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

June 21, 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-84/09, 13C, and 14B)
AECM-84/0318

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

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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 |

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 are a portion of the changes necessary to render the Grand Gulf Nuclear Station Technical Specifications consistent in all material respects with the as-built plant, the SER, and the FSAR and supporting documents.

The description of, technical justification for, and safety evaluation of the proposed changes to the Grand Gulf Technical Specifications are included in Attachments 1 and 2. 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 | Core Performance |
| 2 | Power Systems |

It should be noted that the changes requested for the Power Systems 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.

- 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.

- 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.

- 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. These proposed changes include the first portion of the changes to the Grand Gulf Technical Specifications identified by the TSRP and determined by the NRC staff to require resolution prior to issuance of the full power amendment to License No. NPF-13.

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)

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**MISSISSIPPI POWER & LIGHT COMPANY**

cc: Mr. J. B. Richard (w/o)
Mr. R. B. McGehee (w/o)
Mr. N. S. Reynolds (w/o)
Mr. G. B. Taylor (w/o)

Mr. Richard C. DeYoung, Director (w/a)
Office of Inspection & Enforcement
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Mr. J. P. O'Reilly, Regional Administrator (w/a)
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Atlanta, Georgia 30323

Dr. Alton B. Cobb (w/a)
State Health Officer
State Board of Health
Box 1700
Jackson, Mississippi 39205

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BEFORE THE

UNITED STATES REGULATORY COMMISSION

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LICENSE NO. NPF-13

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DOCKET NO. 50-416

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IN THE MATTER OF

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

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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 21st day of June, 1984.

(SEAL)

Paul J 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: CORE PERFORMANCE

TS48rg1

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

| TSPS No. | Item Nos.* |
| --- | --- |
| 049 | 1.C.8 |
| 051 | 1.A.1 |
| 124 | 1.C.2 |
| 154 | 1.A.2 |
| 155 | 1.C.3 |
| 157 | 1.D.1 |
| 182 | 1.A.3 |
| 183 | 1.C.4 |
| 184 | 1.A.4 |
| 241 | 1.C.1 |
| 251 | 1.C.9 |
| 265 | 1.C.11 |
| 281 | 1.B.1 |
| 282 | 1.C.5 |
| 307 | 1.C.10 |
| 352 | 1.C.6 |
| 354 | 1.A.5 |
| 371 | 1.C.7 |
| 375 | 1.C.8 |
| 382 | 1.D.2 |

*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

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

| | TSPS No. | TS Page No. |
|---|---|---|
| 1. | 051 | 3/4 1-15<br>B 3/4 1-4 |
| 2. | 154 | 3/4 1-7 |

EDITORIAL CHANGES

Proposed editorial changes to the technical specifications are discussed below:

3. (TSPS 182) "All Rods In" Versus "One-Rod-Out" Terminology, Technical Specification 3.9.1

   Technical Specification 3.9.1.b.1 should be revised to read "One-rod-out", instead of "All rods in". The current wording implies that "All rods in" is a refueling position interlock. "All rods in" is a signal (i.e., condition) that, if present in conjunction with the reactor mode switch being in refuel, will allow withdrawal of one control rod. With one control rod withdrawn, a "one-rod-out"

interlock will prevent the withdrawal of any additional control rods. The proposed editorial change is considered administrative in that it renders the Technical Specification consistent with the terminology used to describe the interlock which prevents the withdrawal of more than one rod when the reactor mode switch is in the refuel position. (Page 3/4 9-1)

4. (TSPS 184) "Rod-Out-Notch Override" versus "Continuous Withdrawal", Technical Specification 3.10.3

   The proposed change is an editorial change to replace the subject specification's current reference to the "rod-out-notch override" control with a reference to "continuous withdrawal" control. The term "rod-out-notch override" refers to the switch used in older BWR's to allow withdrawal of a control rod continuously, instead of on a notch by notch basis. This terminology is not used in the BWR-6 control rod withdrawal instrumentation. Continuous control rod withdrawal in a BWR-6 is accomplished using the "continuous withdrawal" pushbutton. This change is a purely administrative change to the Technical Specifications to make the nomenclature consistent with BWR-6 nomenclature. (Page 3/4 10-3)

CLARIFICATIONS

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

5. (TSPS 354) Control Rod Block Equation, Technical Specification Table 3.3.6-2

   The proposed change to the subject technical specification table adds the T factor adjustment to the flow-biased neutron flux control rod block setpoint equation. The proposed change provides consistency with the APRM flow-biased neutron flux control rod block setpoint as indicated in Technical Specification 3.2.2. The proposed change is purely administrative and represents a clarification of the technical specifications. (Page 3/4 3-52)

B. TECHNICAL SPECIFICATIONS/AS-BUILT PLANT CONSISTENCY

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

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

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

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

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

1. (TSPS 281) Fuel Assemblies, Technical Specification 5.3.1

   This proposed change corrects the maximum average fuel assembly enrichments provided for the initial core and subsequent core loadings. This change is consistent with the as-loaded GGNS-1 cycle 1 core design and safety analysis. The proposed change is purely administrative in nature and does not affect any previously calculated safety margins, since the revised values are consistent with those used in the reactor core analyses. Therefore, the proposed change does not adversely impact plant safety and renders the technical specification consistent with the as-built plant. (Page 5-5)

## 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 241), Control Rod Operability Clarification, Technical Specification 3.1.3.1

   A clarification to the Control Rod Operability Technical Specification is requested to achieve consistency with plant design and operation. "ACTION a" should be restructured to ensure that the appropriate response to a LIMITING CONDITION FOR OPERATION is taken. "ACTION b" should be revised to indicate that this separation requirement applies only to inoperable withdrawn control rods. Inoperable control rods which are fully inserted do not contribute to the rod withdrawal error event analysis and are bounded by the rod drop analysis. An additional ACTION Statement "d" should be provided to address scram discharge volume (SDV) drain and vent valve OPERABILITY. The addition of an action statement that requires the closing of the SDV vent and drain valve ensures that the potential drain and vent paths from the reactor to the containment are isolated in a timely manner. The 24 hours allowed to return the valve to operable status is justified based on the redundant function of the SDV high level reactor trip. The proposed changes have no adverse impact to the safe operation of GGNS as they represent operational enhancements which are within the safety analysis. (Pages 3/4 1-3 and 3/4 1-4)

2. (TSPS 124) Shutdown Margin Demonstration, Technical Specification 4.1.1.c

   The proposed change to the subject technical specification increases the time allowance for verifying adequate shutdown margin in the event of an immovable withdrawn control rod from one hour to twelve hours. This change will allow sufficient time to permit the use of accepted nucleonics codes to calculate control rod reactivity worths. Upon

detection of an immovable withdrawn control rod in OPERATIONAL CONDITION 1 or 2, Technical Specification 3.1.3.1 must be satisfied. The provisions of this specification, as revised by TSPS 241 (item 1 above), ensure that the plant will be placed in HOT SHUTDOWN within 12 hours, unless the immovable control rod is at least two control cells away from any other inoperable withdrawn control rod. This requirement provides adequate assurance that the increased time allowance for verifying shutdown margin will not significantly affect the ability of the control rod reactivity control system to perform its design function. In OPERATIONAL CONDITIONS 3,4, or 5, the existing restrictions on control rod withdrawals preclude situations in which the immovable control rod would adversely affect shutdown margin. Therefore, the proposed change represents no adverse impact to the safe operation of GGNS and allows adequate time to use an enhanced calculational model to satisfy the surveillance requirement. (Page 3/4 1-1)

3. (TSPS 155) Control Rod Position Indication, Technical Specification 3.1.3.5

   This proposed change revises Technical Specification 3.1.3.5 to indicate that the provisions of Technical Specification 3.0.4 are not applicable, and that directional control valves may be rearmed to permit testing associated with restoring the control rod to OPERABLE status. These revisions are requested to ensure that plant availability is not unnecessarily affected by the present specification. An exception to Technical Specification 3.0.4 will allow entry into an applicable Operational Condition with an inoperable control rod position indicator, as long as the requirements of the applicable ACTION statements are met. The proposed footnote to allow intermittent testing of directional control valves that have been disarmed is needed in order to restore an inoperable control rod to operable status. These changes are considered to be enhancement items that do not adversely impact plant safety because they are consistent with present safety analyses. (Page 3/4 1-12)

4. (TSPS 183) Rod Pattern Control System Surveillance Procedure, Technical Specification 4.10.2.a

   The proposed change to the subject surveillance requirement is requested to clearly specify when the surveillances are to be performed. The proposed change supplements the existing specification by adding a more restrictive requirement to verify proper control rod sequence operation prior to bypassing a control rod constraint. The proposed change represents no adverse impact to the safe operation of GGNS as it represents an enhancement which adds additional restrictions and clarifies the surveillance requirement. (Page 3/4 10-2)

5. (TSPS 282) Control Rod Assemblies, Technical Specification 5.3.2

   This proposed change revises Design Feature 5.3.2 to reflect that control rods have a design 143.7 inches of boron carbide powder. This value is consistent with Figure 4.2-6b in the Fuel System Design section of the FSAR. This proposed change is a clarification that promotes consistency with the FSAR, thus the change has no adverse impact on plant safety. (Page 5-5)

6. (TSPS 352) Loose-Part Detection System Bases, Technical Specification Bases 3/4.3.7.10

The proposed change adds additional information to the loose-part detection system bases. The loose-part detection system consists of eight channels comprised of sixteen sensors of which eight sensors are active to provide alarm and indication functions. The remaining eight sensors are passive at similar locations that are available for use if an active sensor fails. These passive sensors can be interchanged with the active sensors by swapping the individual connectors at the loose-part monitor panel located in the lower cable spreading room. A change to Bases 3/4.3.7.10 is proposed to distinguish between active and passive sensors. This proposed change is an enhancement and does not adversely impact plant safety because it is consistent with the safety analyses and the licensing basis. (Page B 3/4 3-5)

7. (TSPS 371) Control Rod Drive Coupling, Technical Specification 3.1.3.4

This proposed change revises Technical Specification 3.1.3.4 to indicate that the provisions of Technical Specification 3.0.4 are not applicable. This revision is requested to ensure that plant availability is not unnecessarily affected by Technical Specification 3.1.3.4. An exception to Technical Specification 3.0.4 would allow entry into an applicable Operational Condition with a control rod not coupled to its associated drive mechanism provided the requirements of the applicable ACTION statement are met. This change is considered to be an enhancement item that does not adversely impact plant safety because it is consistent with present safety analyses. (Page 3/4 1-10)

8. (TSPS 049, 375), Power Distribution Limits, Technical Specifications 4.2.1, 4.2.2, 4.2.3, 4.2.4, Bases B 3/4.2.1, B 3/4.2.2, B 3/4.2.3, B 3/4.2.4

Revisions to the GGNS Technical Specifications are proposed to add Specification 4.0.4 exemptions to the power distribution limits surveillance requirements. These exemptions are requested as an operational enhancement which provides consistency with the accepted power distribution limit calculational methods. The existing specifications require the power distribution limits to be verified to be within limits prior to exceeding 25% of RATED THERMAL POWER. The surveillance procedure necessary to verify the power distribution limits, which involves performance of an LPRM calibration via the TIP system, requires a minimum of two hours to complete. This surveillance requirement represents an unnecessary operational restriction, since the prescribed control rod movement sequences and recirculation flow control limitations, which are in effect at low reactor powers, prevent design limits from being exceeded. The proposed exemptions would eliminate this surveillance requirement below 25% of RATED THERMAL POWER. The design of the rod pattern control and the recirculation flow control systems ensures that design limits are not exceeded prior to entry into their applicable OPERATIONAL CONDITION.

Revisions to the subject technical specification bases are proposed to provide summaries of the justifications for the surveillance frequencies associated with the power distribution limits. These proposed changes supplement the information presently contained in the bases.

A revision to the ACTION statement in Specification 3.2.2 is proposed to change the restoration time for APRM setpoints from 2 hours to 8 hours. This revision is necessary to provide sufficient time to permit the APRM setpoints to be physically adjusted to satisfy the LCO. The 8 hours allowance provides one hour to recalibrate each of the 8 APRMs. This change supercedes a previously submitted proposed change (AECM 84/0147, dated February 27, 1984).

A revision to the * footnote in Specification 3.2.2 is proposed to delete the 10% of RATED THERMAL POWER APRM gain adjustment restriction. This proposed change is necessary to facilitate compliance with the intent of the ACTION statement by permitting the APRM flow-biased trip setpoints to be satisfied using gain adjustments rather than the adjustment of the control rod block and scram setpoints themselves. This revision will enable the specified APRM setpoints to be restored in a more timely manner. This change supercedes the previously submitted proposed change (AECM 84/0147).

The proposed changes represent no adverse impact to the safe operation of GGNS as they are operational enhancements which are bounded by the plant safety analysis. (Pages 3/4 2-1, 3/4 2-5, 3/4 2-6, 3/4 2-9, B 3/4 2-1, B 3/4 2-2, B 3/4 2-4 and B 3/4 2-6)

9. (TSPS 251), SRM Minimum Countrate, Technical Specification 4.9.2.c

The proposed changes to the subject surveillance requirement make the SRM minimum countrate requirements consistent throughout the GGNS Technical Specifications. Technical Specification 3.3.7.6 and Table 3.3.6-2, item 3.d were revised to lower the SRM minimum countrate from 3 counts per second to 0.7 cps in Amendment 12 to the GGNS-1 Operating License [letter from A. Schwencer (NRC) to J. P. McGaughy (MP&L) dated February 21, 1984]. This proposed change will make the SRM minimum countrate applicable during refueling operations consistent with the applicable value during startup operations. The proposed change represents no adverse impact to the safe operation of GGNS as it is a revision which is consistent with the plant safety analysis. (Page 3/4 9-4)

10. (TSPS 307) Multiple Control Rod Removal, Technical Specification 3/4.9.10.2

The proposed change adds an additional requirement to suspend all fuel loading operations when any control rod or control rod drive mechanism is removed from the core or the reactor pressure vessel. The proposed change is considered a safety enhancement because it is an additional restriction that is consistent with present safety analyses. (Pages 3/4 9-14 and 3/4 9-15)

11. (TSPS 265) Exposure Units Clarification, Technical Specification 4.1.2.b, Table 4.3.1-1

The proposed change to the subject technical specifications changes the units for associated surveillance frequencies from effective full power hours (EFPH) to megawatt days per ton (MWD/T) for consistency with the procedures used to comply with the requirements. The 1000 MWD/T surveillance frequency is consistent with the frequency for the control rod sequence exchange, which is the basis used in the analysis for reactivity anomaly and LPRM calibration. The proposed changes will ensure the surveillance is performed at least once each analyzed operating control rod sequence. The proposed changes are consistent with the fuel cycle analysis and rod pattern prediction of vendor core management and have no adverse impact on plant safety. (Pages 3/4 1-2 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:

- 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 157), Illegible Figures, Technical Specification Figures 3.1.5-1, 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.3-1, 3.2.3-2, B 3/4.2.3-1, B 3/4 3-1, and B 3/4.4.6-1, Technical Specification 3/4.2.1

   Legible copies of the subject figures are submitted to replace the figures currently found in the technical specifications. The subject figures have been redrawn to show only the major chart divisions, to utilize more legible font, and to clearly identify areas of acceptable and unacceptable operation, where applicable. The MAPLHGR graphs for the three anticipated fuel types (Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3) have been incorporated into a single figure (3.2.1-1). Specification 3/4.2.1 is revised to refer to only this figure. The "75% Rod Line" and the "Maximum Flow Control Line for Transfer to 100% Speed" are added to Figure B 3/4.2.3-1 for enhancement. Neither the limits of operation or scale ranges of the figures are affected by this submittal. This is an administrative change that enhances the readability of the technical specifications, and is submitted in response to NRC requests. (Pages 3/4 1-20, 3/4 2-1, 3/4 2-2, 3/4 2-3, 3/4 2-4, 3/4 2-7, 3/4 2-8, B 3/4 2-5, B 3/4 3-7 and B 3/4 4-7)

2. (TSPS 382), Reactor Core Hydraulic Instability, Technical Specification 3.4.1.1

   This proposed change revises ACTION statements a and b of Technical Specification 3.4.1.1 to incorporate the additional requirement of immediately reducing THERMAL POWER to less than or equal to 80% of the power specified by the 100% rod line of Figure B 3/4 2.3-1 upon determination that one or both recirculation loops are inoperable. Certain test data obtained by the NSSS vendor indicates that with less than two recirculation loops operating, postulated local flow and

neutron flux oscillations could create the potential for portions of the core to operate outside the limits of the power/flow operating map of Figure B 3/4 2.3-1. Reduction in reactor power to 80% of that specified by the 100% rod line provides assurance that local power/ flow conditions will remain within the operating limits established by the plant safety analysis. This change is a result of recent NRC guidance to avoid high power/low flow core regions in the event of the loss of one or more recirculation loops. Therefore, the proposed change represents an enhancement to plant safety since it imposes more stringent restrictions on plant operation than those contained in the current specification. (Page 3/4 4-1)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 SHUTDOWN MARGIN

#### LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or
b. 0.28% delta k/k with the highest worth rod determined by test.

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS* and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#### SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within ~~one hour~~ 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod. 124

*Except movement of IRMs, SRMs or special moveable detectors.

## 3/4.1.2 REACTIVITY ANOMALIES

### LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k:

a. Within 12 hours, perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per ~~31 effective full power days~~ 1000 MWD/T during POWER OPERATION. 263



Amendment No. — 1

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within one hour:

      a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

      b) Disarm the associated directional control valves** either:

         1) Electrically, or

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

      ~~c) Comply with Surveillance Requirement 4.1.1.c.~~

INSERT → ~~Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.~~

   ~~2. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour verify:

      a) That the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

      b) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*.

      Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

      a) Electrically, or

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

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

**May be rearmed intermittently under administrative control to permit testing associated with restoring the control rod to OPERABLE status.

Insert to Technical Specification 3.1.3.1, Page 3/4 1-3

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. Comply with Surveillance Requirement 4.1.1.c within 12 hours.

3. Comply with Surveillance Requirement 4.1.3.1.2.b.

4. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

## LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

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

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. The provisions of Specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

INSERT → 241

## SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open,* and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RPCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.3, 4.1.3.4 and 4.1.3.5.

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*These valves may be closed intermittently for testing under administrative controls.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

Insert to Technical Specification 3.1.3.1, Page 3/4 1-4

d. With the scram discharge volume vent valve and/or scram discharge volume drain valve inoperable, close the open inoperable valve(s) within one hour, restore the closed inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

   1. Declare the "slow" control rod(s) inoperable, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

   1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

   2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

   3. The eight adjacent control rods surrounding each "slow" control rod are:

      a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2, and

      b) OPERABLE.

   4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed 7.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

   a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,

   b. For specifically affected individual control rods** following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

   c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

*Except movement of SRM, IRM, or special removable detectors or normal control rod movement.

**The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance is completed prior to entry into OPERATIONAL CONDITION 1.

## CONTROL ROD DRIVE COUPLING

### LIMITING CONDITION FOR OPERATION

3.1.3.4 All control rods shall be coupled to their drive mechanisms.

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

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   1. If permitted by the RPCS, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      a) Observing any indicated response of the nuclear instrumentation, and

      b) Demonstrating that the control rod will not go to the overtravel position.

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RPCS, then until permitted by the RPCS, declare the control rod inoperable, insert the control rod, and disarm the associated directional control valves** either:

      a) Electrically, or

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

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

   1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

   2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or

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

c. The provisions of Specification 3.0.4 are not applicable. 371

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.5 At least one control rod position indication system shall be OPERABLE.

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

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within one hour:
   1. Determine the position of the control rod by the alternate control rod position indicator, or
   2. Move the control rod to a position with an OPERABLE position indicator, or
   3. When THERMAL POWER is:
      a) Within the low power setpoint of the RPCS:
         1) Declare the control rod inoperable, and
         2) Verify the position and bypassing of control rods with inoperable "Full-in" and/or "Full-out" position indicators by a second licensed operator or other technically qualified members of the unit technical staff.
      b) Greater than the low power setpoint of the RPCS, declare the control rod inoperable, insert the control rod, and disarm the associated directional control valves** either: 155
         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with both control rod position indicators for a withdrawn control rod inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 are not applicable. 155

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10 or 3.9.10.2.

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. 155

## 3/4.1.4 CONTROL ROD PROGRAM CONTROLS

### CONTROL ROD WITHDRAWAL

### LIMITING CONDITION FOR OPERATION

3.1.4.1 Control rods shall not be withdrawn.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, when the main turbine bypass valves are not fully closed and when THERMAL POWER is greater than the low power setpoint of the rod control and information system (RC & IS).

ACTION:

With any control rod ~~withdrawal~~ withdrawn, when the main turbine bypass valves are not fully closed and THERMAL POWER is greater than the low power setpoint of the RC & IS, immediately return the control rod(s) to the position prior to control rod ~~withdrawn~~ withdrawal.

### SURVEILLANCE REQUIREMENTS

4.1.4.1 Control rod withdrawal shall be prevented when the main turbine bypass valves are not fully closed and THERMAL POWER is greater than the low power setpoint of the RC & IS, by a second licensed operator or other technically qualified member of the unit technical staff.

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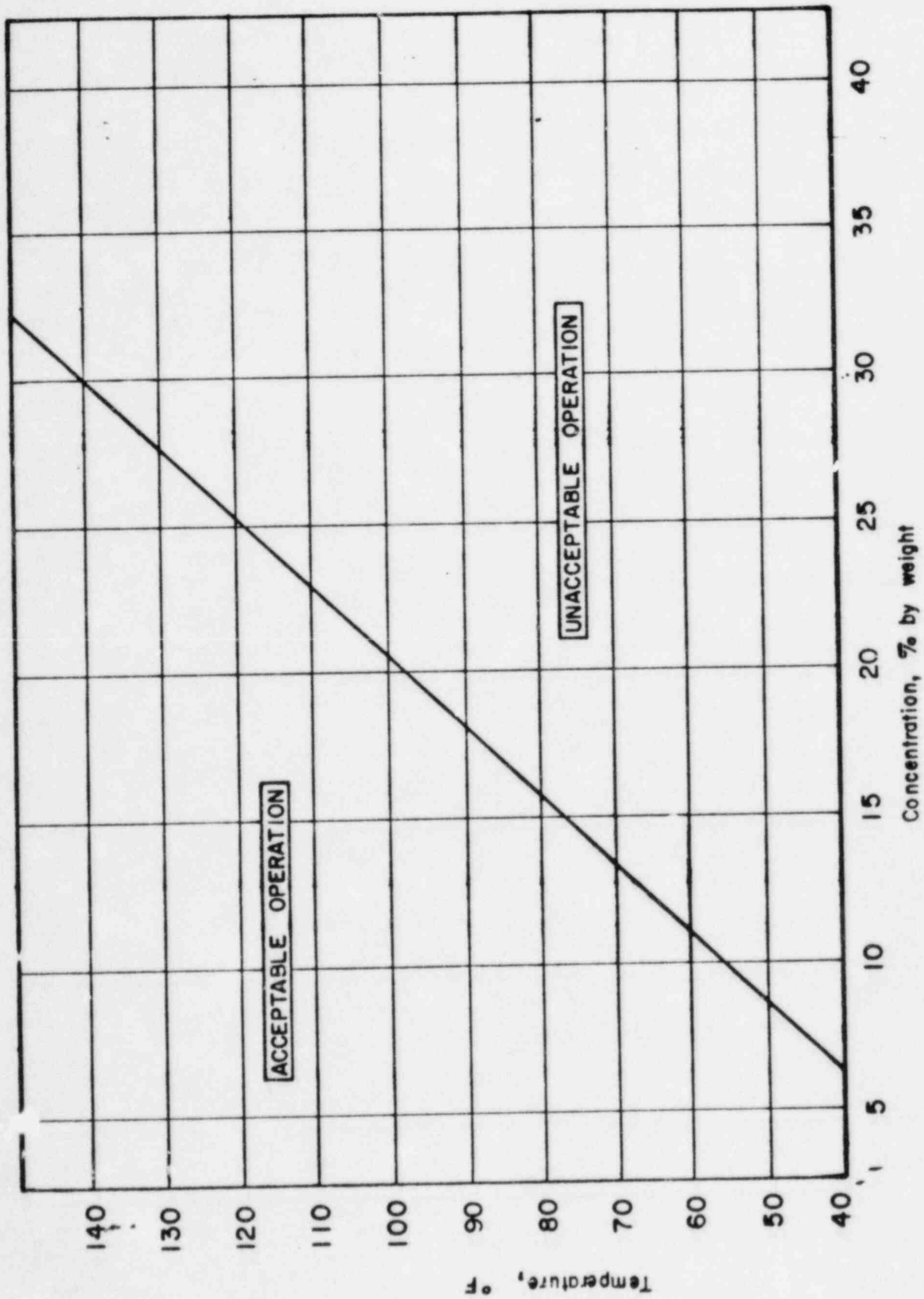



SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3..

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3, initiate corrective action within 15 minutes and restore APLHGR 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

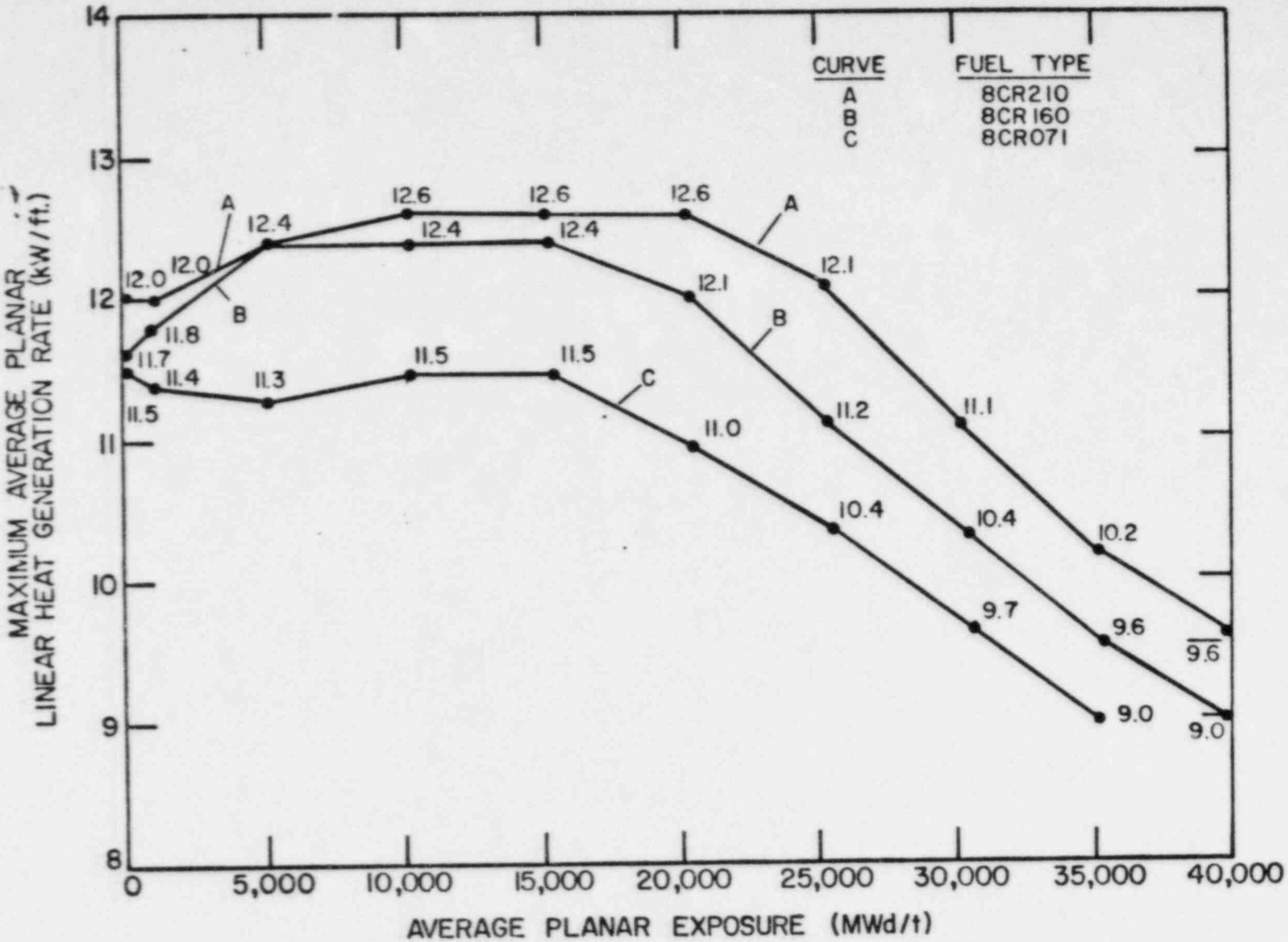
4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3:

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 a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable. 040

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MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES 8CR210, 8CR160 AND 8CR071

Figure 3.2.1-1

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.

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~~ 8 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours. 049

## 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.

d. The provisions of Specification 4.0.4 are not applicable. 049

*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. 049

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

### LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both $MCPR_f$ and $MCRP_p$ limits at indicated core flow and THERMAL POWER as shown in Figures 3.2.3-1 and 3.2.3-2.

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

ACTION:

With MCPR less than the applicable MCPR limits determined from Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR 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

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limits determined from Figures 3.2.3-1 and 3.2.3-2:

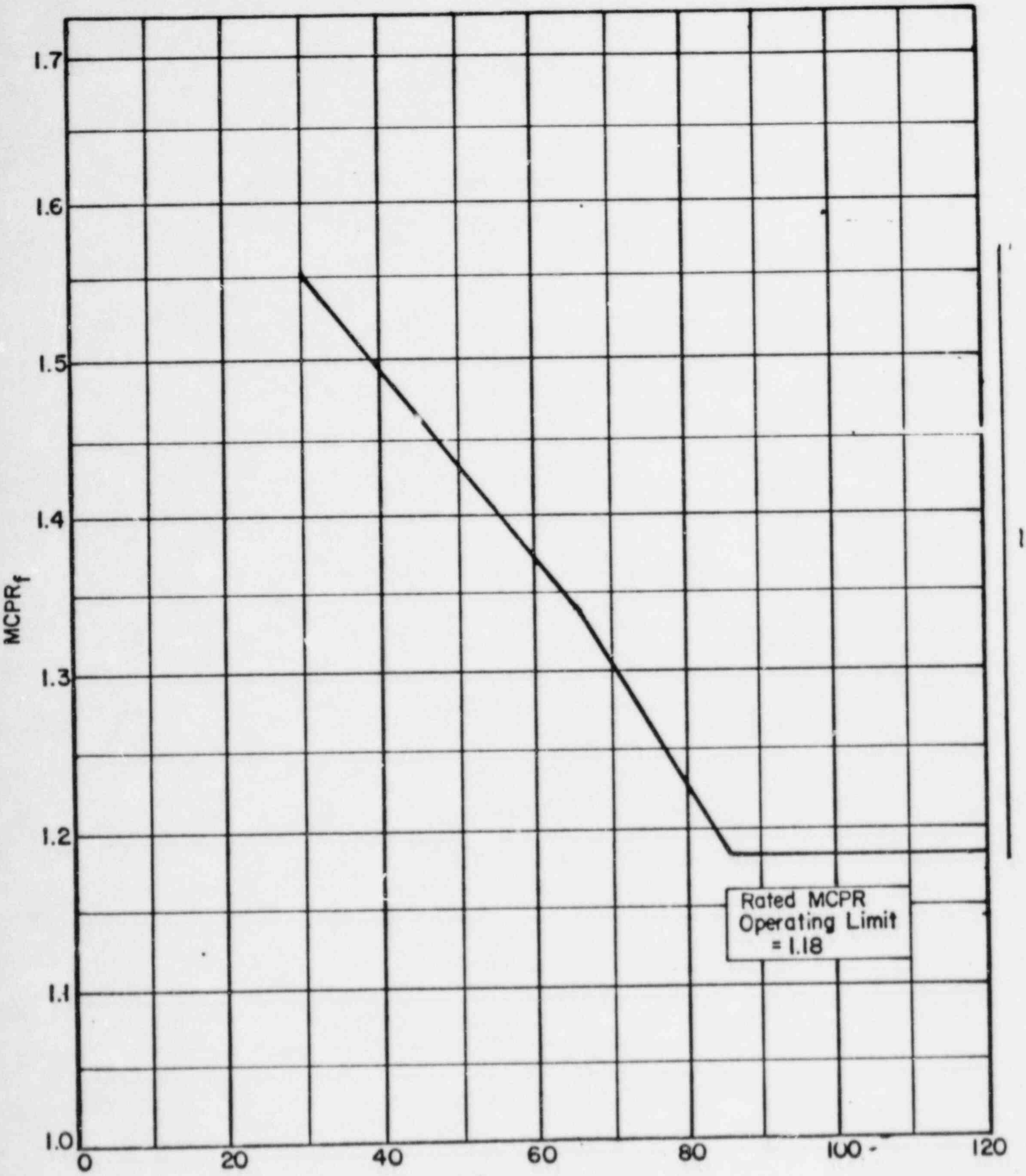
a. At least once per 24 hours,

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 a LIMITING CONTROL ROD PATTERN for MCPR.

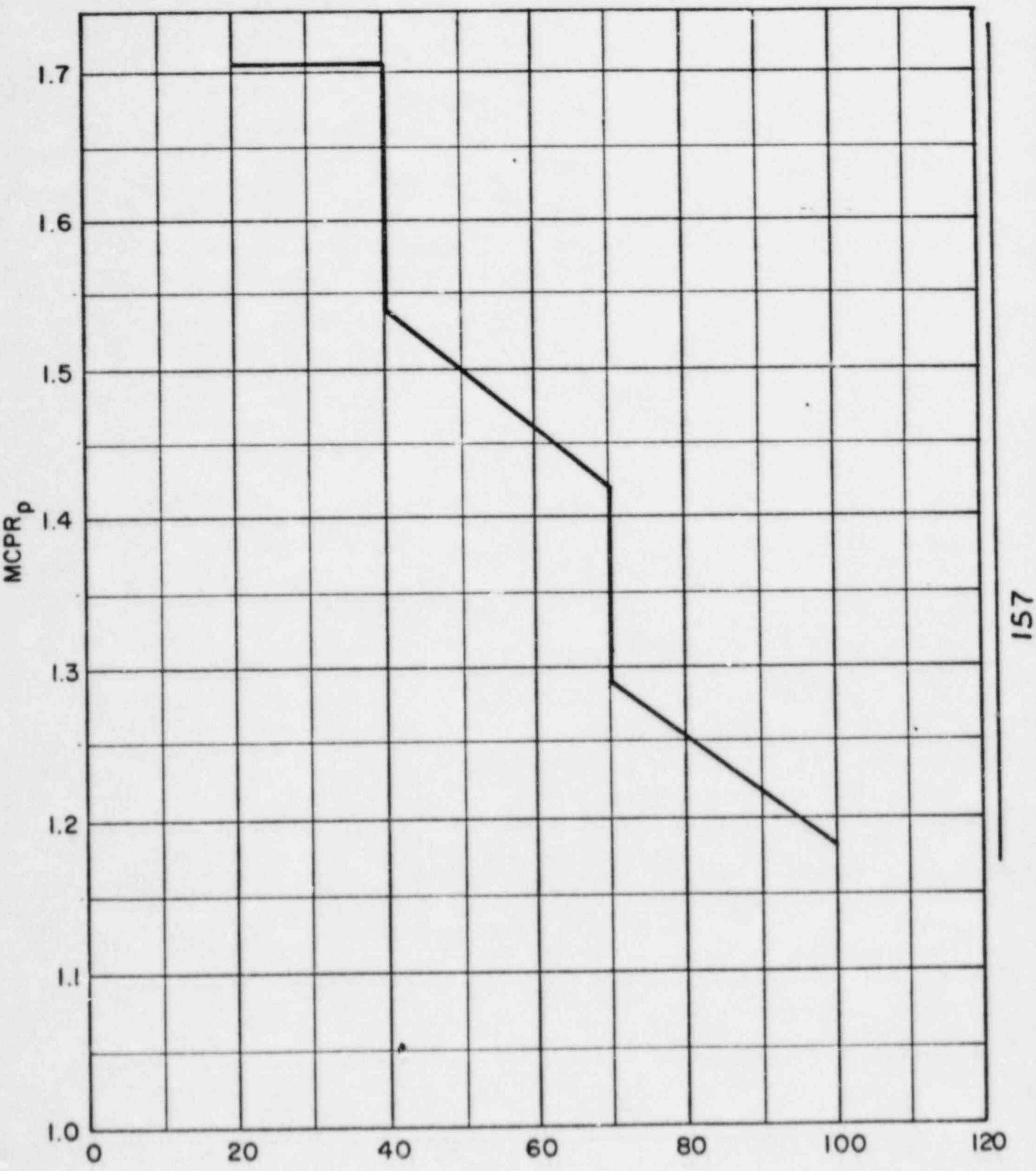
d. The provisions of Specification 4.0.4 are not applicable.

Amendment No. ——




MCPR$_f$

Figure 3.2.3-1



Thermal Power, % of Rated Thermal Power -

MCPR$_p$

Figure 3.2.3-2

## 3/4.2.4 LINEAR HEAT GENERATION RATE

### LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

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 on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable. 049

Amendment No. —

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 |
| 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 |

(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.

(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.

(f) The LPRMs shall be calibrated at least once per 1000 MWD/T ~~effective full power hours (EFPH)~~ 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.

## 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. Intermediate Rod Withdrawal Limiter Setpoint | $\le$ 70% of RATED THERMAL POWER | $\le$ 70% of RATED THERMAL POWER |
| 2. APRM | | |
| a. Flow Biased Neutron Flux-Upscale | $\le (0.66\ W + 42\%)T$* | $\le (0.66\ W + 45\%)T$* |
| b. Inoperative | NA | NA |
| c. Downscale | $\ge$ 5% of RATED THERMAL POWER | $\ge$ 3% of RATED THERMAL POWER |
| d. Neutron Flux - Upscale Startup | $\le$ 12% of RATED THERMAL POWER | $\le$ 14% of RATED THERMAL POWER |
| 3. SOURCE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | $\le 1 \times 10^5$ cps | $\le 1.5 \times 10^5$ cps |
| c. Inoperative | NA | NA |
| d. Downscale | $\ge$ 3 cps | $\ge$ 2 cps |
| 4. INTERMEDIATE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | $\le$ 108/125 of full scale | $\le$ 110/125 of full scale |
| c. Inoperative | NA | NA |
| d. Downscale | $\ge$ 5/125 of full scale | $\ge$ 3/125 of full scale |
| 5. SCRAM DISCHARGE VOLUME | | |
| a. Water Level-High | $\le$ 32 inches | $\le$ 33.5 inches |
| 6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW | | |
| a. Upscale | $\le$ 108% of rated flow | $\le$ 111% of rated flow |

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W) and the ratio of FRACTION of RATED THERMAL POWER to the MAXIMUM FRACTION of LIMITING POWER DENSITY (T factor). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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## 3/4.4 REACTOR COOLANT SYSTEM

### 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 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

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.

immediately initiate an orderly reduction of THERMAL POWER to less than or equal to 80% of the 100% Rod Line as specified in Figure B 3/4.2.3-1 and

## 3/4.9 REFUELING OPERATIONS

### 3/4.9.1 REACTOR MODE SWITCH

#### LIMITING CONDITION FOR OPERATION

**3.9.1** The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. ~~All rods in.~~ One-rod-out 182
2. Refuel platform position.
3. Refuel platform main hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

* See Special Test Exceptions 3.10.1 and 3.10.3.##

# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

##The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

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 ~~3~~ 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 ~~3~~ 0.7 cps. 251

2. Prior to and during spiral loading, until sufficient fuel has been loaded to maintain at least ~~3~~ 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 ~~3~~ 0.7 cps. 251

## MULTIPLE CONTROL ROD REMOVAL

### LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended unless all control rods are inserted in the core. L03

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

## SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations are SUSPENDED UNLESS ALL CONTROL rods are inserted in the core.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod pattern control system (RPCS) per Specification 3.1.4.2 may be suspended by means of the individual rod position bypass switches for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RPCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RPCS are bypassed, verify;

~~a.~~ ~~That~~ movement of the control rods from 75% ROD DENSITY to the RPCS low power setpoint is limited to the established control rod sequence for the specified test, and

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed, that

## 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:

   1. RPS circuitry "shorting links" removed, or

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

b. 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"~~ "continuous withdrawal" 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:

   1. The "shorting links" removed, or

   2. The rod pattern control system OPERABLE.

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.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RPCS provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted. 150

The analysis of the rod drop accident is presented in Section 15.4 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum available quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet a shutdown requirement of 3%. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing and the filling of other piping systems connected to the reactor vessel. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

## 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondar on the rod to rod power distribution within an assembly. The peak clad temperatu is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3.

Insert →

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The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.



Amendment No. —

Insert to Bases 3/4.2.1, Page B 3/4 2-1

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation exceeding a thermal limit.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.06 or that ≥ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a peak power distribution to ensure than an LHGR transient would not be increased in degraded conditions.

INSERT →

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Insert to Bases 3/4.2.2, Page B 3/4 2-2

The daily requirement to verify the APRM control rod block and scram setpoints when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to verify the APRM setpoints within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement to verify the APRM setpoints once per 12 hours after initially determining MFLPD to be greater than FRTP ensures that the consequences of an LHGR transient would not be increased in degraded conditions.

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figure 3.2.3-1. The power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

INSERT

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured. 375

~~The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.~~

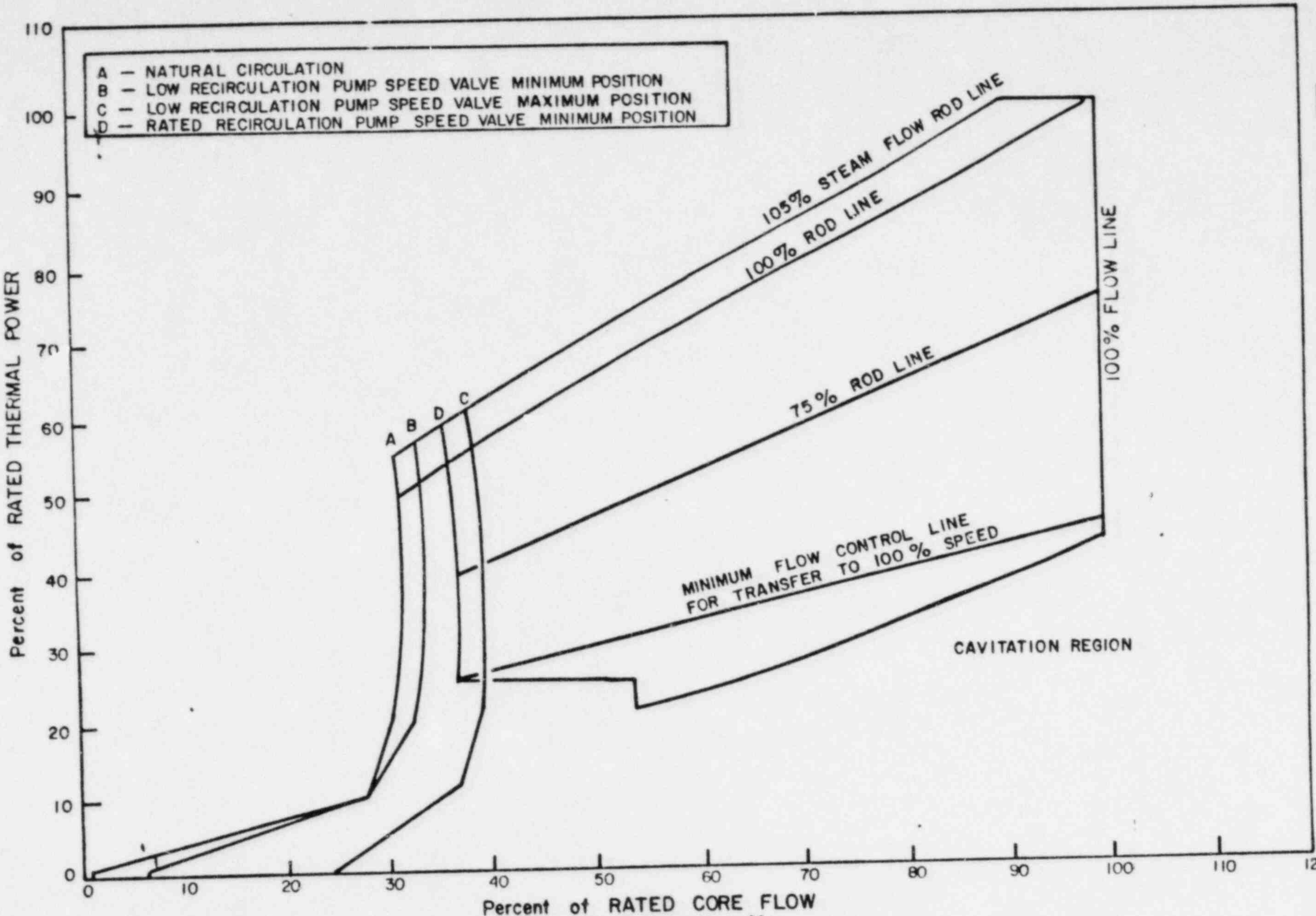
~~The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level.~~ 375

## Insert to Bases 3/4.2.3, Page B 3/4 2-4

The reference core flow increase event used to establish the $MCPR_f$ is a hypothesized slow flow runout to maximum, that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1 item 2). With this basis the $MCPR_f$ curve is generated from a series of steady state core thermal hydraulic calculations performed at several core power and flow conditions along the steepest flow control line. This corresponds to the 105% steamflow flow control line (Figure B 3/4 2.3-1). In the actual calculations a conservative highly steep generic representation of the 105% steamflow flow control line has been used. Assumptions used in the original calculations of this generic flow control line were consistent with a slow flow increase transient duration of several minutes: a) the plant heat balance was assumed to be in equilibrium, and b) core xenon concentration was assumed to be constant. The generic flow control line is used to define several core power/flow states at which to perform steady-state core thermal-hydraulic evaluations.

The first state analyzed corresponded to the maximum core power at maximum core flow (102.5% of rated) after the flow runout. Several evaluations were performed at this state iterating on the normalized core power distribution input until the limiting bundle MCPR just exceeded the safety limit Specification (2.1.2). Next, similar calculations of core MCPR performance were determined at other power/flow conditions on the generic flow control line, assuming the same normalized core power distribution. The result is a definition of the $MCPR_f$ performance requirement such that a flow increase event to maximum (102.5%) will not violate the safety limit. (The assumption of constant power distribution during the runout power increase has been shown to be conservative. Increased negative reactivity feedback in the high power limiting bundle due to doppler and voids would reduce the limiting bundle relative power in an actual runout.)

The $MCPR_p$ is established to protect the core from plant transients other than core flow increase including the localized rod withdrawal error event. Core power dependent setpoints are incorporated (incremental control rod withdrawal limits) in the Rod Withdrawal Limiter (RWL) System Specification (3.3.6). These setpoints allow greater control rod withdrawal at lower core powers where core thermal margins are large. However, the increased rod withdrawal requires higher initial MCPR's to assure the MCPR safety limit Specification (2.1.2) is not violated. The analyses that establishes the power dependent MCPR requirements that support the RWL system are presented in GESSAR II, Appendix 15B. Since the severity of other (core-wide) transients at off-rated conditions is limited by the requirement to setdown the APRM flow biased simulated thermal power-high scram trip setpoint, Specification (3.2.2), the rod withdrawal error is the limiting transient and establishes $MCPR_p$ requirements.




Power - Flow Operating Map

Figure B 3/4 2.3-1

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157
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MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. ~~The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.~~

Replace with Insert

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### 3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

Insert

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References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

3. Qualification of the One Dimensional Core Transient Model For Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program For The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

Insert to Bases 3/4.2.3, Page B 3/4 2-6

The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, that could place operation exceeding a thermal limit.

Insert to Bases 3/4.2.4, Page B 3/4 2-6

The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that, could place operation exceeding a thermal limit.

## 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 the SRMs and they can be retracted.

## 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 of the reactor core.

## 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.

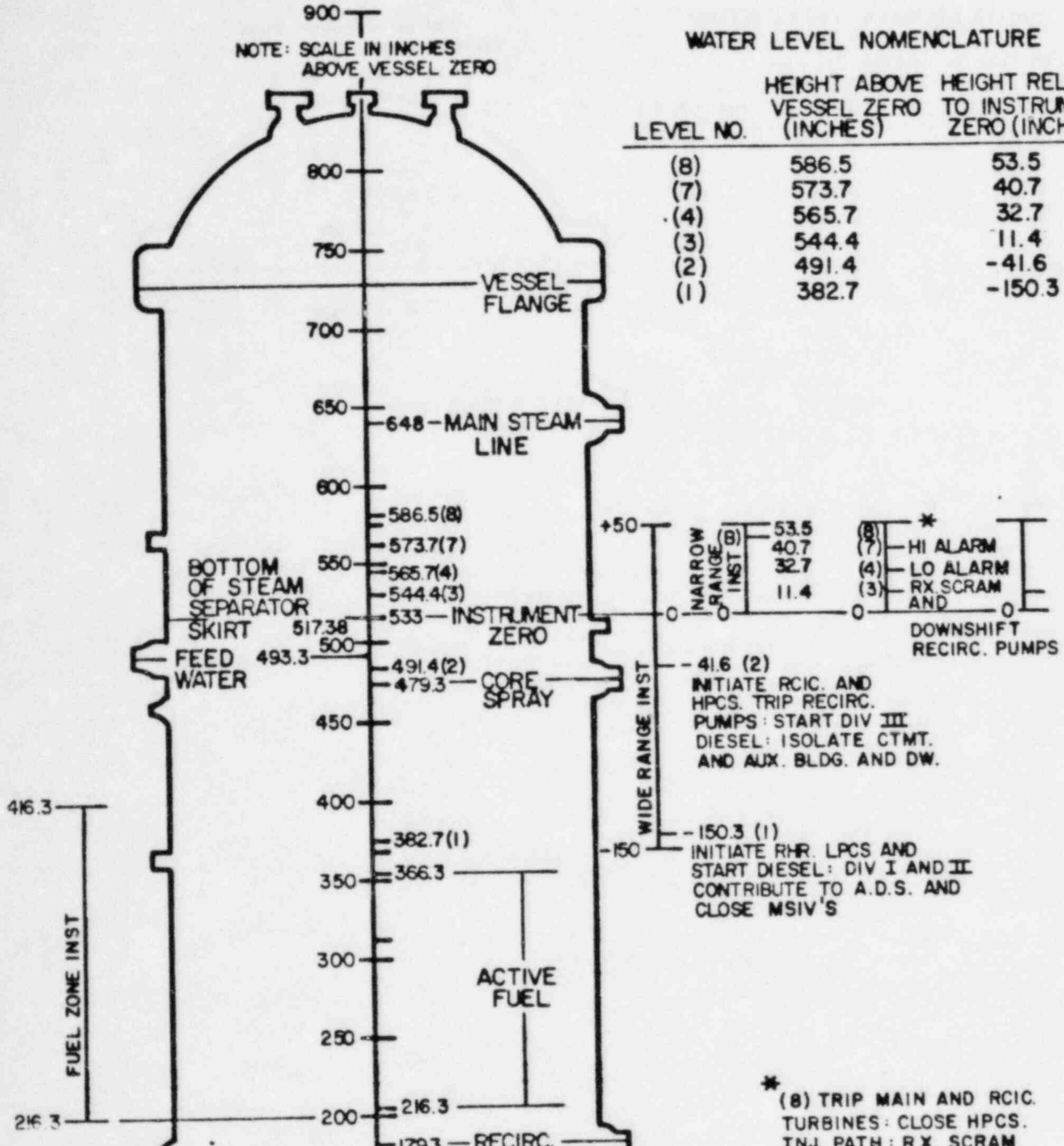
See Attached Insert

Insert for 3/4.3.7.10, Page B 3/4 3-5

The system consists of 16 sensors, of which only 8 are selected and need to be OPERABLE at a time, to provide the inputs to the 8 monitoring channels. The remaining 8 sensors may be used as replacement sensor inputs for failed sensors or to provide a change in location of the area being monitored.

## WATER LEVEL NOMENCLATURE

| LEVEL NO. | HEIGHT ABOVE VESSEL ZERO (INCHES) | HEIGHT RELATIVE TO INSTRUMENT ZERO (INCHES) |
|---|---|---|
| (8) | 586.5 | 53.5 |
| (7) | 573.7 | 40.7 |
| (4) | 565.7 | 32.7 |
| (3) | 544.4 | 11.4 |
| (2) | 491.4 | -41.6 |
| (1) | 382.7 | -150.3 |

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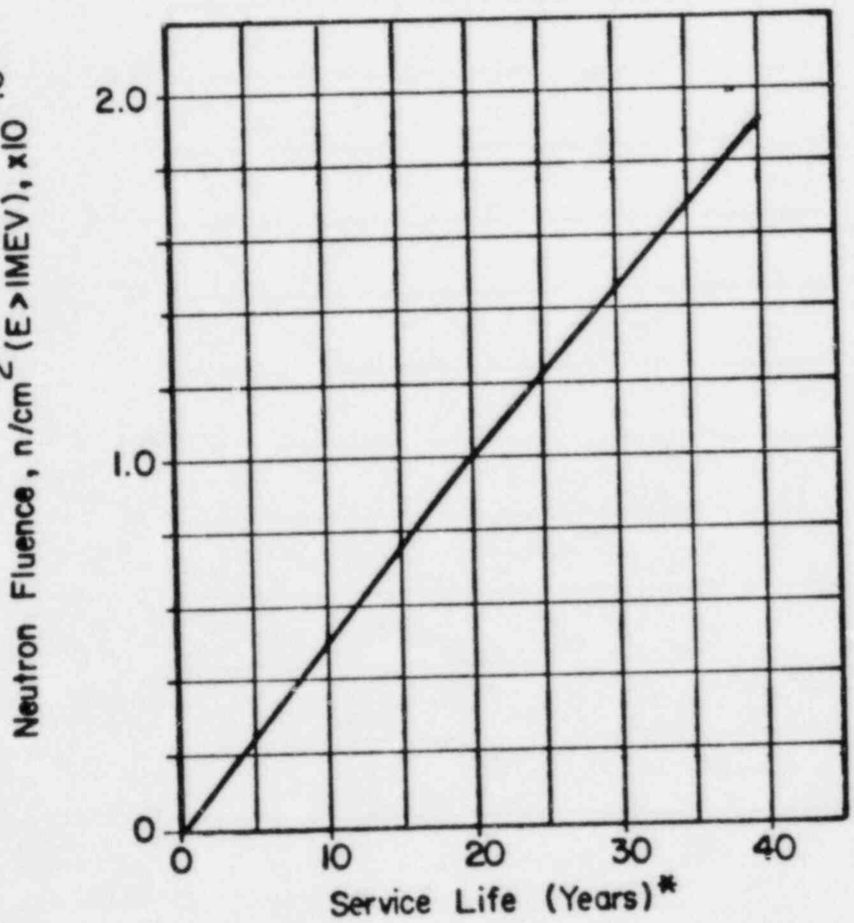
Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL




Fast Neutron Fluence (E>1 MeV) at 1/4 T As a Function of Service Life *

Bases Figure B 3/4.4.6-1

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*At 90% of RATED THERMAL POWER and 90% availability.

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a design nominal active fuel length of 150 inches. The initial core loading shall have a ~~maximum average enrichment of 1.70 weight~~ design nominal enrichment of 1.708 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. ~~and shall have a maximum average enrichment of 2.82 weight percent U-235~~. 182

### CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 193 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing ~~113~~ a design nominal 143.7 inches of boron carbide, $B_4C$, powder surrounded by a cruciform shaped stainless steel sheath. 182

## 5.4 REACTOR COOLANT SYSTEM

### DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

### VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal $T_{ave}$ of 533°F.

ATTACHMENT 2

PROPOSED CHANGES TO THE
GRAND GULF NUCLEAR STATION
TECHNICAL SPECIFICATIONS

NRC TECHNICAL REVIEW BRANCH: POWER SYSTEMS

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

| TSPS No. | Item Nos.* |
|---|---|
| 007 | 2.D.01 |
| 026 | 2.A.01 |
| 043 | 2.C.01 |
| 060 | 2.D.02 |
| 137 | 2.B.01 |
| 145 | 2.D.03 |
| 148 | 2.D.04 |
| 174 | 2.A.02 |
| 175 | 2.D.07 |
| 177 | 2.A.03 |
| 179 | 2.A.04 |
| 180 | 2.C.02 |
| 181 | 2.C.03 |
| 227 | 2.D.05 |
| 228 | 2.D.06 |
| 302 | 2.C.04 |
| 342 | 2.C.05 |

*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:

- 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:

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 page for exact change proposed.)

| | TSPS No. | TS Page No. |
|---|---|---|
| 1. | 026 | 3/4 8-4 |

CLARIFICATIONS

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

2. (TSPS 174), Diesel Generator Useable Fuel, Bases Section 3/4.8.1, 3/4.8.2, and 3/4.8.3

   The proposed change is purely administrative to add a sentence to Bases section 3/4.8.1, 3/4.8.2, and 3/4.8.3 in order to clarify that the values specified for each Diesel Fuel Storage System represent useable fuel and not the minimum number of gallons in each fuel storage tank. This change is for clarification only and does not affect plant operation or safety. (Page B 3/4 8-1)

3. (TSPS 177), DC Sources Action Statement, Technical Specification 3.8.2.2

   A revision to Action Statement "c" of the subject Technical Specification is requested to clarify the required actions associated with inoperable full capacity charger(s). The proposed change clarifies the intent of the specification to assure that if any of the required chargers are inoperable, the associated battery bank will be verified operable. As such, this is purely an administrative change. (Page 3/4 8-14)

4. (TSPS 179), Valve Designation on RCIC Turbine, Technical Specification Table 3.8.4.2-1

   The proposed change is purely administrative to clarify specifically that the valve listed in the Specification as "Valve on Turbine Q1E51C002" is actually the "RCIC Trip and Throttle Valve on Turbine Q1E51C002." (Page 3/4 8-39)

B. TECHNICAL SPECIFICATION/AS-BUILT PLANT CONSISTENCY

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

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

- 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 137), Channel Functional Test Requirements of the MOV Thermal Overloads, Technical Specification 3/4.8.4.2

   The proposed change is to alter Surveillance Requirement 4.8.4.2.1.a to allow the full channel test for the ECCS portion of the channel to be performed once per 18 months. The current wording of Surveillance 4.8.4.2.1.a requires this test be performed every 92 days. Performance of this Surveillance requires the injection of a simulated signal into the LOCA signal transmitter, causing a false ECCS actuation signal disrupting plant operation every 92 days. The proposed change will retain the test of the individual valve bypass circuitry every 92 days and the ECCS portions of the channel every 18 months. This change provides an enhancement which is consistent with current surveillance philosophy with respect to individual circuitry and channel OPERABILITY requirements of similar logic arrangements and adequately assures system OPERABILITY. By reducing the number of false ECCS actuation signals disrupting plant operations, this change improves plant safety. (Page 3/4 8-38)

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 043), Diesel Generator Fuel Oil Pump Surveillance, Technical Specification 3/4.8.1.1

   The proposed change is purely administrative to delete Surveillance Requirement 4.8.1.1.2.d.14, since the existing Surveillance Requirement 4.8.1.1.2.a.3, requires a more frequent test of the fuel transfer system every 31 days, and since surveillance requirement 4.8.1.1.2.d.14 is written for a design that contains cross-connects between fuel storage tanks, which the GGNS design does not utilize. Therefore, this change will not result in a decrease in the diesel generator Surveillance Requirements nor affect plant operation or safety and is consistent with GGNS design. (Page 3/4 8-7)

2. (TSPS 180), Reactor Protection System (RPS) Electric Power Monitoring, Technical Specification 3/4.8.4.3

   These changes revise the over-voltage and under-voltage setpoint values for Bus A and Bus B of the protective instrumentation in Surveillance Requirement 4.8.4.3.b. The changes are made to reflect system design by accounting for line voltage drops to determine precise over-voltage and under-voltage setpoints. These new setpoints meet the voltage requirements of individual components served by RPS Bus A and Bus B. These changes are enhancements to plant operation and are made to prevent unnecessary or spurious trips to the reactor protection system power supply which could cause unwarranted challenges to the reactor protection system, thereby improving plant safety. (Page 3/4 8-46)

3. (TSPS 181), Reactor Protection System (RPS) Electric Power Monitoring, Technical Specification 3/4.8.4.3 & Bases 3/4.8.4

Surveillance Requirement 4.8.4.3.a is changed to require a CHANNEL FUNCTIONAL TEST of the RPS electric power monitoring assemblies only when the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months. Testing conducted during cold shutdown provides adequate assurance that the assemblies will operate when required. This change will lessen the potential of an accidental reactor trip from power as a result of switching to the alternate power supply during testing. The BASES are changed to add a clarification of the purpose of the RPS electric power monitoring assemblies. The change to Specification 3/4.8.4.3 is an enhancement but does not change the intent of the technical specification. By reducing the likelihood of unnecessary challenges to safety equipment following accidental trips, the change improves plant safety. (Page 3/4 8-46 and B 3/4 8-3)

4. (TSPS 302), Electrical Equipment Protective Devices, Technical Specification 3/4.8.4.1 and Table 3.8.4.1-1

The change to Surveillance 4.8.4.1, Section a.2 is proposed to clarify that when circuit breakers are inoperable, they shall be restored to OPERABLE prior to resuming operation of the affected equipment. As presently worded, the surveillance requirement could be misinterpreted to imply that plant operation could not resume if a circuit breaker is found inoperable. The 125 VDC and 120 VAC circuit breakers have been added to Table 3.8.4.1-1 only to ensure that appropriate surveillances are performed to detect degradation of these devices over the life of the plant. These surveillances are required because of the possibility of these circuits developing sufficient fault current to cause damage to containment penetrations. The response time of the 480 VAC K600S type circuit breakers in Technical Specification Table 3.8.4.1-1 is increased from 0.05 seconds to 0.07 seconds. This change is necessary to ensure that the response time is long enough to include all breaker trips when subjected to a test current of 120% of the instantaneous trip setpoint. The Gould time - current characteristic curves for this type of breaker indicate that a response time of less than .07 seconds ensures a satisfactory breaker response test. Also as shown in FSAR figure 40.5-7 (No. 4/0 AWG penetration cable time-current characteristic curves), a response time of 0.07 seconds is much lower than the maximum fault current versus time limit of .52 seconds which the penetration and cable must not exceed to avoid possible degradation. The change in response time is therefore an enhancement to the technical specifications and involves no safety significance. The change to Surveillance 4.8.4.1, Section a.2 and the addition of the 125 VDC and 120 VAC circuit breakers provides an enhancement to plant safety in that it clarifies the intent of the surveillance and adds more stringent requirements to the technical specifications. (Page 3/4 8-20, 3/4 8-21, 3/4 8-37a, 3/4 8-37b, 3/4 8-37c, 3/4 8-37d, 3/4 8-37e, 3/4 8-37f, 3/4 8-37g, 3/4 8-37h, and 3/4 8-37i)

5. (TSPS 342), High Pressure Core Spray (HPCS) Diesel Generator Testing, Technical Specification 3/4.8.1.1

The proposed change is to add the "*" footnote to ACTION statements a., b., d., and e. of Specification 3.8.1.1 and to the bottom of the affected pages, in order to clearly indicate that Diesel Generator 13 testing per Surveillance Requirement 4.8.1.1.2.a.4 is required only when the HPCS system is operable as Diesel General 13 supplies power only to the HPCS system. This is an operational enhancement which clarifies the specification and prevents unnecessary testing of Diesel Generator 13. By decreasing the number of unnecessary diesel generator tests for Diesel Generator 13, plant safety is enhanced. (Pages 3/4 8-1 and 3/4 8-2)

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 007) Diesel Generator Air Starts, Technical Specification 3/4.8.1.1

   The proposed change deletes Surveillance Requirement 4.8.1.1.2.d.13 which contains a requirement to verify that the specified diesel generators start at least 5 times with the starting air receivers pressurized to less than or equal to 256 psig and the compressors secured. The 5 start test of the diesel is an initial preoperational test (already performed at GGNS) performed to verify that the air start receivers are sized correctly. As such this verification is not required periodically. Therefore, its removal does not adversely impact plant safety. In addition, these proposed changes are consistent with proof and review comments provided to MP&L by the NRC staff. (Page 3/4 8-6)

2. (TSPS 060), Diesel Generator Requirement, (A.C. Sources - Shutdown) Technical Specification 3/4.8.1.2

   AECM-83/0565, item 11, previously submitted proposed changes to delete the "and" condition for Diesel Generators 11 "and/or" 12 in Specification 3.8.1.2.b, and to delete the "or" condition for Diesel Generators 11 "and/or" 12 in line 2 of ACTION statement a., as well as the superfluous phrase "of the above required A.C. electrical power sources." Additional proposed changes are to delete "each" from 3.8.1.2.b.2.a, to replace the "and" condition with an "or" condition in Specification 3.8.1.2.b.2.a, and to delete the

superfluous phrase "of the above required A.C. electrical power sources" in ACTION statement b. These proposed changes provide clarification to the specification and are consistent with specification 3.5.2 and bases 3/4.8.1, 3/4.8.2 and 3/4.8.3. The proposed changes are consistent with NRC guidance and are also consistent with the requirements for offsite AC power sources of Specification 3.8.1.2, DC power sources of Specification 3.8.2.2, and onsite power distribution Specification 3.8.3.2. In addition, these proposed changes are consistent with proof and review comments provided to MP&L by the NRC staff and improve plant safety by clearly specifying the appropriate requirement. (Page 3/4 8-9)

3. (TSPS 145), Electrical Power Systems, Technical Specification 3/4.8.1.1

   This change deletes Surveillance Requirement 4.8.1.1.2.d.6, addressing simulated loss of diesel generators with offsite power not available. Generic Letter No. 83-30 identified this Surveillance Requirement as being in excess of the scope of General Design Criterion 17 (Appendix A, 10CFR50). With this change the technical specification requirement will continue to be in conformance with General Design Criterion 17, Regulatory Guide 1.108, and the NRC Standard Review Plan. (Page 3/4 8-5)

4. (TSPS 148), Turbine Overspeed Protection System, Technical Specification 3/4.3.9 (New Specification)

   A revision to the GGNS Technical Specification is requested to add a new Specification 3/4.3.9 and associated Bases addressing the turbine overspeed protection system. The new specification is necessary to ensure continued conformance with the requirements of General Design Criterion 4 (Appendix A, 10 CFR 50) and the guidance provided in Standard Review Plan 10.2. The proposed change adds Limiting Conditions for Operations, Action Statements, and Surveillance Requirements for the turbine overspeed protection system along with the bases for this protection system. The surveillance testing intervals in Specification 3/4.3.9 follow vendor recommendations. The 40 month interval for disassembling the stop and control valves is assumed to begin when the turbine is first placed on line with nuclear steam. This proposed change helps assure that plant response will be consistent with the current safety analysis. The proposed specification has been evaluated and conforms to GGNS plant design. In addition, these proposed changes are consistent with proof and review comments provided to MP&L by the NRC staff. (Page 3/4 3-101 and B 3/4 3-6)

5. (TSPS 227), Battery Charger Surveillance Requirement, Technical Specification 3/4.8.2.1

   This proposed change revises Surveillance Requirement 4.8.2.1.c.4.a to require that the battery charger is to supply 400 amperes at a minimum of 125 volts for at least 10 hours. For Division 3, Surveillance Requirement 4.8.2.1.c.4.b is changed to require that the battery charger is to supply 50 amperes at a minimum of 125 volts for 3 hours. This change provides more stringent testing requirements for the

battery chargers and is in conformance with associated requirements in the FSAR. In addition, these proposed changes are consistent with proof and review comments provided to MP&L by the NRC staff. (Page 3/4 8-11)

6. (TSPS 228), MOV Thermal Overload Protection, Table 3.8.4.2-1

These changes to Table 3.8.4.2-1 delete the divisional power identifiers for the E51 and B21F065 and B21F098 valves (of the RCIC and Nuclear Boiler Systems, respectively) identified on page 3/4 8-39, since this identifier is not a part of the valve number. Also, the "A" and "B" designation on the valve numbers for the Drywell Monitoring System (D23) valves are being deleted because they were erroneously indicated as part of the valve number. These are purely administrative changes to achieve consistency within the Table and with the plant valve identification. No Technical Specification requirements are affected by these changes. Therefore, the proposed changes have no adverse effects on safety. In addition, these proposed changes are consistent with proof and review comments provided to MP&L by the NRC staff. (Page 3/4 8-39)

7. (TSPS 175), Diesel Generators Surveillance Requirement, Technical Specification 3/4.8.1.1

The proposed change to ACTION statements a and d will decrease, from three hours to two, the time required to demonstrate OPERABILITY of the operable diesel generators when less than the minimum number of A.C. Power Sources required by the technical specification are OPERABLE. Also, the requirement to perform Surveillance Requirement 4.8.1.1.2.a.4 within one hour in ACTION statement e. is changed to two hours, to provide consistency with the above change to the specification. These changes to ACTION a. and d. will reduce the time allowed for operation before verifying a sufficient number of A.C. power sources are available, but will still allow sufficient time to perform the required surveillance. The proposed change is in compliance with NRC guidance, represents a more stringent surveillance requirement and promotes consistency throughout the specification. (Pages 3/4 8-1 and 3/4 8-2)

E. PROPOSED TECHNICAL SPECIFICATION CHANGES

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

INSTRUMENTATION

## 3/4.3.9 TURBINE OVERSPEED PROTECTION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.3.9 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam line inoperable and/or with one stop valve or one control valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam line. Otherwise isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

### SURVEILLANCE REQUIREMENTS

4.3.9.1 The provisions of Specification 4.0.4 are not applicable.

4.3.9.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 14 days by cycling each of the following valves through at least one complete cycle from the running position using the manual test or Automatic Turbine Tester (ATT):
   1) Four high pressure turbine stop valves,
   2) Four high pressure turbine control valves.
   3) Six low pressure turbine stop valves, and
   4) Six low pressure turbine control valves.

b. At least once per 14 days by testing of the two mechanical overspeed devices using the Automatic Turbine Tester or manual test.

c. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

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## 3/4.8 ELECTRICAL POWER SYSTEMS

### 3/4.8.1 A.C. SOURCES

### A.C. SOURCES - OPERATING

### LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

   1. Separate day fuel tanks containing a minimum of 220 gallons of fuel.

   2. A separate fuel storage system containing a minimum of:

      a) 48,000 gallons of fuel each for diesel generators 11 and 12, and

      b) 39,000 gallons of fuel for diesel generator 13.

   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator 11 or 12 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4* for one diesel generator at a time, within ~~three~~ two hours and at least once per 8 hours therafter; restore at least two offsite circuits and diesel generators 11 and 12 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 one offsite circuit and diesel generator 11 or 12 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4* for one diesel generator at at time, within two hours and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators 11 and 12 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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*4.8.1.1.2.a.4 must be performed for Diesel Generator 13 only when the HPCS system is operable.

## LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With either diesel generator 11 or 12 of the above required A.C. electrical power sources inoperable, in addition to ACTION a or b, above as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining diesel generator 11 or 12 as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4* for one diesel generator at a time, within ~~three~~ two hours and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. 342

e. With diesel generators 11 and 12 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a* and 4.8.1.1.2.a.4* within ~~one~~ two hours and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 11 and 12 to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators 11 and 12 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. 342/175

within one hour

f. With diesel generator 13 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4, for one diesel generator at a time, within two hours and at least once per 8 hours thereafter; 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 3.5.1.

* 4.8.1.1.2.a.4 must be performed for Diesel Generator 13 only when the HPCS system is operable. 342

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c. At least once per 92 days and from new oil prior to addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-~~1975~~ has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @ 40°C of greater than or equal to 1.9 but less than or equal to 4.1 when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70, except that the test of new fuel for impurity level shall be performed within 7 days after addition of the new fuel to the storage tank.

1965 (re-approved 1980)

d. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1200 kW (LPCS Pump) for diesel generator 11, greater than or equal to 550 kW (RHR B/C Pump) for diesel generator 12, and greater than or equal to 2180 kW (HPCS Pump) for diesel generator 13 while maintaining less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 7000 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13 without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

a) For Divisions 1 and 2:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. [DELETED]

~~6. Verifying that on a simulated loss of the diesel generator, with offsite power not available:~~

~~a. For Divisions 1 and 2:~~

~~1. The loads are shed from emergency busses associated with Diesel Generators 11 and 12.~~

~~2. Subsequent loading of the diesel generators is in accordance with design requirements.~~

~~b. For Division 3:~~

~~1. The associated output breaker for Diesel Generator 13 opens automatically.~~

~~2. Subseqent loading of the diesel generator is in accordance with design requirements.~~

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7. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

a) For Divisions 1 and 2:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads within 10 seconds and the autoconnected emergency loads within 20 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

8. Verifying that all automatic diesel generator trips are automatically bypassed upon an ECCS actuation signal except:
   a) For Divisions 1 and 2, engine overspeed, generator differential current, low lube oil pressure, and generator ground overcurrent.
   b) For Division 3, engine overspeed and generator differential current.
9. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 7700 kW for diesel generators 11 and 12 and 3630 kW for diesel generator 13 and during the remaining 22 hours of this test, the diesel generator shall be loaded to 7000 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.7.a).2) and b).2)*.
10. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 7000 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13.
11. Verifying the diesel generator's capability to:
    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
    b) Transfer its loads to the offsite power source, and
    c) Be restored to its standby status.
12. Verifying that with the diesel generator operating in a test mode and connected to its bus that a simulated ECCS actuation signal:
    a) For Divisions 1 and 2, overrides the test mode by returning the diesel generator to standby operation.
    b) For Division 3, overrides the test mode by bypassing the diesel generator automatic trips per Surveillance Requirement 4.8.1.1.2.d.8.b).

[DELETED]

13. ~~Verifying that with all diesel generator air start receivers pressurized to less than or equal to 256 psig and the compressors secured, the diesel generator starts at least 5 times from ambient conditions and accelerates 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.~~ 007

*If Surveillance Requirement 4.8.1.1.2.d.4.a)2) or b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at rated load for one hour or until operating temperatures have stabilized.

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.~~ [DELETED] 043

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 pursuant to Specification 6.9.1. 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.

## 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 or ~~and~~ 12.
      b) 39,000 gallons of fuel for diesel generator 13.
   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

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

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

## SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

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

1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
2. Total battery terminal voltage is greater than or equal to 129-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,
2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than $150 \times 10^{-6}$ ohms, and
3. The average electrolyte temperature of every sixth connected cells is above 60°F.

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

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,
2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,
3. The resistance of each cell and terminal connection is less than or equal to $150 \times 10^{-6}$ ohms, and
4. The battery charger will supply:

   a) For Divisions 1 and 2, at least 400 amperes at a minimum of ~~105~~ 125 volts for at least ~~2~~ 10 hours.

   b) For Division 3, at least 50 amperes at a minimum of ~~105~~ 125 volts for at least ~~2~~ 3 hours.

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AMENDMENT No. —

## D.C. SOURCES - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division 1 or Division 2, and, when the HPCS system is required to be OPERABLE, Division 3, of the D.C. electrical power sources shall be OPERABLE with:

a. Division 1 consisting of:
   1. 125 volt battery 1A3.
   2. 125 volt full capacity charger 1A4 or 1A5.

b. Division 2 consisting of:
   1. 125 volt battery 1B3.
   2. 125 volt full capacity charger 1B4 or 1B5.

c. Division 3 consisting of:
   1. 125 volt battery 1C3.
   2. 125 volt full capacity charger 1C4.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With both Division 1 battery and Division 2 battery of the above required D.C. electrical power sources inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment and operations with a potential for draining the reactor vessel.

b. With Division 3 battery of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. With any of the above required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour and at least once per 8 hours thereafter. If any Category A limit in Table 4.8.2.1-1 is not met, declare the battery inoperable.

d. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. For the lower voltage circuit breakers the nominal trip setpoint and short circuit response times are listed in Table 3.8.4.1-1. Testing of these circuit breakers shall consist of injecting a current in excess of 120% of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

of the affected equipment.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

## TABLE 3.8.4.1-1

## PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

| DEVICE NUMBER AND LOCATION | TRIP SETPOINT (Amperes) | RESPONSE TIME (Cycles) | SYSTEM/ COMPONENT AFFECTED |
|---|---|---|---|
| **a. 6.9 kV Circuit Breakers** | | | |
| 252-1103-B | 7200/45/± 10%# | 60 | Reactor Recir. Pump |
| 252-1103-C | 7200/45/± 10%# | 60 | Pump B33C001A |
| 252-1205-B | 7200/45/± 10%# | 60 | Reactor Recir. Pump |
| 252-1205-C | 7200/45/± 10%# | 60 | Pump B33C001B |

b. 480 VAC Circuit Breakers

Stored Energy Type K600S with SS3G3 Tripping Device

| BREAKER NUMBER | TRIP SETPOINT (Amperes) | RESPONSE TIME (Seconds) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-12202 | 1200 | ~~0.05~~ 0.07 | CONTAINMENT COOLING FILTER TRAIN HEATERS (N1M41D002B-N) |
| 52-12209 | 2000 | ~~0.05~~ 0.07 | CNTMT POLAR CRANE (Q1F13E001-N) |
| 51-11502 | 1200 | ~~0.05~~ 0.07 | CNTMT CLG. FILTER TRAIN HEATER (N1M41D002A-N) |
| 52-15105 | 2000 | ~~0.05~~ 0.07 | DRYWELL PURGE COMPRESS. (Q1E61C001A-A) |
| 52-16204 | 2000 | ~~0.05~~ 0.07 | DRYWELL PURGE COMPRESS. (Q1E61C001B-B) |
| 52-16404 | 1200 | ~~0.05~~ 0.07 | HYDROGEN RECOMBINER (Q1E61C003B-B) |
| 52-15205 | 1200 | ~~0.05~~ 0.07 | HYDROGEN RECOMBINER (Q1E61C003A-A) |

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#Primary current/setpoint.

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(d) 125V DC BREAKERS
GE-E-150 LINE TYPE THED

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 72-11A-23 | 30 | 5.0 | AUTOMATIC DEPRESSURIZATION SYSTEM VALVES |
| 72-11A-28 | 15 | 5.0 | REMOTE SHUTDOWN PANEL/AUTOMATIC DEPRESSURIZATION SYSTEM VALVES |
| 72-11A-30 | 15 | 5.0 | REACTOR PROTECTION SYSTEM/BACKUP SCRAM VALVE |
| 72-11A-33 | 15 | 5.0 | CONTAINMENT & DRYWELL ISOLATION SYSTEM ANNUNCIATION |
| 72-11A-38 | 15 | 5.0 | RESIDUAL HEAT REMOVAL SYSTEM VALVES |
| 72-11B-14 | 50 | 5.0 | RESIDUAL HEAT REMOVAL SYSTEM |
| 72-11B-28 | 15 | 5.0 | REMOTE SHUTDOWN PANEL/ADS VALVES |
| 72-11B-30 | 15 | 5.0 | REACTOR PROTECTION SYSTEM/ BACKUP SCRAM VALVE |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(d) 125V DC BREAKERS
GE-E-150 LINE TYPE THED

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 72-11B-34 | 30 | 5.0 | AUTOMATIC DEPRESSURIZATION SYSTEM VALVES |
| 72-11B-37 | 15 | 5.0 | CONTAINMENT & DRYWELL ISOLATION SYSTEM |
| 72-11D-39 | 15 | 5.0 | CONTAINMENT PURGE ISOLATION VALVE F010 |
| 72-11D-71 | 15 | 5.0 | CHARCOAL FILTER TRAIN N1M41D002A-N ALARMS |
| 72-11D-72 | 15 | 5.0 | FLOOR & EQUIPMENT DRAIN SYSTEM |
| 72-11D-73 | 15 | 5.0 | CONDENSATE AND REFUELING WATER STORAGE AND TRANSFER SYSTEM |
| 72-11D-79 | 15 | 5.0 | FILTER DEMIN CONT. VB G36-P002 |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(d) 125V DC BREAKERS
GE-E-150 LINE TYPE THED

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 72-11E-36 | 15 | 5.0 | FIRE PROTECTION PANEL |
| 72-11E-69 | 15 | 5.0 | FLOOR & EQUIPMENT DRAIN SYSTEM |
| 72-11E-73 | 15 | 5.0 | CHARCOAL FILTER TRAIN N1M41D002B-N ALARM |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS
GE TYPE TQB

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P112-12 | 40 | 4.0 | RWCU REACTOR SAMPLE STATION (CONSTANT TEMP BATH) |
| 52-1P112-13 | 35 | 4.0 | RWCU SYSTEM FILTER DEMON CONT |
| 52-1P112-14 | 15 | 4.0 | CONT POWER SUPPLY NSSSS (1G33TSN008) |
| 52-1P112-17 | 15 | 4.0 | RWCU REACTOR SAMPLE STATION (INST POWER) |
| 52-1P112-20 | 15 | 4.0 | RWCU SYS DUSTER COLLECTOR TANK (N1G36D016) |
| 52-1P112-22 | 15 | 4.0 | RWCU SYS RESIN PUMP (N1G36C003-N) |
| 52-1P112-23 | 15 | 4.0 | AREA RAD MONIT SYSTEM CTMT BLDG. ALARMS |
| 52-1P151-20 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYSTEM (N1B33D003A1-N) |

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## TABLE 3.8.4.1-1 (Continued)

## PRIMARY CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS (Continued)
GE TYPE TQB

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P151-22 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYSTEM (N1B33D003A2-N) |
| 52-1P151-23 | 15 | 4.0 | CTMT CLG SYSTEM CHARCOAL FLTR TRAIN HEATER (N1M41D002A-N) |
| 52-1P151-24 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYS (N1B33D003A3-N) |
| 52-1P151-25 | 15 | 4.0 | MAIN STEAM PIPING AREA DRWL COOLER SERVICE WATER CONT. TRANSMITTER (TT-N041) |
| 52-1P151-26 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYSTEM (N1B33D003A4-N) |
| 52-1P151-37 | 15* | 4.0 | DRWL PERSONNEL LOCK (120'-10" ELEV) |
| 52-1P151-38 | 15* | 4.0 | CTMT PERSONNEL LOCK (LOWER) |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS (Continued)
GE TYPE TQB

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P222-17 | 15 | 4.0 | CTMT CLG SYSTEM CHARCOAL FLT TRAIN HTR (N1M41D002B-N) |
| 52-1P222-24 | 15 | 4.0 | CTMT & DRWL PERSONNEL AIR LOCK MONITORING SYSTEM IN CONT ROOM |
| 52-1P222-27 | 15 | 4.0 | DRWL COOLERS SERVICE WATER CONT TRANSMITTER (TT - N044) |
| 52-1P251-13 | 15 | 4.0 | PUMP VALVE SOLENOID CONT CKT & TEMPERATURE FOR REACTOR WATER CLEAN UP SYS |
| 52-1P251-37 | 15 | 4.0 | CONTAINMENT EQUIP HATCH (Q1M23Y007-1) |
| 52-1P251-38 | 15* | 4.0 | CONTAINMENT EQUIP HATCH (Q1M23Y007-2) |

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## PRIMARY CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS (Continued)
GE TYPE TQB

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P411-19 | 15 | 4.0 | PLANT SERVICE WATER SYS CONTROL VALVE INDICATION (1P44ZLR001) |
| 52-1P412-22 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYS (N1B33D003B1-N) |
| 52-1P412-23 | 20 | 4.0 | UTILITY POWER FOR REMOTE SIGNAL CONDITIONING PANEL |
| 52-1P412-24 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYS (N1B33D003B2-N) |
| 52-1P412-25 | 20 | 4.0 | UTILITY POWER FOR REMOTE SIGNAL CONDITIONING PANEL |
| 52-1P412-26 | 15 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYS (N1B33D003B3-N) |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS (Continued)
GE TYPE TQB

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P412-28 | 16 | 4.0 | MOTOR SPACE HEATER FOR REACTOR RECIRC SYS (N1B33D003B4-N) |
| 52-1P511-10 | 15 | 4.0 | MOTOR SPACE HEATER FOR DRYWL PURGE COMPRESSOR (Q1E61C001A-A) |
| 52-1P511-21 | 15 | 4.0 | MOTOR SPACE HEATER FOR SLCS (Q1C41C001A-A) |
| 52-1P531-19 | 30 | 4.0 | HYDROGEN IGNITOR CONTROL |
| 52-1P531-21 | 30 | 4.0 | HYDROGEN IGNITOR CONTROL |
| 52-1P621-25 | 15 | 4.0 | MOTOR SPACE HEATER FOR DRWL PURGE COMPRESSOR (Q1E61C001B-B) |
| 52-1P631-15 | 30 | 4.0 | HYDROGEN IGNITOR CONTROL |
| 52-1P631-17 | 30 | 4.0 | HYDROGEN IGNITOR CONTROL |

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

(e) 208/120V AC CIRCUIT BREAKERS (Continued)
GE TYPE TQB/TQL

| BREAKER NO. | TIME O.C. PICKUP (AMPERES) | RESPONSE TIME (SECONDS) | SYSTEM/COMPONENT AFFECTED |
|---|---|---|---|
| 52-1P631-21 | 15 | 4.0 | MOTOR SPACE HEATER FOR SLCS (Q1C41C001B-B) |
| 52-1DP641-07** | 15 | 4.0 | CTMT CLG SMOKE DETECTOR POWER SUPPLY |

* 3 Pole Breaker
** GE Type TQL

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## 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).

### 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. For ~~At least once per 92 days for~~ those thermal overloads which are normally in force during plant operation and bypassed under accident conditions: 1. At least once per 92 days for the individual valve bypass circuitry.
2. At least once per 18 months for the ECCS portion of the channel. 137

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.

## TABLE 3.8.4.2-1

## MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

| VALVE NUMBER | BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO) | SYSTEM(S) AFFECTED | |
|---|---|---|---|
| Q1E51F010~~-A~~ | Continuous | RCIC System | |
| Q1E51F013~~-A~~ | Continuous | RCIC System | |
| Q1E51F019~~-A~~ | Continuous | RCIC System | |
| Q1E51F022~~-A~~ | Continuous | RCIC System | |
| Q1E51F031~~-A~~ | Continuous | RCIC System | |
| Q1E51F045~~-A~~ | Continuous | RCIC System | 228 |
| Q1E51F046~~-A~~ | Continuous | RCIC System | |
| Q1E51F059~~-A~~ | Continuous | RCIC System | |
| Q1E51F068~~-A~~ | Continuous | RCIC System | |
| Valve on Turbine Q1E51C002 RCIC TRIP AND THROTTLE | Continuous | RCIC System | 178 |
| Q1B21F065A~~-A~~ | No | Reactor Coolant System | |
| Q1B21F065B~~-A~~ | No | Reactor Coolant System | |
| Q1B21F098A~~-B~~ | No | Reactor Coolant System | 228 |
| Q1B21F098B~~-B~~ | No | Reactor Coolant System | |
| Q1B21F098C~~-B~~ | No | Reactor Coolant System | |
| Q1B21F098D~~-B~~ | No | Reactor Coolant System | |
| Q1B21F019 | Continuous | Reactor Coolant System | |
| Q1B21F067A | Continuous | Reactor Coolant System | |
| Q1B21F067B | Continuous | Reactor Coolant System | |
| Q1B21F067C | Continuous | Reactor Coolant System | |
| Q1B21F067D | Continuous | Reactor Coolant System | |
| Q1B21F016 | Continuous | Reactor Coolant System | |
| Q1B21F147A | Continuous | MSL Drain Post LOCA Leakage Control | |
| Q1B21F147B | Continuous | MSL Drain Post LOCA Leakage Control | |
| Q1B33F019 | Continuous | Recirculation System | |
| Q1B33F020 | Continuous | Recirculation System | |
| Q1B33F125 | Continuous | Recirculation System | |
| Q1B33F126 | Continuous | Recirculation System | |
| Q1B33F127 | Continuous | Recirculation System | |
| Q1B33F128 | Continuous | Recirculation System | |
| Q1D23F591~~B~~ | * | Drywell Monitoring System | |
| Q1D23F592~~A~~ | * | Drywell Monitoring System | 228 |
| Q1D23F593~~B~~ | * | Drywell Monitoring System | |
| Q1D23F594~~A~~ | * | Drywell Monitoring System | |
| Q1E12F040 | Continuous | RHR System | |
| Q1E12F023 | Continuous | RHR System | |
| Q1E12F006A | Continuous | RHR System | |
| Q1E12F052A | Continuous | RHR System | |
| Q1E12F008 | Continuous | RHR System | |

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

## LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring system to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring assemblies for an inservice RPS MG set or alternate power supply inoperable; restore at least one electric power monitoring assembly to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

## SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring assemblies shall be determined OPERABLE:

a. ~~At least once per six months by performance of a CHANNEL FUNCTIONAL TEST, and~~ By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months. 181

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints:

1. Over-voltage ~~≤ 132 VAC,~~ Bus A ≤ 132.9 VAC, Bus B ≤ 133.0 VAC
2. Under-voltage ~~≥ 117 VAC, and~~ Bus A ≥ 115.0 VAC, Bus B ≥ 115.9 VAC 180
3. Under-frequency ~~≥ 57 Hz.~~ Bus A ≥ 57 Hz, Bus B ≥ 57 Hz

### 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.

### 3/4.3.9 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures. 148



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## 3/4.8 ELECTRICAL POWER SYSTEMS

## BASES

### 3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

(The following bases are developed for low power operation while diesel generator 11 is out of service for disassembly and inspection.)

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy these systems requirements for capacity, capability, and redundancy needed for safe plant shutdown. The volumes of fuel specified for each fuel storage system represent useable fuel. 174

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least Division 2 of the onsite A.C. or the gas turbine generator system and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. Division 3 supplies the high pressure core spray (HPCS) system only.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator 12 or gas turbine generator system is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator 12 or gas turbine generator system as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period diesel generator 12 or gas turbine generator system is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Regulatory Guide 1.137" Fuel-Oil Systems for Standby Diesel Generators", ~~Revision 1, October 1979.~~ JANUARY, 1978, as addressed in the FSAR. 174

## 3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturers brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are than tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The OPERABILITY or bypassing of the motor operated valve thermal overload protection continuously or under accident conditions by integral bypass devices ensures that the thermal overload protection during accident conditions will not prevent safety related valves from performing their function. The surveillance requirements for demonstrating the OPERABILITY or bypassing of the thermal overload protection continuously and or during accident conditions are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

The reactor protection system (RPS) electric power monitoring assemblies provide redundant protection to the RPS and other systems which receive power from the RPS buses by acting to disconnect the RPS from the power source circuits in the presence of an electrical fault in the power supply. The BASES for the functional requirements of the RPS are discussed in the BASES for Specification 3/4.3.1. 181