MISSISSIPPI POWER & LIGHT COMPANY
Helping Build Mississippi
P. O. BOX 1640, JACKSON, MISSISSIPPI 39205

June 22, 1984

JAMES P. McGAUGHY, JR
VICE PRESIDENT

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-13
File 0260/0840/L-860.0
Proposed Amendment to the Operating License (PCOL's-84/11B, 13D, 14C, and 16)
AECM-84/0319

Mississippi Power & Light Company (MP&L) completed its review of the Grand Gulf Nuclear Station Technical Specifications in accordance with the Technical Specification Review Program (TSRP) submitted to the NRC on March 18, 1984 (AECM-84/0183). The results of the TSRP were submitted to the NRC on April 9, 1984 (AECM-84/0217) and on April 19, 1984 (AECM-84/0229). Findings of the TSRP, which require changes to the Grand Gulf Technical Specifications, were identified on Technical Specification Problem Sheets (TSPS). A number of revised Problem Sheets were submitted to the NRC on May 1, 1984 (AECM-84/0251) and on May 8, 1984 (AECM-84/0286). Eleven additional problem sheets summarizing items identified by the NRC were included in the May 1, 1984 letter. Since submittal of the final TSRP results, MP&L and the NRC staff have met numerous times to discuss the TSRP findings, and the justification for, and safety significance of any proposed changes to the Grand Gulf Technical Specifications identified during the TSRP.

On April 18, 1984, the NRC issued an Order Restricting Conditions for Operation of Grand Gulf Unit No. 1, in which twenty-two changes were made to the Grand Gulf Technical Specifications. These twenty-two changes were those identified by MP&L in its TSRP as being necessary to support restart and full power operations for Unit No. 1. With the implementation of these changes to the Grand Gulf Technical Specifications, MP&L was authorized to restart and operate Unit No. 1 under its operating license up to five percent power.

On May 24, 1984 MP&L submitted a proposed amendment to the Grand Gulf Nuclear Station Technical Specifications. This amendment included a revised organization and modified terminology in the administrative section of the technical specifications. This amendment

A001
3/40

tsG1

8406270091 840622
PDR ADOCK 05000416
P PDR

also resolved two technical specification problem sheets as noted in a letter from MP&L to the NRC dated May 25, 1984 (AECM-84/0303). Further proposed changes associated with the TSRP which have been submitted are listed below:

DATE	MP&L CORRESPONDENCE NUMBER
June 17, 1984	AECM-84/0330
June 18, 1984	AECM-84/0336
June 19, 1984	AECM-84/0338
June 20, 1984	AECM-84/0315
June 21, 1984	AECM-84/0318

As a follow-up to the TSRP and the NRC Order of April 18, 1984, MP&L was notified by letter dated May 9, 1984 from Mr. T. M. Novak of the methods to be used in resolving the findings of the TSRP. In accordance with that letter and with 10CFR 50.59 and 10CFR 50.90, MP&L requests that the proposed changes to the Grand Gulf Technical Specifications, set forth in the attachments to this letter be incorporated into the full power amendment to License No. NPF-13. All of these proposed changes to the Grand Gulf Technical Specifications have been reviewed and evaluated by both MP&L and the NRC staff as part of and in conjunction with MP&L's TSRP. The proposed changes in the attachments to this letter, in conjunction with the other proposed changes discussed above, complete the formal submittals of proposed Technical Specification changes planned by MP&L in response to the May 9, 1984 letter from Mr. T. M. Novak.

The description of, technical justification for, and safety evaluation of the proposed changes to the Grand Gulf Technical Specifications are included in Attachments 1 through 6. Each attachment contains all of the proposed technical specification changes which are within the purview of a single branch of the Office of Nuclear Reactor Regulation. The attachments to this letter and the responsible branches for each are listed below:

Attachment	NRC Technical Review Branch
1	Auxiliary Systems
2	Instrumentation and Control Systems
3	Licensee Qualifications
4	Mechanical Engineering
5	Reactor Systems
6	Standardization and Special Projects

It should be noted that the changes requested for the Standardization and Special Projects Branch which involve technical specification pages issued by the May 22, 1984 Order Requiring Diesel Generator Inspection are based on the technical specification pages existing prior to the order and not on the "revised interim technical specifications appended" to the order.

The proposed changes to the Grand Gulf Technical Specifications have been divided into four categories as described below. This categorization was made to assist the NRC staff in expediting its review of the proposed changes to the Grand Gulf Technical Specifications:

TECHNICAL SPECIFICATION CHANGE CATEGORIES

- o TYPOGRAPHICAL ERRORS, EDITORIAL CHANGES, CLARIFICATIONS: Changes which correct obvious typographical errors, implement editorial changes such as correction of spelling errors, punctuation errors and grammatical errors or merely provide clarification of, without changing, the basic meaning and intent of the technical specification being changed.

- o TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY: Changes which 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.

- o ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES: Changes 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.

- o REGULATORY REQUIREMENTS/REQUESTS/RECOMMENDATIONS: Changes or enhancements 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 the NRC.

The enclosed changes to the Grand Gulf Technical Specifications have been reviewed and approved by the Plant Safety Review Committee and the Safety Review Committee. All of the proposed changes have been determined to be conservative with respect to the Grand Gulf safety analyses and, based on the guidelines set forth in 10CFR 50.92, involve no significant hazards considerations.

In accordance with provisions of 10 CFR 50.30, three (3) signed originals and forty (40) copies of the proposed changes to the Grand Gulf Technical Specifications, as described in the attachments to this letter, are hereby formally provided for your review and approval. Based upon MP&L's evaluation of the proposed changes and upon discussions with members of your staff, MP&L has concluded that there should be no additional fee for the proposed technical specification changes.

Yours truly,

James P McGaughy

JPM:lm
Attachments

cc: (See Next Page)

tsG3

BEFORE THE

UNITED STATES REGULATORY COMMISSION

LICENSE NO. NPF-13

DOCKET NO. 50-416

IN THE MATTER OF

MISSISSIPPI POWER & LIGHT COMPANY
and
MIDDLE SOUTH ENERGY, INC.
and
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

AFFIRMATION

I, J. P. McGaughy, Jr., being duly sworn, stated that I am Vice President - Nuclear Support of Mississippi Power & Light Company; that on behalf of Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association I am authorized by Mississippi Power & Light Company to sign and file with the Nuclear Regulatory Commission, this application for amendment of the Operating License of the Grand Gulf Nuclear Station; that I signed this application as Vice President - Nuclear Support of Mississippi Power & Light Company; and that the statements made and the matters set forth therein are true and correct to the best of my knowledge, information and belief.

J P McGaughy

J. P. McGaughy, Jr.

STATE OF MISSISSIPPI
COUNTY OF HINDS

SUBSCRIBED AND SWORN TO before me, a Notary Public, in and for the County and State above named, this 22nd day of June, 1984.

(SEAL)

Paul A Richardson

Notary Public

My commission expires:

My Commission Expires Oct. 27, 1987

ATTACHMENT 1

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: AUXILIARY SYSTEMS

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

TSPS No.	Item Nos.*
002	1.A.01
017	1.A.03
035	1.B.01
058	1.C.05
094	1.D.01
129	1.B.02
132	1.C.06
156	1.A.02
173	1.C.03
176	1.A.04
195	1.C.01
214	1.C.07
229	1.B.03
255	1.A.05
258	1.B.04
267	1.C.02
287	1.C.04
312	1.A.06
313	1.B.05
826	1.C.08

*Item number format: 1.A.02

Item number within category

Category designator

Attachment number

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

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

	TSPS No.	TS Page No.
1.	002	3/4 7-3 3/4 7-30

EDITORIAL CHANGES

A proposed editorial change to the technical specifications is discussed below:

2. (TSPS 156), Standby Liquid Control System, Technical Specification 3/4.1.5

   The subject technical specification specifies the Limiting Conditions for Operation (LCO), ACTION statements, and Surveillance Requirements for the Standby Liquid Control System. This change is proposed to clearly define the requirements for all redundant system components. To clarify the requirements, "pump and/or one explosive valve" is changed to "subsystem" and "The Standby Liquid Control System" is reworded to reflect two subsystems. These changes are purely administrative as they represent an editorial change to clarify the intent of the subject specification. (Page 3/4 1-18, 3/4 1-19)

CLARIFICATIONS

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

3. (TSPS 017), Standby Service Water (SSW), Cooling Tower Fan Surveillance Requirements, Technical Specification 3/4.7.1.3

   The proposed change adds a provision to the 31 day surveillance requirement, Surveillance Requirement 4.7.1.3.b, for the SSW cooling tower fans to clarify that only fans not already in operation are to be started from the control room. The existing requirement could be misinterpreted to mean that a fan that is already running must be shut down in order to start it from the control room. This proposed change is purely administrative as it represents a clarification of the intent of the surveillance requirement. (Page 3/4 7-4)

4. (TSPS 176), Requirements to Suspend Crane Operations, Technical Specification 3/4.8.1.2

   The proposed change is to clarify that crane operations over the upper containment pool as well as the spent fuel storage pool should be suspended whenever all offsite circuits are inoperable and/or with diesel generators 11 and 12 inoperable. Since the upper containment pool may be used for the temporary storage of spent fuel, this addition to the specification will further minimize the possibility of dropping a heavy load onto spent fuel when power is not available. The proposed change is a safety enhancement in that it constitutes an additional limitation not presently included in the technical specifications. (Page 3/4 8-9)

5. (TSPS 255), Control Rod Drive Bases, Technical Specification Bases B 3/4.1.3

   Several revisions to the subject Bases are proposed to provide supplemental information. The summary of Technical Specification 3/4.1.3 has been expanded to indicate that the technical specification limits the effects of the Rod Withdrawal Error event and that the safety analysis performed to support the required control rod insertion times includes a non-accident and a transient analysis. The description of the limitation on the number of inoperable control rods has been clarified by addition of the qualifier "but trippable". This change is made to reflect that more than eight control rods can be inoperable as long as they are trippable. The discussion of the appropriate response for an inoperable scram accumulator will be changed to reflect that in such an event the affected control rod may be inserted with normal drive pressure or scrammed using reactor pressure. The proposed changes are enhancements that represent clarifications of the design bases but do not change the intent of the specifications or adversely impact plant safety. (Page B 3/4 1-2)

6. (TSPS 312), Fuel Pool Gate Removal Requirements, Technical Specification 3/4.6.3.4

This proposed change to Surveillance Requirement 4.6.3.4.b requires verification that both refueling gates are in the stored position or otherwise removed from the upper containment pool in OPERATIONAL CONDITIONS 1, 2, and 3. This provides assurance that an adequate source of water exists for the suppression pool makeup system. This change is a safety enhancement that represents an additional requirement not presently included in the technical specifications.
(Page 3/4 6-26)

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:

- 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 35) Refueling Equipment Bases, Technical Specification Bases 3/4.9.6

   This change is proposed to revise Bases 3/4.9.6 to make it consistent with the refueling equipment used at Grand Gulf. The change reflects that only the main hoist of either the refueling platform or the fuel handling platform will be used to handle irradiated fuel assemblies. Additionally, the Bases are revised to indicate that all platform hoists have sufficient load capacity for handling fuel assemblies and/or control rods. This change is an addition to the proposed Technical Specification changes submitted in a letter from L. F. Dale to H. R. Denton, dated June 9, 1984 (AECM-83/0314, Item 17). The proposed change does not impact plant safety in that it affects the Bases only and does not affect any technical specification requirements. (Page B 3/4 9-1)

2. (TSPS 129), Standby Service Water (SSW) Systems, Technical Specification 3/4.7.1.1

   The proposed change to Technical Specification 3.7.1.1.b is to delete the reference to specific equipment and replace it with a more general reference to "associated plant equipment" and to insert phrases identifying the SSW subsystems required for each Operational Condition. Also, "Operational Condition" has been capitalized in ACTION statement 3.7.1.1.e to be consistent with the standard format of the Technical Specifications. As presently written, Specification 3.7.1.1.b requires two independent SSW subsystems to be OPERABLE under all Operational Conditions. The revision provides clarification by

requiring two SSW subsystems to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3. For OPERATIONAL CONDITIONS 4, 5, and *, the revision requires the SSW subsystems to be OPERABLE consistent with the requirements of Technical Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 or 3.9.11.2. The change to surveillance requirement 4.7.1.1 involves terminology corrections to reflect that only the standby service water subsystem(s) required OPERABLE by 3.7.1.1 are required to be demonstrated OPERABLE by the surveillance requirements. The proposed changes provide consistency with the as-built plant and, as such, are consistent with the plant safety analysis and will not adversely impact plant safety. (Page 3/4 7-1 and 3/4 7-2)

3. (TSPS 229), Main Steam Isolation Valve (MSIV) Leakage Control System Surveillance Requirements, Technical Specification 3/4.6.1.4

   Surveillance Requirement 4.6.1.4 requires that OPERABILITY of the heaters for each MSIV Leakage Control System (MSIV-LCS) subsystem be demonstrated by verifying that the heaters draw 7.8 - 9.5 amperes per phase. However, since only the inboard MSIV-LCS has heaters, a change from "heater" to "inboard heater" has been proposed. Also, the amperage range for demonstrating heater OPERABILITY has been added to the 31 day Surveillance Requirement and reworded to read "8.65 amperes ± 10%" instead of "7.8 to 9.5 amperes". These proposed changes represent a safety enhancement in that they make the technical specification consistent with the as-built plant as specified in FSAR Section 6.7.1 and add surveillance requirements not presently in the technical specifications. (Page 3/4 6-7)

4. (TSPS 258), Spent Fuel Storage Pool Drainage, Technical Specification 5.6.2

   This change is proposed to revise the Spent Fuel Pool elevation in Technical Specification 5.6.2 from 202'6" to 202'5¼". The 202'5¼" elevation corresponds to the elevation of the Residual Heat Removal System suction piping penetration. This is the lowest elevation to which the pool could inadvertently drain with the Spent Fuel Storage Pool gate installed. This change is considered an administrative change which will render the technical specification consistent with the as-built design. (Page 5-6)

5. (TSPS 313) Standby Liquid Control System Surveillance Requirement, Technical Specification 3/4.1.5

   Surveillance Requirement 4.1.5.d.4 currently refers to "the storage tank heaters." The proposed change is to make the reference singular, i.e., "the storage tank heater." The current Grand Gulf design contains only one Standby Liquid Control (SLC) tank heater used for maintaining the liquid temperature above its saturation temperature, as described in FSAR Sections 9.3.5.2 and 9.3.5.3. This change is purely administrative in nature and is proposed to reflect actual plant design. (Page 3/4 1-19)

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 195) MSIV Leakage Control System, Technical Specification 3/4.6.1.4

   This proposed change clarifies MSIV Leakage Control System Surveillance Requirement 4.6.1.4.c.2, and adds a note to clarify the valve lineup during the blower surveillance. The new pressure and flow values were determined during pre-operational and start-up testing using the specified valve lineup. These changes are considered enhancements that have no adverse safety impact because are consistent with the safety analysis and with the philosophy and intent of the technical specifications. (Page 3/4 6-7)

2. (TSPS 267), Horizontal Fuel Transfer System, Technical Specification 3/4.9.12

   A revision to Technical Specification 3/4.9.12 is proposed to add the room number and elevation to Limiting Condition for Operation (LCO) 3.9.12.a and to add a corresponding surveillance requirement to require periodic verification that the room through which the Transfer System penetrates is sealed during Transfer System operation. Existing Surveillance Requirements "a" and "b" are redesignated "b" and "c", respectively. Access to this area during fuel transfer operations could result in personnel exposure in excess of 10 CFR 20 requirements. The proposed revision will insure compliance with Technical Specification 3.9.12.a and is considered to be a safety enhancement in that it represents an additional control not presently in the technical specifications. (Page 3/4 9-18)

3. (TSPS 173), Standby Service Water (SSW) System, Technical Specification 3/4.7.1

This change is proposed to add ACTION statement 3.7.1.1.f to Service Water System Technical Specification 3/4.7.1. This change is requested to ensure that in all OPERATIONAL CONDITIONS, when an SSW subsystem is declared inoperable, the associated Diesel Generator will also be declared inoperable and the ACTIONS required by Specification 3.8.1.1 or 3.8.1.2 will be taken. This proposed change is an enhancement to safety in that it represents an additional restriction not presently contained in the technical specifications. (Page 3/4 7-1)

4. (TSPS 287), High Pressure Core Spray (HPCS) Service Water, Technical Specification 3/4.7.1.2

Technical Specification 3.7.1.2 ACTION statement requires that the HPCS System be declared inoperable when the HPCS Service Water system is inoperable. A technical specification change is proposed to require that the HPCS diesel generator as well as the HPCS system be declared inoperable when the HPCS service water system is declared inoperable, so that the ACTION statements in Specification 3.8.1.1 or 3.8.1.2 will be taken. This proposed change is an enhancement to safety in that it represents an additional restriction not presently contained in the technical specifications. (Page 3/4 7-3)

5. (TSPS 058), Spent Fuel Pool Temperature, Technical Specification 3/4.7.9

Technical Specification 3.7.9 requires that spent fuel pool temperature be maintained at less than or equal to 150°F. Presently, the surveillance requires the fuel pool cooling system inlet temperature to be determined. The proposed change to Surveillance Requirement 4.7.9.1 will allow other acceptable methods of determining the bulk pool temperature. This change does not adversely impact plant safety because it is consistent with the safety analysis and is a clarification of the intent of the surveillance requirement. (Page 3/4 7-45)

6. (TSPS 132), Control Room Temperature Limits, Technical Specification Table 3.7.8-1

The proposed changes revise the temperature limit for the control room from 77°F to 90°F and deletes the "Equipment Not Operating" column and the heading "Equipment Operating" heading from Table 3.7.8-1.

The 77°F control room temperature limit is increased to 90°F because the present limit of 77°F is derived from human factors considerations rather than equipment qualification data. NUREG-0700, "Guidelines for Control Room Design Reviews", requires that the control room HVAC system be capable of maintaining the dry bulb temperature between 73°F and 77°F. This is a system performance standard for maintaining the comfort zone for personnel occupancy. The proposed control room

temperature limit of 90°F is based upon a review of the control room equipment qualification data sheets. The lowest environmental qualification temperature of any equipment in the control room was found to be 90°F. Control room temperatures exceeding this limit for more than eight hours requires an evaluation of the impact on the qualified life of the affected equipment as required by the present technical specification ACTION statement. The use of the lowest qualification temperature for the control room is consistent with the limits established for other areas listed in this table. Since the existing control room technical specification limit of 77°F is based on a human factors performance standard and not an equipment performance standard, as is the intent of Table 3.7.8-1, a change to 90°F is fully justified.

The proposed change to delete the present area temperature limits when equipment is not operating is made because they may not be the limiting temperature for the affected areas. The change to a single temperature limit will also eliminate confusing and possibly conflicting requirements.

The proposed change to delete the table heading "Equipment Operating" is purely administrative in nature since these temperature limits will apply at all times.

These changes will not adversely impact plant safety because they serve only to clarify the intent of the specifications. (Page 3/4 7-44)

7. (TSPS 214), Control Rod Drive (CRD) Scram Accumulators, Surveillance Requirement 4.1.3.3.b.2

The proposed change deletes Surveillance Requirement 4.1.3.3.b.2. This Surveillance requires that measurements be taken once per 18 months for each individual control rod scram accumulator check valve of the time, for up to 10 minutes, that each check valve maintains its associated accumulator pressure above the alarm setpoint with no control rod drive pump operating. The purpose of the accumulator check valve is to prevent accumulator gross leakage from the accumulator back into the CRD system (rather than to the CRD drives) and provide back pressure during a reactor scram. These are hard material ball check valves and as such are not designed to prevent slow leakage from the accumulator to the CRD system which might occur following a CRD pump trip. In the event that the CRD accumulator check valves were unable to maintain sufficient pressure in the accumulator to stay above the low pressure alarm setpoint during a CRD pump trip, the reactor mode switch would be required to be placed in the SHUTDOWN position on the receipt of two or more CRD accumulator low pressure alarms. ACTION statement 3.1.3.3 requires that the reactor be scrammed on receipt of two or more control rod accumulator low pressure alarms and no control rod drive pumps operating in order to prevent a pattern of inoperable control rods (declared inoperable by Specification 3.1.3.3.a.1) which may result in less negative reactivity insertion on a scram than has been analyzed. Provisions to

declare the control rod(s) inoperable, insert the inoperable control rods, and disarm associated directional control valves are provided in Specification 3.1.3.3. ACTION 3.1.3.1.c allows up to eight control rods to be inoperable. This proposed change is recommended in a memo from R. C. Lewis to D. G. Eisenhut dated February 9, 1984. This proposed change is an enhancement that will not adversely impact plant safety because the surveillance requirement does not test the design function of the accumulator check valve. (Page 3/4 1-9)

8. (TSPS 826), Spent Fuel Pool Temperature, Technical Specification 3/4.7.9

The ACTION statement for exceeding the spent fuel storage pool (SFSP) temperature limit requires that the pool temperature be reduced to less than or equal to 150°F within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The proposed change to this ACTION statement is to replace the requirement for SHUTDOWN with a requirement to submit a Special Report whenever the 150°F limit has been exceeded for longer than 72 hours. This is acceptable, as the temperature of the SFSP has no effect on the safety of operation of the plant, and plant SHUTDOWN would not affect the temperature of the SFSP or aid in cooling it any faster. Surveillance Requirement 4.7.9.2 is also added to ensure fuel pool cooling pump OPERABILITY. These proposed changes constitute an enhancement to the technical specification without adversely affecting plant safety. (Page 3/4 7-45)

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:

- 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 094), High Pressure Core Spray (HPCS) Service Water System, Technical Specification 3/4.7.1.2

   This proposed change revises Surveillance Requirement 4.7.1.2 to read "The HPCS service water system shall be demonstrated OPERABLE:" with the remainder of the present requirement becoming Surveillance Requirement 4.7.1.2.a. This change also adds Surveillance Requirement 4.7.1.2.b, "At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a service water actuation test signal". "Each automatic valve servicing safety-related equipment" refers to the HPCS Service Water Pump discharge valve. These proposed changes are in response to an NRC proof and review comment and result in a more stringent Surveillance Requirement than is presently in the technical specifications. (Page 3/4 7-3)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than the alarm setpoint unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

   a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

   b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of 1520 + 30, -0 psig on decreasing pressure.

2. ~~Measuring and recording the time, for up to 10 minutes, that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point, starting at normal system operating pressure, with no control rod drive pump operating.~~ 214

REACTIVITY CONTROL SYSTEMS

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 ~~The~~ BOTH standby liquid control ~~system~~ SUBSYSTEMS shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one ~~pump and/or one explosive valve~~ SUBSYSTEM inoperable, restore the inoperable ~~pump and/or explosive valve~~ SUBSYSTEM to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With ~~the~~ both standby liquid control ~~system otherwise~~ SUBSYSTEMS inoperable, restore ~~the system~~ at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one ~~pump and/or one explosive valve~~ SUBSYSTEM inoperable, restore the inoperable ~~pump and/or explosive valve to~~ SUBSYSTEM TO OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With ~~the~~ both standby liquid control ~~system otherwise~~ SUBSYSTEMS inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 ~~The~~ EACH standby liquid control ~~system~~ SUBSYSTEM shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is greater than or equal to 4587 gallons.

3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Starting both pumps and recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charge.

3. Determining that the available weight of sodium pentaborate is greater than or equal to 5500 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

4. 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.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1220 psig is met.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control SUBSYSTEMS ~~system loops~~, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both ~~injection loops~~ SUBSYSTEMS shall be tested in 36 months. 156 156

2. Demonstrating that the pump relief valve setpoint is less than or equal to 1386 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

3. **Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water.

4. Demonstrating that the storage tank heater~~s~~ IS ~~are~~ OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heater~~s~~ IS ~~are~~ energized. 313

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSIV leakage control 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 and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. Blower OPERABILITY by starting the blowers from the control room and operating the blowers for at least 15 minutes.

2. ~~Heater~~ INBOARD HEATER OPERABILITY by demonstrating electrical continuity of the heating element circuitry, by VERIFYING THE INBOARD HEATER DRAWS 8.65 ± 10% AMPERES PER PHASE. 229

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor operated valve through at least one complete cycle of full travel.

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position, the blowers start and the INBOARD heater draws ~~7.8 to 9.5 amperes~~ 8.65 ± 10% AMPERES per phase. 229

2. Verifying that the blower developed at least the below required vacuum at the rated capacity.*

   a) Inboard ~~valves, 15" ± 1" H_2O at 100 scfm~~ SUBSYSTEM, 10" ± 1" H_2O VACUUM AT ≥ 100 SCFM.

   b) Outboard ~~valves, 50" ± 2" H_2O at 200 scfm~~ SUBSYSTEM, ≥ 15" H_2O VACUUM AT ≥ 200 SCFM. 195

d. By verifying the inboard flow, inboard and outboard pressure, and inboard temperature instrumentation to be OPERABLE 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.

* THE SUBSYSTEMS ARE NOT LINED-UP TO THE MAIN STEAMLINES DURING THE PERFORMANCE OF THIS TEST. 195

CONTAINMENT SYSTEMS

SUPPRESSION POOL MAKEUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.4 The suppression pool makeup system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool makeup line inoperable, restore the inoperable makeup line 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 the upper containment pool water level less than the limit, restore the water level to within the limit within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With upper containment pool water temperature greater than the limit, restore the upper containment pool water temperature to within the limit 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

4.6.3.4 The suppression pool makeup system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the upper containment pool water:

   1. Level to be greater than or equal to 23'3", and

   2. Temperature to be less than or equal to 125°F.

b. 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 secure in position, is in its correct position, and both refueling gates are in the stored position or are otherwise removed from the upper containment pool. 312

c. At least once per 18 months by performing 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 makeup of water to the suppression pool may be excluded from this test.

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 ~~Two~~ Each of the following independent standby service water (SSW) system subsystems, ~~shall be OPERABLE~~ with each subsystem comprised of: 129

a. One OPERABLE SSW pump, and
b. An OPERABLE flow path capable of taking suction from the associated SSW cooling tower basin and transferring the water through the RHR heat exchangers, ~~ECCS pump room seal coolers, and associated coolers and pump heat exchangers.~~ See attached (INSERT) 129

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:
1. With one SSW subsystem inoperable, restore the inoperable subsystem 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.
2. With both SSW subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the SSW subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 4 or 5 with the SSW subsystem, which is associated with an ECCS pump required OPERABLE by Specification 3.5.2, inoperable, declare the associated ECCS pump inoperable and take the ACTION required by Specification 3.5.2.

d. In OPERATIONAL CONDITION 5 with the SSW subsystem, which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

e. In ~~Operational Condition~~ OPERATIONAL CONDITION *, with the SSW subsystem, which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable. 129

f. In OPERATIONAL CONDITION 1,2,3,4,5, or * with a SSW subsystem inoperable, declare the associated Diesel Generator inoperable and take the action required per Specification 3.8.1.1 or 3.8.1.2. 173

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

** Whenever both SSW 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.

Insert to 3.7.1.1.b

and to associated plant equipment, as required, shall be OPERABLE as follows:

1. In OPERATIONAL CONDITIONS 1, 2, and 3: two subsystems; and

2. In OPERATIONAL CONDITIONS 4, 5, and *: the subsystems associated with the systems and components required to be OPERABLE by Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 or 3.9.11.2.

SURVEILLANCE REQUIREMENTS

AT LEAST THE ABOVE REQUIRED SUBSYSTEM(S)

4.7.1.1 ~~Each~~ standby service water system ~~subsystem~~ shall be demonstrated OPERABLE: 129

a. At least once per 31 days by:

1. Verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
2. Verifying that the valves isolating service to the spent fuel storage pool cooler are locked closed.

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a actuation test signal.

HIGH PRESSURE CORE SPRAY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The high pressure core spray (HPCS) service water system shall be OPERABLE with:

a. one OPERABLE HPCS service water pump, and

b. An OPERABLE flow path capable of taking suction from the associated SSW cooling tower basin and transferring the water through the HPCS service water system heat exchangers.

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

ACTION:

With the HPCS service water system inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1 or 3.5.2, as applicable~~.~~, and declare the associated diesel INOPERABLE and take the action required by Specification 3.8.1.1 or 3.8.1.2. 287

SURVEILLANCE REQUIREMENTS

4.7.1.2 The HPCS service water system shall be demonstrated OPERABLE:

a. At ~~at~~ least once per 31 days by verifying that each valve, manual, power operated or automatic, servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a service water actuation test signal. 094

*When the ~~HPSCS~~ HPCS system is required to be OPERABLE. 002

Amendment No. —

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 At least the following independent SSW cooling tower basins, each with:

a. A minimum basin water level at or above elevation 130'3" Mean Sea Level, USGS datum, equivalent to an indicated level of ≥ 87".

b. Two OPERABLE cooling tower fans,#

shall be OPERABLE:

a. In OPERATIONAL Condition 1, 2 and 3, two basins,

b. In OPERATIONAL Condition 4, 5 and *, the basins associated with systems and components required OPERABLE by Specifications 3.7.1.1 and 3.7.1.2.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, 3, 4, 5 and * with one SSW cooling tower basin inoperable, declare the associated SSW subsystem inoperable and, if applicable, declare the HPCS service water system inoperable, and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2, as applicable.

b. In OPERATIONAL CONDITION 1, 2, 3, 4 or 5 with both SSW cooling tower basins inoperable, declare the SSW system and the HPCS service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In Operational Condition * with both SSW cooling tower basins inoperable, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 At least the above required SSW cooling tower basins shall be determined OPERABLE at least once per:

a. 24 hours by verifying basin water level to be greater than or equal to 87".

b. 31 days by starting FROM THE CONTROL ROOM each SSW cooling tower fan NOT ALREADY IN OPERATION, ~~from the control room~~ and operating ~~the~~ EACH fan for at least 15 minutes.

c. 18 months by verifying that each SSW cooling tower fan starts automatically when the associated SSW subsystem is started.

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

#The basin cooling tower fans are not required to be OPERABLE for HPCS service water system OPERABILITY.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.6.1.3 The diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each cell in each battery is above the plates, and

2. The overall battery set voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity for each cell is appropriate for continued service of the battery. The specific gravity, corrected to 77°F and full electrolyte level, shall be ~~is~~ greater than or equal to 1.20. 002

c. At least once per 18 months by verifying that:

1. The battery case and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. Battery terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA		TEMPERATURE LIMIT (°F) ~~EQUIPMENT NOT OPERATING~~	~~EQUIPMENT OPERATING~~
a.	Containment		
	Inside Drywell	135	150
	CRD Cavity	135	185
	Outside Drywell	80	105
	Steam Tunnel	125	125
b.	Auxiliary Building		
	General	104	104
	ECCS Rooms	105	150
	ESF Electrical Rooms	104	104
	Steam Tunnel	125	125
c.	Control Building		
	ESF Switchgear and Battery Rooms	104	104
	Control Room	77	~~77~~ 90
d.	Diesel Generator Rooms	125	125
e.	SSW Pumphouse	104*	104*

132

*For this area, the limit shall be the greater of 104°F or outside ambient temperature plus 20°F, not to exceed 122°F for greater than one hour.

PLANT SYSTEMS

3/4.7.9 SPENT FUEL STORAGE POOL TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.7.9 The spent fuel storage pool temperature shall be maintained at less than or equal to 150°F.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel storage pool.

ACTION: With the spent fuel storage pool temperature greater than 150°F, ~~restore the pool temperature to less than or equal to 150°F within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~ 826

SURVEILLANCE REQUIREMENTS

4.7.9.1 The spent fuel storage pool temperature shall be verified to be less than or equal to 150°F ~~by determining the pool cooling system inlet temperature~~ at least once per 12 hours. 826 058

for longer than 72 hours, prepare and submit a Special Report pursuant to Specification 6.9.2 within the next 14 days outlining the cause of the high temperature condition and the plans for restoring the spent fuel storage pool temperature to normal for existing plant conditions.

4.7.9.2 Start each fuel pool cooling and cleanup pump not already running at least once per 92 days and run for at least 15 minutes. 826

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:

AND THE UPPER CONTAINMENT POOL

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 above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical. 176

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.

3/4.9.12 HORIZONTAL FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The horizontal fuel transfer system (HFTS) may be in operation provided that:

a. Room 1A525, Auxiliary Building, elevation 182'0", the room ~~The room~~ through which the transfer system penetrates, is sealed. 267

b. All interlocks with the refueling and fuel handling platforms are OPERABLE.

c. All HFTS primary carriage position indicators are OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 4* and 5*.

ACTION:

With the requirements of the above specification not satisfied, suspend HFTS operation with the HFTS at either the Spent Fuel Building pool or the Reactor Containment Building pool terminal point.

SURVEILLANCE REQUIREMENTS

4.9.12 Within 24 hours prior to the operation of HFTS and at least once per 7 days thereafter, verify that:

b ~~a~~. All interlocks with the refueling and fuel handling platforms are OPERABLE. 267

c ~~b~~. All HFTS primary carriage position indicators are OPERABLE.

*When the reactor mode switch is in the Refuel position.

a. Room 1A525, Auxiliary Building, elevation 182'0", the room through which the transfer system penetrates, is sealed. 267

3/4.1.3 CONTROL RODS

nonaccident, and transient

The specification of this section ensuresthat (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident, analyses, and (3) ~~limit~~ the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis. 255 255

and rod withdrawal error event are limited.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

but trippable

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem. 255

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.06 during the limiting power transient analyzed in Section 15.4 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than 1.06. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

slowly scrammed via reactor pressure or

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor. 255

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING ~~PLATFORM~~ Equipment

The OPERABILITY requirements ensure that (1) only the main hoist of the refueling platform or the main hoist of the fuel handling platform will be used for handling ~~control rods and~~ irradiated fuel assemblies ~~within the reactor pressure vessel~~, (2) ~~each crane and~~ Platform hoists ~~has~~ have sufficient load capacity for handling fuel assemblies and for control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. 35

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

platform hoists have

only on the main hoist of

EQUIPMENT

or the main hoist of the fuel handling platform

3/4.9.6 REFUELING ~~PLATFORM~~

irradiated

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling ~~control rods and~~ fuel assemblies ~~within the reactor pressure vessel~~, (2) ~~each crane and hoist has~~ sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. 035

/or

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.2-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 4.3 of the FSAR.

b. A nominal 12 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~202'6"~~ 202' 5 1/4". 258

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1270 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

ATTACHMENT 2

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: INSTRUMENTATION & CONTROL SYSTEMS

MTS50mm1

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

TSPS No.	Item Nos.*	TSPS No.	Item Nos.*
003	2.A.06	238	2.A.02
004	2.B.01	253	2.C.06
009	2.C.01	257	2.C.07
010	2.B.02	278	2.A.10
011	2.B.03	280	2.A.11
013	2.A.04	286	2.A.12
022	2.B.04	298	2.C.08
023	2.B.05	303	2.B.07
040	2.C.02	308	2.B.13
047	2.A.07	315	2.B.08
050	2.B.02	323	2.C.09
074	2.A.03, 2.A.01	334	2.A.13
077	2.C.03	345	2.B.09
079	2.D.01	346	2.B.10
083	2.D.02	350	2.B.11
158	2.A.08	355	2.A.14
196	2.A.09	356	2.A.15
197	2.C.04	360	2.B.12
201	2.B.06	363	2.C.10
206	2.A.05	364	2.B.07
237	2.C.05	367	2.C.11
		369	2.C.12

*Item number format: 1.A.02

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

MTS50mm2

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.	074	B 2-2 B 3/4 4-6 3/4 3-14 3/4 3-23a
2.	238	3/4 3-19

EDITORIAL CHANGES

Proposed editorial changes to the technical specifications are discussed below:

3. (TSPS 074), Footnotes and Editorial Changes, Technical Specifications Table 3.3.3-2, Table 4.8.2.1-1, Technical Specifications 6.2.2.c and 6.2.2.e

- Presently an unreferenced footnote appears at the bottom of Table 3.3.3-2. The origin of this footnote is the GE BWR/6 STS. The type of relays to which it applies were not installed at Grand Gulf. In the present technical specification, the information in Table 4.8.2.1-1 tends to run together which makes it hard to read. The proposed change is to add lines to divide the information.

The footnote to 6.2.2.c and 6.2.2.e defines certain minimum staffing requirements for the unit organization. The present wording of the footnote leaves the subject (i.e. "the number of personnel") unclear. A change is proposed to correct the ambiguity of the present technical specifications. The changes to delete the footnote on Table 3.3.3-2, to add lines to divide the information on Table 4.8.2.1-1 into sections, and to modify the footnote on Specification 6.2.2.e are purely administrative in nature. (Pages 3/4 3-29, 3/4 8-13, and 6-1)

4. (TSPS 013), Group 5 Isolation Logic, Technical Specification Table 3.3.2-1

This proposed change adds a footnote (o) to items 1.c and 1.e of Table 3.3.2-1, Isolation Actuation Instrumentation. This footnote explains that four Group 7 valves are also isolated by the Group 5 isolation signals (i.e., Reactor Vessel Water Level-Low Low Low, Level 1 and Drywell Pressure-High). This footnote is being added to supply additional information concerning Grand Gulf design and does not affect technical specification requirements. The proposed change does not affect plant safety since the proposed change does not alter any technical specification requirements or surveillance requirements. (Pages 3/4 3-10 and 3/4 3-14a)

5. (TSPS 206), LPCI/LPCS Injection Valve Interlocks, Technical Specification Table 4.3.3.1-1

This proposed change deletes footnotes (c) and (d), Note 1, and the reference to footnotes (c) and (d) in Items A.1.d and B.1.d from Table 4.3.3.1-1, Emergency Core Cooling System Actuation Instrumentation Surveillance Requirements. This change is proposed to reflect that the LPCI/LPCS injection valve interlocks are not presently installed in the plant. This proposed change will not adversely affect plant safety since the proposed change is consistant with the as-built plant design and the safety analysis. These interlocks are required to be installed and OPERABLE prior to restart after the first refueling outage by License Condition 2.c(2.1). Technical specification changes to reflect this design change will be submitted prior to implementation of the design change. This proposed change is considered administrative in nature. (Pages 3/4 3-31 and 3/4 3-33)

CLARIFICATIONS

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

6. (TSPS 003), Isolation Actuation Signal, Technical Specifications 3/4.6.6.3 and 3/4.7.2

Surveillance Requirement 4.6.6.3.d.3.c & d require the Standby Gas Treatment System to be tested for receipt of an actuation signal from simulated high radiation. The actual actuation signal is

high-high, as the high signal only activates an alarm. The same criterion also applies for the actuation signals required for Control Room Emergency Filtration System isolation in Surveillance Requirement 4.7.2.d.2.a (high-high radiation) and 4.7.2.d.2.d (low-low reactor water level). These proposed changes are purely administrative in nature and are proposed to clarify the criteria in the surveillance requirements. (Pages 3/4 6-54 and 3/4 7-6)

7. (TSPS 047), EOC-RPT Response Time Testing, Technical Specification 3/4.3.4.2

The proposed change to Surveillance Requirement 4.3.4.2.3 will require response time testing of the turbine control valve function from one trip system as well as the turbine stop valve function from the redundant trip system during one 18 month period. In the subsequent 18 month period, the functions from the opposite trip system will be tested, such that all channels are tested within the required 36 month period.

The current specification is ambiguous. This change is needed in order to clarify the intent of the specification. The surveillance schedule in the proposed change assures:

a. All channels are tested within a 36 month period.

b. Trip functions from both trip systems (turbine stop and control valves) are tested every 18 months.

c. Redundant trip functions (i.e., turbine stop valve closure as well as turbine control valve closure) are tested every 18 months.

The proposed change does not adversely affect plant safety since the change represents an enhancement which will clarify the intent of the surveillance requirements. (Page 3/4 3-39)

8. (TSPS 158), APRM Setpoint Calculation, Technical Specification 3/4.2.2

This proposed change revises Technical Specification 3.2.2 to indicate that the factor T, which is the lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by MAXIMUM FRACTION OF LIMITING POWER DENSITY, is applied only if it is less than or equal to 1.0. The current specification indicates that T is always less than or equal to 1.0; however, the calculated value of T will routinely be greater than one. In order to maintain the conservatism in the flow biased scram setting, the actual value of T used in the calculation is never greater than 1.0. This change is a clarification of the existing specification to eliminate possible discrepancies between the calculated value of T and the value of T applied to the flow biased scram trip setting. The proposed change does not affect plant safety since the limits identified in the safety analyses are not exceeded. (Page 3/4 2-5)

9. (TSPS 196), Meteorological System Instrument OPERABILITY, Technical Specification Table 3.3.7.3-1

The proposed change replaces the term "INSTRUMENTS" with "CHANNELS" in order to clarify terminology and clearly indicate that the requirements apply to an entire channel and not to individual instruments. The proposed change is purely administrative in nature in that it clarifies the technical specifications. (Page 3/4 3-64)

10. (TSPS 278), LPCI and LPCS Manual Initiation Actuation Footnote, Technical Specification Table 3.3.3-1

This proposed change revises Table 3.3.3-1 to add footnote b to the Manual Initiation entries for both Division I and II of ECCS actuation instrumentation (A.1.d and B.1.d) to indicate that manual initiation will also start the associated diesel generator. This change is considered to be a administrative and is consistent with the as-built plant design. (Page 3/4 3-25)

11. (TSPS 280), Friction Testing in OPERATIONAL CONDITION 5*, Technical Specification 3/4.9.3

The proposed change to the Control Rod Position requirements applied during refueling operations moves the one-rod-out allowance from the ACTION statement to footnote "*". This will prevent unnecessary entry into the ACTION statement during single control rod withdrawals for friction testing, subcriticality checks, and instrumentation response checks. The proposed change does not affect the requirements of the specification and represents a clarification of the one-rod-out provision of the technical specification. The proposed change does not adversely impact plant safety since the change is consistent with existing safety analysis. (Page 3/4 9-5)

12. (TSPS 286) Reactor Core Isolation Cooling (RCIC) System Action Statement Clarification, Technical Specification 3/4.7.3

This proposed change clarifies the ACTION statement of Technical Specification 3/4.7.3 to indicate those actions that should be taken in the event that RCIC and High Pressure Core Spray (HPCS) are inoperable simultaneously. As currently worded, the specification could be interpreted to imply that Specification 3.0.3 should be invoked in the event that both HPCS and RCIC are inoperable simultaneously. If invoked, Specification 3.0.3 would require the plant to be in STARTUP within 6 hours and HOT SHUTDOWN within the following 6 hours. The intent of this proposed change is to more clearly indicate that the requirements of the ACTION statement of Technical Specification 3.7.3 (Hot Shutdown within 12 hours), rather than the requirements of Specification 3.0.3, should be followed if HPCS and RCIC are simultaneously inoperable. This change is considered to be purely administrative in nature and will not adversely affect plant safety since it is proposed for the purpose of clarification. (Page 3/4 7-7)

13. (TSPS 334) Rod Pattern Control System, Technical Specification 3/4.1.4.2, Table 3.3.6-1

The proposed changes to the subject technical specifications for the Rod Pattern Control System (RPCS) are clarifications requested to achieve consistency with the system design. The proposed changes are:

a. A revision to Technical Specification 3/4.1.4.2 ACTION statement a to refer to the Low Power Setpoint rather than 20% of RATED THERMAL POWER. This proposed change is more conservative than the existing specifications since more stringent restrictions are placed on control rod movement. The Low Power Setpoint is required by Technical Specification 3.3.6 to be 20% +15 -0% of RATED THERMAL POWER. Increasing the applicable power level for ACTION statement a to as high as 35% of RATED THERMAL POWER does not affect the rod withdrawal accident (RWA) analysis since the limiting RWA occurs between 70% and 100% of RATED THERMAL POWER. (Page 3/4 1-16)

b. A revision to Technical Specification 3/4.1.4.2 ACTION statement b to refer to the rod action control system (RACS) rather than the rod gang drive system (RGDS). As stated in GEK-73677A, February 1983, the RACS is a subsystem of the rod control and information system (RC&IS). The RACS contains both the rod position information system (RPIS) and the rod pattern control system (RPCS) as subsystems. When a control rod has its position bypassed, a bit in the position word from the RPIS is set. This bit, when received by the RPCS, tells the RPCS to allow movement of the bypassed control rod (reference Design and Performance Specification for the Rod Pattern Controller, 22A4540). The RACS, therefore, is the appropriate terminology to be used in the specification. The RGDS, another subsystem of the RC&IS, receives input from the RACS in the form of a drive command which is relayed to the hydraulic control unit(s). Because the signal to bypass the control rod is input in the RACS upstream of the RGDS, the proposed revision is appropriate. (Page 3/4 1-16)

c. A revision to Technical Specification 3/4.1.4.2 ACTION statement 2.a to change "control rod" to "control rod(s)" since more than a single control rod may be bypassed and the ACTION to be taken is the same for each rod being bypassed. (Page 3/4 1-16)

d. A minor revision to Technical Specification 4.1.4.2 is proposed for clarification. Item a.1 has "control rod" changed to "control rod or gang" to indicate that verification is to be performed for either individual or gang withdrawal modes. (Page 3/4 1-17)

e. Minor revisions to Technical Specification 4.1.4.2 items (a) and (b) are proposed to indicate that two separate functions of the Rod Pattern Control System are being addressed. (Page 3/4 1-17)

f. A revision to Tables 3.3.6-1, 3.3.6-2, and 4.3.6-1 to refer to the High Power Setpoint rather than the Intermediate Rod Withdrawal Limiter Setpoint. The High Power Setpoint determines when the rod withdrawal limiter function is enforcing. Grand Gulf design does not include an Intermediate Rod Withdrawal Limiter between the Low and High Power Setpoints. Proper terminology for the Intermediate Rod Withdrawal Limiter Setpoint is the High Power Setpoint. (Pages 3/4 3-50, 3/4 3-52, 3/4 3-53)

As described above, these proposed changes are clarifications or additional restrictions which will provide consistency with the RPCS design. The proposed changes do not adversely effect plant safety since the proposed changes represent nomenclature revisions for clarification purposes to be consistent with the plant design and terminology.

14. (TSPS 355), Surveillance Frequency Nomenclature, Technical Specification Table 4.3.1.1-1

The proposed change deletes redundant and/or inappropriate requirements to perform CHANNEL FUNCTIONAL TESTS prior to startup from Table 4.3.1.1-1. Footnote C, which requires testing 7 days prior to startup, is redundant and all references to it have been deleted from the table. Also, for Items 2.b and 2.c, all requirements for CHANNEL FUNCTIONAL TESTS prior to startup have also been deleted because, as indicated in Table 3.3.1-1, these particular functional units are only required to be OPERABLE in OPERATIONAL CONDITION 1. This change does not adversely impact plant safety because it deletes only redundant and superfluous requirements and does not alter the intent of the subject technical specification. (Pages 3/4 3-7 and 3/4 3-8)

15. (TSPS 356), Rod Block Frequency Nomenclature, Technical Specification Table 4.3.6-1

The proposed changes to Table 4.3.6-1 will:

a. Delete Note e and delete the reference to Note e from Item 1.a, since the BWR/6 design does not use a reactor manual control system but instead uses a Rod Pattern Control System.

b. Delete the reference to Note b from Items 2, 3, 4, and 6 to remove the redundant requirement for surveillance testing. Note b requires the surveillances to be performed within 7 days prior to startup; however, the items for which Note b is being deleted already require weekly surveillance testing. In

addition, the reference to a surveillance being required within 24 hours prior to startup has been revised to clarify the intent of the specification which is to ensure surveillance within 7 days prior to startup.

c. Delete the startup surveillance requirement for Items 2.a, 2.c, and 6.a since these functions are only applicable in to OPERATIONAL CONDITION 1.

d. Reword Note c to clarify that the intent of the specification is to perform the required surveillance within 24 hours prior to control rod movement and as each power range above the RPCS low power setpoint is entered for the first time during any 24 hour period during power increase or decrease.

The proposed changes remove redundant and superfluous requirements, thus provide clarity and improved understanding. These proposed changes do not adversely affect plant safety since they do not alter the meaning or intent of the technical specifications. (Page 3/4 3-53 and 3/4 3-54)

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:

- 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 004), Suppression Pool Makeup System, Technical Specification 3/4.6.3.4 and Bases 3/4.6.3

   A change is proposed to revised Surveillance Requirement 4.6.3.4.a.1 to clearly indicate that the reference point for the normal water level of the upper containment pool is the dryer/separator storage area of the pool floor. In addition, Bases 3/4.6.3 has been revised to clarify both the hazard associated with and the safeguards used to prevent the inadvertent opening of the suppression pool makeup dump valves. The proposed change does not adversely affect plant safety and is considered an enhancement in that it will provide additional information that is consistent with the as-built design. (Pages 3/4 6-26 and B 3/4 6-5)

2. (TSPS 010, 050), Traversing In-Core Probe (TIP) System OPERABILITY, Technical Specification 3/4.3.7.7 and Bases 3/4.3.7.7

   A change is proposed to revise Technical Specification 3/4.3.7.7 to indicate that there are five, rather than three, movable detectors in the TIP system. The surveillance requirement has also been revised to clarify that normalizing is only required for LPRM calibration and is not required for individual detectors used to monitor core thermal limits such as APLHGR or MFLPD. The applicable bases has also been expanded to indicate that the normalizing process (for LPRM calibration) and the comparison of individual detector data with normalized data (for monitoring) are the basic methods of demonstrating TIP System OPERABILITY. The proposed changes do not adversely affect plant safety since the changes only render the technical specification consistent with the as-built plant and system usage. (Pages 3/4 3-74 and B3/4 3-5)

3. (TSPS 011), Control Rod Block Instrumentation, Technical Specification Table 3.3.6-1

The proposed change deletes the reference to footnote (d) in Trip Function 4.a, which indicates that the IRM "Detector not full in" rod block function is bypassed when the IRM channels are on Range 1. The design of the Grand Gulf neutron monitoring system does not include this bypass capability. As discussed in FSAR Section 7.7.1.2.3.2.3.2, the only automatic bypass on range 1 involving the IRM's is the downscale function. This change does not adversely affect plant safety since the revised technical specification will be consistent with the plant as-built design and the FSAR. (Page 3/4 3-50)

4. (TSPS 022), Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip, Technical Specification 3/4.3.4.1

A revision to the ACTION statements of the subject technical specification is proposed to achieve consistency with the as-built ATWS recirculation pump trip (RPT) system logic design. The GGNS ATWS RPT logic consists of two trip systems with each system having a single recirculation pump trip function. The logic within each of the two trip systems includes two pairs of trip units (reactor water level and reactor vessel pressure) arranged such that a signal from any unit will trip the trip system and corresponding recirculation pump. The proposed change to the ACTION statements will provide a 14 day restoration period when the number of OPERABLE channels is not satisfied for one or both trip systems. This provision incorporates single failure criteria and redundant system design features to satisfy the design intent of the ATWS RPT system (i.e., to trip both recirculation pumps upon high reactor vessel pressure or reactor vessel water level-low low, level 2). The proposed revision is necessary to ensure that the intent of the technical specifications is satified without subjecting the plant to unnecessary transients caused by recirculation pump trips. In the Grand Gulf trip logic design, any channel placed in the tripped condition will result in a recirculation pump trip. The proposed change will allow the recirculation pump(s) to continue to operate while the inoperable channel(s) is repaired or the plant is taken to STARTUP. The proposed change promotes consistency among the as-built plant design, operation and the technical specifications. The proposed change enhances plant safety since it reduces the number of unnecessary recirculation pump trips which would be imposed on the plant by existing technical specification. Revised definitions for the terms "Channels", "Trip Systems", and "Trip Functions" for the ATWS recirculation pump trip system instrumentation are included on the last page of Category B of this attachment to reflect the proposed changes described above. This information supercedes the definitions provided in a letter from Mr. L. F. Dale to Mr. H. R. Denton on May 8, 1984 (AECM-84/0093). (Page 3/4 3-34)

5. (TSPS 023), Relief Valve and Low-Low Set Function, Technical Specifications 3.4.2.1 and 3.4.2.2

Revisions to the subject technical specification are proposed to provide additional ACTION statements to reflect the pressure actuation trip system redundancy in the GGNS design. The proposed changes are necessary because the present specification could be interpreted to require plant shutdown within 12 hours when more than one relief valve has an inoperable low-low set pressure relief function. This revision is justified because redundant trip logic is provided for the low-low set pressure relief function. These proposed changes do not adversely affect plant safety and represent operational enhancements which are within the safety analyses. (Pages 3/4 4-5 and 3/4 4-6)

6. (TSPS 201), Secondary Containment Isolation - Manual Initiation, Technical Specification Table 3.3.2-1.

The proposed change is to delete the reference to footnote (f) from Table 3.3.2-1 Item 3.e because it is not consistent with the as-built plant. Footnote (f) incorrectly states that manual initiation of secondary containment isolation automatically trips the main condenser mechanical vacuum pumps, thus isolating this effluent path. The mechanical vacuum pumps are only tripped automatically on a main steam line high radiation signal, but can be manually tripped if excessive radioactivity is detected at the turbine building vent. This change (deletion) does not adversely affect plant safety and is consistent with the isolation actuation instrumentation logic design and the safety analysis. (Page 3/4 3-11)

7. (TSPS 303, 364) High Pressure Core Spray (HPCS) Initiation Instrumentation, Technical Specification Table 3.3.3-1

This proposed change revises Table 3.3.3-1 and associated ACTION statement 33 to reflect the as-built HPCS initiation instrumentation design. Currently ACTION 33 implies a design of more than one trip system, while actual as-built design of the instrumentation consists of a single trip system. This change will make ACTION 33 consistent with the single trip system design. This change also deletes the word "system" from Table 3.3.3-1 Item C.1.f, Manual Initiation, since there is only one trip system and reference to "system" is already implied. This change does not adversely affect plant safety in that it does not affect the actual logic or number of required OPERABLE channels, but only revises the terminology of the table and its associated ACTION statement to reflect as-built plant and trip system logic design. (Pages 3/4 3-26 and 3/4 3-27)

8. (TSPS 315) RCIC/RHR Hi Flow Isolation Instrumentation, Technical Specification Table 3.3.2-2

This proposed change reduces the Allowable Value for the RCIC/RHR isolation high flow function, item 5.k of Table 3.3.2-2 from less

than or equal to 160" H_2O to less than or equal to 151" H_2O based upon a revision to GE design documentation. This change is based on refined calculations that were performed as a result of vendor verification of design documents. This change is consistent with the as-built plant and represents a more stringent requirement than is presently included in the technical specification. (Page 3/4 3-17)

9. (TSPS 345) Plant Systems Actuation Instrumentation Setpoints, Technical Specification Table 3.3.8-2

The proposed change reduces the Allowable Value for the Reactor Vessel Water Level-High, Level 8, for the Feedwater System/Main Turbine Trip System from less than or equal to 55.7 inches to less than or equal to 54.1 inches. This change is consistent with vendor design documents. The present Allowable Value was determined assuming wide range instrumentation as the reference base whereas narrow range instrumentation is the actual instrumentation providing the trip signal. The reduction in the Allowable Value is proposed to reflect the accuracy associated with the narrow range instrumentation. The proposed change is an enhancement to plant safety in that the revised Allowable Value is more conservative than the value presently included in the technical specifications and is consistent with the as-built plant design. (Page 3/4 3-99)

10. (TSPS 346) Chlorine Detection System, Technical Specifications 3.3.7.8 and 4.3.7.8

The proposed change replaces the term "system" with the more appropriate term "channel" in five places. The revised terminology is consistent with the definition of "channel" in the chlorine detection system. At Grand Gulf, there are two channels of chlorine detection, one of which actuates control room isolation logic A, and one of which actuates control room isolation logic B. Either control room isolation logic actuation will provide control room isolation and start the respective Control Room Emergency Filtration System. This change is considered a clarification of the technical specification in that it more clearly represents the isolation logic utilized in the as-built plant and therefore enhances plant safety. (Page 3/4 3-75)

11. (TSPS 350) Isolation Actuation Instrumentation Valve Closure Verification, Technical Specification Table 3.3.2-1

The proposed change revises ACTION Statement 28 in Table 3.3.2-1 to allow alternative methods of ensuring penetration isolation. The present ACTION requires that affected system isolation valves be locked closed within one hour after determining that the requried RHR system isolation actuation instrument is inoperable. However, this requirement is not practical in the case of the RHR shutdown cooling inboard isolation valve (E12-F009), as this valve is located in the drywell and can not be locked closed with the reactor at power because of radiological considerations. The revised ACTION

statement will allow, as an alternative, the use of remote indication to verify that the valve is closed. When the valve is verified closed by remote indication, it must then be electrically disarmed. The proposed change has no adverse impact on plant safety since the change makes the technical specification more consistent with plant design and meets the intent of the technical specifications by ensuring penetration isolation. (Page 3/4 3-14)

12. (TSPS 360) Reactor Core Isolation Cooling (RCIC) Trip Systems, Technical Specification Table 3.3.5-1

The proposed changes revise Table 3.3.5-1 and associated ACTION statements to more accurately reflect RCIC actuation instrumentation design. Currently, the wording in the subject table and ACTION statement imply a design of more than one trip system. Logic system definitions in use at Grand Gulf are based upon a single-trip system design for this instrumentation. The proposed changes revise the terminology of the table and its associated ACTION statements to reflect the single trip system design. These changes represent clarification of terminology only and do not affect the content or requirements of the specification. The changes therefore have no adverse impact on plant safety since the changes more accurately reflect plant design. (Pages 3/4 3-45 and 3/4 3-46)

13. (TSPS 308), Setpoints for Equipment Area Temperature-High, Technical Specification Table 3.3.2-2

The proposed change to table 3.3.2-2 is to delete items 4.c.4, 4.c.5, and 4.c.6 (Equipment Area Temperature-High) and 4.d.4, 4.d.5, and 4.d.6 (Equipment Area delta Temp-High), Trip Setpoints and Allowable Values for the Reactor Water Clean Up (RWCU) Demineralizer Rooms, Receiving Tank Room and Demineralizer Valve Room. All these rooms are downstream of the RWCU heat exchangers in normal modes of RWCU operation, so temperature of the coolant in these pipes is less than or equal to 120°F. Analysis reveals that a leak in these rooms would cause the temperature of the rooms to decrease because the cooling effect of evaporation would exceed the release of sensible heat caused by a pipe break. Furthermore, GE analysis, as found in GE document 22A3735AA, Section 3.1.6.8, states that temperature monitoring for leakage in these rooms would not be responsive because of the relatively low temperature of the reactor coolant in these rooms, and for this reason the RWCU delta Flow-High instrumentation was installed. This specification represents an unnecessary Surveillance Requirement that provides no meaningful function. This change is submitted to make the technical specifications consistent with system design. (Page 3/4 3-16)

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION TABLE 3.3.4.1-1

Trip Unit	Parmeter	Logic	
B21-LIS-N699A	RPV Level - Lvl 2	Any One	Trip Pump C001A
B21-PIS-N658A	*RPV Press - HI		
B21-LIS-N699E	RPV Level - Lvl 2		
B21-PIS-N658E	RPV Press - HI		
		TRIP SYSTEM	TRIP FUNCTION

Trip Unit	Parmeter	Logic	
B21-LIS-N699B	RPV Level - Lvl 2	Any One	Trip Pump C001B
B21-PIS-N658B	RPV Press - HI		
B21-LIS-N699F	RPV Level - Lvl 2		
B21-PIS-N658F	RPV Press - HI		
		TRIP SYSTEM	TRIP FUNCTION

* One Channel (Typical of 8 shown on this page)

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 009), Source Range Monitor (SRM) OPERABILITY Requirements, Technical Specification Table 3.3.6-1, Technical Specification 3/4.3.7.6, and Bases 3/4.3.6 and 3/4.3.7.6

   These proposed changes resolve inconsistencies among the technical specifications and bases regarding the number of required SRM's for various OPERATIONAL CONDITIONS.

   a) Technical Specification 3.9.2 requires that two SRM's be OPERABLE during refueling (OPERATIONAL CONDITION 5). One of the operable SRM's must be located in the quadrant where the core is being altered and the other SRM must be located in an adjacent quadrant. Two SRM's OPERABLE with one in the quadrant where the core is being altered and one in an adjacent quadrant during refueling ensures that redundant monitoring capability will be available to detect changes in the reactivity condition of the core. The "Minimum Operable Channels per Trip System" for items 3a, b, c and d has been changed from four to two for OPERATIONAL CONDITION 5, and a footnote (**) has been added to Table 3.3.6-1 to ensure that the requirements of Technical Specification 3.9.2 are met.

   b) Technical Specification 3.3.7.6 has been revised to require four instead of three OPERABLE SRM's in OPERATIONAL CONDITION 2, to be consistent with the requirements of Technical Specification 3.3.6. Furthermore, the requirements for OPERATIONAL CONDITIONS 3 and 4 in Technical Specification 3.3.7.6 have been changed from three OPERABLE SRM's (with ACTION b required

when two or more of the three are inoperable) to two OPERABLE SRM's (with ACTION b required when one or more of the two are inoperable). During OPERATIONAL CONDITION 3 and 4 the reactor core is homogenous; therefore, one SRM channel can monitor the core without channel redundancy. Requiring two SRM's OPERABLE is conservative since only one channel is required to provide adequate core monitoring.

c) The Bases, B 3/4.3.6 and B 3/4.3.7.6, have been revised to reflect the the changes to the above Specifications. The OPERABILITY of the control rod block instrumentation (including SRM's) in OPERATIONAL CONDITION 5 provides diversity of rod block protection to the one-rod-out interlock. The SRM's are required OPERABLE in OPERATIONAL CONDITION 2 to provide for rod block capability, and are required OPERABLE in OPERATIONAL CONDITION 3 and 4 to provide monitoring capability which provides diversity of protection to the reactor mode switch interlock.

These changes are proposed to achieve consistency throughout the technical specifications and will ensure adequate core monitoring capabilities. The proposed changes are administrative in nature. (Pages 3/4 3-50, 3/4 3-51, 3/4 3-73, B 3/4 3-3, and B 3/4 3-5)

2. (TSPS 040), Isolation System Instrumentation Response Time, Technical Specification Table 3.3.2-3

The proposed technical specification change decreases the response times for certain isolation system instrumentation listed in Table 3.3.2-3 from less than or equal to 13 seconds to less than or equal to 10 seconds. A response time of less than or equal to 3 seconds is also added with footnote "***" to the present response requirement for the fuel handling area ventilation and pool sweep exhaust radiation instrumentation, Items 3.c and 3.d. The change to less than or equal to 10 seconds is made to reflect the design time required for the diesel generators to restore A.C. power on a loss of all offsite power as specified in diesel generator OPERABILITY Surveillance Requirement 4.8.1.1.2. The response time requirement of less than or equal to 3 seconds is added to Items 3.c and 3.d because the accident analysis for a fuel handling accident (FSAR sections 15.7.4.5 and 15.7.6.2) assumes that the isolation system instrumentation response time for air operated dampers is 3 seconds with no diesel generator delay. These changes are enhancements which achieve consistency within the Technical Specifications, provide more stringent Surveillance Requirements, and provide consistency with the accident analysis. The changes represent improvements in plant safety. (Pages 3/4 3-18 and 3/4 3-19)

3. (TSPS 077), Remote Shutdown System, Technical Specification 3/4.3.7.4, Table 3.3.7.4-1, and Bases 3/4.3.7.4

This proposed change adds remote shutdown system control switches and control circuits to the subject technical specification and table. Limiting Conditions For Operation, ACTION Statements, and Surveillance Requirements for these controls are comparable to the existing requirements for the remote shutdown system instrumentation. This proposed change would also revise Table 3.3.7.4-1 to add the necessary control requirements and to clearly indicate the remote shutdown system divisional requirements. The Bases for the remote shutdown system is revised to expand its scope to include control switches and control circuits. These proposed changes enhance plant safety since they represent additional requirements which broaden the scope of the GGNS Technical Specifications. These proposed changes also provide a more accurate reflection of system design. (Pages 3/4 3-66, 3/4 3-67, 3/4 3-67a, and B 3/4 3-4)

4. (TSPS 197), Reactor Protection System Instrumentation, Technical Specification Table 3.3.1-1; Control Rod Block Instrumentation, Technical Specification Tables 3.3.6-1, 3.3.6-2 and 4.3.6-1

The proposed change to the reactor protection system instrumentation Table 3.3.1-1 increases the number of minimum OPERABLE channels per trip system for the reactor mode switch shutdown from one channel per trip system to two channels per trip system. (Page 3/4 3-3)

The proposed change also adds the reactor mode switch shutdown position to the control rod block instrumentation tables as follows:

a. Control rod block instrumentation:

   1) Adds the reactor mode switch shutdown position to the Table 3.3.6-1 as Item 7 and provides the minimum OPERABLE channels per trip function, applicable OPERATIONAL CONDITIONS and ACTION statement. (Page 3/4 3-50)

   2) Adds new ACTION statement 63 for the added reactor mode switch shutdown position. (Page 3/4 3-51)

b. Control rod block instrumentation setpoints: Adds the reactor mode switch shutdown position to Table 3.3.6-2 as Item 7 and states that the Trip Setpoint and Allowable Value are not applicable to this function. (Page 3/4 3-52)

c. Control rod block instrumentation surveillance requirements: Adds the reactor mode switch shutdown position to Table 4.3.6-1 as Item 7 and provides frequencies for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION and OPERATIONAL CONDITIONS for which the surveillance is required. (Page 3/4 3-53)

The proposed changes ensure consistency with the GGNS trip system definitions for the logic associated with the reactor mode switch shutdown position. These trip system definitions were presented in a letter from L. F. Dale to H. R. Denton, dated May 8, 1984 (AECM-84/0093). Subsequent to transmittal of this letter, it has been determined that the number of channels per trip system for the reactor mode switch shutdown position should be two instead one as identified in the previous submittal. The proposed changes are more conservative than the existing technical specifications and are considered to be enhancements to safety that are consistent with the philosophy and intent of the technical specifications.

5. (TSPS 237), Control Rod Block Instrumentation Setpoints, Technical Specification Table 3.3.6-2

The proposed change reduces the setpoint for the Average Power Range Monitor (APRM) downscale trip function from greater than or equal to 5% of RATED THERMAL POWER to greater than or equal to 4% which is consistent with vendor design documents. Lowering the APRM downscale setpoint provides additional operating margin to the APRM neutron flux-high setdown of 15% in OPERATIONAL CONDITION 2. The lowered setpoint value also satisfies the design intent of the APRM downscale control rod block function. The proposed change represents an operational enhancement which does not adversely impact plant safety because it is within the previously analyzed safety design basis and ensures consistency with vendor design document. (Page 3/4 3-52)

6. (TSPS 253), Minimum Number of IRM Channels While in Shutdown, Technical Specification Table 3.3.1-1

The proposed change increases the number of IRM Channels required to be OPERABLE during OPERATIONAL CONDITIONS 3 and 4 from two channels per trip system to three channels per trip system. The IRM's are necessary in OPERATING CONDITIONS 3 and 4 because Table 1.2 of the Technical Specifications allows a control rod to be withdrawn in these operating conditions. Of the four channels per trip system available, the NSSS vendor has designed the system such that three in each system will monitor the entire core (any one can be bypassed). Although the Standard Technical Specifications requires two minimum OPERABLE channels per trip system, two channels per trip system is not sufficient to monitor the entire core. Since there is a possiblity that a control rod could be withdrawn from an

unmonitored section of the core without prior shutdown margin demonstration, the minimum OPERABLE channels should be three per trip system to ensure full core monitoring. This change is consistent with FSAR Section 7.6.1.5.4, Technical Specification Bases 2.2.1.1, and Technical Specification 3/4.3.1. The proposed change will ensure the minimum OPERABLE channels requirement is satisfied for all reactor mode switch positions permitted by the footnotes to Table 1.2. The proposed change represents an operational enhancement within the safety analyses and licensing basis. The proposed change is more restrictive than the existing technical specification requirement. (Page 3/4 3-2)

7. (TSPS 257), Safety/Relief Valve Tailpipe Pressure Switches, Technical Specification 3/4.4.2

The proposed change adds a requirement for the safety/relief tail-pipe pressure switches to be OPERABLE in the Limiting Condition for Operation. The ACTION statement which is applicable when these pressure switches are inoperable is presently included in the technical specification. This change is considered an enhancement to the technical specifications that more accurately defines the Limiting Condition for Operation and ensures compliance with the safety analyses. (Page 3/4 4-5)

8. (TSPS 298) Reactor Protection System Instrumentation Setpoints, Technical Specification Bases 2.2.1

The Bases for Technical Specification 2.2.1 describes the reactor protection system instrumentation setpoints associated with the limiting safety system settings. The proposed changes are enhancements or clarifications to the bases for four of the reactor protection system instrumentation trip setpoints, as follows:

a. Drywell Pressure High - An enhancement is proposed to add an explanation for the primary function of this setpoint which is to mitigate the transient caused by a loss of drywell cooling.

b. Turbine Control Valve Fast Closure, Trip Oil Pressure Low - An enhancement is proposed to identify the trip setpoint of 43.3 psig in the bases.

c. Reactor Mode Switch Shutdown Position - A clarification is proposed to explain the redundant trip system inputs provided by the reactor mode switch shutdown position.

d. Manual Scram - A clarification is proposed to identify the redundant trip system inputs provided by the Manual Scram Pushbutton Switches.

The proposed changes provide more accurate descriptions in the technical specification bases of features which are already included in the current licensing bases, are covered by existing safety analysis, and do not affect the design intent for these parameters. (Pages B 2-8 and B 2-9)

9. (TSPS 323), Rod Pattern Control System (RPCS), Technical Specifications 3/4.9.2 and 3/4.10.3

A revision of Refueling Operations Technical Specification 3.9.2.c is proposed to require the Source Range Monitor (SRM) shorting links to be removed from the Reactor Protection System (RPS) Circuitry prior to and during any control rod withdrawals until adequate shutdown margin has been demonstrated. This proposed change will also eleminate the existing provision which allows operability of the RPCS to be substituted for removal of the SRM shorting links. The proposed change sare requested to incorporate vendor recommendations to prevent inadventent criticality with the reactor head removed during core alternations. Removal of the SRM shorting links from the RPS circuitry enables a reactor scram signal to be initiated upon SRM high-high trip. The RPCS is not an appropriate alternative for SRM shorting link removal since the RPCS rod withdrawal restrictions do not prevent the withdrawal of a single control rod in OPERATIONAL CONDITION 5.

A revision to the Shutdown Margin Demonstration Technical Specification 3/4.10.3 is proposed to require the SRM shorting links to be removed from the RPS circuitry prior to the circuitry prior to and during any control rod withdrawals to demonstrate shutdown margin.

The proposed change is requested to provide the same protection required when a single control rod is withdrawn. The proposed change will require either the RPCS to be OPERABLE or compliance with the shutdown margin procedure to be verified by a second qualified person. The change is necessary for conformance with the two methods available to demonstrate shutdown margin: the in-sequence demonstration and the out-of-sequence demonstration. An OPERABLE RPCS provides adequate assurance for an in-sequence demonstration that the prescribed control rod movement sequence is followed. The out-of-sequence demonstration requires the RPCS to be bypassed, so second person verification is necessary to provide assurance that the control rod movement sequence is followed. These changes to Specification 3/4.10.3 will ensure the open vessel shutdown margin demonstration is performed within the established control rod withdrawal sequence and protect against inadvertent criticality. The proposed changes represent the application of more stringent controls than the existing technical specifications and the proposed changes are within the safety analyses and licensing basis. These proposed changes are considered enhancements to the existing technical specification requirements. (Pages 3/4 9-3, 3/4 9-4, and 3/4 10-3)

10. (TSPS 363), Clarification of ACTION Statements, Technical Specifications 3/4.3.1 and 3/4.3.2

The proposed changes revise footnote (**) of Technical Specifications 3.3.1.b and 3.3.2.c to clarify the proper ACTION which must be implemented when the number of OPERABLE channels is less than the number required by the Minimum OPERABLE Channels per Trip System requirement. The proposed changes are required to prevent unnecessary system actuations/trips and to allow for controlled implementation of ACTION statements. The controlled implementation of ACTION statements is achieved by specifying that the trip system need not be placed in the tripped condition if this action would cause the trip function to occur. These changes are consistent with the existing safety analyses and licensing basis and are considered operational enhancements. These changes improve plant safety by decreasing the number of challenges to the trip system. (Pages 3/4 3-1 and 3/4 3-9)

11. (TSPS 367), Accident Monitoring Instrumentation, Technical Specification Table 3.3.7.5-1

The proposed change to Table 3.3.7.5-1 increases the minimum number of OPERABLE channels from one to two for Item 13, containment/drywell area radiation monitors. This proposed change requires that all four containment/drywell area radiation monitors be OPERABLE. Interpretation of Generic Letter 83-86, which provides guidance for the NUREG-0737 items, requires a minimum of two containment and two drywell radiation monitors to be OPERABLE at all times. This proposed change is considered to be an enhancement to the technical specifications and is consistent with the as-built plant. The proposed change imposes more stringent OPERABILITY requirements than are contained in the existing technical specifications. (Page 3/4 3-70)

12. (TSPS 369), Reactor Protection System Instrumentation Surveillance Requirements, Technical Specification Table 4.3.1.1-1

Table 4.3.1.1-1 specifies Surveillance Requirements for instrumentation listed in Table 3.3.1-1, but does not contain the same footnotes for the affected items. This in effect requires a surveillance to be performed when the instrumentation is not required to be OPERABLE. A Technical Specification change is proposed to add footnotes to Table 4.3.1.1-1 such that a surveillance will be applicable only when the affected instrumentation is required to be OPERABLE. This proposed change is administrative in that it promotes consistency between the technical specification limiting conditions for operation and the surveillance requirements. (Pages 3/4 3-7 and 3/4 3-8)

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 079), Reactor Protection System Instrumentation Bases, Technical Specification Bases 3/4.3.1

   The proposed change deletes the reference to IEEE-279 in the Bases for the reactor protection system instrumentation. Section 7.2 of the Grand Gulf Safety Evaluation Report contains a detailed discussion on the conformance of the GGNS design to regulatory requirements. It is unnecessary to incorporate this type of information in the technical specification bases. The proposed change is administrative and does not affect the design bases for the reactor protection system. (Page B 3/4 3-1)

2. (TSPS 083), Suppression Pool Makeup System (SPMU), Technical Specification Tables 3.3.8-1, 3.3.8-2, 4.3.8.1-1, and Bases 3/4.3.8

   Technical Specification 3/4.6.3.4 presently contains the OPERABILITY requirements for the suppression pool makeup system but does not address the associated actuation instrumentation. A proposed change to Technical Specification 3/4.3.8 involves adding the suppression pool makeup system instrumentation to Tables 3.3.8-1, 3.3.8-2 and 4.3.8.1-1. The actuation instrumentation along with associated definitions for the terms "Channels", "Trip Systems" and "Trip Functions" were submitted in a letter from Mr. L. F. Dale to Mr. H. R. Denton on May 8, 1984 (AECM 84/0093). A copy of these definitions associated with this change are included on the last page of Category D of this attachment.

The definitions describe the instrumentation necessary to ensure OPERABILITY of the system, show the number of instruments installed in the plant and allow determination of the minimum OPERABLE channels per Trip System and required ACTION for inoperable instruments. The proposed Applicable Operational Conditions of 1, 2, and 3 for the suppression pool makeup system instrumentation is consistent with the requirements of the associated suppression pool makeup system per Specification 3.6.3.4.

The proposed ACTION statements for Table 3.3.8-1 include allowance for restring redundant instruments and restoration times consistent with the allowances in Specification 3.6.3.4. A footnote (a) is added to Table 3.3.8-1 to permit a channel to be placed in an inoperable status for up to 2 hours for performance of a required surveillance provided a redundant channel is OPERABLE. This allowance is consistent with current provisions of the Technical Specifications (i.e., as permitted by footnote (a) to Table 3.3.3-1 for ECCS actuation instrumentation) and applies to all instrumentation in the Table. An exemption to Specification 3.0.4 requirements is provided with proposed ACTION 135 to facilitate the appropriate response to a degraded Trip Function condition.

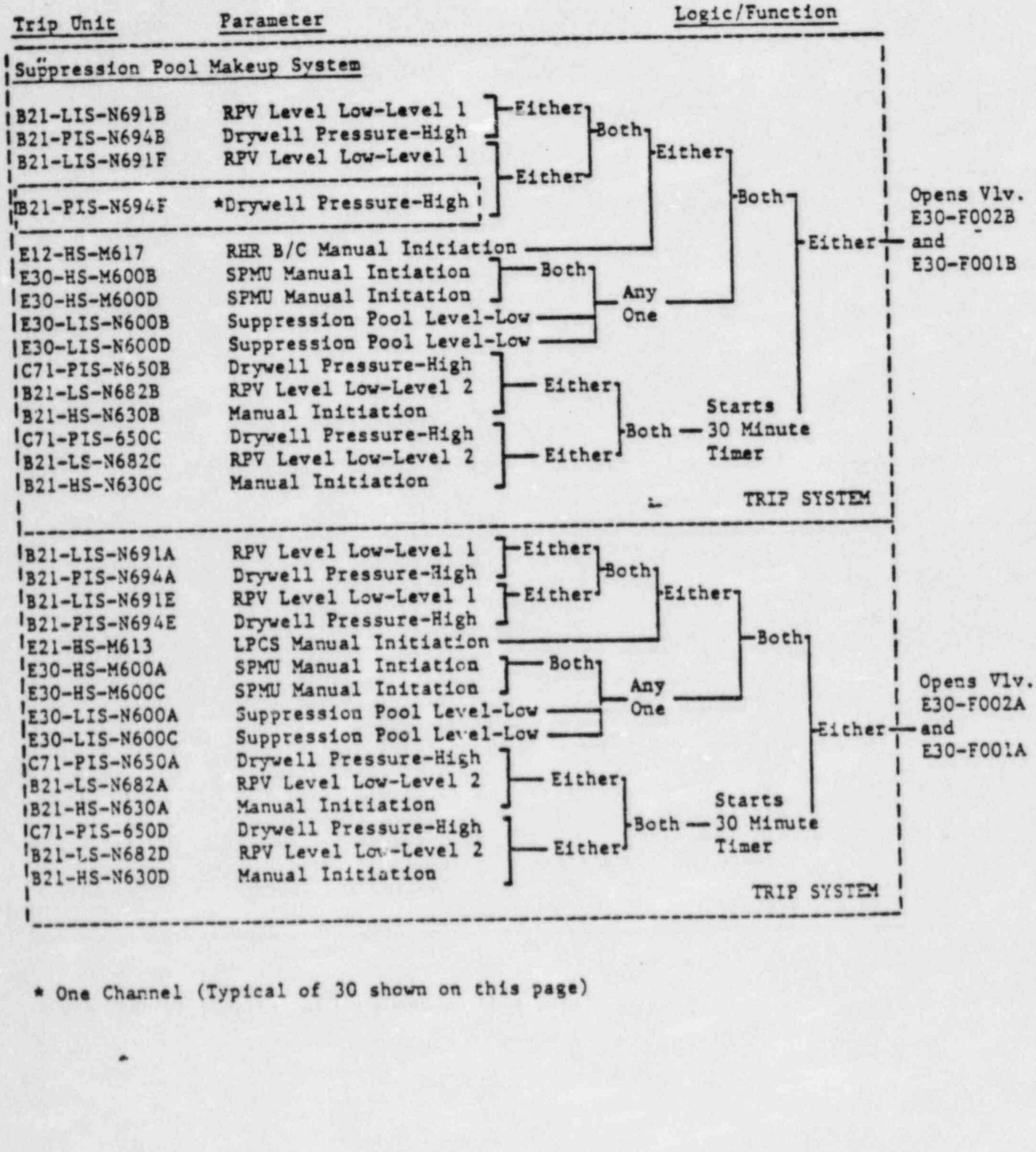
The proposed instrument setpoints and allowable values added to Table 3.3.8-2 are consistent with the settings for the same parameters (for example, Drywell Pressure-High) throughout the technical specifications and are also consistant with the FSAR analyses. An explanation for the suppression pool water level - low setpoints is added to Bases 3/4.3.8. Suppression Pool Makeup Timer Setpoint and Allowable Value are consistent with FSAR analyses of requirements for suppression pool makeup following a LOCA (FSAR Section 6.2.7.3.4).

The Channel Check, Channel Functional Test, and Channel Calibration frequencies added to Table 4.3.8.1-1 are consistent with comparable surveillance requirements in the technical specifications and vendor recommended surveillance frequencies. A footnote (a) is added to Table 4.3.8.1-1 to indicate that the SPMU system instrumentation having a once per 18 months CHANNEL CALIBRATION frequency must have the associated trip unit calibrated at least once per 31 days. This increased calibration frequency is consistent with other Rosemont trip units in the technical specifications and applies to all Rosemont trip units in the Table.

A proposed change to Bases 3/4.3.8 would add an explanation of the suppression pool water level-low instrumentation setpoints. The proposed change will provide complete descriptions within the Bases for the SPMU system actuation instrumentation setpoints.

The proposed changes reflect the as-built plant design and are consistent with the safety analyses. The changes represent enhancement to the technical specifications which inolve imposition of more restrictive requirements on plant operation. These proposed changes are presented in response to an NRC request to include SPMU system instrumentation in the Grand Gulf Technical Specifications. (Pages 3/4 3-98, 3/4 3-98a, 3/4 3-99, 3/4 3-100, B 3/4 3-6)

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR PLANT SYSTEMS ACTUATION INSTRUMENTATION TABLE 3.3.8.1 (Continued)



* One Channel (Typical of 30 shown on this page)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL, correlation. The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

---

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Amendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level, is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run position because at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

7. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

8. Drywell Pressure-High

High pressure or a loss of drywell cooling in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected ~~as low as possible without causing~~ to be as low as possible to minimize heat loads of equipment located in the primary containment and to avoid spurious trips. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high.

LIMITING SAFETY SYSTEM SETTING

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

9. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of 26 gallons of water.

10. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 40 psig, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient assuming the turbine bypass valves fail to operate.

11. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by a low EHC fluid pressure in the control valve and in less than 100 milliseconds after the start of control valve fast closure. This loss of pressure is sensed by pressure transmitters which output to trip units whose contacts form the one-out-of-two twice logic input to the Reactor Protection System. This trip setting and a different valve characteristic from that of the turbine stop valve combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section 15.2.2 of the Final Safety Analysis Report.

The trip setpoint is 43.3 psig.

12. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position ~~is a redundant channel~~ to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

provides trip signals into system trip channels which are redundant

13. Manual Scram

The Manual Scram ~~is a redundant channel~~ to the automatic protective instrumentation channels and provides manual reactor trip capability.

pushbutton switches introduce trip signals into system trip channels which are redundant

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod pattern control system (RPCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#.

ACTION

a. With the RPCS inoperable or with the requirements of ACTION b, below, not satisfied and with:

   1. THERMAL POWER less than or equal to ~~20% of RATED THERMAL POWER~~ the Low Power Setpoint, control rod movement shall not be permitted, except by a scram. HEE

   2. THERMAL POWER greater than ~~20% of RATED THERMAL POWER~~ the Low Power Setpoint, control rod withdrawal shall not be permitted. HEE

b. With an inoperable control rod(s), OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RPCS provided that:

   1. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable, this inoperable control rod may be bypassed in the rod ~~gang drive~~ Action control system (~~RGDS~~ RACS) provided that the SHUTDOWN MARGIN has been determined to be equal to or greater than required by Specification 3.1.1. HEE

   2. With up to eight control rods inoperable for causes other than addressed in ACTION b.1, above, ~~one of~~ these inoperable control rods may be bypassed in the ~~RGDS~~ RACS provided that: HEE

      a) The control rod(s) to be bypassed is inserted and the directional control valves are disarmed either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      b) All inoperable control rods are separated from all other inoperable control rods by at least two control cells in all directions.

      c) There are not more than 3 inoperable control rods in any RPCS group.

   3. The position and bypassing of an inoperable control rod(s) is verified by a second licensed operator or other technically qualified member of the unit technical staff.

*See Special Test Exception 3.10.2

#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RPCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality

4.1.4.2 The RPCS shall be demonstrated OPERABLE by verifying the OPERABILITY of the:

a. Rod pattern controller function when THERMAL POWER is less than the low power setpoint by selecting and attempting to move an inhibited control rod:

1. After withdrawal of the first insequence control rod or gang for each reactor startup.

2. As soon as the rod inhibit mode is automatically initiated at the RPCS low power setpoint, 20 +15, -0% of RATED THERMAL POWER, during power reduction.

3. The first time only that a banked position, N1, N2, or N3, is reached during startup or during power reduction below the RPCS low power setpoint.

b. Rod withdrawal limiter function when THERMAL POWER is greater than or equal to the low power setpoint by selecting and attempting to move a restricted control rod in excess of the allowable distance:

1. As each power range above the RPCS low power setpoint is entered during a power increase or decrease.

2. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power—high scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

Trip Setpoint	Allowable Value
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER.

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 112.5 million lbs/hr.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD). T is ~~always less than or equal to 1.0.~~ Applied only if less than or equal to 1.0. 158

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power—high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the allowable value-column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} to within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP AND MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power—high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**~~If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.~~

REPLACE WITH INSERT ON NEXT PAGE

Amendment No. ___

Insert for ** Footnote, Page 3/4 3-1

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place he trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION	
1. Intermediate Range Monitors:				
a. Neutron Flux - High	2	3	1	153
	3, 4	~~2~~ 3	2	
	$5^{(b)}$	3	3	
b. Inoperative	2	3	1	153
	3, 4	~~2~~ 3	2	
	5	3	3	
2. Average Power Range Monitor (c):				
a. Neutron Flux - High, Setdown	2	3	1	
	3	3	2	
	$5^{(b)}$	3	3	
b. Flow Biased Simulated Thermal Power - High	1	3	4	
c. Neutron Flux - High	1	3	4	
d. Inoperative	1, 2	3	1	
	3	3	2	
	5	3	3	
3. Reactor Vessel Steam Dome Pressure - High	1, $2^{(d)}$	2	1	
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1	
5. Reactor Vessel Water Level-High, Level 8	$1^{(e)}$	2	4	
6. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	4	
7. Main Steam Line Radiation - High	1, $2^{(d)}$	2	5	
8. Drywell Pressure - High	1, $2^{(f)}$	2	1	

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
9. Scram Discharge Volume Water Level - High	1, 2 $5^{(g)}$	2 2	1 3
10. Turbine Stop Valve - Closure	$1^{(h)}$	4	6
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(h)}$	2	6
12. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	~~1~~ 2 ~~1~~ 2 ~~1~~ 2	1 7 3
13. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

197

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED	
1. Intermediate Range Monitors:					
a. Neutron Flux - High	S/U,S,(b) S	S/U~~(c)~~, W W	R R	2 3, 4, 5	355
b. Inoperative	NA	W	NA	2, 3, 4, 5	
2. Average Power Range Monitor:(f)					
a. Neutron Flux - High, Setdown	S/U,S,(b) S	S/U~~(c)~~, W W	SA SA	2 3, 5	355
b. Flow Biased Simulated Thermal Power - High	S, D(h)	~~S/U(c),~~ W	W(d)(e), SA, R(i)	1	355
c. Neutron Flux - High	S	~~S/U(c),~~ W	W(d), SA	1	
d. Inoperative	NA	W	NA	1, 2, 3, 5	
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(g)	1, 2 (j)	329
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(g)	1, 2	
5. Reactor Vessel Water Level - High, Level 8	S	M	R(g)	1	
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1	
7. Main Steam Line Radiation - High	S	M	R	1, 2 (j)	329
8. Drywell Pressure - High	S	M	R(g)	1, 2 (k)	

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High	S	M	$R^{(g)}$	1, 2, $5^{(l)}$
10. Turbine Stop Valve - Closure	S	M	$R^{(g)}$	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	S	M	$R^{(g)}$	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

379

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.
(c) ~~Within 24 hours prior to startup, if not performed within the previous 7 days.~~ [Deleted] 355
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal. 265
(f) The LPRMs shall be calibrated at least once per 1000 ~~effective full power hours (EFPH)~~ megawatt days per ton (Mwd/T) using the TIP system.
(g) Calibrate trip unit at least once per 31 days.
(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.
(i) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant. 367
(j)
(k) } SEE INSERT
(l)

INSERT TO TABLE 4.3.1.1-1, PAGE 3/4 3-8

(j) Not applicable when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(k) Not applicable when DRYWELL INTEGRITY is not required.

(l) Applicable with any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

~~**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.~~ 36

REPLACE WITH INSERT ON NEXT PAGE

Amendment No. —

Insert for ** Footnote, Page 3/4 3-9

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	6A, 7, 8, $10^{(c)(d)}$	2	1, 2, 3 and #	20
b. Reactor Vessel Water Level-Low Low Level 2 (ECCS - Division 3)	6B	4	1, 2, 3 and #	29
c. Reactor Vessel Water Level-Low Low Low, Level 1 (ECCS - Division 1 and Division 2)	$5^{(n)(o)}$	2	1, 2, 3 and #	29
d. Drywell Pressure - High	6A, $7^{(c)(d)}$	2	1, 2, 3	20
e. Drywell Pressure-High (ECCS - Division 1 and Division 2)	$5^{(n)(o)}$	2	1, 2, 3	29
f. Drywell Pressure-High (ECCS - Division 3)	6B	4	1, 2, 3	29
g. Containment and Drywell Ventilation Exhaust Radiation - High High	7	$2^{(e)}$	1, 2, 3 and *	21
h. Manual Initiation	6A, 7, 8, $10^{(c)(d)}$	2	1, 2, 3 and *#	22
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	1	2	1, 2, [illegible]	20
b. Main Steam Line Radiation - High	1, $10^{(f)}$	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	1	2	1	24
d. Main Steam Line Flow - High	1	8	1, 2, 3	23
e. Condenser Vacuum - Low	1	2	1, 2,** 3**	23

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. MAIN STEAM LINE ISOLATION (Continued)				
f. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp.- High	1	2	1, 2, 3	23
h. Manual Initiation	1, 10	2	1, 2, 3	22
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	N.A.(c)(d)(h)	2	1, 2, 3, and #	25
b. Drywell Pressure - High	N.A.(c)(d)(h)	2	1, 2, 3	25
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	N.A.(j)	2	1, 2, 3, and *	25
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	N.A.(j)	2	1, 2, 3, and *	25
e. Manual Initiation	N.A.(c)(d)~~(f)~~(h) N.A.(c)(d)~~(f)~~(h)	2 2	1, 2, 3 *	26 25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	8	1	1, 2, 3	27
b. Δ Flow Timer	8	1	1, 2, 3	27
c. Equipment Area Temperature - High	8	1/room	1, 2, 3	27
d. Equipment Area Δ Temp. - High	8	1/room	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	8	2	1, 2, 3	27

102

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

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

b. In ~~Operational Condition~~ OPERATIONAL CONDITION *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel. 470

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - ~~Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.~~ INSERT 350

ACTION 29 - Close the affected system isolation valves within one hour and declare the affected system or component inoperable or:

a. In OPERATIONAL CONDITION 1, 2 or 3 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 CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

ACTION 30 - Declare the affected SLCS pump inoperable.

NOTES

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

** The low condenser vacuum MSIV closure may be manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\#\# With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

Amendment No. —
Order

APR 1 8 1984

INSERT FOR TABLE 3.3.2-1 ACTION 28

Within one hour, lock the affected system isolation valves closed, or verify, by remote indication, that the valve is closed and electrically disarmed, or isolate the penetration(s) and declare the affected system inoperable.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

NOTES (Continued)

(c) Also actuates the standby gas treatment system.
(d) Also actuates the control room emergency filtration system in the isolation mode of operation.
(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.
(f) Also trips and isolates the mechanical vacuum pumps.
(g) Deleted.
(h) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.
(i) Closes only RWCU system isolation valves G33-F001, G33-F004, and G33-F251.
(j) Actuates the Standby Gas Treatment System and isolates Auxiliary Building penetration of the ventilation systems within the Auxiliary Building.
(k) Closes only RCIC outboard valves. A concurrent RCIC initiation signal is required for isolation to occur.
(l) Valves E12-F037A and E12-F037B are closed by high drywell pressure. All other Group 3 valves are closed by high reactor pressure.
(m) Valve Group 9 requires concurrent drywell high pressure and RCIC Steam Supply Pressure-Low signals to isolate.
(n) Valves E12-F042A and E12-F042B are closed by Containment Spray System initiation signals.
(o) Also isolates valves E61-F009, E61-F010, E61-F056, and E61-F057 from Valve Group 7.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. MAIN STEAM LINE ISOLATION (Continued)		
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**
h. Manual Initiation	NA	NA
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.23 psig	≤ 1.43 psig
c. Fuel Handling Area Ventilation Exhaust Radition - High High	≤ 2.0 mR/hr**	≤ 4.0 mR/hr**
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	≤ 18 mR/hr**	≤ 35 mR/hr**
e. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 79 gpm	≤ 89** gpm
b. Δ Flow Timer	≤ 45 seconds	≤ 57 seconds
c. Equipment Area Temperature - High		
1. RWCU Hx Room	≤ 120°F	≤ 126°F
2. RWCU Pump Rooms	≤ 170°F	≤ 176°F
3. RWCU Valve Nest Room	≤ 135°F	≤ 141°F
~~4. RWCU Demin. Rooms~~	~~≤ 139°F~~	~~≤ 145°F~~
~~5. RWCU Rec. Tank Room~~	~~≤ 139°F~~	~~≤ 145°F~~
~~6. RWCU Demin. Valve Room~~	~~≤ 135°F~~	~~≤ 141°F~~
d. Equipment Area Δ Temp. - High		
1. RWCU Hx Room	≤ 65°F	≤ 66°F
2. RWCU Pump Rooms	≤ 115°F	≤ 118°F
3. RWCU Valve Nest Room	≤ 70°F	≤ 73°F
~~4. RWCU Demin. Rooms~~	~~≤ 70°F~~	~~≤ 73°F~~
~~5. RWCU Rec. Tank Room~~	~~≤ 70°F~~	~~≤ 73°F~~
~~6. RWCU Demin. Valve Room~~	~~≤ 71°F~~	~~≤ 74°F~~

308

308

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)		
e. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -41.6 inches*	$\geq$ -43.8 inches
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ 101°F**	$\leq$ 104°F**
h. SLCS Initiation	NA	NA
i. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	$\leq$ 363" H_2O	$\leq$ 371" H_2O
b. RCIC Steam Supply Pressure - Low	$\geq$ 60 psig	$\geq$ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig
d. RCIC Equipment Room Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
e. RCIC Equipment Room Δ Temp. - High	$\leq$ 125°F**	$\leq$ 128°F**
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ 101°F**	$\leq$ 104°F**
h. Main Steam Line Tunnel Temperature Timer	$\leq$ 30 minutes	$\leq$ 30 minutes
i. RHR Equipment Room Ambient Temperature - High	$\leq$ 165°F**	$\leq$ 171°F**
j. RHR Equipment Room Δ Temperature - High	$\leq$ 99°F**	$\leq$ 102°F**
k. RHR/RCIC Steam Line Flow - High	$\leq$ 145" H_2O	$\leq$ ~~160"~~ H_2O 151

515

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#	
1.	**PRIMARY CONTAINMENT ISOLATION**		
a.	Reactor Vessel Water Level - Low Low, Level 2	≤ ~~13~~ 10(a)	040
b.	Reactor Vessel Water Level - Low Low, Level 2 (ECCS - Division 3)	≤ ~~13~~ 10(a)	
c.	Reactor Vessel Water Level-Low Low Low, Level 1 (ECCS - Division 1 and Division 2)	≤ ~~13~~ 10(a)	
d.	Drywell Pressure - High	≤ ~~13~~ 10(a)	
e.	Drywell Pressure-High (ECCS - Division 1 and Division 2)	≤ ~~13~~ 10(a)	
f.	Drywell Pressure-High (ECCS - Division 3)	≤ ~~13~~ 10(a)	
g.	Containment and Drywell Ventilation Exhaust Radiation - High High(b)	≤ ~~13~~ 10(a)**	
h.	Manual Initiation	NA	
2.	**MAIN STEAM LINE ISOLATION**		
a.	Reactor Vessel Water Level - Low Low Low, Level 1	≤ 1.0*/≤ ~~13~~ 10(a)**	040
b.	Main Steam Line Radiation - High(b)	≤ 1.0*/≤ ~~13~~ 10(a)**	
c.	Main Steam Line Pressure - Low	≤ 1.0*/≤ ~~13~~ 10(a)**	
d.	Main Steam Line Flow - High	≤ 0.5*/≤ ~~13~~ 10(a)**	
e.	Condenser Vacuum - Low	NA	
f.	Main Steam Line Tunnel Temperature - High	NA	
g.	Main Steam Line Tunnel Δ Temp. - High	NA	
h.	Manual Initiation	NA	
3.	**SECONDARY CONTAINMENT ISOLATION**		
a.	Reactor Vessel Water Level - Low Low, Level 2	≤ ~~13~~ 10(a)	040
b.	Drywell Pressure - High	≤ ~~13~~ 10(a)	
c.	Fuel Handling Area Ventilation Exhaust Radiation - High High(b)	≤ 3.0***/≤ ~~13~~ 10(a)	
d.	Fuel Handling Area Pool Sweep Exhaust Radiation - High High(b)	≤ 3.0***/≤ ~~13~~ 10(a)	
e.	Manual Initiation	NA	
4.	**REACTOR WATER CLEANUP SYSTEM ISOLATION**		
a.	Δ Flow - High	NA	
b.	Δ Flow Timer	NA	
c.	Equipment Area Temperature - High	NA	
d.	Equipment Area Δ Temp. - High	NA	
e.	Reactor Vessel Water Level - Low Low, Level 2	≤ ~~13~~ 10(a)	040
f.	Main Steam Line Tunnel Ambient Temperature - High	NA	
g.	Main Steam Line Tunnel Δ Temp. - High	NA	
h.	SLCS Initiation	NA	
i.	Manual Initiation	NA	

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#	
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ ~~13~~ 10 (a)###	040
b. RCIC Steam Supply Pressure - Low	≤ ~~13~~ 10 (a)	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	
d. RCIC Equipment Room Ambient Temperature - High	NA	
e. RCIC Equipment Room Δ Temp. - High	NA	
f. Main Steam Line Tunnel Ambient Temp. - High	NA	
g. Main Steam Line Tunnel Δ Temp. - High	NA	
h. Main Steam Line Tunnel Temperature Timer	NA	
i. RHR Equipment Room Ambient Temperature - High	NA	
j. RHR Equipment Room Δ Temp. - High	NA	
k. RHR/RCIC Steam Line Flow - High	NA	
l. Manual Initiation	NA	
m. Drywell Pressure - High (ECCS Division 1 and Division 2)	≤ ~~13~~ 10 (a)	040
6. RHR SYSTEM ISOLATION		
a. RHR Equipment Room Ambient Temperature - High	NA	
b. RHR Equipment Room Δ Temp. - High	NA	
c. Reactor Vessel Water Level - Low, Level 3	≤ ~~13~~ 10 (a)	040
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	
e. Drywell Pressure - High	NA	
f. Manual Initiation	NA	

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

\#\#\# Isolation system instrumentation response time for air operated dampers. No diesel generator delays assumed. 040

\#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.~~5~~.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve. 040

\#\#\#Without 13 second time delay.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (Continued)				
e. Drywell Pressure - High	S	M	R[(c)]	1, 2, 3
f. Manual Initiation	NA	M[(a)]	NA	1, 2, 3

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

**The low condenser vacuum MSIV closure may be ~~the~~ manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint. | 404

#During CORE ALTERATION and operations with a potential for draining the reactor vessel.

##With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

(c) Calibrate trip unit at least once per 31 days.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION	
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) & LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3, 4*, 5*	30	
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30	
c. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31	
d. Manual Initiation	1/system[(b)]	1, 2, 3, 4*, 5*	32	278
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3	30	
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30	
c. ADS Timer	1	1, 2, 3	31	
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31	
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	31	
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	31	
g. Manual Initiation	2/system	1, 2, 3	32	
B. DIVISION 2 TRIP SYSTEM				
1. RHR B & C (LPCI MODE)				
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2[(b)]	1, 2, 3, 4*, 5*	30	
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30	
c. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31	
d. Manual Initiation	1/system[(b)]	1, 2, 3, 4*, 5*	32	278
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3	30	
b. Drywell Pressure - High	2[(b)]	1, 2, 3	[illegible]	
c. ADS Timer	1	1, 2, 3	31	
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31	
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31	
f. Manual Initiation	2/system	1, 2, 3	32	

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	$4^{(b)}$	1, 2, 3, 4*, 5*	33
b. Drywell Pressure - High##	$4^{(b)}$	1, 2, 3	33
c. Reactor Vessel Water Level-High, Level 8	$2^{(c)}$	1, 2, 3, 4*, 5*	31
d. Condensate Storage Tank Level-Low	$2^{(d)}$	1, 2, 3, 4*, 5*	34
e. Suppression Pool Water Level-High	$2^{(d)}$	1, 2, 3, 4*, 5*	34
f. Manual Initiation##	1/~~system~~	1, 2, 3, 4*, 5*	32
D. LOSS OF POWER			
1. Division 1 and 2			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	4	1, 2, 3, 4**, 5**	30
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30
2. Division 3			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only.

(d) Provides signal to HPCS pump suction valves only.

(e) One out-of-two taken.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## Prior to STARTUP following the first refueling outage, the injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function ~~requirement:~~

~~a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable.~~

~~b. For both trip systems, declare the HPCS system inoperable.~~

3o3

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

requirement, place the inoperable channel(s) in the tripped condition within one hour* or declare the HPCS system inoperable.

APR 1 8 1984

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER		
1. Division 1 and 2		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 2912 volts	2912 +0, -291 volts
	2. 120 volt Basis 83.2 volts	83.2 +0, -8.3 volts
	3. Time Delay 0.5 seconds	0.5 +0.5, -0.1 seconds
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	1. 4.16 kV Basis 3328 volts	3328 +0, -167 volts
	2. 120 volt Basis 95.1 volts	95.1 +0, -4.8 volts
	3. Time delay 0.5 seconds	0.5 +0.5, -0.1 seconds
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	1. 4.16 kV Basis 3744 volts	3744 +93.6, -0 volts
	2. 120 volt Basis 107 volts	107 +2.7, -0 volts
	3. Time Delay 9.0 seconds	9.0 ± 0.5 seconds
2. Division 3		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 3045 volts	3045 ± 61 volts
	2. 120 volt Basis 87 volts	87 ± 1.7 volts
	3. Time Delay 2.3 seconds	2.3 + 0.2, -0.3 seconds

*See Bases Figure B 3/4 3-1.
~~#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.~~

0714

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED	
A. DIVISION I TRIP SYSTEM					
1. RHR-A (LPCI MODE) AND LPCS SYSTEM					
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*	
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3	
c. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*	
d. Manual Initiation	NA	R(b)~~(c)~~	Q~~(d)~~	1, 2, 3, 4*, 5*	206
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#					
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3	
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3	
c. ADS Timer	NA	M	Q	1, 2, 3	
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3	
e. LPCS Pump Discharge Pressure-High	S	M	R	1, 2, 3	
f. LPCI Pump A Discharge Pressure-High	S	M	R(a)	1, 2, 3	
g. Manual Initiation	NA	R(b)	NA	1, 2, 3	
B. DIVISION 2 TRIP SYSTEM					
1. RHR B AND C (LPCI MODE)					
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*	
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3	
c. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*	
d. Manual Initiation	NA	R(b)~~(c)~~	Q~~(d)~~	1, 2, 3, 4*, 5*	206

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTATION

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## Prior to STARTUP following the first refueling outage, the injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

(a) Calibrate trip unit at least once per 31 days.

(b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.

~~(c) Manual initiation test shall include verification of the OPERABILITY of the LPCS and LPCI injection valve interlocks. (See Note 1)~~ 206

~~(d) This calibration shall consist of the CHANNEL CALIBRATION of the LPCS and LPCI injection valve interlocks with the interlock setpoint verified to be ≤ 150 psig. (See Note 1)~~

(e) Functional Testing of Time Delay Not Required

~~Note 1: Until restart after the first refueling outage, the requirements of (c) and (d) above do not apply.~~ 206

(c) Deleted

(d) Deleted

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

Insert →

~~b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.~~

~~c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:~~

~~1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour.~~ 022

~~2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.~~

~~d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.~~

~~e. With both trip systems inoperable, restore at least one trip sytem to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.~~

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

Insert for Technical Specification 3/4.3.4.1, Page 3/4 3-34

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE per Trip System for one or both Trip Systems, restore the inoperable channel(s) to OPERABLE status within 14 days or be in at least STARTUP within the next six hours.

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. ~~Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months.~~ The time allotted for breaker arc suppression, 50 ms, shall be verified at least once per 60 months.*

---

*Prior to STARTUP after the first refueling outage, the breaker arc suppression time of 12 ms, as determined by the manufacturer, shall apply.

Each test shall include two turbine control valve channels from one trip system and two turbine stop valve channels from the other trip system such that all channels are tested at least once per 36 months.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS ~~PER TRIP SYSTEM~~ ~~(a)~~ (a)	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4	50
b. Reactor Vessel Water Level - High, Level 8	2~~(b)~~	51
c. Condensate Storage Tank Water Level - Low	2~~(c)~~	52
d. Suppression Pool Water Level - High	2~~(c)~~	52
e. Manual Initiation	1~~/system~~~~(d)~~	53

360

360

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

~~(b) One trip system with two-out-of-two logic.~~

~~(c) One trip system with one-out-of-two logic.~~

~~(d) One trip system with one channel.~~

360

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels ~~per Trip System~~ requirement ~~for one trip system~~; place the inoperable channel(s) or ~~that~~ trip system in the tripped condition within one hour or declare the RCIC system inoperable. 390

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels ~~per Trip System~~ requirement, declare the RCIC system inoperable. 390

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels ~~per Trip System~~ requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable. 390

ACTION 53 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels ~~per Trip System~~ requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable. 390

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION	
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	2	1, 2	60	334
b. ~~Intermediate Rod Withdrawal~~ High Power ~~Limiter~~ Setpoint	2	1, 2	60	
2. APRM				
a. Flow Biased Neutron Flux-Upscale	6	1	61	
b. Inoperative	6	1, 2, 5	61	
c. Downscale	6	1	61	
d. Neutron Flux - Upscale, Startup	6	2, 5	61	
3. SOURCE RANGE MONITORS				600
a. Detector not full in(a)	4	2, ~~5~~	61	
	2**	5	62	
b. Upscale(b)	4	2, ~~5~~	61	
	2**	5	62	
c. Inoperative(b)	4	2, ~~5~~	61	
	2**	5	62	
d. Downscale(c)	4	2, ~~5~~	61	
	2**	5	62	
4. INTERMEDIATE RANGE MONITORS				110
a. Detector not full in ~~(d)~~	6	2, 5	61	
b. Upscale	6	2, 5	61	
c. Inoperative	6	2, 5	61	
d. Downscale(d)	6	2, 5	61	
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	2	1, 2, 5*	62	
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	3	1	62	
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63	167

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RPCS inoperable and take the ACTION required by Specification 3.1.4.2.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(a) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(b) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(c) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

** OPERABLE channels must be associated with SRMs required OPERABLE per Specification 3.9.2.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

Amendment No. —

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
1. ROD PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	20 + 15, -0% of RATED THERMAL POWER	20 + 15, -0% of RATED THERMAL POWER	
b. High Power ~~Intermediate Rod Withdrawal Limiter~~ Setpoint	≤ 70% of RATED THERMAL POWER	≤ 70% of RATED THERMAL POWER	334
2. APRM			
a. Flow Biased Neutron Flux-Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*	
b. Inoperative	NA	NA	
c. Downscale	≥ ~~5~~ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER	137
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER	
3. SOURCE RANGE MONITORS			
a. Detector not full in	NA	NA	
b. Upscale	≤ 1 x 10^5 cps	≤ 1.5 x 10^5 cps	
c. Inoperative	NA	NA	
d. Downscale	≥ 0.7 cps	≥ 0.5 cps	
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	NA	NA	
b. Upscale	≤ 108/125 of full scale	≤ 110/125 of full scale	
c. Inoperative	NA	NA	
d. Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale	
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	≤ 32 inches	≤ 33.5 inches	
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow	
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA	167

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED	
1. ROD PATTERN CONTROL SYSTEM					
a. Low Power Setpoint	NA	S/U(b)~~(e)~~, D(c)~~(e)~~, M(d)~~(e)~~	Q	1, 2	356
b. High Power ~~Intermediate Rod Withdrawal Limiter~~ Setpoint	NA	S/U(b)~~(e)~~, D(c)~~(e)~~, M(d)~~(e)~~	Q	1, 2	334 356
2. APRM					
a. Flow Biased Neutron Flux-Upscale	NA	~~S/U(h)~~, W	W(f)(g), SA	1	356
b. Inoperative	NA	S/U~~(h)~~, W	NA	1, 2, 5	
c. Downscale	NA	~~S/U(h)~~, W	W(h), SA	1	
d. Neutron Flux - Upscale, Startup	NA	S/U~~(h)~~, M	Q	2, 5	
3. SOURCE RANGE MONITORS					
a. Detector not full in	NA	S/U~~(b)~~, W	NA	2, 5	356
b. Upscale	NA	S/U~~(b)~~, W	Q	2, 5	
c. Inoperative	NA	S/U~~(b)~~, W	NA	2, 5	
d. Downscale	NA	S/U~~(b)~~, W	Q	2, 5	
4. INTERMEDIATE RANGE MONITORS					
a. Detector not full in	NA	S/U~~(b)~~, W	NA	2, 5	356
b. Upscale	NA	S/U~~(b)~~, W	Q	2, 5	
c. Inoperative	NA	S/U~~(b)~~, W	NA	2, 5	
d. Downscale	NA	S/U~~(b)~~, W	Q	2, 5	
5. SCRAM DISCHARGE VOLUME					
a. Water Level-High	NA	M	R	1, 2, 5*	
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW					
a. Upscale	NA	~~S/U~~(b), M	Q	1	356
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4	197

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 7 days ~~24 hours~~ prior to startup. ~~if not performed within the previous 7 days.~~

c. Within 24 hours ~~one hour~~ prior to control rod movement, ~~unless performed within the previous 24 hours,~~ and as each power range above the RPCS low power setpoint is entered for the first time during any 24 hour period during power increase or decrease. 356

d. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

e. ~~Includes reactor manual control multiplexing system input.~~ DELETED

f. This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM Channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

g. This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

h. This calibration shall consist of verifying the trip setpoint only.

* With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS ~~INSTRUMENTS~~ OPERABLE
a. Wind Speed	
1. Elev. 33 ft and 162 ft	1 each
b. Wind Direction	
1. Elev. 33 ft and 162 ft	1 each
c. Air Temperature	
1. Elev. 33 ft	1
d. Air Temperature Difference	
1. Elev. 33/162 ft	1

REMOTE SHUTDOWN SYSTEM ~~MONITORING~~ INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system ~~monitoring~~ instrumentation and controls ~~channels~~ shown in Table 3.3.7.4-1 shall be OPERABLE. ~~with readouts on the remote shutdown panel~~.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown system ~~monitoring~~ instrumentation less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c ~~b~~. The provisions of Specification 3.0.4 are not applicable. 077

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown system ~~monitoring~~ instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

4.3.7.4.2 Each of the above remote shutdown control switches and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

b. With the number of OPERABLE remote shutdown system controls less than required by Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

Amendment No. ——

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	1
2. Reactor Vessel Water Level	1
3. Suppression Pool Water Level	1
4. Suppression Pool Water Temperature	1
5. RHR System Flow	1
6. Standby Service Water System Flow	1
7. RCIC Turbine Speed	1
8. Condensate Storage Tank Level	1

TABLE 3.3.7.4-1

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE DIV 1	MINIMUM CHANNELS OPERABLE DIV 2
1. Reactor Vessel Pressure	1	1
2. Reactor Vessel Water Level	1	1
3. Suppression Pool Water Level	1	1
4. Suppression Pool Water Temperature	1	1
5. RHR System Flow	1	1
6. Standby Service Water System Flow	1	1
7. RCIC Turbine Speed	1	NA
8. Condensate Storage Tank Level	1	NA

REMOTE SHUTDOWN SYSTEM CONTROLS

CONTROL	MINIMUM CHANNELS OPERABLE DIV 1	MINIMUM CHANNELS OPERABLE DIV 2
1. SSW Pump	1	1
2. SSW Pump Discharge Vlv	1	1
3. SSW Basin Transfer Vlv	1	1
4. SSW Pump Recirc Vlv	1	1
5. SSW Return Vlv to Cooling Tower	1	1
6. SSW Cooling Tower Fans	2[b]	2[b]
7. SSW Bypass Vlvs	NA	2[a]
8. RHR HX Inlet/Outlet/Bypass Vlvs	5[a]	5[a]
9. RHR Pump	1	1
10. RHR Pump Suction Vlv	1	1
11. RHR Shutdown Cooling-Vlv	3[a]	3[a]

Z50sd27

TABLE 3.3.7.4-1 (continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

CONTROL	MINIMUM CHANNELS OPERABLE DIV 1	DIV 2
12. RHR Injection Vlvs	2[a]	2[a]
13. RHR Test Line Vlv	1	1
14. RHR Hx Cond to RCIC Vlv	1	1
15. RHR Hx Flow to Suppr Pool Vlv	1	1
16. RHR Discharge To Radwaste Vlv	1	1
17. RCIC Steam to RHR Hx Vlv	2[a]	2[a]
18. Diesel Generator Hx Inlet Vlv	1	1
19. Safety/Relief Vlvs	6[a]	6[a]
20. RHR to RCIC Head Spray Line Vlv	NA	1
21. RCIC Turbine Flow Controller	1	NA
22. RCIC Suct Flow Suppr Pool Vlv	1	NA
23. RCIC Inj Shutoff Vlv	1	NA
24. RCIC Suct from CST	1	NA
25. RCIC Recirc Main Flow Byp Vlv	1	NA
26. RCIC Test FCV to CST	1	NA
27. RCIC Test RTN to CST Vlv	1	NA
28. Steam to RCIC Turb Vlv	1	NA
29. RCIC Turbine Trip & Throttle Vlv	1	NA
30. RCIC Turb Cooling Wtr Vlv	1	NA
31. RCIC Turb Local Cont Sel Sw	1	NA
32. RCIC Gland Seal Compressor	1	NA
33. Shutdown Cooling Isolation Vlv Reset Sw	1	1

NOTE: a. 1 per valve
b. 1 per cooling tower fan

TABLE 3.3.7.5-1
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	APPLICABLE OPERATIONAL CONDITIONS	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Reactor Vessel Pressure	1, 2, 3	2	1	80
2. Reactor Vessel Water Level	1, 2, 3, 4, 5	2	1	82
3. Suppression Pool Water Level	1, 2, 3	2	1	80
4. Suppression Pool Water Temperature	1, 2, 3	6, 1/sector	6, 1/sector	80
5. Drywell/Containment Differential Pressure	1, 2, 3	2	1	80
6. Drywell Pressure	1, 2, 3	2	1	80
7. Drywell and Control Rod Drive Cavity Temperature	1, 2, 3	2 (each)	1 (each)	80
8. Containment Hydrogen Concentration Analyzer and Monitor	1, 2, 3	2	1	80
9. Drywell Hydrogen Concentration Analyzer and Monitor	1, 2, 3	2	1	80
10. Containment Pressure (wide and narrow range)	1, 2, 3	2 (each)	1 (each)	80
11. Containment Air Temperature	1, 2, 3	2	1	80
12. Safety/Relief Valve Tail Pipe Pressure Switch Position Indicators	1, 2, 3	1/valve	1/valve	80
13. Containment/Drywell Area Radiation Monitors	1, 2, 3, 4, 5	2#	2 ~~1~~#	81
14. Containment Ventilation Exhaust Radiation Monitor	1, 2, 3, 4, 5	1	1	81
15. Off-gas and Radwaste Bldg. Ventilation Exhaust Radiation Monitor	1, 2, 3, 4, 5	1	1	81
16. Fuel Handling Area Ventilation Exhaust Radiation Monitor	1, 2, 3, 4, 5	1	1	81
17. Turbine Bldg. Ventilation Exhaust Radiation Monitor	1, 2, 3	1	1	81
18. Standby Gas Treatment System A & B Exhaust Radiation Monitors	*	1/each	1/each	81

#Each for containment and drywell.
*When its associated train of the standby gas treatment system is required operable (Ref. 3.6.6.3).

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least ~~three~~ the following source range monitor channels shall be OPERABLE.

a. In OPERATIONAL CONDITION 2*, four

b. In OPERATIONAL CONDITION 3 or 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least ~~3~~ 4 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with ~~two~~ one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

a) 12 hours in CONDITION 2*, and

b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 0.7 cps with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. ~~Three~~ Five movable detectors, drives and readout equipment to map the core, and 010

b. Indexing equipment to allow all ~~three~~ five detectors to be calibrated in a common location. 010

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

With the traversing in-core probe system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use when required for the ~~above applicable monitoring or~~ LPRM calibration function~~s~~. 050

*Only the detector(s) in the location(s) of interest are required to be OPERABLE.

INSTRUMENTATION

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 Two independent chlorine detection ~~systems~~ CHANNELS shall be OPERABLE with their trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm. 346

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one CHANNEL chlorine detection ~~system~~ CHANNEL inoperable, restore the inoperable detection ~~system~~ to OPERABLE status within 7 days, or within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the isolation mode of operation. 346

b. With both chlorine detection ~~systems~~ CHANNELS inoperable, within one hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the isolation mode of operation. 346

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine detection ~~systems~~ CHANNELS shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 6 months. 346

TABLE 3.3.8-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CONTAINMENT SPRAY SYSTEM			
a. Drywell Pressure-High	2	1, 2, 3	130
b. Containment Pressure-High	1	1, 2, 3	131
c. Reactor Vessel Water Level-Low Low Low, Level 1	2	1, 2, 3	130
d. Timers			
1) System A	1	1, 2, 3	131
2) System B	1	1, 2, 3	131
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High, Level 8	3	1	132
3. SUPPRESSION POOL MAKEUP SYSTEM			
a. Drywell Pressure-High (ECCS)	2	1, 2, 3	135
b. Drywell Pressure-High (RPS)	2	1, 2, 3	135
c. Reactor Vessel Water Level-Low Low Low, Level 1	2	1, 2, 3	135
d. Reactor Vessel Water Level-Low Low, Level 2	2	1, 2, 3	135
e. Suppression Pool Water Level-Low	1	1, 2, 3	133
f. Suppression Pool Makeup Timer	1	1, 2, 3	133
g. SPMU Manual Initiation	2	1, 2, 3	134

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance provided at least one other OPERABLE channel in the same trip system is

083

PLANT SYSTEMS ACTUATION INSTRUMENTATION

ACTION

ACTION 130 - a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within one hour; otherwise, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE channels per Trip System requirement, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

ACTION 131 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the channels to OPERABLE status within one hour; otherwise, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

ACTION 132 - For the feedwater system/main turbine trip system:

a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

INSERT

083

* The provisions of Specification 3.0.4 are not applicable.

083

Insert for Technical Specification Page 3/4 3-98a

ACTION 133 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated suppression pool makeup system inoperable and take the action required by Specification 3.6.3.4

ACTION 134 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channels to OPERABLE status within 8 hours; otherwise, declare the associated suppression pool makeup system inoperable and take the action required by Specification 3.6.3.4.

ACTION 135 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

TABLE 3.3.8-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CONTAINMENT SPRAY SYSTEM		
a. Drywell Pressure-High	≤ 1.39 psig	≤ 1.44 psig
b. Containment Pressure-High	≤ 7.84 psig	≤ 8.34 psig
c. Reactor Vessel Water Level-Low Low Low, Level 1	≥ - 150.3 inches	≥ - 152.5 inches
d. Timers		
1) System A	10.85 ± 0.10 minutes	10.26 - 0.00, + 1.18 minutes
2) System B	10.85 ± 0.10 minutes**	10.26 - 0.00, + 1.18 minutes
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level 8	≤ 53.5 inches*	≤ ~~55.7~~ 54.1 inches

*See Bases Figure B 3/4 3-1.
**Setpoint for System B is the sum of E12-K093B plus E12-K116. E12-K116 is not to exceed 10.00 seconds.

3. SUPPRESSION POOL MAKEUP SYSTEM		
a. Drywell Pressure - High (ECCS)	≤ 1.39 psig	≤ 1.44 psig
b. Drywell Pressure - High (RPS)	≤ 1.23 psig	≤ 1.43 psig
c. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
d. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ - 43.8 inches
e. Suppression Pool Water Level - Low	≥ 16 ft 4 inches	≥ 15 ft 6.5 inches
f. Suppression Pool Makeup Timer	≤ 29.0 minutes	≤ 29.5 minutes
g. SPMU Manual Initiation	NA	NA

345

080

TABLE 4.3.8.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CONTAINMENT SPRAY SYSTEM				
a. Drywell Pressure-High	S	M	$R^{(a)}$	1, 2, 3
b. Containment Pressure-High	S	M	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	$R^{(a)}$	1, 2, 3
d. Timers	NA	M	Q	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level 8	S	M	R	1
3. SUPPRESSION POOL MAKEUP SYSTEM				
a. Drywell Pressure - High (ECCS)	S	M	$R^{(a)}$	1, 2, 3
b. Drywell Pressure - High (RPS)	S	M	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	$R^{(a)}$	1, 2, 3
d. Reactor Vessel Water Level - Low Low, Level 2	S	M	$R^{(a)}$	1, 2, 3
e. Suppression Pool Water Level - Low	S	M	$R^{(a)}$	1, 2, 3
f. Suppression Pool Makeup Timer	NA	M	Q	1, 2, 3
g. SPMU Manual Initiation	NA	R	NA	1, 2, 3

(a) Calibrate trip unit at least once per 31 days.

083

083

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

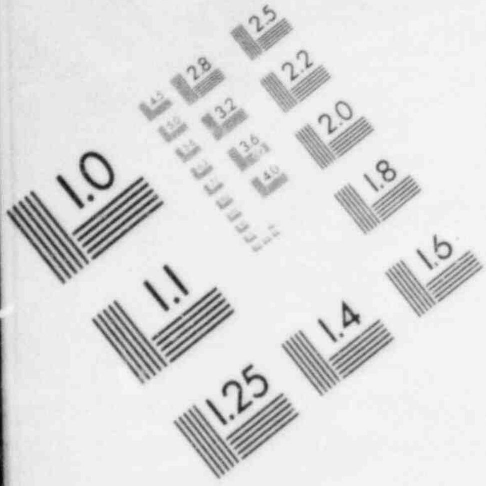
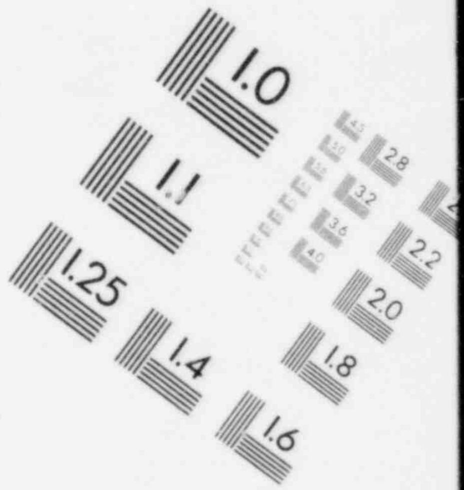
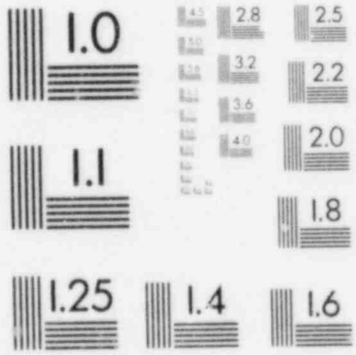
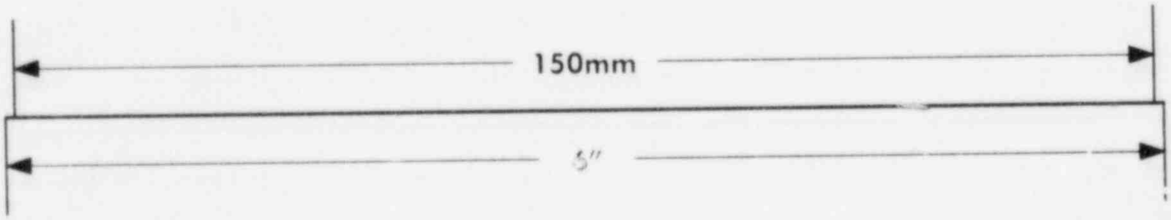
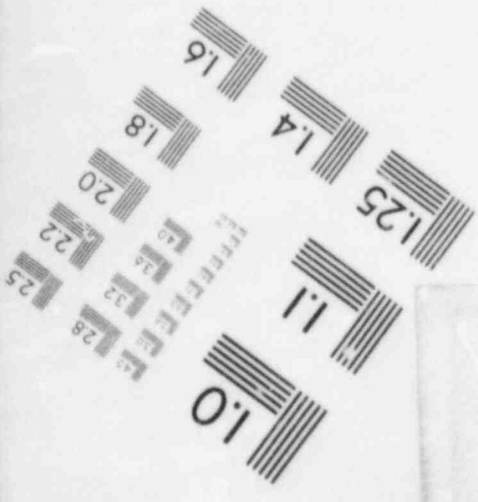


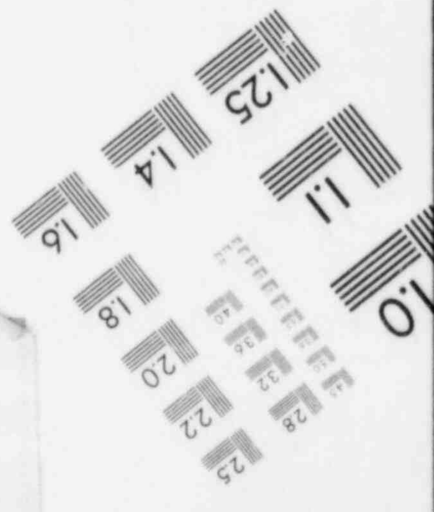
IMAGE EVALUATION TEST TARGET (MT-3)











REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 ~~Of~~ FOR the following safety/relief valves: a) ~~the~~ The safety valve function of at least 7 valves and the relief valve function of at least 6 valves other than those satisfying the specified safety valve function requirement shall be OPERABLE with the specified lift settings; AND 257 257

b) THE SAFETY/RELIEF TAIL-PIPE PRESSURE SWITCHES FOR EACH SAFETY/RELIEF VALVE SHALL BE OPERABLE.

Number of Valves	Function	Setpoint* (psig)
8	Safety	1165 ± 11.6 psi
6	Safety	1180 ± 11.8 psi
6	Safety	1190 ± 11.9 psi
1#	Relief	1103 ± 15 psi
10	Relief	1113 ± 15 psi
9	Relief	1123 ± 15 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 105°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief tail-pipe pressure switches inoperable, restore the inoperable switch(es) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. 013

INSERT

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The tail-pipe pressure switch for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 30 ± 5 psig by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.
b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

#Initial opening of 1B21-F051B is 1103 ± 15 psig due to low-low set function.

Insert for Technical Specification 3.4.2.1, Page 3/4 4-5

d. With either relief valve function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

Z50sd25

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following low-low set function lift settings:

Valve No.	Setpoint* (psig) ± 15 psi Open	Close
F051D	1033	926
F051B	1073	936
F047D	1113	946
F047G	1113	946
F051A	1113	946
F051F	1113	946

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and the low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

INSERT

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

Insert for Technical Specification 3.4.2.2, Page 3/4 4-6

c. With either relief valve/low-low set function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SUPPRESSION POOL MAKEUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.4 The suppression pool makeup system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool makeup line inoperable, restore the inoperable makeup line 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 the upper containment pool water level less than the limit, restore the water level to within the limit within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With upper containment pool water temperature greater than the limit, restore the upper containment pool water temperature to within the limit 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

4.6.3.4 The suppression pool makeup system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the upper containment pool water:

    1. Level to be greater than or equal to 23'3" above the pool bottom in the dryer/separator storage area, and 18

    2. Temperature to be less than or equal to 125°F.

b. 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 secure in position, is in its correct position.

c. At least once per 18 months by performing 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 makeup of water to the suppression pool may be excluded from this test.

Amendment No. ___

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a subsystem flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and
      b) Fuel handling accident.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.2 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%.

   3. Verifying that the filter train and isolation dampers receive the appropriate actuation signal by each of the following test conditions. For at least one of these test conditions, verify that the filter train starts and isolation dampers open on receipt of the actuation signal.

      a. Drywell pressure - high,
      b. Reactor vessel water level - low low, level 2,
      c. Fuel handling area ventilation exhaust radiation - high high ~~high~~,
      d. Fuel handling area pool sweep exhaust radiation - high high ~~high~~, and
      e. Manual initiation from the Control Room.

   4. Verifying that the fan can be manually started.

   5. Verifying that the heaters dissipate 50 ± 5.0 kW when tested in accordance with ANSI N510-1975 (except for the phase balance criteria stated in Section 14.2.3).

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.2 inches Water Gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

2. Verifying that the subsystem receives an appropriate isolation actuation signal by each of the following test conditions. For at least one of the test conditions, verify that the subsystem automatically switches to the isolation mode of operation and the isolation valves close within 4 seconds.

   a) ~~High~~ High high radiation in the outside air intake duct,
   b) High chlorine concentration in the outside air intake duct,
   c) High drywell pressure,
   d) ~~Low~~ Low low reactor water level, and
   e) Manual initiation from the Control Room.

3. Verifying that the heaters dissipate 20.7 ± 2.1 kW when tested in accordance with ANSI N510-1975 (except for the phase balance criteria stated in Section 14.2.3).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 135 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days. Otherwise ~~or~~ be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 135 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water,

2. 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.

3. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 800 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1025 + 20, -80 psig.*

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[b]	> 2.07 volts
Specific Gravity[a]	≥ 1.195	≥ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190

(a) Corrected for electrolyte temperature and level.
(b) May be corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

Unless adequate shutdown margin has been demonstrated, the shorting

c. ~~Prior to and during the time any control rod is withdrawn and shutdown margin demonstrations are in progress, either:~~

links shall be removed from the RPS circuitry prior to and during

~~1. The "shorting links" removed from the RPS circuitry, or~~

the time any control rod is withdrawn.#

~~2. The rod pattern control system OPERABLE per Specification 3.1.4.2.~~

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS** and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

---

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Except movement of IRM, SRM or special movable detectors.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to the start of CORE ALTERATIONS, and

   2. At least once per 7 days.

c. Verifying that the channel count rate is at least ~~2~~ 0.7 cps: 251

   1. Prior to control rod withdrawal,

   2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours,

   except that:

   1. During spiral unloading, the required count rate may be permitted to be less than ~~2~~ 0.7 cps. 251

   2. Prior to and during spiral loading, until sufficient fuel has been loaded to maintain at least ~~2~~ 0.7 cps, the required count rate may be achieved by: 251

      a) Use of portable external source, or

      b) Loading up to 2 fuel assemblies### in cells containing inserted control rods around an SRM.

d. Verifying that the RPS circuitry "shorting links" have been removed ~~or that the rod pattern control system is OPERABLE~~ within 8 hours prior to and at least once per 12 hours during:

   1. The time any control rod is withdrawn,## or

   2. Shutdown margin demonstrations.

##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

###These fuel assemblies may be loaded with the SRM count-rate less than ~~2~~ 0.7 cps.

REFUELING OPERATIONS

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS. ~~except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.~~ 280

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours prior to:

1. The start of CORE ALTERATIONS.

2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. At least once per 12 hours.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2/ OR ONE CONTROL ROD WITHDRAWN UNDER CONTROL OF THE REACTOR MODE SWITCH REFUEL POSITION ONE-ROD-OUT INTERLOCK. 280

**See Special Test Exception 3.10.3.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE per Specification 3.9.2 with ~~the:~~ the RPS circuitry "shorting links" removed.

~~1. RPS circuitry "shorting links" removed, or~~

~~2. The rod patten control system OPERABLE per Specification 3.1.4.2.~~ 323

b. The rod pattern control system is OPERABLE per Specification 3.1.4.2, or

~~b.~~ c. Conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2 ~~with:~~ with the RPS circuitry "shorting links" removed.

~~1. The "shorting links" removed, or~~

~~2. The rod pattern control system OPERABLE.~~ 323

b. The rod pattern control system OPERABLE, or a

~~b.~~ ~~A~~ second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. ~~The system meets the intent of IEEE-279 for nuclear power plant protection systems~~. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted from start of motion of the stop valve or turbine control valve until the sensor relay contact supplying the input to the reactor protection system opens, i.e., 70 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 50 ms, and plant pre-operational test results.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

(*Insert next page here*)

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

INSERT FOR PAGE B 3/4 3-3

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN ~~MONITORING~~ SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown ~~monitoring~~ system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without 009 the SRMs and they can be retracted. (Insert for Bases 3/4.3.7.6)

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution 050 of the reactor core. (Insert for Bases 3/4.3.7.7)

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", Revision 1, January 1977.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

INSERT TO BASES 3/4.3.7.6, PAGE B 3/4 3-5

The SRM's are required OPERABLE in OPERATIONAL CONDITION 2 to provide for rod block capability, and are required OPERABLE in OPERATIONAL CONDITION 3 and 4 to provide monitoring capability which provides diversity of protection to the mode switch interlock.

Insert to Bases 3/4.3.7.7, Page B 3/4 3-5

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to be OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output with data obtained during the previous LPRM calibration.

3/4.3.7.11 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.7.12 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 ad 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action to mitigate the consequences of accidents that are beyond the ability of the operator to control. The LPCI mode of the RHR system is automatically initiated on a high drywell pressure signal and/or a low reactor water level, level 1, signal. The containment spray system will then actuate automatically following high drywell and high containment pressure signals. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell and containment pressure-high. A 10-minute minimum, 13-minute maximum time delay exists between initiation of LPCI and containment spray actuation. A high reactor water level, level 8, signal will actuate the feedwater system/main turbine trip system. The suppression pool makeup system is automatically initiated on a low low suppression pool water level signal with a concurrent LOCA signal or following a specified time delay after receipt of a LOCA signal. The low low suppression pool water level Trip Setpoint and Allowable Value are relative to the surface floor of the suppression pool (93' 6" above mean sea level). 083

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

Beltline Component	Weld Seam I.D. or Material Type	Heat No.-Slab No. or Heat No./Lot No.	Cu %	P (%)	Starting RT_{NDT} (°F)	Maximum** ΔRT_{NDT} (°F)	Minimum Upper Shelf (ft-lb)	Maximum EOL RT_{NDT} (°F)
Plate	SA-533 Gr.B, CL.1 SA-533 Gr.B, CL.1	C2594-2	0.04	0.012	0	+26	96 (C2594-2)	+26 *
Weld	#2 Shell Long. ~~Seam~~ SEAMS	627260/B322A27AE	0.06	0.020	-30	44	N/A	+14

Non-Beltline Component	Material Type or Weld Seam I.D.	Heat No.-Slab No. or Heat No./Lot No.	Highest Starting RT_{NDT} (°F)
Shell Ring	SA-533 Gr.B, CL.1	C2815-2, C2779-2, C2779-1, C2780-2, C2780-1, C2741-1	+10
Bottom Head Dollar Plate	SA-533 Gr.B, CL.1	A1113-1 C2630-2	0
Bottom Head Radial Plates	SA-533 Gr.B, CL.1	C2539-2, A1145-1	+10
Top Head Dollar Plate	SA-533 Gr.B, CL.1	C2440-3	-30
Top Head Side Plates	SA-533 Gr.B, CL.1	C2944-1	+10
Top Head Flange	SA-508 CL.2	4801682	-30
Vessel Flange	SA-508 CL.2	4801141	-30
Feedwater Nozzle	SA-508 CL.2	Forging No. 249A-1, 2, 3, 4, 5, & 6, Q2Q65W	-20
Weld	N/A	N/A	-20***
Closure Stud	SA-540 Gr.B24	84025, 84299	+10

*Combination of the highest starting RT_{NDT} plate and the highest ΔRT_{NDT} plate.

**These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

***Based on purchase spec. requirements.

074

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.~~ INSERT 004

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.

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.

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 3/4.6.3, Page B 3/4 6-5

During refueling, an inadvertent dump would create a radiation hazard due to a loss of shield water if irradiated fuel were in an elevated position. However, GGNS procedures require that both logic trains of the suppression pool makeup dump valves be bypassed whenever the reactor mode switch is in the REFUEL position, thus preventing the valves from opening by either automatic or manual initiation.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Superintendent or, during his absence from the Control Room, a designated individual shall be responsible for the Control Room command function. A management directive to this effect, signed by the Senior Vice President, Nuclear shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include the Shift Superintendent, the STA, the two other members of the minimum shift crew necessary for safe shutdown of the unit, and any personnel required for other essential functions during a fire emergency. At least one AO shall be available to respond to non-fire fighting commands from the control room.

*The number of health physics technicians and Fire Brigade personnel
~~*The health physics technician and Fire Brigade composition~~ may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

Amendment No. ——

ATTACHMENT 3

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: LICENSEE QUALIFICATION

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

TSPS No.	Item Nos.*
052	3.C.1
063	3.C.2
095	3.C.3
096	3.C.4
101	3.C.5
106	3.C.6
289	3.A.2
290	3.A.1

*Item number format: 1.A.02

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

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

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

	TSPS No.	TS Page No.
1.	290	6-10

CLARIFICATIONS

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

2. (TSPS 289), Neutron Monitor/Detector Replacement, Technical Specification 6.2.2.d

   The proposed change is to add a # footnote to the subject specification to require direct supervision by the maintenance foreman and indirect supervision by a Senior Reactor Operator during neutron monitor replacement from under the Reactor Pressure Vessel. The current wording of the specification could be misinterpreted to require direct supervision by a Senior Reactor Operator during neutron monitor replacement as this may be considered a CORE ALTERATION. This is an unnecessary restriction, as neutron monitor cannot result in an inadvertent criticality or fuel damage, but is an ALARA consideration. This change represents a clarification that is within the philosophy and intent of the technical specifications. (Page 6-1)

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

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

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 052), Administrative Controls, Technical Specifications 6.1, 6.2, 6.5, 6.6 and 6.7

   This proposed change makes a revision to Specification 6.5.3.1.a of the Technical Review and Control Section by adding a requirement that all temporary changes be reviewed by the reviewing authority within 14 days of implementation. This proposed change is an enhancement to safety because it includes additional requirements not contained in the current specification. (Page 6-12)

   The proposed revision also changes "Plant Manager" to "GGNS General Manager" and "Senior Vice President - Nuclear" to "Senior Vice President, Nuclear." These changes have been formally submitted to H. R. Denton from J. B. Richard in a letter dated May 24, 1984 (AECM-84/0283). No marked-up technical specification pages are included with this submittal. (Pages 6-1 through 6-4, 6-6 through 6-9, 6-12 and 6-13)

2. (TSPS 063), Composition of Independent Safety Engineering Group (ISEG), Technical Specifications 6.2.3.4 and 6.3.1

   The proposed change is to revise the subject Specification to indicate that the qualification requirements for the ISEG apply only to those members who are used for meeting the minimum complement specified in Specification 6.2.3. The current wording could be misinterpreted to imply that all members of ISEG, including those not required to meet the minimum complement, must meet the qualifications. This is an unnecessary restriction. A revision to Specification 6.2.3.4 to change the title "Assistant Vice President for Nuclear Production" to

"Senior Vice President, Nuclear" is also proposed and has been formally submitted to H. R. Denton from J. B. Richard is a letter dated May 24, 1984 (AECM-84/0283). These changes are enh cements that are consistent with the philosophy and intent of the technical specifications. (Page 6-6)

3. (TSPS 095), Offsite Organization, Technical Specification Figure 6.2.1-1

   This proposed change revises the Offsite Organization Figure 6.2.1-1. This revision has been formally submitted to H. R. Denton from J. B. Richard in a letter dated May 24, 1984 (AECM-84/0283). No marked-up technical specification pages are included with this submittal. (Page 6-3)

4. (TSPS 096), SRC Duties, Technical Specification 6.5.2.7

   It is proposed to change the subject specification from "The SRC shall review:" to "The SRC shall be responsible for the review of:" for clarification. It is further proposed that the number of consultants participating as voting members at any one time during SRC meetings be limited to no more than three. This change will ensure that the primary decisional responsibility of the non-consultant members of the SRC are not diluted excessively. Finally, it is proposed to add a section j. to the specification requiring review of written reports from audits of the ALARA program. This change will make the technical specification consistent with GGNS FSAR Section 12.1.1.2. The changes are enhancements that are consistent with the philosophy and intent of the technical specifications. (Page 6-10)

5. (TSPS 101), Unit Organization, Technical Specification Figure 6.2.2-1

   The proposed change revises the Unit Organization Figure 6.2.2-1. The revision has been formally submitted to H. R. Denton from J. B. Richard in a letter dated May 24, 1984 (AECM-84/0283). No marked-up technical specification pages are included with this submittal. (Page 6-4)

6. (TSPS 106), Change in Plant Safety and Review Committee (PSRC) Membership and Miscellaneous Title Changes, Technical Specification 6.5

   This proposed revision changes "Minimum Quorum of the PSRC" to "Quorum of the PSRC." The words minimum and quorum are redundant terms and the deletion of minimum is for clarification only. (Page 6-7 and 6-10)

   This proposed revision also changes "Assistant Plant Manager" to "Plant Operations Manager," "Nuclear Support Manager" to "Plant Maintenance Manager," "Chemistry/and Radiation Protection Superintendent" to "Chemistry/Radiation Control Superintendent," "Maintenance Superintendent" to "I & C Superintendent," "Nuclear Safety Review Committee" to "Safety Review Committee," "Vice President - Nuclear" to "Vice President, Nuclear Operations," "Senior Vice

President - Nuclear" to Senior Vice President, Nuclear," "Manager of Quality Assurance" to "Director, Quality Assurance," "Manager of Nuclear Services" to "Director, Nuclear Licensing and Safety," "Manager of Radiological and Environmental Services" to "Manager, Radiological and Environmental Services," "Advisor to the Vice President-Nuclear" to "Advisor to the Vice President, Nuclear Operations," and adds an additional member to the PSRC, the Technical Engineering Supervisor. This revision has been formally submitted to H. R. Denton from J. B. Richard in a letter dated May 24, 1984 (AECM-84/0283). No marked-up technical specification pages are included with this submittal. (Pages 6-7, 6-8, 6-9, and 6-12)

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.)

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Superintendent or, during his absence from the Control Room, a designated individual shall be responsible for the Control Room command function. A management directive to this effect, signed by the Senior Vice President, Nuclear shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS# shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. 289

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include the Shift Superintendent, the STA, the two other members of the minimum shift crew necessary for safe shutdown of the unit, and any personnel required for other essential functions during a fire emergency. At least one AO shall be available to respond to non-fire fighting commands from the control room.

*The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

INSERT 289

INSERT TO SPECIFICATION 6.2.2.e # FOOTNOTE, PAGE 6-1

Except neutron monitor replacement from under the reactor pressure vessel which will be observed and directly supervised by the maintenance foreman in charge of the work and indirectly supervised by a Senior Reactor Operator or a Senior Reactor Operator limited to the Control Room.

INDEPENDENT SAFETY ENGINEERING GROUP (ISEG) (Continued)

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the Assistant Vice President for Nuclear Production.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide technical support to the Shift Superintendent in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Chemistry and Radiation Protection Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; the shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure I of the October 30, 1979 NRC letter to all operating nuclear power plants; and ~~the~~ members of the Independent Safety Engineering Group, each of whom shall have a Bachelor of Science degree or be registered as a Professional Engineer and shall have at least two years experience in their field, at least one year of which experience shall be in the nuclear field. 063

those used for meeting the minimum complement specified in Section 6.2.3.2

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Superintendent, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

*Not responsible for sign-off function.

PLANT SAFETY REVIEW COMMITTEE (PSRC) (Continued)

COMPOSITION

6.5.1.2 The PSRC shall be composed of the:

Chairman:	Assistant Plant Manager
Vice Chairman:	Nuclear Support Manager
Member:	Operations Superintendent
Member:	Technical Support Superintendent
Member:	Quality Superintendent
Member:	Chemistry and Radiation Protection Superintendent
Member:	Maintenance Superintendent

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the Plant Manager to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or Vice Chairman.

QUORUM

6.5.1.5 The ~~minimum~~ quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and four members including alternates. 106

RESPONSIBILITIES

6.5.1.6 The PSRC shall be responsible for review of:

a. Station administrative procedures and changes thereto.

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question and all programs required by Specification 6.8 and changes thereto.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

CONSULTANTS

6.5.2.4 Consultants, in addition to those required in Specification ~~6.5.2.3,~~ 6.5.2.2 shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC. *No more than three consultants shall participate as voting members in SRC activities at any one time.* 096

MEETING FREQUENCY

6.5.2.5 The SRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The ~~minimum~~ quorum of the SRC necessary for the performance of the SRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 6 SRC voting members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit. 106

REVIEW

6.5.2.7 The SRC shall ~~review~~ *be responsible for the review of:* 096

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the PSRC. 096

j. *Written reports from audits of the ALARA program.*

AUTHORITY

6.5.2.9 The SRC shall report to and advise the Senior Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures shall be approved as delineated in writing by the Plant Manager. The Plant Manager shall approve administrative procedures, security implementing procedures and emergency plant implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures may be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

Insert

052

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager.

INSERT TO SPECIFICATION 6.5.3.1.a, PAGE 6-12

Temporary changes shall be reviewed by the reviewing authority within 14 days of being issued.

ATTACHMENT 4

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: MECHANICAL ENGINEERING BRANCH

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

TSPS No.	Item Nos.*
123	4.C.1

*Item number format: 1.A.02

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

MTS17rg2

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

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

C. ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES

The following proposed change is an enhancement which is consistent with the safety analyses and the licensing basis and which provides clarification, renders areas consistent with the philosophy and intent of the technical specifications, or provides additional plant operational margin.

Since this proposed change is included in the current licensing bases and is bounded by existing safety analyses, 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 123), Reactor Vessel Steam Coolant Temperature Measurement, Technical Specification 3.4.1.4

   Technical Specification 3.4.1.4 requires that the temperature differential between the reactor vessel steam space coolant and the bottom head drain line coolant should not exceed 100°F when an idle recirculation loop is started.

   The temperature of the vessel steam space coolant is obtained from a pressure measurement signal which is fed into the process computer. The computer converts this signal into a temperature value by reference to a standard pressure/temperature saturation curve. This overall method of measurement thus covers the range from operating pressure to atmospheric boiling conditions with an accuracy of ±6°F. It follows that temperatures of steam space coolant below 212°F are not available.

   By means of the available temperature indication it is not possible to measure the differential temperature of technical specification 3.4.1.4 properly when the reactor is at ambient pressure. A steam space coolant temperature of 212°F will be indicated when in fact the true temperature may be much lower. Thus, the present Technical Specification places unnecessary restrictions on recirculation loop operation. The proposed change will make this technical specification applicable only when the vessel pressure is greater than 25 psig.

   An analysis of Grand Gulf Nuclear Station's Unit 1 Reactor Vessel and components for fatigue usage, performed by GE in accordance with Subsections NA, NB and Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, Summer '76 Addenda and later editions (see GGNS FSAR Section 5.3.3), has conservatively predicted a cumulative

fatigue usage factor of 0.5509 at the end of the Unit's 40 year service life. (End of useful life of a vessel corresponds to a fatigue usage factor of 1.0.)

This analysis considered the possibilities of 100°F step changes in the vessel bottom head temperature for all predicted reactor startup-shutdown and scram cycles, 20 cold liquid injections, and one 348°F step change during preoperational testing. Included in this analysis are 120-100°F step changes that are predicted to occur during normal startup, at operational temperature and pressure, when the recirculation pumps are switched to high speed. This could occur when the coolant in the bottom head is at a conservatively predicted 444°F, and increasing the recirculation pump speed could sweep 544°F coolant into the previously cooler region. No step changes are included in the analysis for normal shutdowns, as decreasing the pump speed results in gradual thermal changes. None are included for initial pump starts at atmospheric pressure, as these occur when the reactor is in a cold condition and at nearly thermal equilibrium, resulting in possible step changes of significantly less than 100°F that don't affect the fatigue usage factor. The predicted reactor vessel transients are in Section 3.9.1.1.1.7 and Table 3.9-1 of the GGNS FSAR.

The 100°F differential temperature of Technical Specification 3.4.1.4 was established on the basis of the combined stress due to the full rated operating pressure of 1050 psig plus the thermal stress caused by the 100°F temperature difference. At 25 psig the stress due to pressure is negligible. At 25 psig in the reactor the pressure instrumentation is sufficiently on scale for the operator to read and provide an accurate input to the computer for conversion to temperature.

The analysis did not initially include the possibility of any 187°F step changes at 25 psig for the liquid control nozzle, which was found to be the most limiting bottom head component. It would be possible for a 187°F step change to occur to the liquid control nozzle at 25 psig if the following highly unlikely sequence of events were to occur:

1) Recirculation pumps removed from service
2) Reactor vessel held stable at 25 psig for an extended period of time
3) The drywell environment cooled to a low enough temperature to allow the bottom head and the moderator therein to cool to 80°F
4) The reactor vessel dome is at saturated condition
5) The establishment of all the above unusual conditions and subsequent start of an idle recirculation pump

Further analysis by GE reveals that the fatigue usage evaluation allows sufficient margin for at least 130 events of this type to occur during the service life of the vessel without exceeding a fatigue usage factor of 1.0. This equates to an increase in the fatigue usage factor of 0.00345, as has been calculated by GE, for each time that this series of unlikely events were to occur. It is anticipated that this postulated event will never occur during the service life of GGNS.

The recirculation pumps are normally started at atmospheric vessel pressure prior to reactor startup, and both pumps are established in service prior to entering the startup mode. On reactor shutdown, the recirculation pumps are normally retained in service for coolant mixing purposes until well after the vessel is depressurized to atmospheric.

This proposed change doesn't affect the predicted fatigue usage factor of 0.5509, since the procedures for starting and operating the recirculation system are not affected, but provides additional operational margin. This proposed change is consistent with the philosophy and intent of the technical specifications. (Page 3/4 4-4)

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.)

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F*, and: 123

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

*Below 25 psig the temperature differential is not applicable. 123

ATTACHMENT 5

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: REACTOR SYSTEMS

MTS52rgl

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

TSPS No.	Item Nos.*
024	5.C.1
032	5.C.7
041	5.C.2
110	5.D.1
162	5.A.1
236	5.C.5
243	5.C.6
256	5.C.3
272	5.A.2
309	5.B.1
310	5.B.1
322	5.C.4

*Item number format: 1.A.02

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

MTS52rg2

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:

- 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 necessary justification for the changes is provided below:

CLARIFICATIONS

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

1. (TSPS 162), Emergency Core Cooling Systems, Technical Specification 3/4.5.1

   This proposed change adds ** footnote which is applicable to ACTION statement a.4 of Technical Specification 3.5.1. The change clarifies the ACTION to be taken whenever two or more RHR subsystems are inoperable. This footnote is identical to the * footnote which is applicable to ACTION statements b.3 and d.3 of Technical Specification 3.5.1. This change is purely administrative and is made to achieve consistency among ACTION statements 3.5.1.a.4, 3.5.1.b.3, and 3.5.1.d.3. (Page 3/4 5-1)

2. (TSPS 272), Alternate Shutdown Cooling, Technical Specification 3/4.4.9.1

   The proposed change incorporates into ACTION Statement 2 of Technical Specification 3.4.9.1 and into Surveillance Requirement 4.4.9.1 the provision for substituting one operating recirculation pump for one operating shutdown cooling mode loop of RHR as is allowed by the subject Limiting Condition for Operation (LCO). These revisions clarify that one operating recirculation pump is an acceptable alternative to having one RHR shutdown cooling mode loop in operation for satisfying the LCO. The proposed change does not adversely impact plant safety as it represents a clarification to achieve consistency within the subject specification. (Page 3/4 4-24)

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 309 and 310), LPCI and LPCS High Pressure Alarm Setpoints, Technical Specification 3/4.5.1

   A revision is proposed to change the high pressure setpoint of the LPCS system from 580 +20, -0 psig to less than or equal to 600 psig. This proposed revision also changes the high pressure setpoint of the LPCI subsystems from 480 +20, -0 psig to less than or equal to 493 psig and clarifies Surveillance Requirement 4.5.1.c.2.a to indicate that both high and low pressure alarms exist for the subject systems. The change to the LPCS setpoint is consistent with the piping design pressure of 600 psig and is conservative with respect to the piping maximum working pressure of 758 psig. This setpoint will allow the high pressure alarm to annunciate prior to activation of the system relief valve which has a setpoint of 585 psig and a water leg relative to the lowest point in the piping of 17 psi, resulting in a maximum pressure of 602 psig at the lowest point when the relief valve lifts. The change to the LPCI subsystems setpoint is conservative, as it decreases the allowable setpoint from that currently prescribed and is less than the piping design piping pressure of 500 psig and the piping maximum working pressure of 758 psig. This setpoint also allows the high pressure alarm to annnunciate prior to activation of the LPCI relief valves which have setpoints of 500 psig and, in the most limiting case, a waterleg relative to the lowest point in the LPCI loop C piping of 11 psi, resulting in a maximum pressure of 511 psig at the lowest point when the relief valve actuates in loop C. The deletion of the "-0" from each requirement is also conservative, as no minimum setpoint is required to insure the alarms function. (Page 3/4 5-4)

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 024), Jet Pump OPERABILITY, Technical Specification 3/4.4.1.2

A revision is proposed to the subject specification to allow present Surveillance Requirement 4.4.1.2 (Renumbered to 4.4.1.2.1) to be performed with THERMAL POWER in excess of 25% of RATED THERMAL POWER instead of the present prior to exceeding 25% of RATED THERMAL POWER. A new Surveillance Requirement (4.4.1.2.2) is added to provide a 4.0.4 exemption, provided the diffuser-to-lower plenum differential pressure of the individual jet pumps are determined to be within 50% of the loop average within 72 hours after entering OPERATIONAL CONDITION 2 and at least once per 24 hours thereafter. The value of 50% of the loop average is qualified, by addition of a "*" footnote, to explain that this is an initial value and that the final value will be determined during the startup test program, with any required change submitted to the Commission within 90 days of test completion.

Justification for not performing the present surveillance requirements below 25% of RATED THERMAL POWER is provided as follows:

1. The current Grand Gulf Unit 1 Technical Specification on Jet Pump OPERABILITY, Section 3/4.4.1.2, requires that all jet pumps be OPERABLE in OPERATIONAL CONDITIONS 1 and 2. The surveillance requirements specify that the jet pumps shall be demonstrated OPERABLE prior to exceeding 25% RATED THERMAL POWER and once per 24 hours. The implication is that in Mode 2, OPERABILITY must be demonstrated every 24 hours.

2. These surveillance requirements are best summarized as monitoring the following 3 correlations:

a. Recirculation loop flow to flow control valve position,

b. Total core flow to recirculation loop flow, and

c. Jet pump dp distribution.

Addressing each of the three surveillance requirements individually, the following information is offered in order to describe effects that will invalidate these tests at low power.

a. Recirculation Loop Flow to Flow Control Valve (FCV) Position Correlation

For any given FCV position, the flow established through the loop is dependent on pump discharge pressure, valve flow coefficient, and coolant density subcooling, with density having some noticeable effects. Natural circulation and core dp effects are minimal; however moderate. The degree of subcooling is regulated by the amount of downcomer heating resulting from steam carry under from the steam dryer. Considering the numerous parameters which regulate this, a baseline correlation would require a wide tolerance band at low power levels.

b. Total Core Flow to Recirculation Loop Flow Correlation.

This correlation is most affected by natural circulation and power level. On the power to flow map (Bases Figure B 3/4 2.3-1), Line C represents a nearly constant recirculation loop flow. The core flow can vary from 25% to 40% along the line. In addition, it must be noted that this power/flow map is representative of nominal conditions. Changes in core dp due to voiding and rod pattern will shift the lines slightly, thereby adding additional uncertainties to the correlation.

c. Jet Pump dp Distribution Correlation

Again, in this correlation the relationship between natural circulation and forced circulation plays an important role. Each jet pump loop will have a unique flow distribution, which is a direct function of the recirculation pump flow rate. However, as power is increased from 0% to 25% the rate of natural circulation to forced circulation is increased. This results in drawing more water through the jet pumps and biasing the established jet pump flow distributions by some unknown percentage. In addition, the subcooling related moderator density changes further complicate the establishment of a valid baseline at low power levels.

3. At power levels below 25% RATED THERMAL POWER, several BWR operating characteristics play important roles in these correlations. The effects of these phenomena render the correlations difficult to quantify, and thus negate the ability to perform a valid, repeatable surveillance test. The three most significant factors are natural circulation, moderator subcooling and core dp, each of which is dependent on reactor power level, control rod pattern core exposure and voiding. Very briefly, the effects of natural circulation tend to bias established flow correlations, moderator subcooling causes changes in moderator densities which directly affect flow rates, and varying core dp changes flow rates by varying the in-core flow resistance.

   To aid in this discussion, refer to Bases Figure B3/4 2.3-1 "Power Flow Operating Map" on page B 3/4 2-5 . Below 30% power the recirculation pumps operate on the Low Frequency Motor Generator sets (LFMG's) at approximately 25% speed. Line C of the Power to Flow map (low speed-FCV maximum position) represents the usual operating configuration while on the LFMG's. From the curve it can be seen that from 0% power to the 100% Rod Line the slope of Line C closely follows the slope of the natural circulation line (Line A). This is to say that, while on the LFMG sets the predominate factor in establishing core flow is natural circulation. This component of core flow will tend to bias only the "established" flow correlation.

   At low core flows and low reactor power levels, the various flow correlations are significantly dependent on numerous secondary parameters, many of which are not distinctly quantifiable or repeatable. The application of flow correlation criteria becomes extremely difficult and would require a very large tolerance band.

   The same arguments can be applied to a lesser degree, above 25% RATED THERMAL POWER when on LFMG's. Any time the recirculation pumps are operating on low speed, natural circulation effects will compound efforts to establish and monitor an accurate baseline. This is true up to the 100% rod line.

4. The most accurate and repeatable measurements will only be available while the recirculation pumps are on high speed (the shift to high speed occurs at approximately 30% RATED THERMAL POWER), and the ratio between natural circulation and forced circulation is minimized.

The proposed Surveillance Requirement 4.4.1.2.2 will allow entry into OPERATIONAL CONDITIONS 1 and 2 without having to perform present Surveillance Requirement 4.4.1.2; however, jet pump OPERABILITY is required to be determined within 72 hours after entering OPERATIONAL CONDITION 2 and at least once per 24 hours thereafter. This jet pump OPERABILITY determination is based on the diffuser-to-lower plenum differential pressures of the individual jet pumps being within 50% of the loop average. The 50% criteria is based on information provided by General Electric in Service Information Letter (SIL) Number 330, dated June 9, 1980. SIL No. 330 discusses jet pump beam cracks and jet pump displacement which have

occurred at three plants. With a displaced jet pump, flow deviation from average was 70%. This flow deviation is determined by differential pressure measurement which is the criteria proposed by this change. The 50% criteria will provide gross failure information while allowing sufficient operating margin, considering the various uncertainties discussed above. The 50% criteria will be verified during the startup test program and any required changes will be submitted to the Commission within 90 days of test completion.

The proposed changes represent a relaxation of the present surveillance requirements but are within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. These changes are necessary to reflect actual system operating characteristics and to more accurately reflect the system design parameters. (Page 3/4 4-2)

2. (TSPS 041), Recirculation Loop Surveillance Requirements, Technical Specification 3/4.4.1.1 and Bases 3/4.4.1

   The proposed changes add Surveillance Requirement 4.4.1.1.1 to verify once every 24 hours that both recirculation loops are in operation while in Operational Conditions 1 and 2 and also add a description of the operation of the recirculation loop flow control valves to Bases 3/4.4.1. The proposed revision to the Bases provides additional detail regarding operation of the recirculation flow control valves. The proposed addition of Surveillance Requirement 4.4.1.1.1 is an enhancement to safety in that it constitutes an additional surveillance requirement which is not presently in the technical specifications. Therefore, these chagnes are considered an enhancement to the present technical specification which is consistent with the safety analysis. (Pages 3/4 4-1 and B 3/4 4-1)

3. (TSPS 256), High Pressure Core Spray, Technical Specification Bases 3/4.5.1 and 3/4.5.2

   A revision to the bases for the ECCS technical specifications is proposed to achieve consistency among the FSAR, design documentation and the technical specifications. The stated system operating differential pressure range and HPCS pump capacity values should be revised to be consistent with the information provided in FSAR Chapter 6.3.1, design documents and Technical Specifications 3/4.5.1 and 3/4.5.2. Minor editorial corrections are also proposed to improve the readability of the bases section. The proposed changes do not adversely impact plant safety because they represent an enhancement to the bases which is consistent with the safety analyses and licensing basis. (Pages B 3/4 5-1 and B 3/4 5-2)

4. (TSPS 322), Low Pressure Injection used in Automatic Depressurization System Bases, Technical Specification Bases 3/4.5.1, 3/4 5.2, and 3/4.7.3

   The proposed change revises the stated operating pressures and flow rates of the low pressure core spray (LPCS) and low pressure coolant injection (LPCI) systems. The bases presently state that the low

pressure cooling systems provide adequate core cooling up to a reactor pressure vessel (RPV) pressure of 350 psig. The LPCS and LPCI systems actually provide incipient flow into the RPV at 295 psid and 229 psid, respectively. This change does not adversely impact plant safety because it provides an enhancement that more accurately reflects the plant design, is consistent with design documents and is consistent with the safety analysis and licensing basis. (Page B 3/4 5-2 and B 3/4 7-1)

5. (TSPS 236), Jet Pump OPERABILITY, Technical Specification 3/4.4.1.2

   This proposed change revises Surveillance Requirement 4.4.1.2 by deleting the requirement that the recirculation flow control valves be in the same position when performing the surveillance. The revised requirement specifies that the surveillance is to be performed with the recirculation loop flows within the mismatch limits of Technical Specification 3.4.1.3. This change will provide an operational enhancement in that the surveillance may be conducted with the recirculation flow control in automatic. Further, this change eliminates the problem of unequal flows which could occur, with equal valve positions, due to instrument drift, pump performance changes, resistance differences, etc. This change does not adversely impact plant safety because the specification remains consistent with the safety analysis and with the philosophy and intent of the technical specification. (Page 3/4 4-2)

6. (TSPS 243), Main Steam Isolation Valve (MSIV) Stroke Time Definition, Technical Specification 3/4.4.7

   The proposed change more clearly defines the methodology used when measuring the closure times of the MSIV's. A clear distinction is lacking between ASME Section XI and the design criteria for both minimum and maximum full stroke times for the MSIV's. Section XI requires that valve closure times be measured from initiation of the actuation signal to valve fully closed. This methodology is appropriate for measuring the maximum MSIV closure time but is inappropriate for measuring the minimum MSIV closure time. The minimum MSIV closure time must not be less than 3 seconds from start of valve movement to valve closure in order to assure that the reactor vessel pressurization transient analyzed in the GGNS safety analyses is not exceeded. ASME Section XI, IWV 3413 requirements are not applicable to this minimum stroke time. This change distinguishes the requirements of the NSSS vendor design specifications from the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. This change is an enhancement that does not adversely impact plant safety because the revised specification more accurately reflects the plant design requirements for surveillance of the MSIV's. (Page 3/4 4-22)

7. (TSPS 032), Reactor Coolant System (RCS) Leakage, Technical Specification 3/4.4.3.2 and Bases B 3/4.4.3.1

   The proposed change is to revise ACTION d. of the subject specification and Surveillance Requirement 4.4.3.2.3 to reflect that high/low pressure interface valve interlocks as well as alarms are present in the RCS. It is further proposed to delete the current Table 3.4.3.2-2

and insert new Tables 3.4.3.2-2 and 3.4.3.2-3 to incorporate setpoints for the interlocks and for alarms not presently listed. This change constitutes an additional requirement not presently in the technical specifications. The term "Leakage" is deleted from the description of the monitors in Table 3.4.3.2-2 entitled "Reactor Coolant System Interface Valves Pressure Leakage Monitors," and in ACTION d. ACTION d. is further revised to delete the requirement to restore OPERABILITY of the inoperable monitors within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours. These monitors are not capable of measuring valve leakage. Their purpose is to prevent opening of valves where the high pressure piping is pressurized above the design pressure of the low pressure piping by alerting the operator to vent the piping prior to valve opening for "trapped" pressure conditions and to prevent operating systems above their design pressure via interlocks and annunciations. Therefore, the term "Leakage" in the table titles and ACTION d. is inappropriate.

Furthermore, it is proposed to revise Bases B 3/4.4.3.1 to reflect that the RCS Leakage Detection Systems also measure leakage from fluid systems in the drywell. These proposed changes are an enhancement to the technical specifications that makes them consistent with system design. (Page 3/4 4-8, 3/4 4-9, 3/4 4-10, and B 3/4 4-2)

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:

- 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 110), Reactor Core Isolation Cooling (RCIC) Steam Line Flow-High, Technical Specification 3/4.3.2

   This proposed change adds a time delay with a Trip Setpoint of 5 seconds to Item 5.a of Tables 3.3.2-1 and Item 5.a of 3.3.2-2 and an Allowable Value of 5 ± 2 seconds to Item 5.a of Table 3.2.2-2. This change also adds surveillance requirements for the subject time delay to Item 5.a of Table 4.3.2.1-1. The time delay is required to prevent inadvertent isolation of the RCIC system during system initiation. This change adds to and supercedes a previously submitted change (Item 10 in letter AECM-83/0565, dated September 9, 1983 which was withdrawn by letter AECM-84/0303, dated May 25, 1984) which revised a ### footnote in Table 3.3.2-3, page 3/4 3-19, to read "Includes time delay of 3 to 7 seconds." The proposed change is in response to NRC concerns and is in compliance with Item 9 of Generic Letter 83-02 which provided guidelines for technical specification changes required by NUREG-0737. (Pages 3/4 3-12, 3/4 3-17, 3/4 3-19 and 3/4 3-22)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8(j)	1	1, 2, 5##	30
i. Manual Initiation	8	2	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	~~4~~	~~1~~	~~1, 2, 3~~	~~27~~
b. RCIC Steam Supply Pressure - Low	4, 9(m)	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	4	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temp. - High	4	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	4	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	4	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	4	1	1, 2, 3	27
1) Pressure	4	1	1, 2, 3	27
2) Time Delay	4	1	1, 2, 3	27

110

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)			
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches	
f. Main Steam Line Tunnel Ambient Temperature - High	≤ 185°F**	≤ 191°F**	
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**	
h. SLCS Initiation	NA	NA	
i. Manual Initiation	NA	NA	
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION			
a. RCIC Steam Line Flow - High	~~≤ 363" H_2O~~	~~≤ 371" H_2O~~	110
1) Pressure	≤ 363" H_2O	≤ 371" H_2O	110
2) Time Delay	5±2 seconds	5 ± 2 seconds	
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 53 psig	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig	
d. RCIC Equipment Room Ambient Temperature - High	≤ 185°F**	≤ 191°F**	
e. RCIC Equipment Room Δ Temp. - High	≤ 125°F**	≤ 128°F**	
f. Main Steam Line Tunnel Ambient Temperature - High	≤ 185°F**	≤ 191°F**	
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**	
h. Main Steam Line Tunnel Temperature Timer	≤ 30 minutes	≤ 30 minutes	
i. RHR Equipment Room Ambient Temperature - High	≤ 165°F**	≤ 171°F**	
j. RHR Equipment Room Δ Temperature - High	≤ 99°F**	≤ 102°F**	
k. RHR/RCIC Steam Line Flow - High	≤ 145" H_2O	≤ 160" H_2O	

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	≤ 13[a]###
b. RCIC Steam Supply Pressure - Low	≤ 13[a]
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
e. RCIC Equipment Room Δ Temp. - High	NA
f. Main Steam Line Tunnel Ambient Temp. - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temp. - High	NA
k. RHR/RCIC Steam Line Flow - High	NA
l. Manual Initiation	NA
m. Drywell Pressure - High (ECCS Division 1 and Division 2)	≤ 13[a]
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Room Ambient Temperature - High	NA
b. RHR Equipment Room Δ Temp. - High	NA
c. Reactor Vessel Water Level - Low, Level 3	≤ 13[a]
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
e. Drywell Pressure - High	NA
f. Manual Initiation	NA

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

###~~Without 13 second time delay.~~
Includes time delay of 3 to 7 seconds.

110

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	A	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	A	1, 2, 3
h. SLCS Initiation	NA	M(b)	NA	1, 2, 5##
i. Manual Initiation	NA	M(a)	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	~~S~~	~~M~~	~~R(c)~~	~~1, 2, 3~~
b. RCIC Steam Supply Pressure - Low	S	M	R(c)	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R(c)	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	M	A	1, 2, 3
e. RCIC Equipment Room Δ Temp. - High	S	M	A	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	A	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	A	1, 2, 3
1) Pressure	S	M	R(c)	1, 2, 3
2) Time Delay	S	M	R	1, 2, 3

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.2 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by: 140

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic unit, and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

*See Special Test Exception 3.10.4.

4.4.1.1.1 Both reactor coolant system recirculation loops shall be verified to be in operation at least once per 24 hours. 165

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE with ~~prior to~~ THERMAL POWER in excess of ~~exceeding~~ 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when ~~the recirculation loops are operating at the same flow control valve position.~~ both indicated recirculation loop flows are in compliance with Specification 3.4.1.3. 020 236

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

4.4.1.2.2 The provisions of Specification 4.0.4 are not applicable provided the diffuser-to-lower plenum differential pressures of the individual jet pumps are determined to be within 50%* of the loop average within 72 hours after entering OPERATIONAL CONDITION 2 and at least once per 24 hours thereafter. 020

*Initial value, Final value to be determined during startup test program. Any required changes to this value shall be submitted to the Commission within 90 days of test completion. 020

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 30 gpm total leakage.

d. 1 gpm leakage at a reactor coolant system pressure of 1050 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one and/or interlocks or more high/low pressure interface valve ~~leakage~~ pressure monitors inoperable, ~~restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours,~~ restore the inoperable monitor(s) and/or interlock(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. 032

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 4 hour.

b. Monitoring the drywell floor and equipment drain sump level and flow rate at least once per 4 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 4 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

In addition, until the LPCS system and the RHR system injection valve reactor coolant pressure-low permissive is modified during or before the first refueling outage, the LPCS system check valve 1E21-F006 and the RHR system check valves 1E12-F041 A, B, and C shall also be demonstrated OPERABLE by verifying leakage to be within its limit:

1. Whenever the unit has been in COLD SHUTDOWN or REFUELING, after the last valve disturbance prior to reactor coolant system temperature exceeding 200°F.

2. Within 24 hours following valve disturbance except when in COLD SHUTDOWN or REFUELING.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valves leakage pressure monitors shall be demonstrated OPERABLE with alarm and interlock setpoints per Table 3.4.3.2-2 and Table 3.4.3.2-3 by performance of a:

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

b. CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
E21-F005	LPCS
E21-F006	
E22-F004	HPCS
E22-F005	
E12-F008	RHR
E12-F009	
E12-F023	
E12-F041	
E12-F042	
E12-F050	
E12-F053	
E12-F308	
E51-F063	RCIC
E51-F064	
E51-F065	
E51-F066	
E51-F076	
E51-F013	

(Insert)

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E21-F005 to E21-F006	LPCS	≤ 50
E12-F008 to E12-F006	RHR	≤135
E12-F041 to E12-F042	RHR	≤ 50
E12-F052 to F51-F064	RCIC	≤480

032

INSERT TO TABLE 3.4.3.2-1, PAGE 3/4 4-10

TABLE 3.4.3.2-2

Reactor Coolant System Interface Valves
Pressure Monitors - Alarm

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E21-F005 to E21-F006	LPCS	≤ 50
E12-F008 to E12-F006A	RHR	≤ 183
E12-F008 to E12-F006B	RHR	≤ 183
E12-F041A to E12-F042A	RHR	≤ 50
E12-F041B to E12-F042B	RHR	≤ 50
E12-F041C to E12-F042C	RHR	≤ 50

TABLE 3.4.3.2-3

Reactor Coolant System Interface Valves
Pressure Interlocks

VALVE NUMBER	SYSTEM	INTERLOCK SETPOINT (psig)
E12-F052 to E51-F064	RCIC	≤ 465
E12-F041A to E12-F042A	RHR	≤ 50
E12-F041B to E12-F042B	RHR	≤ 50
E12-F041C to E12-F042C	RHR	≤ 50
E21-F005 to E21-F006	LPCS	≤ 50

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

    1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

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

        b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds* when tested pursuant to Specification 4.0.5. 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 3 seconds is the time measured from start of valve motion to complete valve closure. The 5 seconds is the time measured from initiation of the actuating signal to complete valve closure.

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

1. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. 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 loop. Be in at least COLD SHUTDOWN within 24 hours.**

2. With no RHR shutdown cooling mode loop or recirculation pump in operation, immediately initiate corrective action to return at least one loop or one recirculation pump to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour. 272

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours. 272

One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

** Whenever two or more 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.
2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.
3. Eight OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.
2. Eight OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* # and 3*.

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.
2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.
3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours**. |162

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig. |162

#See Special Test Exception 3.10.5.

** Whenever two or more 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.

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS division 1, 2 and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Performance of a CHANNEL FUNCTIONAL TEST of the:

a) Discharge line "keep filled" pressure alarm instrumentation, and

b) Header delta P instrumentation.

3. Verifing 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. Verifing that, when tested pursuant to Specification 4.0.5, each:

1. LPCS pump develops a flow of at least 7115 gpm with a total developed head of greater than or equal to 290 psid.

2. LPCI pump develops a flow of at least 7450 gpm with a total developed head of greater than or equal to 125 psid.

3. HPCS pump develops a flow of at least 7115 gpm with a total developed head of greater than or equal to 445 psid.

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months:

1. Performing 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 injection of coolant into the reactor vessel may be excluded from this test.

2. Performing a CHANNEL CALIBRATION of the:

a) Discharge line high pressure and "keep filled" low pressure alarm instrumentation and verifying the:

1) High pressure setpoint of the:

(a) LPCS system to be ≤600 ~~500 + 20, - 0~~ psig.

(b) LPCI subsystems to be ≤493 ~~400 + 20, - 0~~ psig.

APR 1 8 1984

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F.

INSERT →

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 13 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 6 SRVs operating in the relief mode and 7 SRVs operating in the safety mode is acceptable.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 6 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

Insert to Bases 3/4.4.1, Page B 3/4 4-1

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is". This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in the opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the FSAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These systems provide the ability to measure leakage from fluid systems in the drywell.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that certain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Flow and total developed head values for surveillance testing include system losses to ensure design requirements are met. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Flow and total developed head values for surveillance testing include system losses to ensure design requirements are met. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of ~~1160~~ 1177 psid, differential pressure between the reactor vessel and the HPCS suction source, to 0 psid. 256

ECCS-OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to ~~1440/5010~~ 1650/7115 gpm at differential pressures of ~~1160/200 psi.~~ 1147/200 psid Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water. 256

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Flow and total developed head values for surveillance testing include system losses to ensure design requirements are met. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 135 psig even though ~~low pressure core cooling systems provide adequate core cooling up to 350 psig~~ LPCS has incipient flow into the reactor pressure vessel (RPV) at 295 psid and 7115 gpm rated flow at 128 psid and LPCI has incipient flow into the RPV at 229 psid and 7450 gpm rated flow at 24 psid. 322

ADS automatically controls eight selected safety-relief valves although the safety analysis only takes credit for seven valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Cumulative operation of the system for 10 hours with the heaters OPERABLE over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

The surveillance testing for verifying heat dissipation for the Control Room Emergency Filtration 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.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 135 psig even though the LPCI mode of the residual heat removal (RHR) system provides ~~adequate core cooling up to 225 psig~~ incipient flow into the RPV at 229 psid and 7450 gpm rated flow at 24 psid. 322

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 135 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCICS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

ATTACHMENT 6

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: STANDARDIZATION AND
SPECIAL PROJECTS

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

TSPS No.	Item Nos.*
006	6.D.01
053	6.D.02
057	6.D.03
061	6.A.01
093	6.D.04
097	6.A.02

*Item number format: 1.A.02

Item number within category

Category designator

Attachment number

ZZ3mm2

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:

- 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 necessary justification for the changes is provided below:

EDITORIAL CHANGES

A proposed editorial change to the technical specifications is discussed below:

1. (TSPS 061) Motor Operated Valves Thermal Overload Protection, Technical Specification 3/4.8.4.2

   The proposed change to the technical specification ACTION statement is to delete the words "take administrative action to". This is an editorial change to the technical specifications to remove an extraneous and possibly confusing instruction from the ACTION statement. The proposed change does not adversely affect plant safety since the proposed change does not alter the actions to be taken. (Page 3/4 8-38)

2. (TSPS 097), Technical Specification Index

   The proposed changes correct the index to reflect extensive revisions proposed to the technical specifications during the Grand Gulf Technical Specification Review Program. It is Mississippi Power and Light's understanding, based on discussions with the NRC, that additional page number changes may be required after compiling all proposed revisions and that the NRC will revise the index as necessary to reflect all incorporated changes to the technical specifications. There are no marked-up index pages attached with this problem sheet.

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

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

C. ENHANCEMENTS THAT ARE CONSISTENT WITH THE SAFETY ANALYSES

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

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:

- 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 006), Deletion of Snubber List, (Snubber Specification Replacement), Technical Specification 3/4.7.4

   The proposed change is a general revision to Technical Specification 3/4.7.4 and associated bases. Specification 6.10.2.1 is also revised to make it consistent with the general revision in Technical Specification 3/4.7.4. The proposed major changes to Technical Specification 3/4.7.4 are as follows:

   a. Reference to Technical Specification Tables 3.7.4-1 and 3.7.4-2 is replaced by "All hydraulic and mechanical snubbers" in the Limiting Condition for Operation. Furthermore, Tables 3.7.4-1 and 3.7.4-2 are deleted from the specification and the bases are revised to require that a list of individual snubbers with detailed information be available at the site. Provision is included in the revised bases to exclude the requirement for snubbers on nonsafety-related equipment where failure would not adversely affect safety-related equipment. These changes will make the specification consistent with the requirements of Sections 50.71(c) and 50.59 of 10 CFR Part 50 and standard industry practice. The proposed revision to Specification 6.10.2.1 deletes a reference to the subject tables.

b. The term "Inspection Type" is defined in proposed Surveillance Requirement 4.7.4.a and further described in the Bases to allow individual surveillance of snubbers according to design and manufacturer, and further enhance the ability to evaluate generic snubber performance.

c. Present Surveillance Requirement 4.7.4.a (redesignated 4.7.4.b) is changed to require the second inservice visual inspection to be performed at the next refueling outage on any system where all snubbers were found OPERABLE at the first inspection. This is roughly equivalent to the current requirement that the second inspection be performed 12 months ±25% from the date of the first inspection, if less than 2 snubbers are found inoperable, which coincides approximately with the first refueling outage. Furthermore, it is proposed to require accelerated reinspection of only the affected system when failures are detected, and to revise the * footnote to the inspection schedule to allow the inspection interval to be increased two steps for each successful inspection after the first successful inspection following correction of a generic defect. This will eliminate unnecessary testing of reliable snubbers when a known cause of failure has been corrected. Other requirements of this paragraph are unchanged.

d. The reference to "fluid port of a hydraulic snubber is found to be uncovered..." (The current revision reads "hydraulic part", but this is a typographical error) is deleted from Surveillance Requirement 4.7.4.b (redesignated 4.7.4.c) and included in the revised bases, as this is an example presented merely to clarify the intent of the visual inspection acceptance criteria. A requirement is added to this surveillance requirement to consider the OPERABILITY of snubbers common to more than one system in assessing the surveillance schedule for each of the related systems.

e. An additional requirement for Transient Event Inspection, a requirement not presently in the technical specifications, is added as Surveillance Requirement 4.7.4.d. This new requirement states that an inspection of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients will be performed within 6 months following such an event. This constitutes an additional restriction not presently included in the technical specifications.

f. Present Surveillance Requirement 4.7.4.c (redesignated 4.7.4.e) is revised to replace the individual functional test requirements for hydraulic and mechanical snubbers (which make reference to Tables 3.7.4-1 and 3.7.4-2 which this revision proposes to delete) with three separate plans for testing a representative sample of the snubbers. An additional requirement is added to notify the NRC Regional Administrator in writing of changes in the sample plan prior to their use. All three of these sample plans will insure

that a reliable representative sample of snubbers are functionally tested during each test period. These plans have already been approved by the NRC for similar application at other commercial nuclear power plants.

g. Present Surveillance Requirement 4.7.4.d and 4.7.4.e, Acceptance Criteria for the Hydraulic and Mechanical Snubber Functional Tests, are replaced by Surveillance Requirement 4.7.4.f to make the acceptance criteria consistent with the functional tests. All provisions of the existing acceptance criteria are retained in the revision.

h. Surveillance Requirement 4.7.4.g, Functional Test Failure Analysis, incorporates the analysis criteria currently stated in Surveillance Requirement 4.7.4.c. The criteria is reworded to improve clarity and to assure consistency with standard industry practice. The intent and philosophy of this part of the surveillance requirement is retained in the revision.

i. Surveillance Requirement 4.7.4.h constitutes a new requirement not presently in the technical specifications to functionally test repaired or replacement snubbers before installation in the unit. This change constitutes an additional restriction.

j. Surveillance Requirement 4.7.4.i, Snubber Service Life Program, is proposed to replace existing Surveillance Requirement 4.7.4.f, Snubber Service Life Monitoring. The revision makes this part of the surveillance requirement consistent with standard industry practice and easier to understand. The intent and philosophy of this part of the specification is retained in the revision.

k. Figure 4.7.4-1 is added to the surveillance requirement to be used when Surveillance Requirement 4.7.4.e.2 is chosen as the option for functional testing of the snubbers. This figure was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan. This reference is included in the bases.

The remainder of this proposed revision makes the subject specification easier to understand and consistent with industry standards and GGNS design without altering the intent or philosophy of the technical specifications. Requirements of Regulatory Guides 8.8 and 8.10 are incorporated in the bases. The proposed changes do not adversely affect plant safety since the revisions to the technical specification taken as a whole impose more restrictive requirements on plant operation than the current technical specifications. The proposed change is submitted in response to NRC requests. (Pages 3/4 7-9 through 3/4 7-25, B 3/4 7-2, and 6-23)

2. (TSPS 053), Reactor Vessel Water Level, Technical Specification 2.1.4

This proposed change to the ACTION statement clarifies depressurization requirements for the reactor water level safety limit. The existing statement can be misinterpreted if the "if required" phrase is considered to apply to the entire sentence. The "if required" phrase is only applicable to the depressurization requirement. The reactor vessel will only require depressurization if high pressure coolant injection systems are incapable of restoring the vessel water level. The proposed change clarifies that reactor vessel depressurization should only be required to support ECCS operation, (i.e., to support low pressure systems, if high pressure systems are ineffective in restoring water level). The proposed change does not adversely affect plant safety since the proposed change represents a clarification which is consistent with the assumptions used in the safety analysis. (Page 2-2)

3. (TSPS 057), 10 CFR Appendix J Leakrate Testing, Technical Specification 3/4.6.1

A revision to the subject Surveillance Requirement 4.6.1.2.k is requested to clarify that the 25% surveillance interval extension permitted by Specification 4.0.2 is not applicable to those time periods specified by 10 CFR 50 Appendix J. The proposed change is an enhancement to safety in that it represents a clarification of the technical specifications for the purpose of helping ensure compliance with 10 CFR 50 Appendix J. (Page 3/4 6-4)

4. (TSPS 093), Reporting Requirement, Generic

Generic Letter 83-43 was issued by the NRC on December 19, 1983, and provides policy guidance concerning the implementation of technical specification changes required as a result of the addition of Section 50.73 to 10 CFR 50, "Licensee Event Reporting System." In addition to proposing the specific changes requested by this generic letter, a complete review of the Grand Gulf Technical Specifications was conducted to identify any additional specifications requiring change. The major change proposed as a result of this review is the deletion of reference (both specific and implied) to Specification 6.9.1.11, 6.9.1.12, and 6.9.1.13. During this review process it also became evident that additional changes should be proposed to a) add clarification concerning Section 50.73 requirements, and b) reflect plant specific requirements.

The following pages to the technical specifications are affected in order to implement the proposed changes and have been marked to be consistent with the new regulatory requirements.

ii	3/4 4-14	3/4 11-12	3/4 12-11
XX	3/4 6-9	3/4 11-13	6-8
1-7	3/4 6-17	3/4 11-14	6-10
3/4 3-60	3/4 7-27	3/4 11-15	6-13
3/4 3-63	3/4 7-43	3/4 11-16	6-15
3/4 3-76	3/4 8-7	3/4 11-18	6-19
3/4 3-81	3/4 11-5	3/4 11-20	6-20
3/4 3-82	3/4 11-6	3/4 12-1	6-21
3/4 3-87	3/4 11-7	Insert to 3/4 12-6	6-22

Some of the proposed changes included with this submittal were included in the Radiological Assessment Branch (RAB) submittal. Their inclusion with this package (along with problem sheets 070 and 071) will indicate all necessary changes associated with reporting requirements. The items submitted in the RAB package are indicated in the marked up technical specification pages with problem sheet numbers other than 093.

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

SECTION

DEFINITIONS (Continued) PAGE

093

INDEX

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~~ EVENT

1.35 A REPORTABLE ~~OCCURRENCE~~ EVENT shall be any of those conditions specified in ~~Specifications 6.9.1.12 and 6.9.1.13.~~ Section 50.73 To 10 CFR Part 50.

093

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, 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.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level. ~~after depressurizing the reactor vessel, if required.~~ Comply with the requirements of Specification 6.7.1. 053

DEPRESSURIZE THE REACTOR VESSEL AS NECESSARY FOR ECCS OPERATION.

INSTRUMENTATION

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.2 The seismic monitoring instrumentation shown in Table 3.3.7.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status. 093

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.2-1.

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. ~~In lieu of any other report required by Specification 6.9.1, a~~ A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety. 093

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more meteorological monitoring instrumentation channels inoperable for more than 7 days, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status. 093

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.9-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.9-1:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, steam tunnel or drywell, then inspect the primary containment at least once per 8 hours or monitor the containment, steam tunnel and/or drywell air temperature at least once per hour at the locations listed in Specification 3.7.8, 4.6.1.8 and 4.6.2.6.

b. Restore the minimum number of instruments to OPERABLE status within 14 days or, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.9.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.9.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status. 093

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

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

c. CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.11 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.11 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.11-1.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limit of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3~~,~~ and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.12-1.

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

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

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours cumulative operating time in a consecutive 12-month period, or greater than 4.0 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. In OPERABLE CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A Special Report ~~REPORTABLE OCCURRENCE~~ shall be prepared and submitted to the Commission pursuant to Specification 6.9.~~1~~2 within 30 days. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

093

Amendment No. —

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~~. Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2.d, 4.6.1.2.e, and 4.6.1.2.g.

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

Amendment No. ——

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.

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. 093

in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days.

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.

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. 093

in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days.

3/4 7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers listed in Tables 3.7.4-1 and 3.7.4-2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7.4-1 and 3.7.4-2. If less than two snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ±25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.
#The provisions of Specification 4.0.2 are not applicable.

200

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) that attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of these visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Surveillance Requirements 4.7.4.d or 4.7.4.e, as applicable. However, when a fluid part of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE by functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of at least:

1. 10% of the total of the hydraulic snubbers listed in Table 3.7.4-1 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Surveillance Requirement 4.7.4.d, an additional 10% of the hydraulic snubbers shall be functionally tested.

2. That number of mechanical snubbers which follows the expression $35\ (1 + \frac{c}{2})$, where c = 2, the allowable number of snubbers not meeting the acceptance criteria, shall be functionally tested either in-place or in a bench test. For each number of snubbers above c which does not meet the functional test acceptance criteria of Specifications 4.7.4.e, an additional sample selected according to the expression $35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2\ (a - c)$ shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample.

   Functional testing shall continue according to the expression $b\ [35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2]$ where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers in Table 3.7.4-2 have been functionally tested.

Functional Tests (Continued)
The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle
2. Each snubber within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)
3. Each snubber within 10 feet of the discharge from a safety relief valve

Tables 3.7.4-1 and 3.7.4-2 may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber, if it is repaired and installed in another position, and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

006

SURVEILLANCE REQUIREMENTS (Continued)

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Drag force shall not have increased more than 50% since the last surveillance test.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to display under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

006

SURVEILLANCE REQUIREMENTS (Continued)

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintanance records for each snubber listed in Tables 3.7.4-1 and 3.7.4-2 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

006

900

TABLE 3.7.4-1

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	AREA	ELEVATION	SNUBBER NO.	AREA	ELEVATION
a. RECIRCULATION SYSTEM			RECIRCULATION SYSTEM (Continued)		
Q1B33G006S354A	11	125	Q1B33G006S305A	11	131
Q1B33G006S354B	11	125	Q1B33G006S305B	11	131
Q1B33G006S356A	11	125	Q1B33G006S306A	11	131
Q1B33G006S356B	11	125	Q1B33G006S306B	11	131
Q1B33G006S357A	11	125	Q1B33G006S351A	11	134
Q1B33G006S357B	11	125	Q1B33G006S351B	11	134
Q1B33G006S358A	11	110	Q1B33G006S352A	11	125
Q1B33G006S358B	11	110	Q1B33G006S352B	11	125
Q1B33G006S359A	11	110	Q1B33G006S353A	11	134
Q1B33G006S359B	11	110	Q1B33G006S353B	11	134
Q1B33G006S360A	11	107	Q1B33G006S369A	11	123
Q1B33G006S360B	11	107	Q1B33G006S369B	11	123
Q1B33G006S361A	11	101	Q1B33G006S370A	11	125
Q1B33G006S361B	11	101	Q1B33G006S370B	11	125
Q1B33G006S362A	11	110	Q1B33G006S371A	11	123
Q1B33G006S362B	11	110	Q1B33G006S371B	11	123
Q1B33G006S363A	11	102	Q1B33G006S372A	11	101
Q1B33G006S363B	11	102	Q1B33G006S372B	11	101
Q1B33G006S301A	11	111	Q1B33G006S373A	11	101
Q1B33G006S301B	11	111	Q1B33G006S373B	11	101
Q1B33G006S302A	11	103	Q1B33G006S374A	11	101
Q1B33G006S302B	11	103	Q1B33G006S374B	11	101
Q1B33G006S303A	11	107	Q1B33G006S375A	11	101
Q1B33G006S303B	11	107	Q1B33G006S375B	11	101
Q1B33G006S304A	11	111	Q1B33G006S376A	11	107
Q1B33G006S304B	11	111	Q1B33G006S376B	11	107

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7.4-1 provided that a revision to Table 3.7.4-1 is included with the next License Amendment request.

900

TABLE 3.7.4-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	AREA	ELEVATION
MAIN STEAM SYSTEM		
Q1B21G006S102A	11	155
Q1B21G006S103A	11	150
Q1B21G006S104A	11	150
Q1B21G006S105A	11	150
Q1B21G006S101B	11	156
Q1B21G006S102B	11	156
Q1B21G006S103B	11	149
Q1B21G006S104B	11	150
Q1B21G006S105B	11	150
Q1B21G006S106B	11	150
Q1B21G006S107B	11	150
Q1B21G006S108B	11	150
Q1B21G006S101C	11	156
Q1B21G006S102C	11	156
Q1B21G006S103C	11	149
Q1B21G006S104C	11	150
Q1B21G006S105C	11	150
Q1B21G006S106C	11	150
Q1B21G006S107C	11	150
Q1B21G006S108C	11	150
Q1B21G006S102D	11	155
Q1B21G006S103D	11	150
Q1B21G006S104D	11	150
Q1B21G006S105D	11	150

TABLE 3.7.4-2

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

a. RECIRCULATION SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1B33G023R01(2)	11	117
Q1B33G024R01	11	102
Q1B33G024R02(2)	11	102
Q1B33G024R05	11	101
Q1B33G105C01	11	101
Q1B33G105R01	11	101
Q1B33G105R02(2)	11	101
Q1B33G108C01	11	101
Q1B33G108R01(3)	11	101
Q1B33G108R02(2)	11	101
Q1B33G112R01	11	101
Q1B33G122R01	11	108
Q1B33G124R01	11	122

RECIRCULATION SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B33G120C01(2)	11	121
Q1B33G129C01	11	121
Q1B33G262R02	11	103
Q1B33G265C01	11	102
Q1B33G265R04	11	107
Q1B33G265R05	11	112
Q1B33G318R01	11	102
Q1B33G322R01(2)	11	112
Q1B33G331R02	11	111
Q1B33G337R02	11	109
Q1B33G339R01	11	111
Q1B33G346R01	11	105
Q1B33G355R01(2)	11	102

*Snubbers may be added to safety related systems without prior License Amendment to Table 3.7.4-2 provided that a revision to Table 3.7.4-2 is included with the next License Amendment request.

**The number in parentheses is the number of snubbers associated with the component support. If no number in parentheses appears, there is only one snubber associated with the support.

900

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

b. MAIN STEAM SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1B21G021C04	11	141
Q1B21G022R01(2)	11	135
Q1B21G022R03(2)	11	133
Q1B21G022R06(2)	11	124
Q1B21G022R12(2)	11	132
Q1B21G022R13(2)	11	131
Q1B21G022R14	11	126
Q1B21G022R15	11	125
Q1B21G022R16	11	121
Q1B21G023R03	11	137
Q1B21G023R05	11	133
Q1B21G023R06(2)	11	133
Q1B21G023R08	11	126
Q1B21G023R09	11	122
Q1B21G023R10	11	122
Q1B21G023R11(2)	11	120
Q1B21G023R14	11	141
Q1B21G023R15(2)	11	141
Q1B21G023R16	11	133
Q1B21G023R17	11	121
Q1B21G023R18(2)	11	119
Q1B21G023R20	11	120
Q1B21G024C01	11	131
Q1B21G024R04	11	137
Q1B21G024R05(2)	11	132
Q1B21G024R06	11	125
Q1B21G024R07(2)	11	119

MAIN STEAM SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B21G024R11	11	138
Q1B21G024R12(2)	11	127
Q1B21G024R13	11	123
Q1B21G024R17	11	128
Q1B21G025R02	11	128
Q1B21G025R03	11	125
Q1B21G025R04(2)	11	124
Q1B21G025R05	11	120
Q1B21G026C01(2)	11	143
Q1B21G026C02(2)	11	143
Q1B21G026R01	11	143
Q1B21G026R02(2)	11	153
Q1B21G026R03	11	149
Q1B21G026R04(2)	11	153
Q1B21G026R05	11	143
Q1B21G026R06(2)	11	143
Q1B21G026R07	11	143
Q1B21G026R08	11	149
Q1B21G026R03(2)	8	143
Q1B21G030R03	11	129
Q1B21G032R04	11	127
Q1B21G032R05	11	120
Q1B21G123R01	11	165
Q1B21G126R01	11	159
Q1B21G127R01(2)	11	193
Q1B21G127R04	11	186
Q1B21G127R01	11	150

900

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

MAIN STEAM SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B21G139R02	11	150
Q1B21G141R01	11	173
Q1B21G142R01(2)	11	173
Q1B21G144R01	11	173
Q1B21G146C03(2)	11	169
Q1B21G146C04	11	169
Q1B21G146R03	11	173
Q1B21G147C02	11	167
Q1B21G148C01(2)	11	173
Q1B21G1489R01(2)	11	172
Q1B21G153C01	11	174
Q1B21G153C02	11	182
Q1B21G153C03(2)	11	171
Q1B21G153R01	11	181
Q1B21G153R02(2)	11	175
Q1B21G153R03(2)	11	172
Q1B21G153R05(2)	11	170
Q1B21G162R01	11	113
Q1B21G163R01	11	113
Q1B21G163R02	11	113
Q1B21G171R01	11	165
Q1B21G174C01(2)	11	196
Q1B21G174R01	11	197
Q1B21G174R02	11	196
Q1B21G175R01(2)	11	153
Q1B21G175R02(2)	11	158
Q1B21G180R01	11	152

MAIN STEAM SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B21G180R02(2)	11	158
Q1B21G180R03	11	161
Q1B21G181C01	11	158
Q1B21G183R01(2)	11	152
Q1B21G189R02	11	151
Q1B21G189R01	11	153
Q1B21G194R01	11	161
Q1B21G194R02(2)	11	159
Q1B21G195R01	11	161
Q1B21G195R02(2)	11	160
Q1B21G196R01(2)	11	151
Q1B21G197R01(2)	11	157
Q1B21G201R01	11	158
Q1B21G201R02(2)	11	157
Q1B21G204R01	11	152
Q1B21G204R02(2)	11	160
Q1B21G205R01	11	159
Q1B21G205R02(2)	11	160
Q1B21G208R01	11	157
Q1B21G208R03	11	160
Q1B21G210R01(2)	11	157
Q1B21G213R01	11	151
Q1B21G213R02(2)	11	152
Q1B21G217R02	11	159
Q1B21G219R01(2)	11	157
Q1B21G222R01	11	160
Q1B21G224R01	11	152

900

Order APR 18 1984

GRAND GULF-UNIT 1

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

MAIN STEAM SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B21G225R01	11	147
Q1B21G226C03	11	168
Q1B21G226R01(2)	11	173
Q1B21G304R01	11	156
Q1B21G306R01	11	151
Q1B21G311R01(2)	11	152
Q1B21G355R01	11	147
Q1B21G357C03	11	148
Q1B21G359C03	11	148
Q1B21G361C03	11	147
Q1B21G369R01(2)	11	148
Q1B21G372R01(2)	11	148
Q1B21G382R02(2)	11	152
Q1B21G384R01	11	152
Q1B21G423R01	11	147
Q1B21G424R01	11	147
Q1B21G490R03	11	152

c. SLC SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1C41G113C02	11	185
Q1C41G113C03	11	181
Q1C41G113R02	11	181
Q1C41G113R03	11	181
Q1C41G117C02	11	145
Q1C41G117R01	11	151
Q1C41G119R01(2)	11	129
Q1C41G119R03	11	114
Q1C41G119R04	11	112
Q1C41G119R05	11	112
Q1C41G120C05	11	155
Q1C41G124R01	11	159
Q1C41G124R03	11	162

d. RESIDUAL HEAT REMOVAL SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E12G009R03	7	134
Q1E12G009R04	7	134
Q1E12G009R05	8	134

900

Order

ADD.

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G009R06	8	134
Q1E12G010R02	8	105
Q1E12G010R04	8	103
Q1E12G010R05	8	125
Q1E12G010R07	8	133
Q1E12G010R10	8	142
Q1E12G010R11	8	142
Q1E12G010R13(2)	8	113
Q1E12G010R15	8	103
Q1E12G010R16	8	104
Q1E12G010R17(2)	8	104
Q1E12G010R18(2)	8	96
Q1E12G011R02(3)	8	99
Q1E12G012R02(2)	7	114
Q1E12G012R04	7	142
Q1E12G012R05	7	142
Q1E12G012R08	8	104
Q1E12G012R09	8	102
Q1E12G012R13	7	119
Q1E12G012R15	7	133
Q1E12G012R16	7	99
Q1E12G012R18	11	133
Q1E12G012R19	11	133
Q1E12G013C01	7	110
Q1E12G013C02	7	130
Q1E12G013R02(2)	7	115
Q1E12G013R03	7	110

SNUBBER NO.	AREA	ELEVATION
RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G013R04	7	119
Q1E12G013R05(2)	7	100
Q1E12G013R06(3)	7	120
Q1E12G013R07	7	121
Q1E12G013R08	7	105
Q1E12G013R11	7	97
Q1E12G014C01	8	110
Q1E12G014C03	8	106
Q1E12G014C04	8	130
Q1E12G014R01(2)	8	129
Q1E12G014R03(2)	8	98
Q1E12G014R04(3)	8	122
Q1E12G014R05	8	105
Q1E12G014R07	8	106
Q1E12G014R10(2)	8	109
Q1E12G014R11(2)	8	110
Q1E12G015R02	11	156
Q1E12G015R04(2)	11	143
Q1E12G015R06	11	143
Q1E12G015R07	11	214
Q1E12G015R08	11	210
Q1E12G015R11	11	143
Q1E12G015R17	11	210
Q1E12G015R19	11	214
Q1E12G015R20	11	144
Q1E12G015R21(2)	11	140
Q1E12G015R28(3)	11	192

900

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G015R33(2)	11	205
Q1E12G015R38	11	157
Q1E12G016C01	11	143
Q1E12G016R01	11	146
Q1E12G016R02	11	143
Q1E12G016R03	11	143
Q1E12G016R05(2)	11	143
Q1E12G019R05(2)	8	139
Q1E12G019R07	8	149
Q1E12G019R08	7	149
Q1E12G019R09(2)	7	143
Q1E12G020R01(2)	8	148
Q1E12G020R02(2)	7	148
Q1E12G020R03	8	148
Q1E12G020R04(2)	8	148
Q1E12G020R05	7	147
Q1E12G020R07(2)	7	147
Q1E12G020R09	7	147
Q1E12G021R01	8	147

SNUBBER NO.	AREA	ELEVATION
RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G021R03(2)	8	146
Q1E12G025C01(2)	8	95
Q1E12G025R01	8	110
Q1E12G119R02	7	152
Q1E12G159R01	7	126
Q1E12G159R03	7	126
Q1E12G159R04	7	131
e. LPCS SYSTEM		
Q1E21G001R05	9	96
Q1E21G001R07(2)	9	96
Q1E21G002R01	11	150
Q1E21G002R02	11	150
Q1E21G002R03	11	151
Q1E21G002R04	11	153
Q1E21G002R05	11	153
Q1E21G002R06	11	153
Q1E21G002R07	11	150

006

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

f. HPCS SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E22G001R10(2)	8	96
Q1E22G002R02(2)	8	96
Q1E22G002R03	8	96
Q1E22G003R01	11	153
Q1E22G003R02	11	153
Q1E22G003R03	11	149
Q1E22G003R04	11	150
Q1E22G003R05	11	151

g. RCS LEAK DETECTION SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E31G116R01	11	169
Q1E31G122R01(2)	11	149
Q1E31G124R01(2)	11	151
Q1E31G126C01	11	149
Q1E31G140R01	11	159
Q1E31G140R02(2)	11	159
Q1E31G148R01(2)	11	151
Q1E31G149R01(2)	11	151
Q1E31G168R01	11	158
Q1E31G174R01(2)	11	151
Q1E31G176C01	11	147
Q1E31G178R08	11	179
Q1E31G178R09	11	179
Q1E31G181R01	11	156
Q1E31G243R01	11	144
Q1E31G243R02	11	140
Q1E31G246R01(2)	11	144

h. MSIV LEAKAGE CONTROL SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E32G103C01(2)	8	122
Q1E32G106C01	8	121
Q1E32G109C01	8	122
Q1E32G119C01	8	148

i. FEEDWATER LEAKAGE CONTROL SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E38G102R01	8	145

j. RCIC SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1E51G001R05	8	104
Q1E51G001R06	8	109
Q1E51G001R09	11	133
Q1E51G001R10(2)	11	134
Q1E51G001R15	11	178
Q1E51G001R17(2)	11	190
Q1E51G001R18	11	194
Q1E51G001R19(2)	11	194
Q1E51G003R03	7	126
Q1E51G003R04	7	117
Q1E51G003R05(2)	7	127
Q1E51G003R07	8	112
Q1E51G003R08(2)	8	112
Q1E51G003R09(2)	8	109
Q1E51G003R10	8	105
Q1E51G003R11(2)	8	100
Q1E51G003R12(2)	8	106

Order
APR 1 8 1984

900

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
RCIC SYSTEM (Continued)		
Q1E51G004C02(2)	8	97
Q1E51G004R01(2)	8	98
Q1E51G004R05(2)	8	106
Q1E51G004R06(2)	8	96
Q1E51G004R07(2)	8	97
Q1E51G004R08(2)	11	164
Q1E51G004R11	8	97
Q1E51G004R13(2)	11	167
Q1E51G004R14(2)	11	152
Q1E51G158R03(2)	11	143
Q1E51G180R01	8	97
k. COMBUSTIBLE GAS CONTROL SYSTEM		
Q1E61G001R07	11	189
l. RWCU SYSTEM		
Q1G33G002C03(2)	11	113
Q1G33G002R03(2)	8	136
Q1G33G002R05(2)	11	140
Q1G33G002R08(2)	11	102
Q1G33G002R09(3)	11	102
Q1G33G002R10(2)	11	102
Q1G33G002R11	11	102
Q1G33G002R12	11	102
Q1G33G002R13(2)	11	102
Q1G33G002R14(2)	11	102
Q1G33G002R16	11	112
Q1G33G002R17(2)	8	125

SNUBBER NO.	AREA	ELEVATION
RWCU SYSTEM (Continued)		
Q1G33G002R18	8	116
Q1G33G002R19	8	116
Q1G33G002R21(2)	11	102
Q1G33G002R22	11	102
Q1G33G002R24	11	102
Q1G33G011R01	11	140
Q1G33G011R03(2)	11	145
Q1G33G012R01(2)	11	142
Q1G33G012R02	11	152
Q1G33G015R01(3)	11	103
m. FPCC SYSTEM		
Q1G41G006R01	9	114
Q1G41G006R07(3)	7	99
Q1G41G015R09	11	204
Q1G41G016C08	11	163
Q1G41G016R04	11	166
Q1G41G016R24	11	163
Q1G41G016R27(2)	11	203
Q1G41G016R28(2)	11	206
Q1G41G016R32	11	197
Q1G41G018R06	9	197
n. SSW SYSTEM		
Q1P41G001R14(2)	7	98
Q1P41G002R10(2)	8	106
Q1P41G002R12(2)	8	106
Q1P41G006C01	8	99

Order
APR 1 8 1984

900

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

1. SAFETY RELATED MECHANICAL SNUBBERS

SSW SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1P41G006C17	8	99
Q1P41G007R19	025A	144
Q1P41G007R20	025A	144
Q1P41G007R23(2)	025A	138
Q1P41G007R24(2)	025A	137

o. CCW SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1P42G002R06(2)	9	193
Q1P42G002R07(2)	9	186
Q1P42G002R11(2)	9	186
Q1P42G002R13(2)	9	186

900

Order
APR 1 8 1984

GRAND GULF-UNIT 1

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*, **

2. NON-Q MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
a. MAIN STEAM SYSTEM		
N1B21G118R01	11	148
N1B21G118R02	11	147
N1B21G191C02	11	137
N1B21G192C03	11	136
N1B21G193R01(2)	11	138
N1B21G193R04	11	136
N1B21G231R01(2)	11	163
b. RECIRCULATION SYSTEM		
N1B33G104R02	11	102
N1B33G105C01	11	101
N1B33G105C03	11	101
N1B33G105C04	11	101
N1B33G105C05	11	101
N1B33G105R01	11	101
N1B33G106R01	11	102
N1B33G107R01	11	102
N1B33G107R02	11	102
N1B33G108C02	11	101
N1B33G108R03(2)	11	101
N1B33G108R05	11	101
N1B33G108R06(2)	11	101
N1B33G108R07	11	101
N1B33G119R04	11	112
N1B33G120R03	11	101
N1B33G123C01	11	102
N1B33G362R03	11	102

SNUBBER NO.	AREA	ELEVATION
c. RESIDUAL HEAT REMOVAL SYSTEM		
N1E12G172R02	11	129
N1E12G212R01	11	136
N1E12G212R03	11	133
d. REACTOR CORE ISOLATING COOLING SYSTEM		
N1E51G120R01	11	127
e. REACTOR WATER CLEANUP SYSTEM		
N1G33G002R01	7	120
N1G33G002R02	8	118
N1G33G002R03	8	123
N1G33G002R04	8	123
N1G33G002R05(2)	11	147
N1G33G002R08(2)	11	164
N1G33G002R10(2)	11	147
N1G33G002R11(3)	11	180
N1G33G002R12(3)	11	180
N1G33G002R13	11	178
N1G33G002R14	8	120
N1G33G002R21	8	120

Order

APR 1 8 1984

900

PLANT SYSTEMS

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

900

No. of Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.4.f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

900

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4.f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

900

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.4e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

900

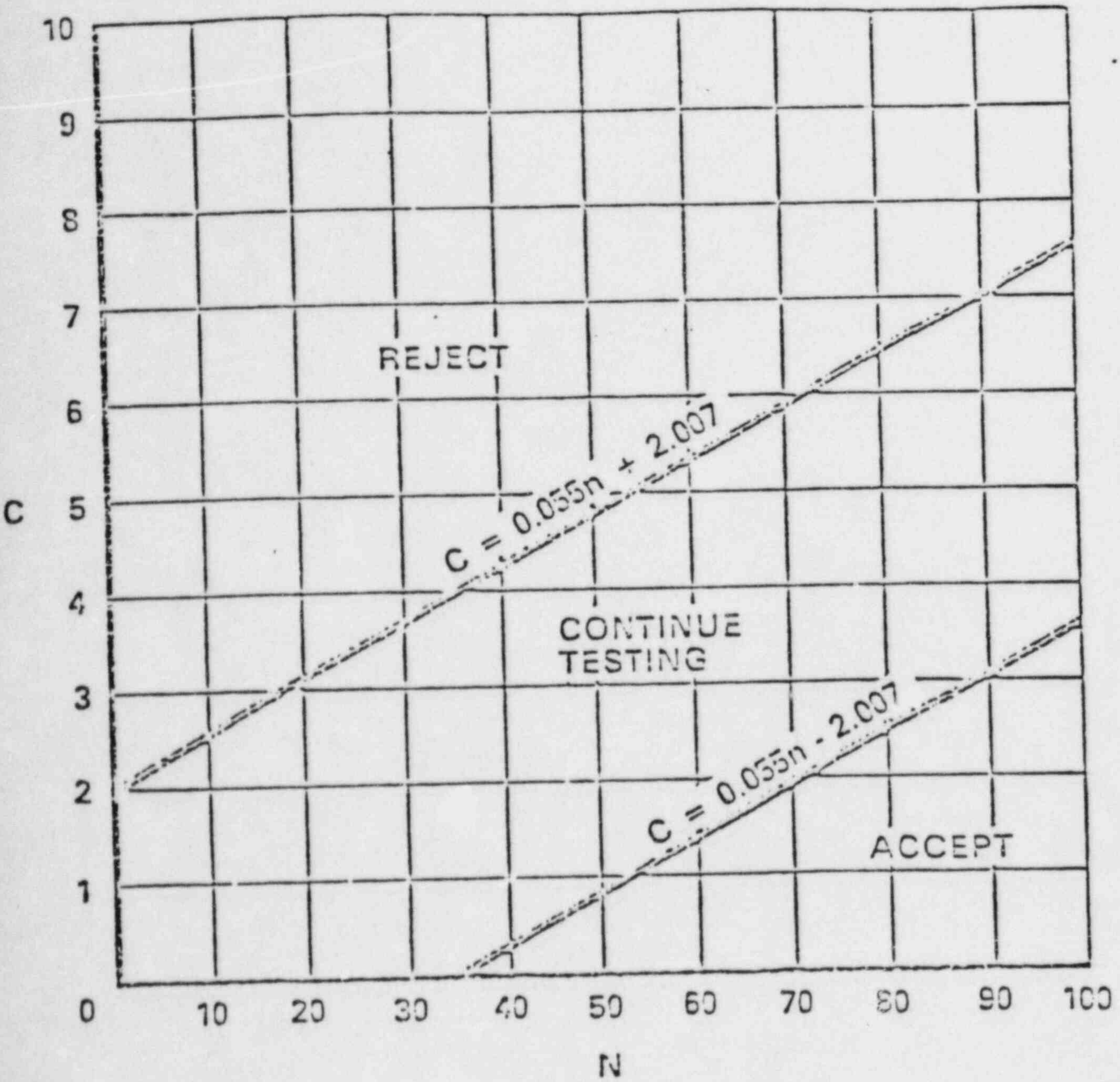


FIGURE 4.7.4-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.5.3 Reports - A Special Report ~~report~~ shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination. 093

3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.8-1 shall be maintained within the limits indicated in Table 3.7.8-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.8-1:

a. For more than eight hours, ~~in lieu of any report required by Specification 6.9.1~~ prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment. 093

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 The temperature in each of the areas shown in Table 3.7.8-1 shall be determined to be within its limit at least once per 12 hours.

14. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed lines.

15. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 11 and 12.

16. Verifying that the following diesel generator lockout features prevent diesel generator starting and/or trip the diesel generator only when required:

    a) Generator loss of excitation.
    b) Generator reverse power.
    c) High jacket water temperature.
    d) Generator overcurrent with voltage restraint.
    e) Bus underfrequency (11 and 12 only).
    f) Engine bearing temperature high (11 and 12 only).
    g) Low turbo charger oil pressure (11 and 12 only).
    h) High vibration (11 and 12 only).
    i) High lube oil temperature (11 and 12 only).
    j) Low lube oil pressure (13 only).
    k) High crankcase pressure.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that the three diesel generators accelerate to at least 441 rpm for diesel generators 11 and 12 and 882 rpm for diesel generator 13 in less than or equal to 10 seconds.

f. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section 11, Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

Within 30 DAYS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8.4.2-1 shall be OPERABLE or shall be bypassed either continuously or only under accident conditions, as indicated, by an OPERABLE bypass device.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not OPERABLE or not bypassed either continuously or only under accident conditions, as indicated in Table 3.8.4.2-1, ~~take administrative action to~~ bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s). 041

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection which is bypassed either continuously or only under accident conditions for the above required valves shall be verified to be bypassed continuously or only under accident conditions, as applicable, by an OPERABLE bypass device (1) by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and (2) by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 92 days for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing.

c. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection which is not bypassed for the above required valves shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

4.8.4.2.3 The thermal overload protection for the above required valves which is continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.1.2. 093

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste system components as specified in the ODCM shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the cumulative projected dose due to the liquid effluent from the site (see Figure 5.1.3-1) in a 31 day period would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report which includes the following information: 093

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093 086

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste system components specified in the ODCM shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquids during the previous 92 days.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tank, not including liners for shipping radwaste, shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above specified tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tanks and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093 249

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in the above specified tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from the radioactive noble gases in gaseous effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.2.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

093

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions from gaseous effluents for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, or radioactive materials in paticulate form, with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.3. 093

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations. Cumulative dose contributions from tritium, radioiodines, and radioactive materials in particulate form with half-lives greater than 8 days for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM components as specified in the ODCM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM inoperable for more than 7 consecutive days, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information: 093

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093 192

SURVEILLANCE REQUIREMENTS

4.11.2.4 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM components specified in the ODCM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM components as described in the ODCM for at least 30 minutes at least once per 92 days unless the system has been utilized to process radioactive gas during the previous 92 days.

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.5 The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected cumulative dose due to gaseous effluent releases from the site (see Figure 5.1.3-1) in a 31 day period would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.*

ACTION:

a. With the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days, or with gaseous waste being discharged without treatment and in excess of the above limits, ~~in lieu of any other report required by Specification 6.9.1,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information: 093

1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 ~~,~~ and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093 080

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days in accordance with the ODCM.

4.11.2.5.2 The VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 30 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*Not applicable to Turbine Building ventilation exhaust unless filtration media is installed

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable. 093

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by maintaining the waste gas in the main condenser off-gas treatment system with the hydrogen monitor OPERABLE as required by Table 3.3.7.12-1 of Specification 3.3.7.12.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system as specified in the PROCESS CONTROL PROGRAM shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirments of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 consecutive days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

3. A description of the alternative used for SOLDIFICATION and packaging of radioactive wastes, and

4. Summary description of action(s) taken to prevent a recurrence.

c. The provisons of Specifications 3.0.3 and 3.0.4 ~~and 6.9.1.11~~ are not applicable.

093

085

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION* of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste.

a. If any test specimen fails to verify SOLIDIFICATION*, the SOLIDIFICATION* of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION* parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION*. SOLIDIFICATION* of the batch may then be resumed using the alternative SOLIDIFICATION* parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION*, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION*. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION* of subsequent batches of waste.

*Except dewatering.

Z64abm1

093 085

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitments over 12 consecutive months to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to: a. less than or equal to 25 mrem to the total body or any organ (except the thyroid), b. less than or equal to 75 mrem to the thyroid.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, ~~in lieu of any other report required by Specification 6.9.1.~~ prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence and exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (i.e., dose) to a member of the public from uranium fuel cycle sources including all effluent pathways and direct radiation for a 12 consecutive month period that includes the release(s) coverd by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.00(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance wth Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4.a.

093

090

INSERT TO TABLE 3.12.1-1, PAGE 3/4 12-6

Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in the table(s) and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All above deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.9.1.6. 249

It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. ~~In lieu of a Licensee Event Report and pursuant to Technical Specification 6.9.1.9,~~ Identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table(s) for the ODCM reflecting the new location(s). 093

F5gkh1

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest permanent residence. Broad leaf vegetation sampling is performed near the site boundary location with the highest projected D/Q in lieu of a garden census.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1, a revised figure in the ODCM reflecting the new location(s) shall be submitted to the Commission as an inclusion to the Monthly Operating Report. 249

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, ~~in lieu of any other report required by Specification 6.9.1~~, a revised figure in the ODCM reflecting the new location(s) shall be submitted to the Commission as an inclusion to the Monthly Operating Report. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. 093

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use cnesus shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, visual or by consulting local agriculture authorities.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12.1-1 shall be followed, 249
including analysis of control samples.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greated than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this 093
land use census was conducted. Identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table(s) for the ODCM reflecting the new location(s).

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed, nominal time less 25%, may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those snubbers which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample of mechanical snubbers according to the expression $35 (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers will require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records, i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . . . The requirement to monitor the snubber service life is included to enusre that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

Plant Safety Review

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the ~~Review of Operations~~ Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.4-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.4-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

006

RESPONSIBILITIES (Continued)

f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. ~~All written reports concerning events requiring 24 hour notification to the Commission.~~ Review of all Reportable Events. 083

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.

~~n. The unexpected offsite release of radioactive material and the report as described in 6.9.1.13(c).~~

n ~~o~~. Changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste systems. 083

AUTHORITY

6.5.1.7 The PSRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under 6.5.1.6(a), (c), (d), (e), (j), and (k), above.

b. Render determinations in writing to the Plant Manager with regard to whether or not each item considered under 6.5.1.6(a), (c) and (d), above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the SRC of disagreement between the PSRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PSRC shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the SRC.

CONSULTANTS

6.5.2.4 Consultants, in addition to those required in Specification 6.5.2.3, shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

MEETING FREQUENCY

6.5.2.5 The SRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The minimum quorum of the SRC necessary for the performance of the SRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 6 SRC voting members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The SRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS. ~~Events requiring 24 hour written notification to the Commission.~~ 093

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the PSRC.

ACTIVITIES (Continued)

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.

d. ~~Occurrences~~ Events reportable pursuant to ~~the Technical Specification 6.9 and violations of Technical Specifications~~ Section 50.73 to 10 CFR Part 50 shall be investigated and a report prepared which evaluates the ~~occurrence~~ events and which provides recommendations to prevent recurrence. Such report shall be approved by the Plant Manager. ~~and forwarded to the Chairman of the Safety Review Committee.~~ 093

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall be members of the plant staff who meet or exceed the qualification requirements of Section 4.4 of ANSI 18.1, 1971, as previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review shall include a determination of whether or not an unreviewed safety question is involved.

g. Records of the above activities shall be provided to the Station Manager, PSRC and/or as necessary for required reviews.

6.6 REPORTABLE ~~OCCURRENCE~~ EVENT ACTION

6.6.1 The following actions shall be taken Pursuant to the Requirements of Section 50.72 of 10CFR Part 50 and by for REPORTABLE ~~OCCURRENCES~~ EVENTS: 093

a. The Commission shall be notified ~~and/or~~ a report submitted pursuant to the requirements of ~~Specification 6.9~~ Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE ~~OCCURRENCE requiring 24 hour notification to the Commission~~ EVENT shall be reviewed by the PSRC and submitted to the SRC and the Senior Vice President - Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Senior Vice President - Nuclear and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence.

PROCEDURES AND PROGRAMS (Continued)

6. Feedwater leakage control system.
7. Post-accident sampling system.
8. Suppression pool level detection portion of the suppression pool makeup system.

The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis,
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS ~~AND REPORTABLE OCCURRENCES~~

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office, unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

MONTHLY OPERATING REPORTS

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days of the date the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the (PSRC).

~~REPORTABLE OCCURRENCES~~

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

~~PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP~~

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

093

 Amendment No. ___

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require unit shutdown, operation of safety systems or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

l. Failure or malfunction of the safety or relief valves.

093

~~THIRTY DAY WRITTEN REPORTS~~

6.9.1.13 The types of events listed below shall be the subject of written reports to the Regional Administrator of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

   1. A description of the event and equipment involved.

   2. Causes(s) for the unplanned release.

   3. Actions taken to prevent recurrence.

   4. Consequences of the unplanned release.

f. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

093

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS. ~~OCCURRENCES submitted to the Commission.~~ 093

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

RECORD RETENTION (Continued)

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the SRC.

l. Records of the service lives of all hydraulic and mechanical snubbers ~~listed on Tables 3.7.9-1 and 3.7.9-2~~ including the date at which the service life commences and associated installation and maintenance records. 900

m. Records of analyses required by the radiological environmental monitoring program.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.