and verifying that each circuit of each Containment and Drywell H_2 igniter subsystem is conducting sufficient current to energize the minimum number of igniter assemblies required as specified on new Table 4.6.7.2-1. New Surveillance Requirement 4.6.7.2.b requires, at every COLD SHUTDOWN but no more frequently than once per 92 days, that each normally inaccessible igniter assembly is verified OPERABLE by energizing the supply breakers and verifying a visible glow from the glow plugs. New Table 3.6.7.2-1 lists the H_2 igniters by electrical division and by circuits within each division. New Table 3.6.7.2-2 lists the H_2 igniters by electrical division/circuit, elevation, azimuth, and distance from the center line of the reactor. New Table 3.6.7.2-2 also lists those igniters in normally accessible, inaccessible, open or enclosed areas within the containment and/or drywell. New Table 4.6.7.2-1 lists the minimum required igniters per circuit.

The proposed changes to the H_2 igniter specification follow NRC guidance and ensure OPERABILITY of the system. The proposed changes to the LCO, ACTION and Surveillance Requirements address igniter assemblies in both enclosed and open areas to ensure that all required areas have OPERABLE igniters. These changes assure H_2 igniter system OPERABILITY and address system design by requiring the normally inaccessible assemblies in the drywell and containment to have a visible glow verification during COLD SHUTDOWN. These igniters are inaccessible due to high levels of radiation and/or high temperatures in these areas during plant operation; however, OPERABILITY is verified for the minimum number required per circuit by electrical current checks at least once per 92 days. The new tables enhance the Specification by providing tabulations for igniter location, electrical division and circuit, and minimum number required for each circuit. The proposed changes provide an increase in safety through more stringent requirements than those currently in the technical specifications and are consistent with the licensing basis. (Pages 3/4 6-57, 3/4 6-57a, 3/4 6-57b, 3/4 6-57c, 3/4 6-57d, 3/4 6-57e, 3/4 6-57f)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

(AFFECTED PAGES ARE PROVIDED IN THE ORDER OF ASCENDING PAGE NUMBERS.)

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of ~~OPERABLE pursuant to~~ Specification 3.6.2.3. 167

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is in compliance with the requirements of ~~OPERABLE pursuant to~~ Specification 3.6.3.1. 167

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE OCCURRENCE

1.35 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.12 and 6.9.1.13.

ROD DENSITY

1.36 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All Auxiliary Building and Enclosure Building penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, rupture disc or deactivated automatic valve or damper, as applicable, secured in its closed position, except as provided in Table 3.6.6.2-1 of Specification 3.6.6.2.

b. All Auxiliary Building and Enclosure Building equipment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.3.

d. The door in each access to the Auxiliary Building and Enclosure Building is closed, except for normal entry and exit.

e. The sealing mechanism associated with each Auxiliary Building and Enclosure Building penetration, e.g., welds, bellows or O-rings, is OPERABLE.

AMENDMENT No. ____

PRIMARY CONTAINMENT INTEGRITY

1.30 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification, 3.6.4.

b. The containment equipment hatch is closed and sealed.

c. Each containment air lock is ~~OPERABLE pursuant to~~ in compliance with the requirements of Specification 3.6.1.3. | 47

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression pool is ~~OPERABLE pursuant to~~ in compliance with the requirements of Specification 3.6.3.1. | 47

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

1.31 The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE - PURGING

1.32 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3833 MWT.

3/4.5.3 SUPPRESSION POOL#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 or 3 with a contained water volume of at least 135,291 ft^3, equivalent to a level of 18'4-~~3/4~~ 1/12". 168

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 93,600 ft^3, equivalent to a level of 12'8", except that the suppression pool level may be less than the limit or may be drained provided that:

   1. No operations are performed that have a potential for draining the reactor vessel,

   2. The reactor mode switch is locked in the Shutdown or Refuel position,

   3. The condensate storage tank contains at least 170,000 available gallons of water, equivalent to a level of 18', and

   4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

   a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression pool water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL CONDITION 4 or 5* with the suppression pool water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.3.1 for pressure suppression requirements.

*The suppression pool is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the upper containment fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water level instrumentation division inoperable, restore the inoperable division to OPERABLE status within 7 days or verify the suppression pool water level to be greater than or equal to 18'4-~~3/4~~ 1/12" or 12'8", as applicable, at least once per 12 hours by an alternate indicator. 168

d. With both suppression pool water level instrumentation divisions inoperable, restore at least one inoperable division to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression pool water level to be greater than or equal to 18'4-~~3/4~~ 1/12" or 12'8", as applicable, at least once per 12 hours by at least one alternate indicator. 168

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. 18'4-~~3/4~~ 1/12" at least once per 24 hours. 168

2. 12'5" at least once per 12 hours.

b. Two suppression pool water level instrumentation divisions, with 1 channel per division, OPERABLE with the low water level alarm setpoint $\geq$ 18'5½" or 12'8", as applicable, by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months.

4.5.3.2 With the suppression pool level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following Type A or B test, by leak rate testing the ~~equipment hatch~~ seals with gas at Pa, 11.5 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La. 144

b. At least once per 31 days by verifying that all containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

c. By verifying each containment air lock is in compliance with the requirements of ~~OPERABLE per~~ Specification 3.6.1.3. 164

d. By verifying the suppression pool is in compliance with the requirements of ~~OPERABLE per~~ Specification 3.6.3.1. 164

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, steam tunnel or drywell and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.437 percent by weight of the containment air per 24 hours at P_a, 11.5 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves** ~~listed in Table 3.6.1-1, except for valves which are hydrostatically leak tested per Table 3.6.4-1,~~ subject to Type B and C tests when pressurized to P_a, 11.5 psig. 294

c. Less than or equal to 100 scf per hour for all four main steam lines through the isolation valves when tested at P_a, 11.5 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ~~ECCS and RCIC~~ containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 12.65 psig. 294

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves** ~~listed in Table 3.6.4-1, except for valves which are hydrostatically leak tested per Table 3.6.4-1,~~ subject to Type B and C tests exceeding 0.60 L_a, or 294

c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines through the isolation valves, or

d. The measured combined leakage rate for all ~~ECCS and RCIC~~ containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves, 294

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

** Includes all valves listed in Table 3.6.4-1, except for those that are hydrostatically leak tested. 294

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. The combined leakage rate for all penetrations and all valves# ~~listed in Table 3.6.4-1, except for valves which are hydrostatically leak tested per Table 3.6.4-1,~~ subject to Type B and C tests to less than or equal to 0.60 L_a, and 294

c. The leakage rate to less than 100 scf per hour for all four main steam lines through the isolation valves, and

d. The combined leakage rate for all ~~ECCS and RCIC~~ containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves, 294

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 11.5 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. ~~Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a~~. 067

Insert, see next page

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be ~~equivalent to at least 25 percent of the total measured leakage at P_a, 11.5 psig.~~ 067

between 0.75 La and 1.25 La.

\# Includes all valves listed in Table 3.6.4-1, except for those that are hydrostatically leak tested. 294

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Confirms the accuracy of the test by verifying that the containment leakage rate, L'_v, calculated in accordance with ANSI N45.4-1972, Appendix C, is within 25 percent of the containment leakage rate, L_v, measured prior to the introduction of the superimposed leak.

d. Type B and C tests shall be conducted with gas at P_a, 11.5 psig,* at intervals no greater than 24 months except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Penetrations using continuous leakage monitoring systems,
4. Valves pressurized with fluid from a seal system,
5. ~~ECCS and RCIC~~ Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and
6. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 11.5 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 12.65 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

i. ~~ECCS and RCIC~~ Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.9.2.

k. The provisions of Specification 4.0.2 are not applicable to 24 month or 40 ± 10 month surveillance intervals.

*Unless a hydrostatic test is required per Table 3.6.4-1.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours# after each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours#, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to Pa, 11.5 psig. 235

b. By conducting an overall air lock leakage test at P_a, 11.5 psig, and verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months#, and

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. By verifying each airlock door inflatable seal system OPERABLE by:

   1. Demonstrating each of the two inflatable seal pressure instrumentation channels per airlock door OPERABLE by performance of a:

      a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

      b) CHANNEL CALIBRATION at least once per 18 months,

      with a low pressure setpoint of $\geq$ 60 psig.

   2. At least once per 7 days, verifying seal air flask pressure to be greater than or equal to 90 psig.

   3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

Amendment No. ___

Order

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification ~~4.6.1.6.~~ 4.6.1.6.1. 276

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without DRYWELL INTEGRITY, restore DRYWELL INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all drywell penetrations** not capable of being closed by OPERABLE drywell automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. By verifying each drywell air lock ~~OPERABLE per~~ is in compliance with the requirements of Specification 3.6.2.3. 167

c. By verifying the suppression pool ~~OPERABLE per~~ is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the drywell or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each drywell air lock shall be demonstrated OPERABLE:

a. Within ~~8~~ 72 hours# after each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours#, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to P_a, 11.5 psig. 031

b. ~~At least once per 6 months by conducting an overall air lock leakage test at P_a, 11.5 psig and by verifying that the overall air lock leakage rate is within its limit.#~~ 031

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. By verifying each airlock door inflatable seal system OPERABLE by:

1. Demonstrating each of the two inflatable seal pressure instrumentation channels per airlock door OPERABLE by performance of a:

    a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

    b) CHANNEL CALIBRATION at least once per 18 months,

    with a low pressure setpoint of $\geq$ 60 psig.

2. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 90 psig.

3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

031

By conducting an overall air lock leakage test at P_a, 11.5 psig and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months#,
2. Prior to establishing DRYWELL INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.

DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification ~~4.6.2.4.~~ 4.6.2.4.1 269

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the drywell not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the drywell shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.2.4.2 Reports Any abnormal degradation of the drywell structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.3 DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL#

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Volume between 135,291 ft^3 and 138,851 ft^3, equivalent to a level between 18'4-~~3/4"~~ 1/12 and 18'~~10"~~ 9-3/4, and a

b. Maximum average temperature of 95°F ~~during OPERATIONAL CONDITION 1 or 2~~, except that the maximum average temperature may be permitted to increase to: 168

1. 105°F during testing which adds heat to the suppression pool.
2. 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.
3. 120°F with the main steam line isolation valves closed following a scram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression pool water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~In OPERATIONAL CONDITION 1 or 2 with~~ With the suppression pool average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above: 168

1. With the suppression pool average water temperature greater than 105°F during testing which adds heat to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the suppression pool average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.
3. With the suppression pool average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water level instrumentation division inoperable and/or with one suppression pool water temperature instrumentation channel in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression pool water level and/or temperature to be within the limits at least once per 12 hours.

d. With both suppression pool water level instrumentation divisions inoperable and/or with both suppression pool water temperature instrumentation channels in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water level division and at least one inoperable water temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying the suppression pool water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours ~~in OPERATIONAL CONDITION 1 or 2~~ by verifying the suppression pool average water temperature to be less than or equal to 95°F, except: | 168

   1. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

   2. At least once per hour when suppression pool average water temperature is greater than or equal to 95°F, by verifying suppression pool average water temperature to be less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

   3. At least once per 30 minutes following a scram with suppression pool average water temperature greater than or equal to 95°F, by verifying suppression pool average water temperature less than or equal to 120°F.

CONTAINMENT SPRAY

LIMITING CONDITION FOR OPERATION

3.6.3.2 The containment spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression pool through a ~~SSW~~ RHR heat exchanger, and the containment spray spargers. | 012 169

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one containment spray loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both containment spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the ~~next~~ following 24 hours. | 012

SURVEILLANCE REQUIREMENTS

4.6.3.2 The containment spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 5650 gpm on recirculation flow through the RHR heat exchange to the suppression pool when tested pursuant to Specification 4.0.5.

c. At least once per 18 months by performance of a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual spraying of coolant into the primary containment may be excluded from this test.

d. ADD INSERT | 169

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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By performance of an air or smoke flow test of the containment spray nozzles and verifying that each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.3.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump; and

b. An OPERABLE flow path capable of recirculating water from the suppression pool through a ~~SSW~~ RHR heat exchanger. |012

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within ~~7 days~~ 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. |169

b. With both suppression pool cooling loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 7450 gpm on recirculation flow through the RHR heat exchangers to the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The containment and drywell isolation valves shown in Table 3.6.4-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.4-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and #.

ACTION:

With one or more of the containment or drywell isolation valves shown in Table 3.6.4-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*Isolation valves, except MSIVs, closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls. 246

#Isolation valves shown in Table 3.6.4-1 are also required to be OPERABLE when their associated actuation instrumentation is required to be OPERABLE per Table 3.3.2-1.

TABLE 3.6.4-1
CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)	
1. Automatic Isolation Valves#					306
a. Containment					
Main Steam Lines	B21-F028A	5(O)*	1	5	
Main Steam Lines	B21-F022A	5(I)*	1	5	
Main Steam Lines	B21-F067A-A	5(O)*	1	~~8~~ 9	306
Main Steam Lines	B21-F028B	6(O)*	1	5	
Main Steam Lines	B21-F022B	6(I)*	1	5	
Main Steam Lines	B21-F067B-A	6(O)*	1	~~8~~ 9	306
Main Steam Lines	B21-F028C	7(O)*	1	5	
Main Steam Lines	B21-F022C	7(I)*	1	5	
Main Steam Lines	B21-F067C-A	7(O)*	1	~~8~~ 9	306
Main Steam Lines	B21-F028D	8(O)*	1	5	
Main Steam Lines	B21-F022D	8(I)*	1	5	
Main Steam Lines	B21-F067D-A	8(O)*	1	~~8~~ 9	306
RHR Reactor Shutdown Cooling Suction	E12-F008-A	14(O)~~(c)~~	3	40	020
RHR Reactor Shutdown Cooling Suction	E12-F009-B	14(I)~~(c)~~	3	40	
Steam Supply to RHR and RCIC Turbine	E51-F063-B	17(I)	4	20	
Steam Supply to RHR and RCIC Turbine	E51-F064-A	17(O)	4	20	
Steam Supply to RHR and RCIC Turbine	E51-F076-B	17(I)	4	20	
RHR to Head Spray	E12-F023-A	18(O)~~(c)~~	3	~~90~~ 94	306 020
Main Steam Line Drains	B21-F019-A	19(O)	1	~~15~~ 20	306

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.
(b) Hydrostatically tested to ASME Section XI criteria.
(c) Hydrostatically tested with water to 1.10 P_a, 12.65 psig.
(d) Hydrostatically tested by pressurizing system to 1.10 P_a, 12.65 psig.
(e) Hydrostatically tested during system functional tests.
(f) Hydrostatically sealed by feedwater leakage control system. Type C test not required.
(g) Normally closed or locked closed manual valves may be opened on an intermittent basis under administrative control. 306

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITIONS 2 or 3 provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.

The "-A, -B, -C, -(A), -(B), -(C)" designators on the valve numbers indicate associated electrical divisions. 306

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
Main Steam Line Drains	B21-F016-B	19(I)	1	~~15~~ 20
~~RHR Heat Exchanger "A" to LPCI~~	~~E12-F042A-A~~	~~20(I)~~ ~~(c)~~	~~5~~	~~22~~
RHR Heat Exchanger "A" to LPCI	E12-F028A-A	20(I) ~~(c)~~	5	~~78~~ 90
RHR Heat Exchanger "A" to LPCI	E12-F037A-A	20(I) ~~(c)~~	3	~~63~~ 74
~~RHR Heat Exchanger "B" to LPCI~~	~~E12-F042B-B~~	~~21(I)~~ ~~(c)~~	~~5~~	~~22~~
RHR Heat Exchanger "B" to LPCI	E12-F028B-B	21(I) ~~(c)~~	5	~~78~~ 90
RHR Heat Exchanger "B" to LPCI	E12-F037B-B	21(I) ~~(c)~~	3	~~63~~ 74
RHR "A" Test Line to Supp. Pool	E12-F024A-A	23(O)[(d)]	5	90
RHR "A" Test Line to Supp. Pool	E12-F011A-A	23(O)[(d)]	5	36
RHR "C" Test Line to Supp. Pool	E12-F021-B	24(O) ~~(d)~~	5	~~101~~ 144
HPCS Test Line	E22-F023-C	27(O)[(d)]	6B	~~60~~ 75
RCIC Pump Suction	E51-F031-A	28(O)[(d)]	4	56
RCIC Turbine Exhaust	E51-F077-A	29(O)[(c)]	9	26
LPCS Test Line	E21-F012-A	32(O) ~~(d)~~	5	144
Cont. Purge and Vent Air Supply	M41-F011-(A)	34(O)	7	4
Cont. Purge and Vent Air Supply	M41-F012-(B)	34(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F034-(B)	35(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F035-(A)	35(O)	7	4
Plant Service Water Return	P44-F070-B	36(I)	6A	33
Plant Service Water Return	P44-F069-A	36(O)	6A	~~24~~ 33
Plant Service Water Supply	P44-F053-A	37(O)	6A	~~24~~ 33
Chilled Water Supply	P71-F150-(A)	38(O)	6A	~~30~~ 12

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)	
Containment (Continued)					
Chilled Water Return	P71-F148-(A)	39(O)	6A	~~30~~ 12	
Chilled Water Return	P71-F149-(B)	39(I)	6A	~~30~~ 12	
Service Air Supply	P52-F105-(A)	41(O)	6A	~~4~~ 6	
Inst. Air Supply	P53-F001-(A)	42(O)	6A	~~4~~ 6	
RWCU to Main Condenser	G33-F034-A	43(O)	8	~~31~~ 35	
RWCU to Main Condenser	G33-F028-B	43(I)	8	~~23~~ 35	
RWCU Backwash to C/U Phase Sep. Tank	G36-F106-(B)	49(I)	6A	~~30~~ 11	
RWCU Backwash to C/U Phase Sep. Tank	G36-F101-(A)	49(O)	6A	~~30~~ 11	
Drywell & Cont. Equip. Drain Sump Disch.	P45-F067-(B)	50(I)	6A	~~4~~ 7	
Drywell & Cont. Equip. Drain Sump Disch.	P45-F068-(A)	50(O)	6A	7	306
Drywell & Cont. Floor Drain Sump Disch.	P45-F061-(B)	51(I)	6A	~~4~~ 7	
Drywell & Cont. Floor Drain Sump Disch.	P45-F062-(A)	51(O)	6A	~~4~~ 7	
Condensate Supply	P11-F075-(A)	56(O)	6A	~~30~~ 10	
FPC & CU to Upper Cont. Pool	G41-F028-A	57(O)	6A	~~44~~ 51	
Upper Cont. Pool to Fuel Pool Drain Tank	G41-F029-A	58(O)	6A	~~40~~ 51	
Upper Cont. Pool to Fuel Pool Drain Tank	G41-F044-B	58(I)	6A	40	
Aux. Bldg. Flr. and Equip. Drn. Tks. to Supp. Pool	P45-F273-A	60(O)	6A	~~23~~ 32	306
Aux. Bldg. Flr. and Equip. Drn. Tks. to Supp. Pool	P45-F274-B	60(O)	6A	~~23~~ 32	

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
Comb. Gas Control Cont. Purge (Outside Air Supply)	E61-F009-(A)	65(O)	7	4
Comb. Gas Control Cont. Purge (Outside Air Supply)	E61-F010-(B)	65(I)	7	4
Purge ~~Rad.~~ Filter Train ~~Detector~~ Isolation	E61-F056-(B)	66(I)	7	4
Purge ~~Rad.~~ Filter Train ~~Detector~~ Isolation	E61-F057-(A)	66(O)	7	4
RHR "B" Test Line To Suppr. Pool	E12-F024B-B	67(O)[d]	5	90
RHR "B" Test Line To Suppr. Pool	E12-F011B-B	67(O)[d]	5	~~27~~ 36
Refueling Water Transf. Pump Suction	P11-F130-(A)	69(O)[c]	6A	~~4~~ 8
Refueling Water Transf. Pump Suction	P11-F131-(B)	69(O)[c]	6A	8
Instr. Air to ADS	P53-F003-A	70(O)	6A	4
RCIC Turbine Exh. Vacuum Breaker	E51-F078-B	75(O)	9	~~7~~ 10
RWCU to Feedwater	G33-F040-B	83(I)	8	~~30~~ 35
RWCU to Feedwater	G33-F039-A	83(O)	8	~~29~~ 35
Chemical Waste Sump Discharge	P45-F098-(B)	84(I)	6A	~~4~~ 8
Chemical Waste Sump Discharge	P45-F099-(A)	84(O)	6A	~~4~~ 8
Supp. Pool Clean-up Return	P60-F009-A	85(O)	6A	8
Supp. Pool Clean-up Return	P60-F010-B	85(O)	6A	~~4~~ 8
Demin. Water Supply to Cont.	P21-F017-A	86(O)	6A	~~10~~ 19
Demin. Water Supply to Cont.	P21-F018-B	86(I)	6A	~~10~~ 19
RWCU Pump Suction	G33-F001-B	87(I)	8	~~30~~ 35

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
RWCU Pump Suction	G33-F252-A	87(I)	8	~~30~~ 35
RWCU Pump Suction	G33-F004-A	87(O)	8	~~30~~ 35
RWCU Pump Disch.	G33-F053-B	88(I)	8	~~22~~ 35
RWCU Pump Disch.	G33-F054-A	88(O)	8	~~22~~ 35
b. Drywell				
Instrument Air	P53-F007-B	~~327~~ 335(O)	6A	~~4~~ 7
Plant Service Water Return	P44-F076-A	331(I)	6A	32
Plant Service Water Return	P44-F077-B	331(O)	6A	32
Plant Service Water Return	P44-F074-B	332(O)	6A	32
RWCU Pump Suction	G33-F250-A	337(I)	8	~~30~~ 35
RWCU Pump Suction	G33-F251-B	337(O)	8	~~30~~ 35
Combustible Gas Con.	E61-F003B-B	338(O)	5	~~60~~ 84
Combustible Gas Con.	E61-F003A-A	339(O)	5	~~60~~ 84
Combustible Gas Con.	E61-F005A-A	340(O)	5	84
Combustible Gas Con.	E61-F005B-B	340(O)	5	84
Combustible Gas Con.	E61-F007-(A)	341(O)	5	9
Combustible Gas Con.	E61-F020-(B)	341(O)	5	18
Drywell Air Purge Supply	M41-F015-(A)	345(I)	7	4
Drywell Air Purge Supply	M41-F013-(B)	345(O)	7	4
Drywell Air Purge Exhaust	M41-F016-(A)	347(I)	7	4
Drywell Air Purge Exhaust	M41-F017-(B)	347(O)	7	4
Equipment Drains	P45-F009-(A)	348(I)	6A	~~4~~ 6
Equipment Drains	P45-F010-(B)	348(O)	6A	~~4~~ 6
Floor Drains	P45-F003-(A)	349(I)	6A	~~4~~ 6
Floor Drains	P45-F004-(B)	349(O)	6A	~~4~~ 6
Service Air	P52-F195-B	363(O)	6A	16
Chemical Sump Disch.	P45-F096-A	364(I)	6A	~~8~~ 9
Chemical Sump Disch.	P45-F097-B	364(O)	6A	~~8~~ 9
RWCU to Heat Exch.	G33-F253-B	366(O)	8	~~30~~ 35
Reactor Water Sample Line	B33-F019-B	465(I)	10	~~28.4~~ 36
Reactor Water Sample Line	B33-F020-A	465(O)	10	~~28.4~~ 36

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	
2. Manual Isolation Valves(g)#			306
a. Containment			
Main Steam Lines	E32-F001A-A	5(O)	
Main Steam Lines	E32-F001E-A	6(O)	
Main Steam Lines	E32-F001J-A	7(O)	
Main Steam Lines	E32-F001N-A	8(O)	
Feedwater Inlet	B21-F065A-A	9(O)(b)	
Feedwater Inlet	B21-F065B-A	10(O)(b)	
RHR Pump "A" Suction	E12-F004A-A	11(O)(d)	
RHR Pump "B" Suction	E12-F004B-B	12(O)(d)	
RHR Pump "C" Suction	E12-F004C-B	13(O)(d)	306
Insert A → RHR Heat Ex. "A" to LPCI	E12-F027A-A	20(O)~~(e)~~	306 020
Insert B → RHR Heat Ex. "B" to LPCI	E12-F027B - B	21(O)~~(e)~~	
RHR Pump "C" to LPCI	E12-F042C-B	22(O)~~(e)~~	
RHR "A" Test Line To Suppr. Pool	E12-F064A-A	23(O)(d)	
RHR "C" Test Line To Suppr. Pool	E12-F064C-B	24(O)~~(d)~~	020
HPCS Suction	E22-F015-C	25(O)(d)	
HPCS Discharge	E22-F004-C	26(O)~~(e)~~	020
HPCS Test Line	E22-F012-C	27(O)(d)	
RCIC Turbine Exh.	E51-F068-A	29(O)(c)	
LPCS Pump Suction	E21-F001-A	30(O)(d)	
LPCS Pump Discharge	E21-F005-A	31(O)~~(e)~~	020
LPCS Min. Flow	E21-F011-A	32(O)~~(d)~~	
CRD Pump Discharge	C11-F083-A	33(O)	
CCW Supply	P42-F066-A	44(O)	
CCW Return	P42-F067-A	45(O)	
CCW Return	P42-F068-B	45(I)	
RCIC Pump Discharge Min. Flow	E51-F019-A	46(O)(d)	
Reactor Recirc. Post Accident Sampling	B33-F128-B	47(I)	

Inserts to Table 3.6.4-1, Page 3/4 6-34

Insert A

RHR Heat Exchanger "A" to LPCI	E12-F042A-A	20(I)

Insert B

RHR Heat Exchanger "B" to LPCI	E12-F042B-B	21(I)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
Reactor Recirc. Post Accident Sampling	B33-F127-A	47(O)
Vent Header to Supp. Pool	E12-F073B-B	48(O)(d)
RHR Pump "B" Test Line	E12-F064B-B	67(O)(d)
RHR "C" Relief Vlv. Vent Hdr. to Suppr. Pool & Post-Acc. Sample Ret.	E12-F346-B	71B(O)(c)
RHR Heat Ex. "A" Relief	E12-F073A-A	77(O)(d)
Reactor Recirc. Accident Sampling	B33-F126-B	81(I)
Reactor Recirc. Accident Sampling	B33-F125-A	81(O)
SSW Supply "A"	P41-F159A-A	89(O)(c)
SSW Return "A"	P41-F168A-A	90(I)(c)
SSW Return "A"	P41-F160A-A	90(O)(c)
SSW Return "B"	P41-F168B-B	91(I)(c)
SSW Return "B"	P41-F160B-B	91(O)(c)
SSW Supply "B"	P41-F159B-B	92(O)(c)
Drywell Press. Inst.	M71-F593-A	101C(O)
Drywell Press. Inst.	M71-F591A-A	101F(O)
Drywell Press. Inst.	M71-F591B-B	102D(O)
Ctmt. Press. Inst.	M71-F592A-A	103D(O)
Ctmt. Press. Inst.	M71-F592B-B	104D(O)
Drywell H_2 Analyzer Sample	E61-F595C-(A)	106A(O)
Drywell H_2 Analyzer Sample	E61-F595D-(B)	106A(I)
Drywell H_2 Analyzer Sample Ret.	E61-F597C-(A)	106B(O)
Drywell H_2 Analyzer Sample Ret.	E61-F597D-(B)	106B(I)
Ctmt. H_2 Analyzer Sample	E61-F596C-(A)	105A(O)
Ctmt. H_2 Analyzer Sample	E61-F596D-(B)	105A(I)
Ctmt. H_2 Analyzer Sample Ret.	E61-F598C-(A)	106E(O)
Ctmt. H_2 Analyzer Sample Ret.	E61-F598D-(B)	106E(I)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
Ctmt. H_2 Analyzer Sample	E61-F596A-(A)	108A(O)
Ctmt. H_2 Analyzer Sample	E61-F596B-(B)	108A(I)
Ctmt. H_2 Analyzer Sample Ret.	E61-F598A-(A)	107B(O)
Ctmt. H_2 Analyzer Sample Ret.	E61-F598B-(B)	107B(I)
Drywell H_2 Analyzer Sample	E61-F595A-(A)	107D(O)
Drywell H_2 Analyzer Sample	E61-F595B-(B)	107D(I)
Drywell H_2 Analyzer Sample Ret.	E61-F597A-(A)	107E(O)
Drywell H_2 Analyzer Sample Ret.	E61-F597B-(B)	107E(I)
Drywell Fiss. Prod. Monitor Sample	D23-F592-A	109A(O)
Drywell Fiss. Prod. Monitor Sample	D23-F591-B	109A(I)
Drywell Fiss. Prod. Mon. Smpl. Ret.	D23-F594-A	109B(O)
Drywell Fiss. Prod. Mon. Smpl. Ret.	D23-F593-B	109B(I)
Ctmt. Press. Inst. (Post Acc. Smpl.)	M71-F594-B	109D(O)
Ctmt. Press. Inst. (Post Acc. Smpl.)	M71-F595-A	109D(I)
Suppr. Pool Level Inst.	E30-F593A-A	113(O)[c]
Suppr. Pool Level Inst.	E30-F592A-A	114(O)
Suppr. Pool Level Inst.	E30-F594A-A	115(O)[c]
Suppr. Pool Level Inst.	E30-F591A-A	116(O)
Suppr. Pool Level Inst.	E30-F593B-B	117(O)[c]
Suppr. Pool Level Inst.	E30-F592B-B	118(O)
Suppr. Pool Level Inst.	E30-F594B-B	119(O)[c]
Suppr. Pool Level Inst.	E30-F591B-B	120(O)

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
b. Drywell		
Cont. Cooling Water Inlet	P42-F114-B	329(O)
Cont. Cooling Water Outlet	P42-F116-A	330(I)
Cont. Cooling Water Outlet	P42-F117-B	330(O)
3. Other Isolation Valves(g) #		306
a. Containment		
Fuel Transfer Tube	F11-E015	4(I)
Feedwater Inlet	B21-F010A	9(I)(f)
Feedwater Inlet	B21-F032A	9(O)(f)
Feedwater Inlet	B21-F010B	10(I)(f)
Feedwater Inlet	B21-F032B	10(O)(f)
RHR "A" Suction	E12-F017A	11(O)(d)
RHR "B" Suction	E12-F017B	12(O)(d)
RHR "C" Suction	E12-F017C	13(O)(d)
RHR Shutdown Cooling Suction	E12-F308	14(I)~~(c)~~
RHR Head Spray	E51-F066(A)	18(I)~~(c)~~ 306
RHR Head Spray	E12-F344	18(I)~~(c)~~
RHR Heat Ex. "A" to LPCI	E12-F044A	20(I)~~(c)~~
RHR Heat Ex. "A" to LPCI	E12-F025A	20(I)~~(c)~~
RHR Heat Ex. "A" to LPCI	E12-F107A	20(I)~~(c)~~ 020
RHR Heat Ex. "B" to LPCI	E12-F025B	21(I)~~(c)~~
RHR Heat Ex. "B" to LPCI	E12-F044B	21(I)~~(c)~~
RHR Heat Ex. "B" to LPCI	E12-F107B	21(I)~~(c)~~

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	
Containment (Continued)			
RHR Heat Ex. "C" to LPCI	E12-F234	22(O)~~(e)~~	020
RHR Pump "C" to LPCI	E12-F041C-B	22(I)~~(e)~~	
RHR Pump "A" Test Line to Suppr. Pool	E12-F259	23(O)(e)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F261	23(O)(e)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F227	23(O)(e)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F262	23(O)(e)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F228	23(O)(e)	
RHR "A" Test Line to Supp. Pool	E12-F290A-A	23(O)(d)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F338	23(O)(c)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F339	23(O)(c)	
RHR Pump "A" Test Line to Suppr. Pool	E12-F260	23(O)(e)	
RHR Pump "C" Test Line to Suppr. Pool	E12-F280	24(O)~~(e)~~	020
RHR Pump "C" Test Line to Suppr. Pool	E12-F281	24(O)~~(e)~~	
HPCS Suction	E22-F014	25(O)(d)	
HPCS Discharge	E22-F005-(c)	26(I)~~(e)~~	306 020
HPCS Discharge	E22-F218	26(I)~~(c)~~	
HPCS Discharge	E22-F201	26(I)~~(c)~~	
HPCS Test Line	E22-F035	27(O)(d)	
HPCS Test Line	E22-F302	27(O)(e)	
HPCS Test Line	E22-F301	27(O)(e)	
LPCS Pump Suction	E21-F031	30(O)(d)	
LPCS Discharge	E21-F006-(A)	31(I)~~(c)~~	306 020
LPCS Discharge	E21-F200	31(I)~~(c)~~	
LPCS Discharge	E21-F207	31(I)~~(c)~~	
LPCS Test Line	E21-F217	32(O)~~(e)~~	
LPCS Test Line	E21-F218	32(O)~~(e)~~	

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
CRD Pump Discharge	C11-F122	33(I)
PSW Supply	P44-F043	37(I)
Plant Chilled Water Supply	P71-F151	38(I)
Service Air Supply	P52-F122	41(I)
Instr. Air Supply	P53-F002	42(I)
CCW Supply	P42-F035	44(I)
RCIC Disch. Min. Flow	E51-F251	46(0)(c)
RCIC Disch. Min. Flow	E51-F252	46(0)(c)
RHR Heat Ex. "B" Relief Vent Header	E12-F055B	48(0)(d)
RHR Heat Ex. "B" Relief Vent Header	E12-F103B	48(0)(d)
RHR Heat Ex. "B" Relief Vent Header	E12-F104B	48(0)(d)
Refueling Wtr. Stg. Tk. to Upper Ctmt. Pool	G41-F053	54(0)
Refueling Wtr. Stg. Tk. to Upper Ctmt. Pool	G41-F201	54(I)
Condensate Supply	P11-F004	56(I)
FPC & CU to Upper Cont. Pool	G41-F040	57(I)
Stby. Liquid Control Sys. Mix. Tk. (future use)	C41-F151	61(I)
Stby. Liquid Control Sys. Mix. Tk. (future use)	C41-F150	61(0)
RHR Pump "B" Test Line	E12-F276	67(0)(e)
RHR Pump "B" Test Line	E12-F277	67(0)(e)
RHR Pump "B" Test Line	E12-F212	67(0)(e)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR Pump "B" Test Line	E12-F213	67(O)(e)
RHR Pump "B" Test Line	E12-F249	67(O)(e)
RHR Pump "B" Test Line	E12-F250	67(O)(e)
RHR Pump "B" Test Line	E12-F334	67(O)(c)
RHR Pump "B" Test Line	E12-E335	67(O)(c)
RHR "B" Test Line To Suppr. Pool	E12-F290B-B	67(O)(d)
Inst. Air to ADS	P53-F006	70(I)
LPCS Relief Valve Vent Header	E21-F018	71A(O)(d)
RHR Pump "C" Relief Valve Vent Header	E12-F025C	71B(O)(d)
RHR Shutdown Vent Header	E12-F036	73(O)~~(d)~~
RHR Shutdown Suction Relief Valve Disch.	E12-F005	76B(O)~~(d)~~
RHR Heat Ex. "A" Relief Vent Header	E12-F055A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F103A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F104A	77(O)(d)
SSW "A" Supply	P41-F169A	89(I)(c)
SSW "B" Supply	P41-F169B	92(I)(c)
Ctmt. Leak Rate Test Inst.	M61-F015	110A(I)
Ctmt. Leak Rate Test Inst.	M61-F014	110A(O)
Ctmt. Leak Rate Test Inst.	M61-F019	110C(I)
Ctmt. Leak Rate Test Inst.	M61-F018	110C(O)
Ctmt. Leak Rate Test Inst.	M61-F017	110F(I)
Ctmt. Leak Rate Test Inst.	M61-F016	110F(O)

020

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	
4. Test Connections(g)			
a. Containment			
Main Steam T/C	B21-F025A	5(O)	
Main Steam T/C	B21-F025B	6(O)	
Main Steam T/C	B21-F025C	7(O)	
Main Steam T/C	B21-F025D	8(O)	
Feedwater T/C	B21-F030A	9(O)(f)	
Feedwater T/C	B21-F063A	9(O)(f)	
Feedwater T/C	B21-F063B	10(O)(f)	
Feedwater T/C	B21-F030B	10(O)(f)	
RHR Shutdown Cool. Suction T/C	E12-F002	14(O)~~(c)~~	020
RCIC Steam Line T/C	E51-F072	17(O)	
RHR to Head Spray T/C	E12-F342	18(O)~~(c)~~	
RHR to Head Spray T/C	E12-F061	18(O)~~(c)~~	020
LPCI "C" T/C	E12-F056C	22(O)~~(c)~~	
RHR "A" Pump Test Line T/C	E12-F322	23(O)(c)	
RHR "A" Pump Test Line T/C	E12-F336	23(O)(c)	
RHR "A" Pump Test Line T/C	E12-F349	23(O)(c)	
RHR "A" Pump Test Line T/C	E12-F303	23(O)(c)	
RHR "A" Pump Test Line T/C	E12-F310	23(O)(c)	
RHR "A" Pump Test Line T/C	E12-F348	23(O)(c)	
RHR"C" Pump Test Line T/C	E12-F311	24(O)~~(c)~~	020
RHR"C" Pump Test Line T/C	E12-F304	24(O)~~(c)~~	
HPCS Discharge T/C	E22-F021	26(O)~~(c)~~	
HPCS Test Line T/C	E22-F303	27(O)(c)	
HPCS Test Line T/C	E22-F304	27(O)(c)	
RCIC Turbine Exhaust T/C	E51-F258	29(O)(c)	
RCIC Turbine Exhaust T/C	E51-F257	29(O)(c)	
LPCS T/C	E21-F013	31(O)~~(c)~~	
LPCS Test Line T/C	E21-F222	32(O)~~(c)~~	020
LPCS Test Line T/C	E21-F221	32(O)~~(c)~~	

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR "B" Test Line T/C	E12-F350	67(O)(c)
RHR "B" Test Line T/C	E12-F312	67(O)(c)
RHR "B" Test Line T/C	E12-F305	67(O)(c)
Refueling Water Transf. Pump Suction T/C	P11-F425	69(O)(c)
Refueling Water Transf. Pump Suction T/C	P11-F132	69(O)(c)
Inst. Air to ADS T/C	P53-F043	70(O)
Cont. Leak Rate T/C	M61-F010	82(I)
RWCU To Feedwater T/C	G33-F055	83(O)
Suppr. Pool Cleanup T/C	P60-F011	85(O)
Suppr. Pool Cleanup T/C	P60-F034	85(O)
RWCU Pump Suction T/C	G33-F002	87(O)
RWCU Pump Discharge T/C	G33-F061	88(O)
SSW T/C	P41-F163A	89(O)(c)
SSW T/C	P41-F163B	92(O)(c)
b. Drywell		
LPCI "A" T/C	E12-F056A	313(O)
LPCI "B" T/C	E12-F056B	314(O)
Instrument Air T/C	P53-F493	335 ~~327~~(O)
SLCS T/C	C41-F026	328(O)
Service Air T/C	P52-F476	363(O)
RWCU T/C	G33-F120	366(I)
Reactor Sample T/C	B33-F021	465(O)

3/4.6.6 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the primary or secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 31 days that:

1. All Auxiliary Building and Enclosure Building equipment hatches and blowout panels are closed and sealed.

2. The door in each access to the Auxiliary Building and Enclosure Building is closed, except for routine entry and exit.

3. All Auxiliary Building and Enclosure Building penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, rupture discs, or deactivated automatic dampers/valves secured in position. 818

b. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 120 seconds, and

2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.266 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 CFM.

*When irradiated fuel is being handled in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.6.6.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

DAMPER/VALVE FUNCTION # ~~(Number)~~	MAXIMUM ISOLATION TIME (Seconds)
a. Dampers	
Auxiliary Building Ventilation Supply Damper (Q1T41F006)-(B)	4
Auxiliary Building Ventilation Supply Damper (Q1T41F007)-(A)	4
Fuel Handling Area Ventilation Exhaust Damper (Q1T42F003)-(B)	4
Fuel Handling Area Ventilation Exhaust Damper (Q1T42F004)-(A)	4
Fuel Handling Area Ventilation Supply Damper (Q1T42F011)-(A)	4
Fuel Handling Area Ventilation Supply Damper (Q1T42F012)-(B)	4
Fuel Pool Sweep Ventilation Supply Damper (Q1T42F019)-(A)	4
Fuel Pool Sweep Ventilation Supply Damper (Q1T42F020)-(B)	4
Containment & Drywell Area Ventilation Supply Damper (Q1M41F007)-(B)	4
Containment & Drywell Area Ventilation Supply Damper (Q1M41F008)-(A)	4
Containment & Drywell Area Ventilation Exhuast Damper (Q1M41F036)-(A)	4
Containment & Drywell Area Ventilation Exhaust Damper (Q1M41F037)-(B)	4

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The "-(A), -(B)" designators on the valve/damper numbers indicate associated electrical divisions.

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TABLE 3.6.6.2-1 (Continued)

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

VALVE FUNCTION ~~(Number)~~	MAXIMUM ISOLATION TIME (Seconds)
b. Valves	
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F306) - (A)	30
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F304) - (A)	30
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F302) - (A)	4
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F300) - (A)	4
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F307) - (B)	30
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F305) - (B)	30
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F303) - (B)	4
Plant Chilled Water System Aux. Bldg. Isol. Valve (P71-F301) - (B)	4
Service Air System Aux. Bldg. Isol. Valve (P52-F221A) - (A)	4
Service Air System Aux. Bldg. Isol. Valve (P52-F160A) - (A)	4
Service Air System Aux. Bldg. Isol. Valve (P52-F221B) - (B)	4
Service Air System Aux. Bldg. Isol. Valve (P52-F160B) - (B)	4
Instrument Air System Aux. Bldg. Isol. Valve (P53-F026A) - (A)	4
Instrument Air System Aux. Bldg. Isol. Valve (P53-F026B) - (B)	4
FPCC Filt-Demin System Backwash Aux. Bldg. Isol. Valve (G46-F253) - (A & B)	30

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TABLE 3.6.6.2-1 (Continued)

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

VALVE FUNCTION ~~(Number)~~	MAXIMUM ISOLATION TIME (Seconds)
Valves (Continued)	
RWCU Backwash RCVG Tk. Aux. Bldg. Isol. Valve (G36-F108) – (A)	30
RWCU Backwash RCVG Tk. Aux. Bldg. Isol. Valve (G36-F109) – (B)	30
Nuclear Boiler System Aux. Bldg. Isol. Valve (B21-F113) – (A)	30
Nuclear Boiler System Aux. Bldg. Isol. Valve (B21-F114) – (B)	30
RWCU Aux. Bldg. Isol. Valve (G33-F235) – (A)	30
RWCU Aux. Bldg. Isol. Valve (G33-F234) – (B)	30
SPCU Aux. Bldg. Isol. Valve (P60-F003) – (A)	30
SPCU Aux. Bldg. Isol. Valve (P60-F004) – (B)	30
SPCU Aux. Bldg. Isol. Valve (P60-F007) – (B)	30
SPCU Aux. Bldg. Isol. Valve (P60-F008) – (A)	30
Fire Protection System Aux. Bldg. Isol. Valve (P64-F282A) – (A)	4
Fire Protection System Aux. Bldg. Isol. Valve (P64-F283A) – (A)	4
Fire Protection System Aux. Bldg. Isol. Valve (P64-F332A) – (A)	4
Fire Protection System Aux. Bldg. Isol. Valve (P64-F282B) – (B)	4
Fire Protection System Aux. Bldg. Isol. Valve (P64-F283B) – (B)	4
Fire Protection System Aux. Bldg. Isol. Valve (P64-F332B) – (B)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F062) – (A)	9

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TABLE 3.6.6.2-1 (Continued)

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

VALVE FUNCTION ~~(Number)~~	MAXIMUM ISOLATION TIME (Seconds)
Valves (Continued)	
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F064)-(A)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F066)-(A)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F047)-(A)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F063)-(B)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F065)-(B)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F067)-(B)	4
Cond. & Refuel Water Transfer Aux. Bldg. Isol. Valve (P11-F061)-(B)	4
Floor and Equipment Drains System Aux. Bldg. Isol. Valve (P45-F158)-(A)	9
Floor and Equipment Drains System Aux. Bldg. Isol. Valve (P45-F160)-(A)	9
Floor and Equipment Drains System Aux. Bldg. Isol. Valve (P45-F163)-(A+B)	9
Floor and Equipment Drains System Aux. Bldg. Isol. Valve (P45-F159)-(B)	9
Floor and Equipment Drains System Aux. Bldg. Isol. Valve (P45-F161)-(B)	9
Makeup Water Treatment Sys. Aux. Bldg. Isol. Valve (P21-F024)-(A)	30
Domestic Water System Aux. Bldg. Isol. Valve (P66-F029A)-(A+B)	4
PSW Aux. Bldg. Isol. Valve (P44-F121)-(A)	100

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TABLE 3.6.6.2-1 (Continued)

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

VALVE FUNCTION ~~(Number)~~	MAXIMUM ISOLATION TIME (Seconds)
Valves (Continued)	
PSW Aux. Bldg. Isol. Valve (P44-F122)-(A)	100
PSW Aux. Bldg. Isol. Valve (P44-F117)-(A)	100
PSW Aux. Bldg. Isol. Valve (P44-F118)-(A)	100
PSW Aux. Bldg. Isol. Valve (P44-F120)-(B)	100
PSW Aux. Bldg. Isol. Valve (P44-F123)-(B)	100
PSW Aux. Bldg. Isol. Valve (P44-F116)-(B)	100
PSW Aux. Bldg. Isol. Valve (P44-F119)-(B)	100
RHR "A" Loop Discharge To Liquid Radwaste Valve (E12-F203)-(A)	30

CONTAINMENT SYSTEMS

3/4.6.7 ATMOSPHERE CONTROL

CONTAINMENT ~~AND DRYWELL~~ HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.7.1 Two independent containment ~~and drywell~~ hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment ~~and drywell~~ hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.1 Each containment ~~and drywell~~ hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Maintain ≥700°F for at least 2 hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all control room recombiner instrumentation and control circuits.

2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3. Verifying during a recombiner system functional test that the heater sheath temperature increases to greater than or equal to 1200°F within 5 hours and is maintained between 1150°F and 1300°F for at least 4 hours.

4. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e., loose wiring or structural connections, deposits of foreign materials, etc.

c. [DELETED]

CONTAINMENT AND DRYWELL HYDROGEN IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.2 Two independent containment and drywell hydrogen ignition system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment and drywell hydrogen ignition subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.2 Each containment and drywell hydrogen ignition subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least 41 glow plugs are energized.

b. At least once per 18 months by:

   1. Verifying the cleanliness of each glow plug by a visual inspection.

   2. Energizing each glow plug and verifying a surface temperature of at least 1700°F.

CONTAINMENT AND DRYWELL HYDROGEN IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.2 The containment and drywell hydrogen ignition system consisting of the following:

a. At least two igniter assemblies in each enclosed area specified on Table 3.6.7.2-2,

b. All igniter assemblies adjacent to any inoperable igniter assembly in each open area specified on Table 3.6.7.2-2, and

c. Two independent containment and drywell hydrogen ignition subsystems each consisting of two circuits (as listed on Table 3.6.7.2-1) with no more than two igniter assemblies inoperable per circuit.

shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2,

ACTION:

a. With less than two igniter assemblies OPERABLE in any enclosed area specified in Table 3.6.7.2-2, restore at least two igniter assemblies to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With any adjacent igniter assemblies within an open area as specified on Table 3.6.7.2-2 inoperable restore the igniter assemblies in that open area so that all igniter assemblies adjacent to an inoperable igniter assembly are OPERABLE within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With one containment and drywell hydrogen igniter subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.2 The containment and drywell hydrogen ignition system shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and:

   1. Verifying a visible glow from the glow plug tip of each normally accessible igniter assembly specified in Table 3.6.7.2-2,

   2. Verifying that each circuit of each containment and drywell hydrogen igniter subsystem is conducting sufficient current to energize the minimum required number of igniter assemblies specified on Table 4.6.7.2-1.

b. At every COLD SHUTDOWN, but no more frequently than once per 92 days, by energizing the supply breakers and verifying a visible glow from the glow plug tip of each normally inaccessible igniter assembly specified in Table 3.6.7.2-2.

c. At least once per 18 months by:

   1. Verifying the cleanliness of each glow plug by a visual inspection.

   2. Energizing each glow plug and verifying a surface temperature of at least 1700°F.

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Table 3.6.7.2-1

Hydrogen Igniter Circuits

Division I		Division II	
Circuit 1	Circuit 2	Circuit 1	Circuit 2
D124	D107	D125	D106
D126	D109	D127	D108
D128	D111	D129	D110
D130	D112	D136	D113
D132	D114	D138	D115
D134	D116	D140	D117
D137	D119	D149	D118
D139	D121	D151	D120
D141	D123	D153	D122
D143	D148	D161	D131
D145	D150	D162	D133
D147	D152	D165	D135
D155	D154	D166	D142
D157	D159	D168	D144
D172	D160	D170	D146
D174	D163	D171	D156
D176	D164	D178	D158
D183	D167	D180	D173
D192	D169	D182	D175
D185	D179	D187	D177
D186	D181	D189	D184
	D188	D191	D193
	D190		D194
	D195		

Table 3.6.7.2-2

Hydrogen Igniters and Locations

*Igniter	Div./Circuit	Elevation	Azimuth	Dist. From Center Line of Reactor
NORMALLY ACCESSIBLE				
Open Areas				
Containment				
D124	I/1	136'-0"	21	57'-0"
D125	II/1	132'-10"	47	53'-0"
D126	I/1	132'-10"	75	51'-9"
D127	II/1	132'-10"	107	51'-9"
D128	I/1	132'-10"	135	51'-9"
D129	II/1	132'-10"	165	51'-9"
D130	I/1	132'-10"	195	51'-9"
D131	II/2	145'-7"	220	60'-0"
D132	I/1	134'-4"	253	51'-9"
D133	II/2	134'-4"	285	51'-9"
D134	I/1	134'-4"	317	52'-8"
D135	II/2	136'-0"	349	51'-9"
D137	I/1	160'-4"	36	53'-6"
D138	II/1	157'-10"	70	51'-9"
D139	I/1	157'-10"	100	51'-9"
D140	II/1	160'-4"	135	51'-2"
D141	I/1	155'-10"	164	51'-9"
D142	II/2	155'-10"	196	51'-9"
D143	I/1	165'-0"	226	61'-4"
D144	II/2	160'-4"	260	54'-2"
D145	I/1	159'-4"	285	51'-5"
D146	II/2	159'-4"	321	51'-5"
D148	I/2	182'-9"	30	61'-0"
D149	II/1	167'-8"	41	42'-0"
D154	I/2	182'-4"	136	51'-9"
D155	I/1	182'-4"	254	55'-9"
D156	II/2	183'-4"	274	48'-0"
D157	I/1	182'-4"	293	58'-11"
D158	II/2	183'-4"	320	53'-2"
D160	I/2	202'-0"	35	46'-0"
D161	II/1	207'-9"	59	44'-0"
D170	II/1	207'-7"	135	55'-8"
D171	II/1	206'-0"	216	46'-9"
D172	I/1	204'-11"	252	26'-0"
D173	II/2	204'-4"	256	53'-8"
D174	I/1	204'-11"	284	53'-8"
D175	II/2	201'-11"	298	26'-8"
D176	I/1	207'-9"	310	56'-6"

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*Igniter	Div./Circuit	Elevation	Azimuth	Dist. From Center Line of Reactor

Open Areas

Containment (Continued)

D178	II/1	262'-0"	6	55'-5"
D179	I/2	262'-0"	48	55'-5"
D180	II/1	262'-0"	91	55'-0"
D181	I/2	262'-0"	140	55'-0"
D182	II/1	262'-0"	183	55'-0"
D183	I/1	262'-0"	225	55'-0"
D184	II/2	262'-0"	268	55'-0"
D185	I/1	262'-0"	333	55'-0"
D186	I/1	283'-10"	349	39'-9"
D187	II/1	283'-10"	34	39'-9"
D188	I/2	283'-10"	81	39'-9"
D189	II/1	283'-10"	127	39'-9"
D190	I/2	283'-10"	152	39'-9"
D191	II/1	283'-10"	199	39'-9"
D192	I/1	283'-10"	242	39'-9"
D193	II/2	283'-10"	286	39'-9"
D194	II/2	295'-0"	349	15'-3"
D195	I/2	295'-0"	158	15'-3"

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NORMALLY INACCESSBILE

Open Areas

Drywell

D106	II/2	146'-3"	0	26'-6"
D107	I/2	145'-7"	63	29'-3"
D108	II/2	146'-2"	120	29'-8"
D109	I/2	147'-1"	180	26'-3"
D110	II/2	145'-7"	240	29'-2"
D111	I/2	145'-7"	313	25'-1 1/4"
D112	I/2	160'-6"	0	27'-4"
D113	II/2	160'-6"	60	29'-9"
D114	I/2	160'-6"	135	27'-1"
D115	II/2	160'-6"	180	26'-10"
D116	I/2	160'-6"	232	26'-1"
D117	II/2	160'-6"	324	26'-5"
D118	II/2	179'-0"	0	26'-4"
D119	I/2	179'-0"	65	26'-4"
D120	II/2	179'-0"	125	26'-4"
D121	I/2	179'-0"	180	26'-4"
D122	II/2	179'-0"	245	26'-4"
D123	I/2	179'-0"	305	26'-4"

*Igniter	Div./Circuit	Elevation	Azimuth	Dist. From Center Line of Reactor
NORMALLY INACCESSIBLE (Continued)				
Enclosed Areas				
Main Steam Tunnel				
D136	II/1	166'-0"	16	51'-9"
D147	I/1	166'-0"	344	51'-9"
RWCU Backwash Room				
D150	I/2	168'-10"	70	42'-0"
D151	II/1	168'-10"	105	42'-0"
D152	I/2	178'-10"	70	46'-2"
D153	II/1	178'-10"	109	51'-5"
RWCU Heat Exchanger Room				
D159	I/2	202'-0"	21	50'-4"
D177	II/2	202'-0"	341	55'-0"
Filter/Demineralizer Room				
D162	II/1	202'-0"	74	55'-8"
D163	I/2	202'-0"	88	48'-0"
D164	I/2	202'-0"	92	48'-0"
D165	II/1	202'-0"	106	55'-8"
RWCU Pump Area				
D166	II/1	202'-0"	93	45'-0"
D167	I/2	202'-0"	86	37'-6"
RWCU Sample Area				
D168	II/1	202'-0"	86	34'-0"
D169	I/2	202'-0"	96	22'-6"

*System Prefix is E61 for all igniters

Table 4.6.7.2-1

NUMBER OF IGNITERS BY CIRCUIT

	Minimum Required	Total on Circuit
Division I		
Circuit 1	19	21
Circuit 2	22	24
Division II		
Circuit 1	20	22
Circuit 2	21	23

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DRYWELL PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.3 Two independent drywell purge system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell purge system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS ~~Continued~~

4.6.7.3 Each drywell purge system subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Starting the subsystem from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by:

   1. Verifying a subsystem flow rate of at least 1000 cfm during subsystem operation for at least 15 minutes.

   2. Verifying the pressure differential required to open the vacuum breakers on the drywell purge compressor discharge lines, from the closed position, to be less than or equal to 1.0 psid.

c. Verifying the OPERABILITY of the drywell purge compressor discharge line vacuum breaker isolation valve differential pressure actuation instrumentation with an opening setpoint of 0.0 to 1.0 psid (Drywell minus Containment) by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,

   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   3. CHANNEL CALIBRATION at least once per 18 months.

3.4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 11.5 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment. Insert, see next page 379

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

Insert to Bases 3/4.6.1.3, Page B3/4 6-1

Verification that each air lock door inflatable seal system is OPERABLE by the performance of a local leak-detection test for a period of less than 48 hours is permissible if it can be demonstrated that the leakage rate can be accurately determined for this shorter period. This is in accordance with Section 6.4 and 7.6 of ANSI N45.4-1972.

CONTAINMENT PURGE SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failures develop. The 0.60 L_a leaking limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by containment spray. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

Insert A

The limitation on drywell bypass leakage rate ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the containment exceeding its design pressure of 15.0 psig. The integrated drywell leakage value is limited to 10% of the ~~design drywell leakage rate.~~ allowable drywell leakage capability, Insert B 172

Insert C The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and containment will result in a significant pressure buildup in the containment due to this bypass leakage.

3/4.6.2.3 DRYWELL AIR LOCKS

The limitations on closure for the drywell air locks are required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the drywell. INSERT 379

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak pressure of 22.0 psig does not exceed the design pressure of 30.0 psig and that the containment peak pressure of 11.5 psig does not exceed the design pressure of 15.0 psig during LOCA conditions. The maximum external drywell pressure differential is limited to +1.0 psid, well below the 2.3 psid at which suppression pool water will be forced over the wier wall and into the drywell. The limit of 2.0 psid for initial positive drywell to containment pressure will ~~limit the drywell pressure to 22.0 psig which is less than the design pressure and~~ is consistent with the safety analysis. 127

not allow clearing of the top vent which

Insert to Bases 3/4.6.2.2, Page B3/4 6-3

Insert "A"

The design drywell leakage rate is expressed as $A/\sqrt{k}$ and has a value of 0.90 ft^2. $A/\sqrt{k}$ is dependent only on the geometry of drywell leakage paths where A = flow area of leakage paths in ft^2 and k is a lumped constant which considers geometric and friction loss coefficients such as discontinuities and Reynolds numbers. At a 3 psi differential pressure from drywell to containment an $A/\sqrt{k}$ has an equivalent mass flow of 35,000 scfm.

Insert "B"

which is equivalent to 3500 scfm at 3 psid drywell to containment.

Insert "C"

The $A/\sqrt{k}$ value of 0.90 ft^2 is derived from the analysis of "bypass capability with containment spray and heat sinks". (FSAR 6.2.1.1.5.5)

Insert to Bases 3/4.6.2.3, Page B3/4 6-3

Verification that each air lock door inflatable seal system is OPERABLE by the performance of a local leak-detection test for a period of less than 48 hours is permissible if it can be demonstrated that the leakage rate can be accurately determined for this shorter period. This is in accordance with Section 6.4 and 7.6 of ANSI N45.4-1972.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of 30 psig and 15 psig, respectively, during primary system blowdown from full operating pressure. 1060 psia

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from ~~1060 psig.~~ 320
Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of 15 psig. Similarly the drywell pressure remains below the design pressure of 30 psig. The maximum and minimum water volumes for the suppression pool are 138,851 cubic feet and 136,146 cubic feet, respectively. These values include the water volume of the containment pool, horizontal INSERT A 168
vents, and weir annulus. Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes.

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation without excessive loads on the containment pool walls will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

In addition to the limits on temperature of the suppression pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

The containment spray system consists of two 100% capacity trains, each with three spray rings located at different elevations about the inside circumference of the containment. RHR A pump supplies one train and RHR pump Insert B 233
B supplies the other. RHR pump C cannot supply the containment spray system. Dispersion of the flow of water is effected by 350 nozzles in each train, Add
enhancing the condensation of water vapor in the containment volume and Insert C 169
preventing overpressurization. Heat rejection is through the RHR heat exchangers. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control.

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing and have corresponding pool water depths of 18'4-1/12" and 18' 9-3/4" respectively.

The minimum suppression pool volume of 135,291 ft^3 is based on satisfying the Mark III suppression pool sizing criteria established in a GE Design Review File (DRF T23-0408). The 135,291 ft^3 and 138,701 ft^3 pool water volumes were used in the Grand Gulf containment analysis (GE DRF 699-0010) to verify their adequacy to perform all required functions following the design basis Limiting Condition of a main steam line break from 105% of full power. (FSAR 6.2.1.1.3.3.3)

The 18'4-1/12" and 18' 9-3/4" pool water depths are nominal values derived analytically, considering pool geometry, from the above pool volumes which were used in the pool design analyses. The pool levels (depth) satisfy criteria or constraints imposed by: (1) 2-foot minimum post-LOCA horizontal vent coverage to assure steam condensation/pressure suppression, (2) adequate ECCS pump NPSH, (3) adequate depth for vortex prevention, (4) adequate depth for minimum recirculation volume, (5) adequate weir-wall free-board for inadvertent upper pool dump and (6) to limit hydrodynamic loads on submerged structures during SRV and vent steam discharges.

The suppression pool temperatures are based as follows:

95°F — Is the initial condition for the analysis determining pool volume adequacy and satisfies the post-LOCA long term peak pool temperature of 185°F.

120°F — Is analytically based and is derived to satisfy the 170°F post-blowdown peak pool temperature assuming a LOCA when the reactor is isolated.

105°F and 110°F — Are derived from the analytically based 95°F and 120°F using engineering judgement considering operator response time, reactor pressure vessel energy and pool heat capacity to meet the 170°F limit and also to avoid unnecessary scrams and/or depressurizations.

Insert B to Bases Section 3/4.6.3, Page B3/4 6-4

The surveillance requirements, which include system losses for surveillance testing, provide adequate assurance that the containment spray system will be OPERABLE when required.

Insert C to Bases 3/4.6.3, Page B 3/4 6-4

Whenever maintenance activities are performed that could result in nozzle obstruction, a surveillance will be performed to verify that flow through the nozzles is unobstructed.

DEPRESSURIZATION SYSTEMS (Continued)

excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the standby service water is accomplished in the RHR heat exchangers.

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow through two 100% capacity dump lines following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes, ensuring the long term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. The minimum freeboard distance above the suppression pool high water level to the top of the weir wall is adequate ~~to~~ preclude flooding of the drywell in the event of an inadvertent dump. During refueling, neither automatic nor manual action can open the make-up dump valves.

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The operability of the drywell isolation valves ensures that the drywell atmosphere will be directed to the suppression pool for the full spectrum of pipe breaks inside the drywell. Since the allowable value of drywell leakage is so large, individual drywell penetration leakage is not measured. By checking valve operability on any penetration which could contribute a large fraction of the design leakage, the total leakage is maintained at less than the design value.

Insert → The maximum isolation times for containment and drywell automatic isolation valves are the times used in the FSAR accident analysis for valves with analytical closing times. For automatic isolation valves not having analytical closing times, closing times are derived by applying margins to previous valve closing test data obtained by using ASME Section XI criteria. Maximum closing times for these valves was determined by using a factor of two times the allowable (from previous test closure to next test closure) ASME Section XI margin and adding this to the previous test closure time. 170

3/4.6.5 DRYWELL POST-LOCA VACUUM BREAKERS

The post-LOCA drywell vacuum breaker system is provided to relieve the vacuum in the drywell due to steam condensation following blow-down. Containment air is drawn through the vacuum breaker check valves in the two branches of the separate post-LOCA vacuum relief line and in a branch of each drywell purge compressor discharge line. Vacuum relief initiates at a differential pressure of one psi. This vacuum relief, in conjunction with the rest of the

Insert to Bases 3/4.6.4, Page B 3/4 6-5

Table 3.6.4-1 lists the Containment and Drywell Isolation Valves in four sections. Section 1 contains the Automatic Isolation Valves which are those valves that receive an automatic isolation signal from Table 3.3.2-1 instrumentation and are located on the Containment or Drywell penetrations. The valves included in Section 2 are Manual Isolation Valves which receive a remote manual signal from a handswitch and are located on the Containment or Drywell Penetrations. Some of the valves in Section 2 may receive automatic signals, but not automatic isolation signals from instrumentation in Table 3.3.2-1. The valves included in Section 3 are those which do not receive isolation signals from instrumentation listed in Table 3.3.2-1 and do not utilize a remote manual handswitch. Section 3 includes check valves, local manual operated valves and power operated valves that do not utilize a handswitch. Section 4 of Table 3.6.4-1 contains test connection valves.

DRYWELL POST-LOCA VACUUM BREAKERS (Continued)

drywell purge system, is necessary to insure that the post-LOCA drywell H_2 concentration does not exceed 4% by volume.

Following vacuum relief, the drywell purge system pressurizes the drywell, forcing noncondensibles through the horizontal vents and into the containment at a rate designed to maintain the H_2 concentration below the flammable limits.

There are two 100% vacuum relief systems so that the plant may continue operation with one system out of service for a limited period of time.

3/4.6.6 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Auxiliary Building and Enclosure Building provide secondary containment during normal operation when the containment is sealed and in service. When the reactor is in COLD SHUTDOWN or REFUELING, the containment may be open and the Auxiliary Building and Enclosure Building then become the only containment.

The maximum isolation times for secondary containment automatic isolation dampers/valves are the times used in the FSAR accident analysis for dampers/valves with analytical closing times. For automatic isolation valves not having analytical closing times, closing times are derived by applying margins to previous valve closing test data obtained by using ASME Section XI criteria. Maximum closing times for these valves was determined by using a factor of two times the allowable (from previous test closure to next test closure) ASME Section XI margin and adding this to the previous test closure time.

Establishing and maintaining a vacuum in the Auxiliary Building and Enclosure Building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, latches, dampers, ~~and~~ valves, blind flanges and rupture discs, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Cumulative operation of the system with the heaters OPERABLE for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters.

The surveillance testing for verifying heat dissipation for the Standby Gas Treatment System heaters is performed in accordance with ANSI N510-1975 with the exception of the 5% current phase balance criteria of Section 14.2.3. The offsite power system for the Grand Gulf Nuclear Station consists of a non-transpositional 500 KV grid. The grid has an inherent unbalanced load distribution which results in unbalanced voltages in the plant. Voltage unbalances exceeding the ANSI N510-1975 5% criteria are not atypical.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

UNRESTRICTED AREA BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The unrestricted area boundary for gaseous effluents and for liquid effluents shall be as shown in Figure 5.1.3-1. The gaseous effluent release points are shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment is a steel lined, reinforced concrete structure composed of a vertical right cylinder and a hemispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head which contains an approximately eighteen to nineteen foot deep water filled suppression pool connected to the drywell through a series of horizontal vents. The containment has a minimum net free air volume of 1,400,000 cubic feet. The drywell has a minimum net free air volume of 270,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell 30 psig.
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external-to-internal differential pressure:
   1. Drywell 21 psid.
   2. Containment 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the ~~Reactor~~ Auxiliary Building and the Enclosure Building, and has a minimum free volume of 3,640,000 cubic feet. 283

ATTACHMENT 3

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: MATERIALS ENGINEERING

Listing of Item Numbers by
Technical Specification Problem Sheet (TSPS) Number

TSPS No.	Item Nos.*
160	3.A.01
219	3.D.01
319	3.B.01

*Item number format: 1.A.02

- 02 — Item number within category
- A — Category designator
- 1 — Attachment number

A. TYPOGRAPHICAL ERRORS, EDITORIAL CHANGES, AND CLARIFICATIONS

This proposed change corrects obvious typographical errors, implements editorial changes such as correction of spelling errors, punctuation errors, and grammatical errors or provides clarification of the basic meaning or intent of the subject technical specification.

MP&L has determined that the proposed change does not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed change does not involve a significant hazards consideration.

A description of this change including necessary justification for the change is provided below:

CLARIFICATIONS

A clarification to the technical specifications to improve understanding and readability is discussed below:

1. (TSPS 160), Reactor Coolant Pressure/Temperature Limits, Technical Specification 3/4.4.6, Bases 3/4.4.6

   The proposed change to this specification is to clarify that reactor vessel metal temperature rather than reactor coolant system temperature is to be used in conjunction with Figure 3.4.6.1-1. This will make the specification consistent with the figure which is based on vessel metal temperature. The specification is further revised to refer specifically to reactor coolant temperature for the maximum heatup/cooldown rates. This is conservative, since the coolant temperature will change more rapidly than the vessel metal temperature. Additionally, it is proposed to revise Surveillance Requirement 4.4.6.1.3 to indicate that reactor vessel material specimens will be removed and examined in accordance with Appendix H of 10 CFR 50 and to renumber Surveillance Requirements 4.4.6.1.3 and 4.4.6.1.4 so that requirements dealing with temperature measurement and requirements dealing with materials testing are grouped together appropriately. This proposed change also adds Specification 4.4.6.1.5 to require removal and examination of the reactor flux wire specimens during the first refueling outage for determination of pressure vessel fluence as a function of time and power level. This is consistent with current industry practice, since flux wires contain fewer impurities than material specimens and therefore provide a more accurate indication of neutron fluence. Also, for further clarification, Figure 3.4.6.1-1 is revised to indicate that the region of acceptable operation is to the right of the curves and that B' and C' are coincident with B and C, respectively. Bases 3/4.4.6 is also revised to maintain consistency with the proposed change and to clarify that the revised figure complies with 10 CFR 50, Appendix G. This proposed revision provides consistency with GGNS FSAR Section 5.3.2.1. This change improves the safety of the plant by clarifying certain requirements, assuring conformance with current regulations, and assuring consistency with the plant as described in the safety analyses. (Pages 3/4 4-17, 3/4 4-18, 3/4 4-19, B 3/4 4-4, and B 3/4 4-5)

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

The following change is proposed to render the technic 1 specification consistent with the as-built plant. In all such cases, the as-built plant is consistent with the safety analyses and the licensing basis.

In that this proposed change is inherently consistent with the safety analyses and the licensing basis, it is concluded that the proposed change does not:

- o Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- o Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- o Involve a significant reduction in a margin of safety.

Therefore, the proposed change does not involve a significant hazards consideration.

A description of this change including justification fr the change is provided below:

1. (TSPS 319), Reference Code for Rx Vessel, Technical Specification Bases 2.1.2 and 2.1.3

   This proposed change modifies Bases 2.1.3 to reflect the appropriate Edition and Addenda of the ASME Code. This proposed change makes the technical specifications consistent with the as-built plant and is consistent with Section 5.3.3.1.1.1 of the FSAR. (Page B 2-5)

C. ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES

No technical specification changes in this category are included with this attachment.

D. REGULATORY REQUIREMENTS/REQUESTS/RECOMMENDATIONS

The following change is proposed to render the technical specification consistent with recent changes in NRC policy and the Code of Federal Regulations, as well as to implement changes or enhancements recently requested or recommended by NRC reviewers.

This proposed change is required to render the technical specification consistent with recent NRC guidance, and it has been concluded based on a review of this item that the proposed change does not:

- Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- Involve a significant reduction in a margin of safety.

Therefore, the proposed change does not involve a significant hazards consideration.

A description of this change including justification for the change is provided below:

1. (TSPS 219) Reactor Pressure vs. Metal Temperature Curves, Technical Specification 3/4.4.6, Bases 3/4.4.6

   This proposed change revises the curves of Figure 3.4.6.1-1 to conform to the revised requirements of 10CFR50, Appendix G, paragraph IV.A.2 and 3, (May, 1983); also, a statement is added to Bases 3/4.4.6 to indicate that the RT_{NDT} for welds and base material in the closure region is equal to or less than 10°F and that the initial hydrostatic test pressure was 1563 psig. The curves presently in the technical specifications conform with the requirements of the previous revision of Appendix G, with the exceptions noted in GE BWR Licensing Topical Report NEDO-21778-A. The proposed curves are in compliance with the revised Appendix G requirements to maintain the appropriate margins (as defined in the revised IV.A.2) above the bolt-up reference temperature at pressures exceeding 20 percent of the preservice system hydrostatic test pressure. For pressures below 20 percent of the preservice hydrostatic test pressure, the proposed curves meet the requirement to maintain a 60°F margin above the bolt-up reference temperature, as defined in paragraph IV.A.3. It is therefore concluded that this change is in compliance with 10CFR50, Appendix G requirements and is in accordance with recent NRC guidance to implement the May, 1983 revisions to Appendix G in the Grand Gulf Technical Specifications. (Page 3/4 4-19 and B 3/4 4-4)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

(AFFECTED PAGES ARE PROVIDED IN THE ORDER OF ASCENDING PAGE NUMBERS.)

SAFETY LIMITS

BASES

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code 1971 ~~1974~~ Edition, including Addenda through Winter 1972 ~~Summer 1975~~, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure, 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The pressure Safety Limit is selected to be the transient overpressure allowed by the ASME Boiler and Pressure Vessel Code, Section III, Class I. 319

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system ~~temperature and~~ pressure And reactor vessel metal temperature shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with: 160

a. A maximum reactor coolant heatup of 100°F in any one hour period, 160

b. A maximum reactor coolant cooldown of 100°F in any one hour period, 160

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and, to the right of the limit lines of Figure 3.4.6.1-1 curves A or ~~B~~, B and B', as applicable, at least once per 30 minutes. 160

the reactor coolant system pressure and reactor vessel metal temperature shall be determined to be

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system ~~temperature and~~ pressure AND REACTOR vessel metal temperature shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup. 160

4.4.6.1.~~3~~4 The reactor vessel material specimens shall be removed and examined ~~to determine reactor pressure vessel fluence~~ as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. ~~The results of these fluence determinations shall be used to update the curves of Figure 3.4.6.1-1. The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:~~ 160

~~a. Actual shift in the RT_{NDT} for materials in the capsules as defined by the CVN impact test.~~ 160

~~b. Predicted shift in RT_{NDT} for plate C2594-2 and weld 627260/B322A27AE (heat/lot) as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials".~~

4.4.6.1.~~4~~3 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F: 160

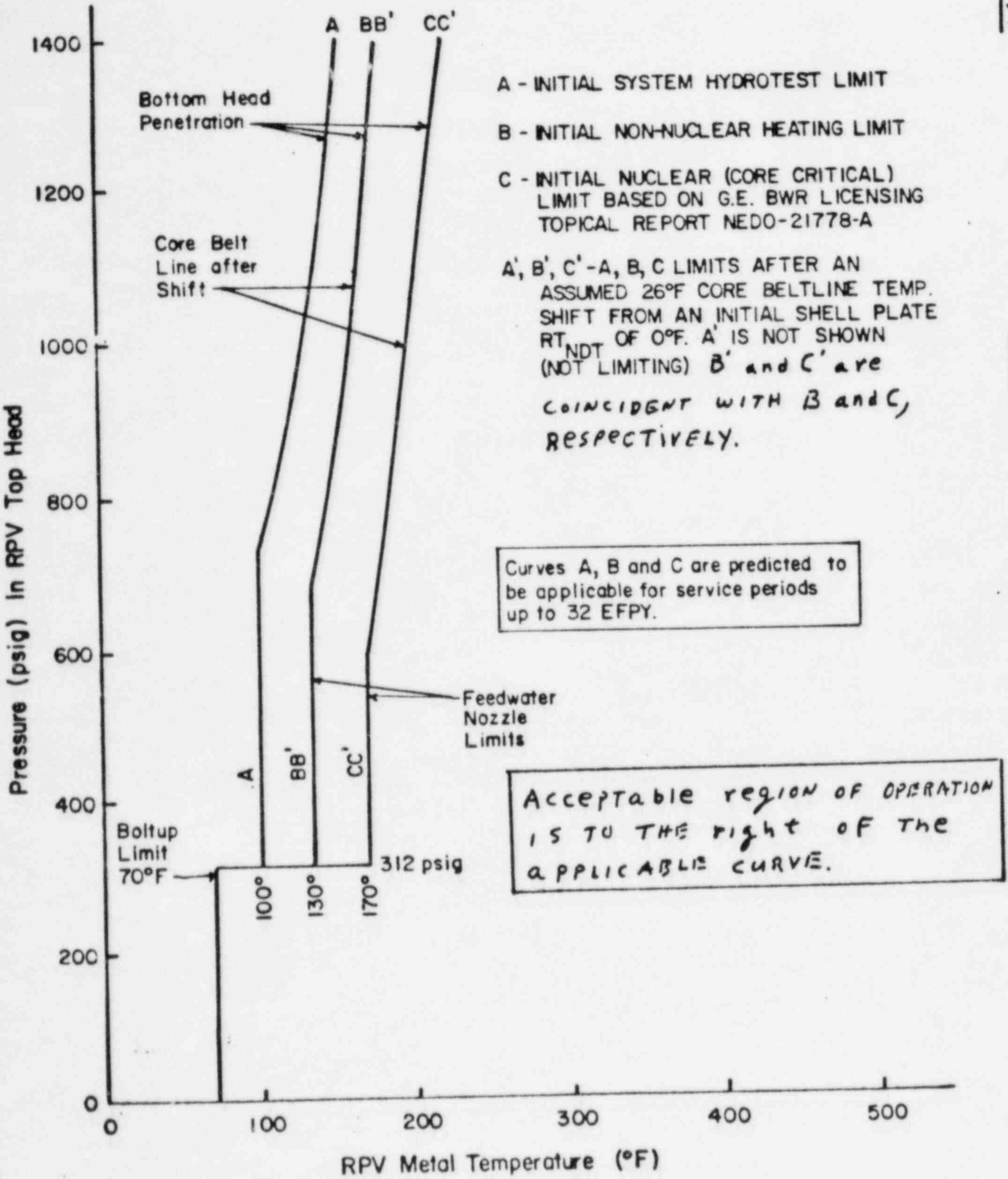
a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

1. $\leq$ 100°F, at least once per 12 hours.

2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

4.4.6.1.5 The REACTOR flux wire specimens shall be removed at the first refueling outage and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4. 4.6-1. The results of the fluence determinations, in conjunction with Figure B 3/4. 4.6-1, shall be used to adjust the curves of Figure 3.4.6.1-1, as required. 160



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

THE RT_{NDT} for welds and base material in the closure flange region is ≤ 10°F. The initial hydrostatic test pressure was 1563 psig.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence as well as adjustments for possible errors in the pressure and temperature sensing instruments. Curves B' and C' are coincident with curves B and C, respectively.

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The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A ~~and A'~~, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing. 160

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, and Addenda through Summer 1978.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.