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Nuclear

10 CFR 50.90

TMI-11-103
October 18, 2011

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
<u>NRC Docket No. 50-289</u>

Subject: License Amendment Request to revise Technical Specifications to incorporate administrative changes

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to the Technical Specifications (TS) for Three Mile Island Nuclear Station, Unit 1 (TMI, Unit 1).

The proposed amendment incorporates several administrative changes. The administrative changes include correcting typographical errors, removing unwarranted formatting, and clarifying symbols/pages that may not copy/print well. Attachment 1 provides a description of the proposed changes. Attachment 2 provides the existing TS page mark-ups and Attachment 3 provides the revised pages showing the proposed changes.

There are no regulatory commitments contained in this letter.

The proposed changes have been reviewed by the TMI, Unit 1 Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendment by October 18, 2012. Once approved, the amendment shall be implemented within 60 days of issuance.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Wendy E. Croft at (610) 765-5726.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of October 2011.

Respectfully,

Michael D. Jesse
Director, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments: 1. Evaluation of Proposed Changes
2. Proposed Technical Specification Pages (Mark-Up)
3. Revised Technical Specification Pages

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, TMI, Unit 1
USNRC Senior Resident Inspector, TMI, Unit 1
Director, Bureau of Radiation Protection, PA Department of Environmental Resources
Chairman, Board of County Commissioners, Dauphin County, PA
Chairman, Board of Supervisors, Londonderry Township, PA
R. R. Janati, Commonwealth of Pennsylvania

ATTACHMENT 1

Evaluation of Proposed Changes

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50

Subject: License Amendment Request to revise Technical Specifications to incorporate administrative changes

1.0 SUMMARY DESCRIPTION

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1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI, Unit 1).

The proposed amendment incorporates several administrative changes. The administrative changes include correcting typographical errors, removing unwarranted formatting, and clarifying symbols/pages that may not copy/print well in the Technical Specifications (TS).

2.0 DETAILED DESCRIPTION

The following proposed changes to the TMI, Unit 1 TS are administrative in nature:

- The following Table of Contents (TOC) TS section references were revised to remove the all capitalization format and replace it with the initial capitalization format consistent with similar sections of the TOC: 1.9, 1.10, 1.11, 4.4.1, 4.4.2, 4.4.3, 4.4.4, 4.5.1, 4.5.2, 4.5.3, 4.5.4, 4.7.1, 4.8, 4.9.1, 4.9.2, 4.12.1, 4.12.4, 4.14, 5.2.1, 5.2.2, 5.3.1, 5.3.2, 5.4.1, 5.4.2, 6.2.1, 6.2.2, 6.5, 6.5.1, 6.5.2, 6.5.3, 6.5.4, 6.9.1, 6.9.2, 6.9.3, 6.9.4, 6.9.5, 6.9.6, 6.15, and 6.16. There were no changes to the content of the text in these sections.

- The following TOC TS section references were previously deleted in the TS TOC and have been revised to state only the word "Deleted" consistent with similar sections of the TOC: 1.14, 3.1.10, 3.5.4, 3.15.2, 3.15.3, 3.18, 3.20, 3.21, 3.21.1, 3.21.2, 3.22, 3.22.1, 3.22.2, 3.22.3, 3.22.4, 3.23, 3.23.1, 3.23.2, 3.23.3, 4.3, 4.7.2, 4.12.2, 4.12.3, 4.18, 4.21, 4.21.1, 4.21.2, 4.22, 4.22.1, 4.22.2, 4.22.3, 4.22.4, 4.23.1, 4.23.2, and 4.23.3.

- The following TOC TS section references were revised to remove the initial capitalization format and replace it with the all capitalization format consistent with similar sections of the TOC: 2.1, 2.2, 2.3, 3.0, 3.1, 3.3, 3.4, 3.5, 3.6, 3.7, 3.8, 3.10, 3.11, 3.12, 3.13, 3.14, and 3.15. There were no changes to the content of the text in these sections.

- The TOC Sections were revised by altering indentations and line breaks to be consistent with the following numbering format:

  <Line Break>
  1. <u>HEADING 1</u>
  1.1 <u>HEADING 2</u>
  1.1.1 Heading 3

- TOC TS section reference 1.2.10, on page i, was revised to state "T_{ave}," consistent with TS Section 1.2.10 on page 1-2.

- The amendment revision section on the footer of page i was revised to add strikethrough formatting. There were no changes to the content of the text in this section.

- A line break was added to the header of page ii to be consistent with similar sections of the TOC.

- TOC TS section references 3.2 and 3.9 on page ii were revised to remove underlining consistent with similar sections of the TOC. There were no changes to the content of the text in these sections.

- The "Table of Contents" header on page iii was revised to add underlining consistent with similar sections of the TOC. There were no changes to the content of the text in this section.

- A line break was removed from the header of page iii to be consistent with similar sections of the TOC.

- The amendment revision section on the footer of page iii was revised to add the text "No." consistent with similar sections of the TOC.

- TOC TS numerical reference 6.20 on page v was revised to match the formatting of the TOC.

- The page number on the footer of page v was revised to remove the two hyphens (-) on either side of the page number consistent with similar sections of the TOC.

- TS 1.2.1, 1.2.2, and 1.2.4.a on page 1-1 are edited to replace the term "Tavg" with "T_{ave}." This change utilizes subscript formatting to correctly reference the arithmetic average of the coolant temperature in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made, as defined in TS Section 1.2.10.

- TS 1.4.2, 1.4.3, and 1.4.4 on page 1-3 are edited to remove boldface formatting. The text in this section remains unchanged.

- TS Figure 2.1-2 on page 2-4b was deleted per Amendment 184; however, the page still exists in the TS with the word "DELETED" printed over the figure. Consistent with other figure deletions in TS (i.e., Figures 3.1-2a, 3.1-3, and 3.5-2a-l) page 2-4b is being replaced with a page containing only the words, "INFORMATION ON THIS PAGE HAS BEEN DELETED" and the footer.

- TS Figure 2.3-2 on page 2-12 was deleted per Amendment 184; however, the page still exists in the TS with the word "DELETED" printed over the figure. Consistent with other figure deletions in TS (i.e., Figures 3.1-2a, 3.1-3, and 3.5-2a-l) page 2-12 is being replaced with a page containing only the words, "INFORMATION ON THIS PAGE HAS BEEN DELETED" and the footer.

- TS 3.1.1 header "OPERATIONAL COMPONENTS" on page 3-1a is reformatted consistent with the TMI, Unit 1 TS format to add underlining.

- TS 3.1.2 header "PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS" on page 3-3 is reformatted consistent with the TMI, Unit 1 TS format to add underlining.

- TS 3.1.3 header "MINIMUM CONDITIONS FOR CRITICALITY" on page 3-6 is reformatted consistent with the TMI, Unit 1 TS format to add underlining.

- TS 3.1.9 header "LOW POWER PHYSICS TESTING RESTRICTIONS" on page 3-18 is reformatted consistent with the TMI, Unit 1 TS format to add underlining.

- TS 3.1.13 header "REACTOR COOLANT SYSTEM VENTS" on page 3-18g is reformatted consistent with the TMI, Unit 1 TS format to add underlining.

- The "*" in TS 3.1.13.1.a and applicable note on page 3-18g are deleted because this information is no longer applicable. This information applied only to installation and initial declaration of operability of RC-V42 and RC-V43 in 1984 via Amendment 97 and no longer applies.

- TS 3.3.1.1.a-g and 3.3.1.2.a on page 3-21 are edited to remove boldface formatting. The text in these sections remains unchanged.

- TS 3.3.1.2.b-e and the Header on page 3-22 are edited to remove boldface formatting. The text in these sections remains unchanged.

- The "*" in TS 3.3.2 and applicable note on page 3-23 are deleted because this information is no longer applicable. This information applied only to TMI, Unit 1 Cycle 13. The station is currently operating in Cycle 19.

- TS 3.5.2.1 and 3.5.2.2.a-c on page 3-33 are edited to remove the symbol "Δ" utilized as a symbol for the term "delta" and replace it with the written term "delta." This change is consistent with other sections of the TS. In addition, this editorial change minimizes confusion over the symbol used and potential printing/copying errors.

- The amendment revision section on the footer of page 3-33 was revised to remove the note, "(5-18-76)."

- TS 3.5.2.5.b.2 on page 3-35 is edited to remove the symbol "Δ" utilized as a symbol for the term "delta" and replace it with the written term "delta." This change is consistent with other sections of the TS. In addition, this editorial change minimizes confusion over the symbol used and potential printing/copying errors.

- TS 3.5.2.5.b.1 and 2 on page 3-35 are edited to remove an erroneous period (.).

- TS Figure 3.5-2M on page 3-36b was deleted per Amendment 168; however, the page still exists in the TS, with the word "DELETED" printed over the figure. To be consistent with other Figure deletions in TS (i.e., Figures 3.1-2a, 3.1-3, and 3.5-2a-l) page 3-36b is being replaced with a page containing only the words, "INFORMATION ON THIS PAGE HAS BEEN DELETED" and the footer.

- TS 3.6.3 on page 3-41 is edited to remove the symbol "Δ" utilized as a symbol for the term "delta" and replace it with the written term "delta." This change is consistent with other sections of the TS. In addition, this editorial change minimizes confusion over the symbol used and potential printing/copying errors.

- The "*" in TS 3.7.2.c and applicable note on page 3-43 are deleted because this information is no longer applicable. This information applied only to TMI, Unit 1 on April 2, 2006 and no longer applies.

- TS 3.15.4.2.b on page 3-62e is edited to remove boldface formatting. The text in this section remains unchanged.

- TS Table 4.1-1, Item 19.e on page 4-5 is edited to remove an erroneous remark for reference "(2)". A review of the TS identified that the reference was added in TMI, Unit 1 Amendment 200. The change request and associated NRC Safety Evaluation Report (SER) for Amendment 200 shows that the addition of the reference was not requested by the station or approved by the SER. In addition, a reference "(2)" remark does not exist on the affected TS page. Therefore, this erroneous TS remark reference is being removed.

- TS Table 4.1-1 Item 25.a on page 4-5 is edited to delete the word "Coolant" in the Channel Description. Item 25a is associated with the Core Flood Tank and the word "Coolant" is misleading and incorrectly used.

- In TS 4.5.3 on page 4-43 the word "Specification" is deleted from the "Objective" statement and is relocated as a heading on the next line. To match TS formatting the heading "Specification" has been underlined. A period (.) has been added to the "Objective" statement consistent with TS formatting.

- TS Figures 5-1 through 5-3 are updated to provided clearer drawings and revise the drawing title boxes to reflect Exelon.

A markup of the proposed administrative TS changes are provided in Attachment 2. Camera-ready pages showing the proposed changes are provided in Attachment 3.

3.0 TECHNICAL EVALUATION

Minor inconsistencies or editorial errors that have no safety impact have been introduced into the TMI, Unit 1 TS. The purpose of this amendment request is to correct those inconsistencies and errors. The proposed changes do not involve any physical changes to the structure, systems, or components (SSCs) in the plant or the way the SCCs are operated or controlled. None of these changes are of a technical nature and they do not require a technical analysis.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

10 CFR 50.36, "Technical specifications," details the information that must be included in each station's TS. The proposed changes incorporate several administrative changes that include correcting typographical errors, removing unwarranted formatting, and clarifying symbols/pages that may not copy/print well. The proposed changes have no impact on current Safety Limits, Limiting Safety System Settings, Limiting Control Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, or Administrative Controls. Therefore, EGC concludes that the methods used to comply with 10 CFR 50.36 are not modified by the proposed changes, and the requirements continue to be met.

4.2 Precedent

Submitting administrative TS changes is a common industry method to maintain the readability of TS.

4.3 <u>No Significant Hazards Consideration</u>

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI, Unit 1). Specifically, the changes incorporate several administrative changes that include correcting typographical errors, removing unwarranted formatting, and clarifying symbols/pages that may not copy/print well.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

No physical changes to the facility will occur as a result of this proposed amendment. The proposed changes will not alter the physical design or operational procedures associated with any plant structure, system, or component. The proposed changes are administrative in nature and have no affect on plant operation.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes are administrative in nature. The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Accordingly, the changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes conform to NRC regulatory guidance regarding the content of plant Technical Specifications. The guidance is presented in Administrative Letter 95-06 and NUREG-1430. The proposed changes are administrative in nature. The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ATTACHMENT 2

Proposed Technical Specifications Pages (Mark-Up)

<u>Three Mile Island Generating Station, Unit 1</u>
Renewed Facility Operating License No. DPR-50

<u>REVISED TECHNICAL SPECIFICATIONS PAGES</u>

i
ii
iii
iv
v
1-1
1-3
2-4b
2-12
3-1a
3-3
3-6
3-18
3-18g
3-21
3-22
3-23
3-33
3-35
3-36b
3-41
3-43
3-62e
4-5
4-43
Figure 5-1 (page N/A)
Figure 5-2 (page N/A)
Figure 5-3 (page N/A)

TABLE OF CONTENTS

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TABLE OF CONTENTS

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TABLE OF CONTENTS

Amendment No. ~~11, 22, 30, 41, 47, 55, 72, 78, 95, 97, 119, 122, 129,~~ ~~137, 146, 197, 242, 245, 246, 248, 261~~, ~~264~~

TABLE OF CONTENTS

1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is a steady state reactor core output of 2568 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 COLD SHUTDOWN

The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and ~~Tavg~~ T_{ave} is no more than 200°F. Pressure is defined by Specification 3.1.2.

1.2.2 HOT SHUTDOWN

The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and ~~Tavg~~ T_{ave} is at or greater than 525°F.

1.2.3 REACTOR CRITICAL

The reactor is critical when the neutron chain reaction is self-sustaining and Keff = 1.0.

1.2.4 HOT STANDBY

The reactor is in the hot standby condition when all of the following conditions exist:

a. ~~Tavg~~ T_{ave} is greater than 525°F

b. The reactor is critical

c. Indicated neutron power on the power range channels is less than two percent of rated power

1.2.5 POWER OPERATION

The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.

1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is ~~described in Section 7.1~~ described in Section 7.1 of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as ~~described in Section 7.1~~ described in Section 7.1 of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as ~~described in Section 7.1~~ described in Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 <u>INSTRUMENTATION SURVEILLANCE</u>

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

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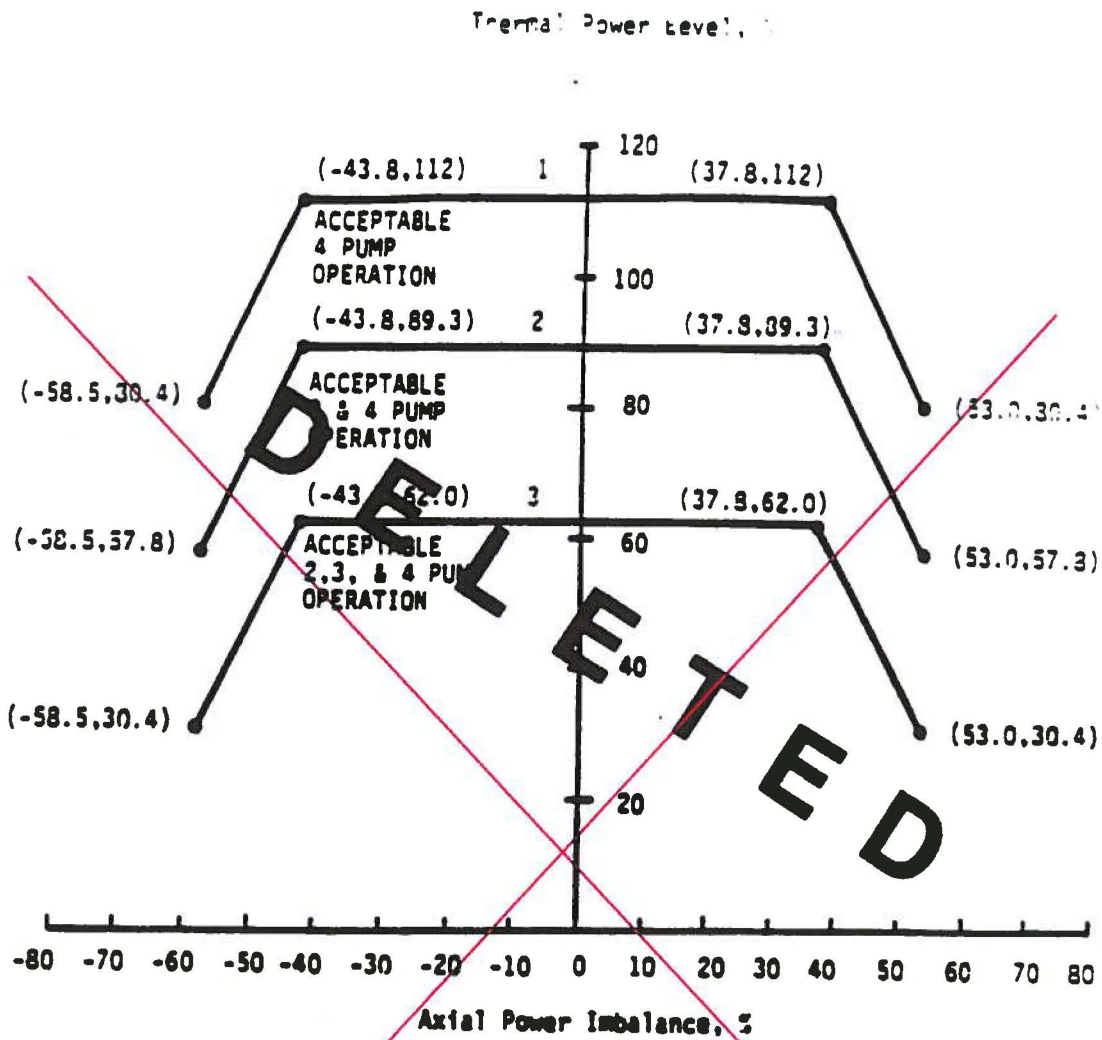

Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

CORE PROTECTION SAFETY LIMITS
TMI-1

Figure 2.1-2

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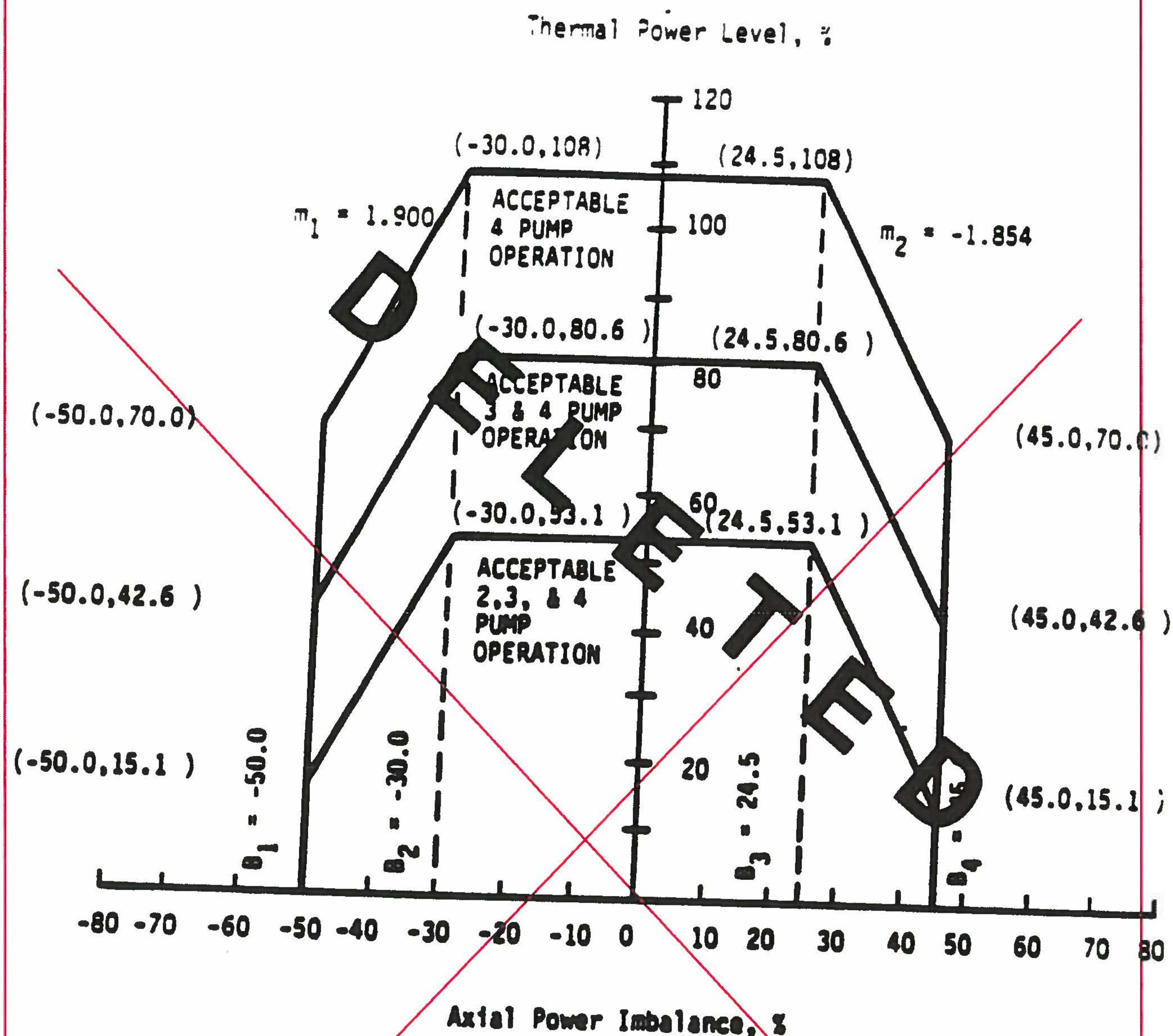

PROTECTION SYSTEM MAXIMUM ALLOWABLE SETPOINTS FOR AXIAL POWER IMBALANCE
TMI-1

Figure 2.3-2

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3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator (SG) Tube Integrity

a. Whenever the reactor coolant average temperature is above 200°F, the following conditions are required:

(1.) SG tube integrity shall be maintained.

AND

(2.) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program. (The Steam Generator Program is described in Section 6.19.)

ACTIONS:

---NOTE---

Entry into Sections 3.1.1.2.a.(3.) and (4.), below, is allowed for each SG tube.

(3.) If the requirements of Section 3.1.1.2.a.(2.) are not met for one or more tubes then perform the following:

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until 29 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding 29 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specification submitted in accordance with 10 CFR 50, Appendix G.

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

c. To assure sufficient pressurizer heater capacity to maintain natural circulation conditions during a loss of offsite power.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 Pressurizer

3.1.3.4.1 The reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.

3.1.3.4.2 A minimum of 107 kw of pressurizer heaters, from each of two pressurizer heater groups shall be OPERABLE. Each OPERABLE 107 kw of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the established manual transfer scheme.

3.1.9 LOW POWER PHYSICS TESTING RESTRICTIONS

Applicability

Applies to Reactor Protection System requirements for low power physics testing.

Objective

To assure an additional margin of safety during low power physics testing.

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protection System Requirements

a. Below 1720 psig Shutdown Bypass trip setting limits shall apply in accordance with Table 2.3-1.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.2 Startup Rate Rod Withdrawal Hold (Reference 1) Shall be operable At All Times.

3.1.9.3 Shutdown margin may not be reduced below 1% delta k/k per 3.5.2.1.

Bases

The above specification provides additional safety margins during low power physics testing, as is also provided for startup (Reference 2).

REFERENCES

(1) UFSAR, Section 7.2.2.1.b - "Reactivity Rate Limits"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

3.1.13 REACTOR COOLANT SYSTEM VENTS

Applicability

Provides the limiting conditions for operation of the Reactor Coolant System Vents. These limiting conditions for operation (LCO) are applicable only when Reactor is critical.

Objective

To ensure that sufficient vent flow paths are operable during the plant operating modes mentioned above.

Specification

3.1.13.1 At least one reactor coolant system vent path consisting of at least two power operated valves in series, powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head~~*~~ (RC-V42 & RC-V43)
b Pressurizer steam space (RC-V28 & RC-V44)
c. Reactor coolant system high point (either RC-V40A and 41A) or (RC-40B and 41B)

Action

3.1.13.2 a. With one of the above reactor coolant system vent paths inoperable, the inoperable vent path shall be maintained closed, with power removed from the valve actuators in the inoperable vent path. The inoperable vent path shall be restored to OPERABLE status within 30 days, or the plant shall be in OT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above reactor coolant system vent paths inoperable, maintain the inoperable vent path closed, with power removed from the valve actuators in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~*~~ ~~This specification becomes binding after installation and initially being declared operable.~~

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the operating status of the emergency core cooling, reactor building emergency cooling, and reactor building spray systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The reactor shall not be made critical unless the following conditions are met:

3.3.1.1 Injection Systems

(BWST)

a. The borated water storage tank ~~(BWST)~~ shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

Makeup and Purification (MU)/High Pressure Injection (HPI) pumps are OPERABLE

b. Two ~~Makeup and Purification (MU)/High Pressure Injection (HPI) pumps are OPERABLE~~ in the engineered safeguards mode powered from independent essential buses. Specification 3.0.1 applies.

OPERABLE.

c. Two decay heat removal pumps are ~~OPERABLE.~~ Specification 3.0.1 applies.

OPERABLE.

d. Two decay heat removal coolers and their cooling water supplies are ~~OPERABLE.~~ (See Specification 3.3.1.4) Specification 3.0.1 applies.

OPERABLE.

e. Two BWST level instrument channels are ~~OPERABLE.~~

OPERABLE.

f. The two reactor building sump isolation valves (DH-V-6A/B) shall be remote-manually ~~OPERABLE.~~ Specification 3.0.1 applies.

~~g. MU Tank (MUT) pressure and level shall be maintained within the Unrestricted Operating Region of Figure 3.3-1.~~

~~1) With MUT conditions outside of the Unrestricted Operating Region of Figure 3.3-1, restore MUT pressure and level to within the Unrestricted Operating Region within 72 hrs. Specification 3.0.1 applies.~~ Insert 1

~~2) Operation with MUT conditions within the Prohibited Region of Figure 3.3-1 is prohibited. Specification 3.0.1 applies.~~

3.3.1.2 Core Flooding System

(CFTs)

a. Two core flooding tanks ~~(CFTs)~~ each containing 940 ± 30 ft^3 of borated water at 600 ± 25 psig shall be available. Specification 3.0.1 applies.

Insert 1

g. MU Tank (MUT) pressure and level shall be maintained within the Unrestricted Operating Region of Figure 3.3-1.

1) With MUT conditions outside of the Unrestricted Operating Region of Figure 3.3-1, restore MUT pressure and level to within the Unrestricted Operating Region within 72 hrs. Specification 3.0.1 applies.

2) Operation with MUT conditions within the Prohibited Region of Figure 3.3-1 is prohibited. Specification 3.0.1 applies.

~~3.3~~ ~~EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)~~

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)

b. ~~CFT~~ CFT boron concentration shall not be less than 2,270 ppm boron. Specification 3.3.2.1 applies.

c. The electrically operated discharge valves from the ~~CFT~~ CFT will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

d. DELETED

e. ~~CFT~~ CFT vent valves CF-V-3A and CF-V-3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The Reactor Building emergency sump pH control system shall be maintained with ≥ 18,815 lbs and ≤ 28,840 lbs of trisodium phosphate dodecahydrate (TSP). Specification 3.3.2.1 applies.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.~~*~~

3.3.2.1 If the CFT boron concentration is outside of limits, or if the TSP baskets contain amounts of TSP outside the limits specified in 3.3.1.3.b, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both CFTs shall be OPERABLE at all times.

b. Both the motor operated valves associated with the CFTs shall be fully open at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be verified to be OPERABLE.

~~*** In accordance with AmerGen License Change Application dated February 14, 2001, and any requirements in the associated NRC Safety Evaluation, a portion of the Nuclear Service Water System piping between valves NR-V-3 and NR-V-5 may be removed from service and Nuclear Services River Water flow realigned through a portion of the Secondary Services River Water System piping for up to 14 days. This note is applicable for one time use during TMI Unit 1 Operating Cycle 13.**~~

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location and sump recirculation switchover setpoint. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

The Reactor Building emergency sump pH control system ensures a sump pH between 7.3 and 8.0 during the recirculation phase of a postulated LOCA. A minimum pH level of 7.3 is required to reduce the potential for chloride induced stress corrosion cracking of austenitic stainless steel and assure the retention of elemental iodine in the recirculating fluid. A maximum pH value of 8.0 minimizes the

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

delta

3.5.2.1 The available shutdown margin shall not be less than one percent ΔK/K with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

delta

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 in the safety or regulating rod banks shall not be permitted. Verify SDM ≥ 1% Δk/k or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent delta Δk/k hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent delta Δk/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, and cannot be aligned per 3.5.2.2.f, power shall be reduced to ≤ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to ≤ 70% of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F_{\Delta H}^{N}$) limits per the COLR have not been exceeded within 72 hours.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ± 5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F^{N}_{\Delta H}$ shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

2. If regulating rods are inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within 2 hours or reduce power to ≤ power allowed by rod insertion limits.

delta

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Plant Manager.

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F^{N}_{\Delta H}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

CONTROLLED COPY

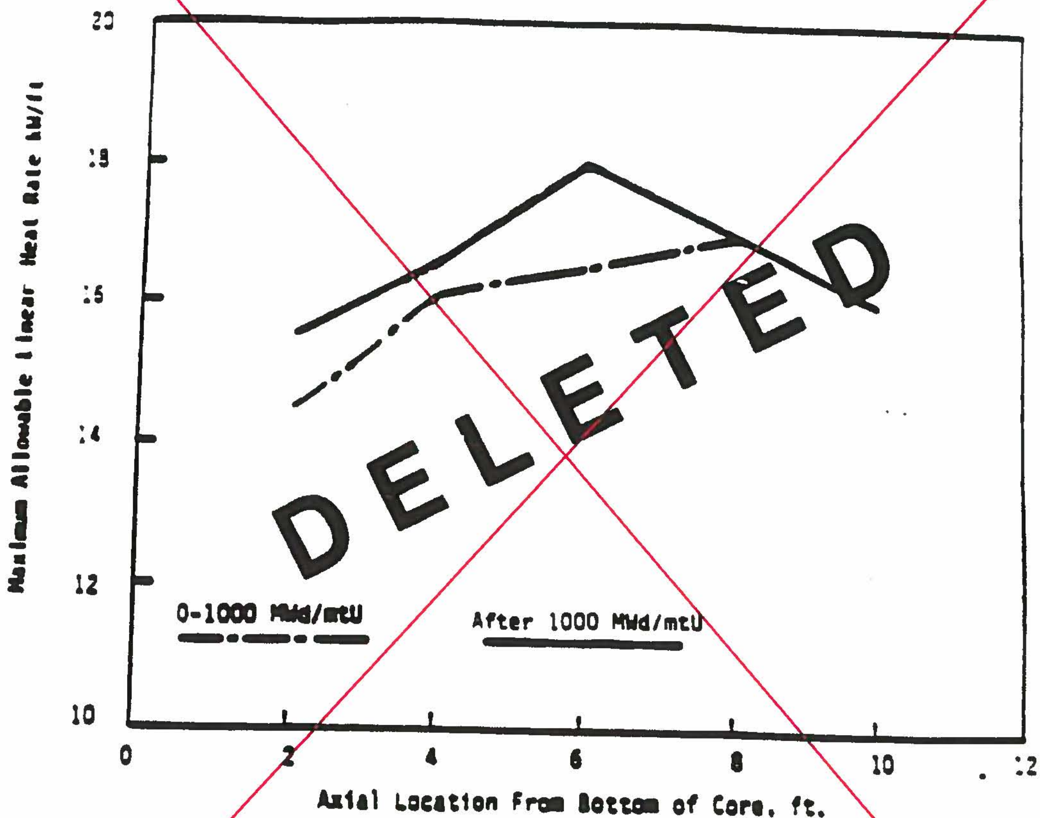


FIGURE DELETED AND INCORPORATED

INTO THE CORE OPERATING LIMITS REPORT.

LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE

TMI-1

Figure 3.5-2M

Amendment No. ~~142~~ ~~152~~, ~~157~~, 168

INFORMATION ON THIS PAGE HAS BEEN DELETED

3.6 REACTOR BUILDING

Applicability

Applies to the CONTAINMENT INTEGRITY of the reactor building as specified below.

Objective

To assure CONTAINMENT INTEGRITY.

Specification

3.6.1 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200 degrees F or greater.

c. Nuclear fuel is in the core.

3.6.2 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

delta

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% Δk/k shall not be made by control rod motion or boron dilution unless CONTAINMENT INTEGRITY is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual Containment Isolation Valves (CIVs) which should be closed are closed and are conspicuously marked.

3.6.6 When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV* in the line will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

b. For lines isolable by one CIV, where the other barrier is a closed system, the line shall be isolated by at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 72 hours or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

* All CIVs required to isolate the penetration.

c. Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days~~*~~ provided that the redundant diesel generator is:

1. verified to be operable immediately;

2. within 24 hours, either:

   a. determine the redundant diesel generator is not inoperable due to a common mode failure; or,

   b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

With one diesel generator inoperable, in addition to the above, verify that: All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

d. If one Unit Auxiliary Transformer is inoperable and a diesel generator becomes inoperable, the unit will be placed in HOT SHUTDOWN within 12 hours. If one of the above sources of power is not made operable within an additional 24 hours the unit shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

e. If Unit 1 is separated from the system while carrying its own auxiliaries, or if only one 230 kV line is in service, continued reactor operation is permissible provided one emergency diesel generator shall be started and run continuously until two transmission lines are restored.

f. The engineered safeguards electrical bus, switchgear, load shedding, and automatic diesel start systems shall be operable except as provided in Specification 3.7.2c above and as required for testing.

g. One station battery may be removed from service for not more than eight hours.

h. If it is determined that a trip of the Unit 1 generator, in conjunction with LOCA loading, will result in a loss of offsite power to Engineered Safeguards buses, the plant shall begin a power reduction within 24 hours and be in HOT SHUTDOWN in an additional 6 hours, except as provided in Specification 3.7.2.e above.

~~* The 7-day allowed outage time of Technical Specification 3.7.2.c, which was entered on April 2, 2006 at 2100 hours, may be extended one time by an additional 3 days to complete repair and testing of EG-Y-1A.~~

3.15.4 Fuel Handling Building ESF Air Treatment System

Applicability

Applies to the Fuel Handling Building (FHB) ESF Air Treatment System and its associated filters.

Objective

To specify minimum availability and efficiency for the FHB ESF Air Treatment System and its associated filters for irradiated fuel handling operations.

Specifications

3.15.4.1 Prior to fuel movement each refueling outage, two trains shall be operable. One train shall be operating continuously whenever TMI-1 irradiated fuel handling operations in the FHB are in progress.

a. With one train inoperable, irradiated fuel handling operations in the Fuel Handling Building may continue provided the redundant train is operating.

b. With both trains inoperable, handling of irradiated fuel in the Fuel Handling Building shall be suspended until such time that at least one train is operable and operating. Any fuel assembly movement in progress may be completed.

3.15.4.2 A FHB ESF Air Treatment System train is operable when its surveillance requirements are met and:

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and carbon absorber banks shall show < 0.05% DOP penetration and < 0.05% halogenated hydrocarbon penetration.

b. The results of laboratory carbon sample analysis shall show ≥ 95% radioactive methyl iodide decontamination efficiency when tested ~~in accordance with ASTM D3803-1989~~ at 30°C, 95% R.H. in accordance with ASTM D3803-1989

c. The fans AH-E-137A and B shall each be shown to operate within ± 10% of design flow (6,000 SCFM).

Bases

Compliance with these specifications satisfies the condition of operation imposed by the Licensing Board as described in NRC's letter dated October 2, 1985, item 1.c.

The FHB ESF Air Treatment System contains, controls, mitigates, monitors and records radiation release resulting from a TMI-1 postulated spent fuel accident in the Fuel Handling Building as described in the FSAR. Offsite doses will be less than the 10 CFR 100 guidelines for accidents analyzed in Chapter 14 (Reference 1).

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
b. RCS Pressure 1600 psig	(1)	(1)	NA	(1) When RCS Pressure > 1800 psig.
c. Deleted				
d. Reactor Bldg. 30 psi pressure switches	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	(1)	(1)~~(2)~~		(1) When CONTAINMENT INTEGRITY is required.
f. Line Break Isolation Signal (ICCW & NSCCW)	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
20. Reactor Building Spray System Logic Channel	NA		NA	
21. Reactor Building Spray 30 psig pressure switches	NA			
22. Pressurizer Temperature Channels		NA		
23. Control Rod Absolute Position	(1)	NA		(1) Check with Relative Position Indicator
24. Control Rod Relative Position	(1)	NA		(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels ~~Coolant~~	NA	NA		
b. Level Channels	NA	NA		
26. Pressurizer Level Channels		NA		

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testig of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable ~~Specification~~ .

Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At the frequency specified in the Surveillance Frequency Control Program and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the R. B. Cooling and Isolation System.

   The test will be considered satisfactory if the spray pumps have been successfully started.

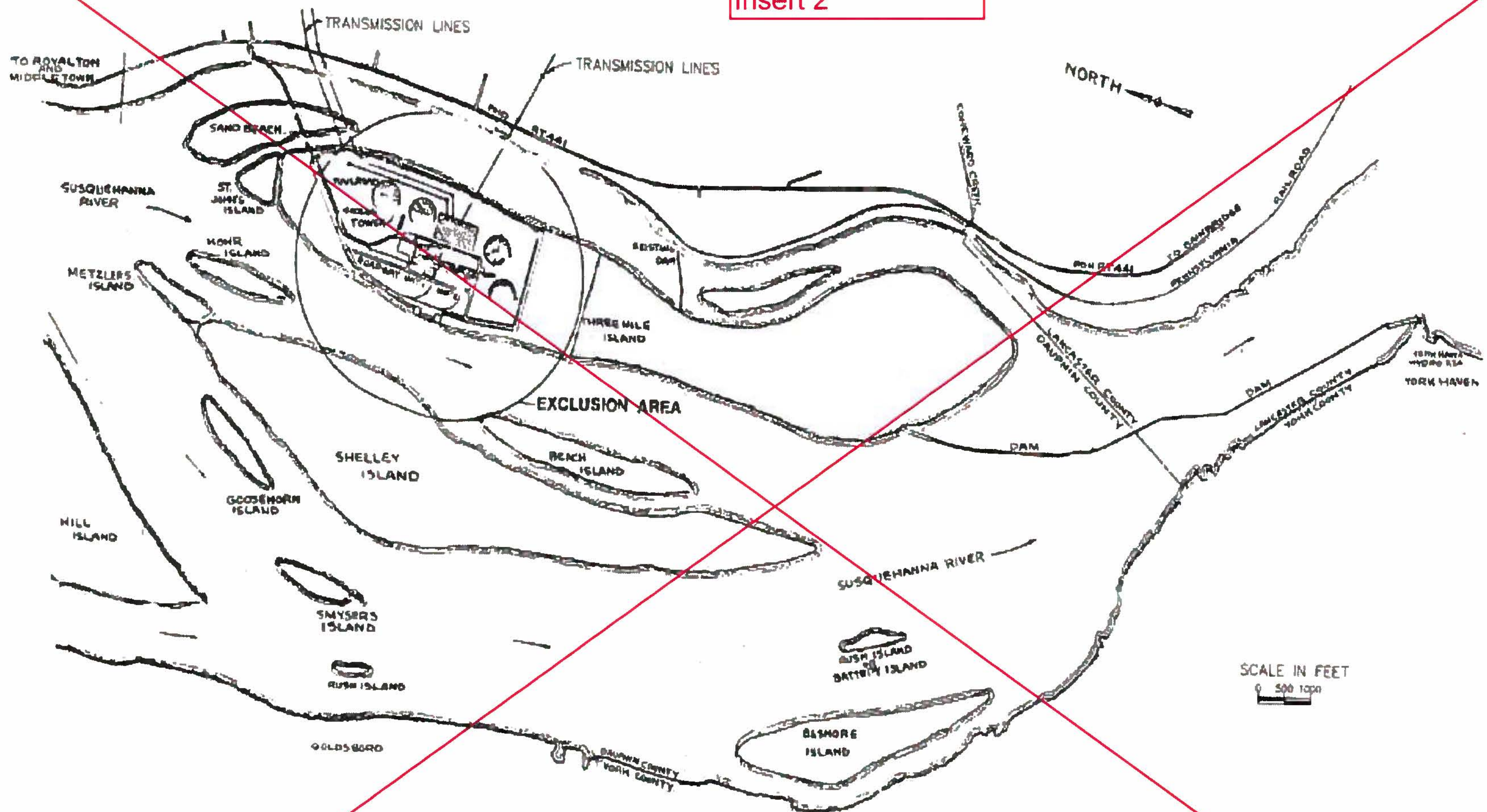
2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at the frequency specified in the Surveillance Frequency Control Program.

b. Reactor Building Cooling and Isolation Systems

1. At the frequency specified in the Surveillance Frequency Control Program, a system test shall be conducted to demonstrate proper operation of the system.

2. The test will be considered satisfactory if measured system flow is greater than accident design flow rate.

Insert 2



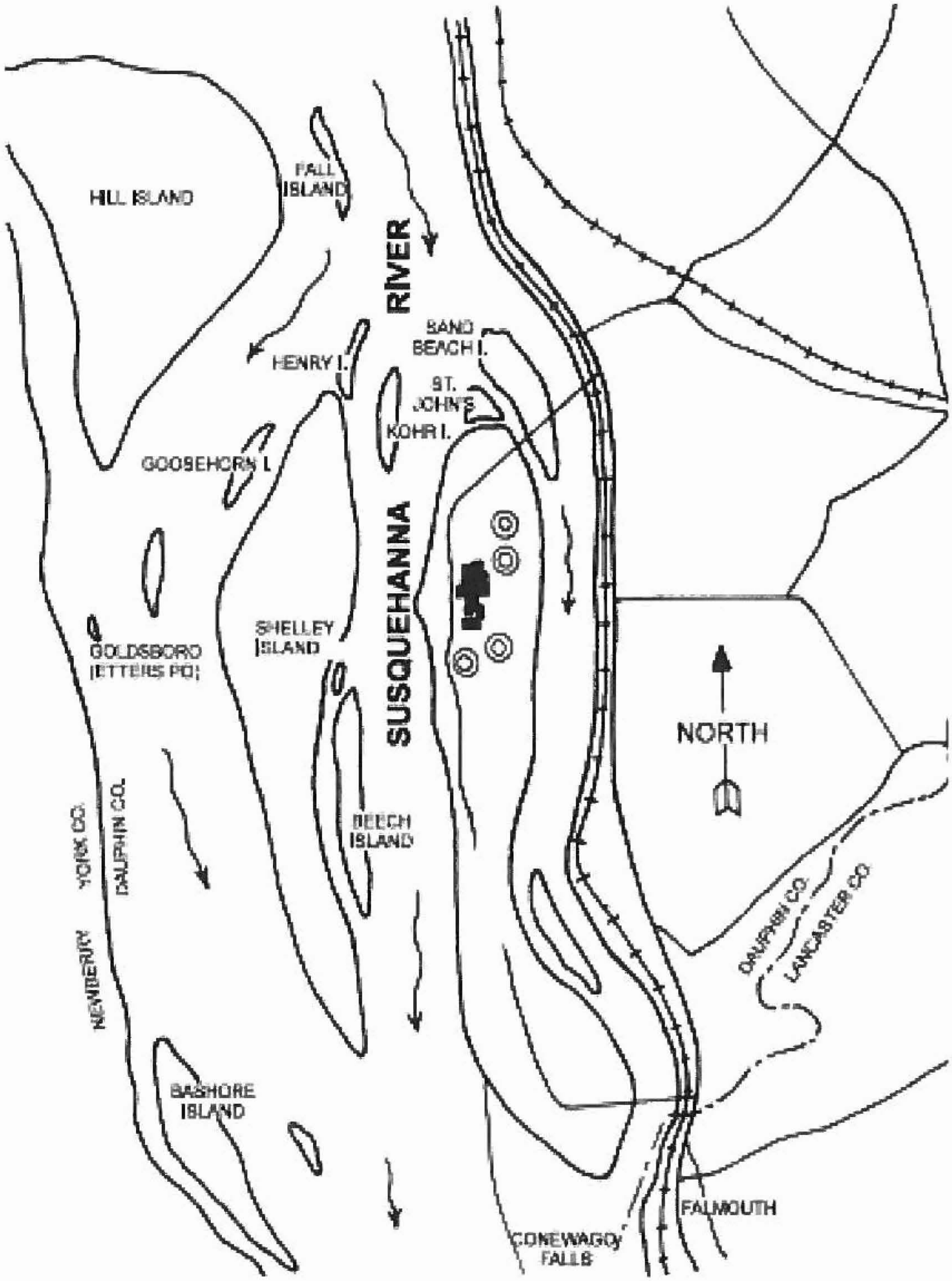
Amendment No. ~~140, 216, 246~~

AmerGen
Three Mile Island Nuclear Station

EXTENDED PLOT PLAN

CAD FILE: 6717R1,DWG FIG 5-1

Insert 2



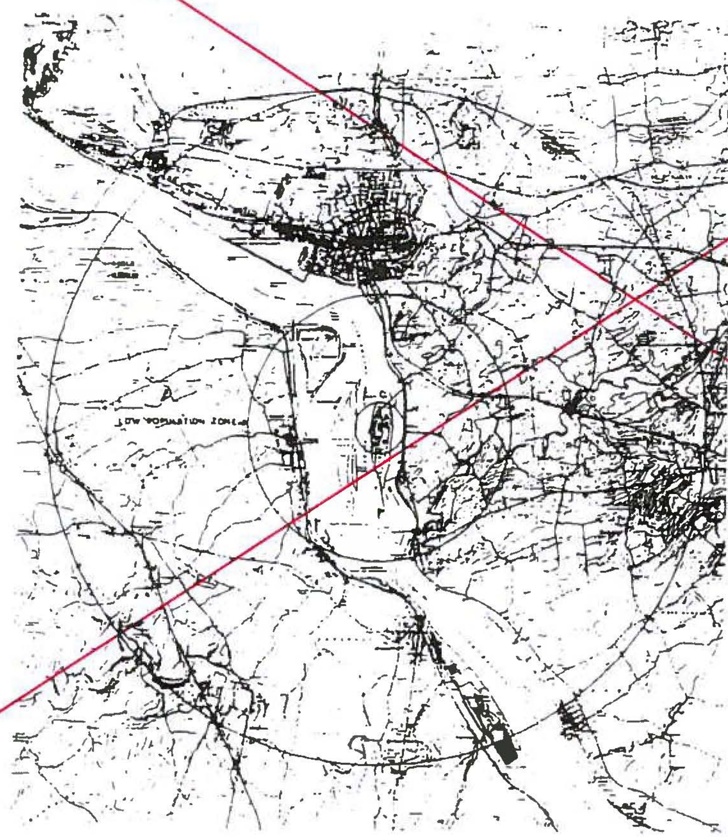
Amendment No. ~~140, 216, 246~~

Exelon
Three Mile Island Nuclear Station

Relative Location of TMI Site

FIG 5-1

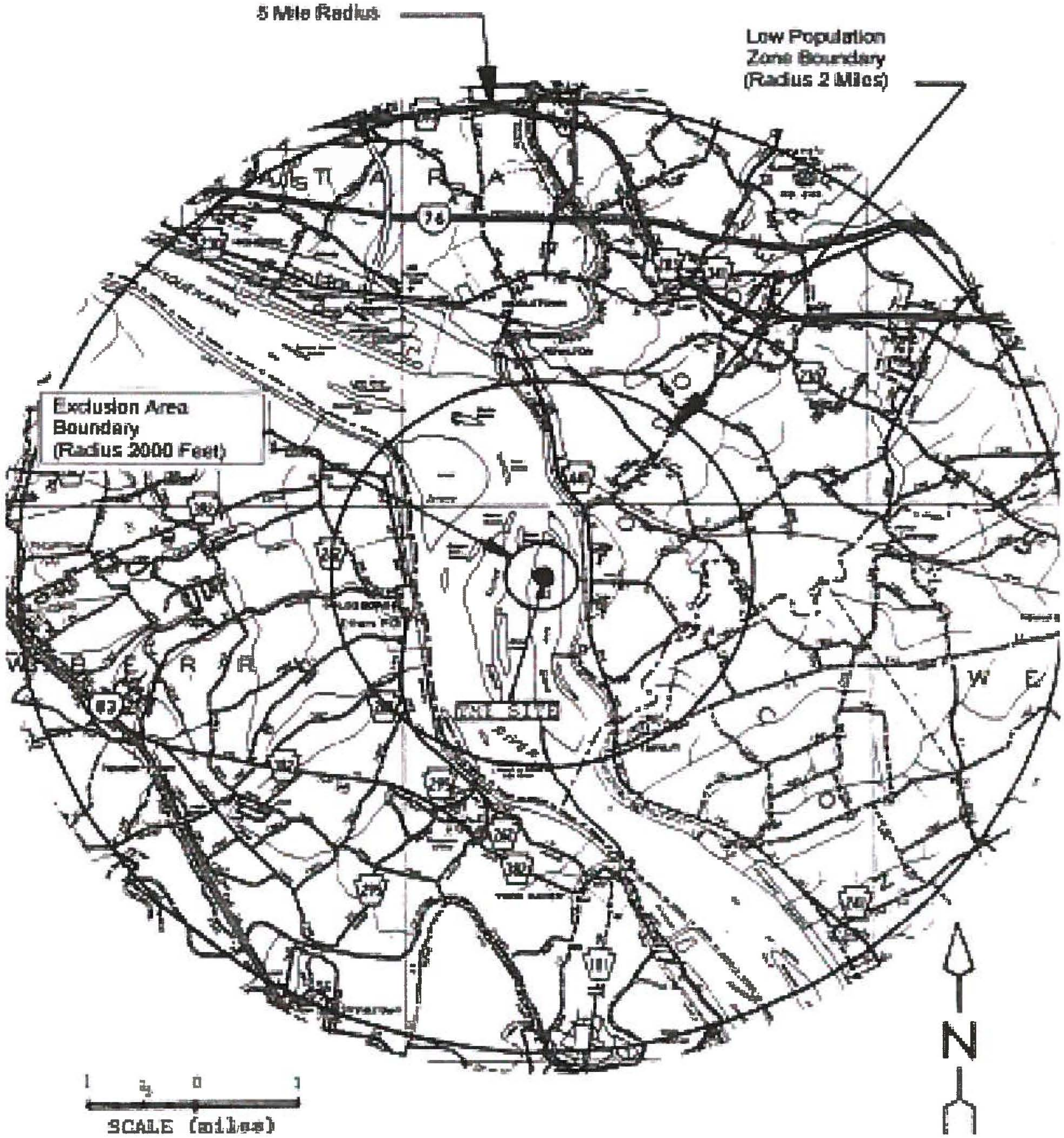
Insert 3



CONTOUR INTERVAL 20 FEET
DATUM IS MEAN SEA LEVEL

AmerGen
Site Topography
5 Mile Radius
Three Mile Island Nuclear Station
Fig. 5-2

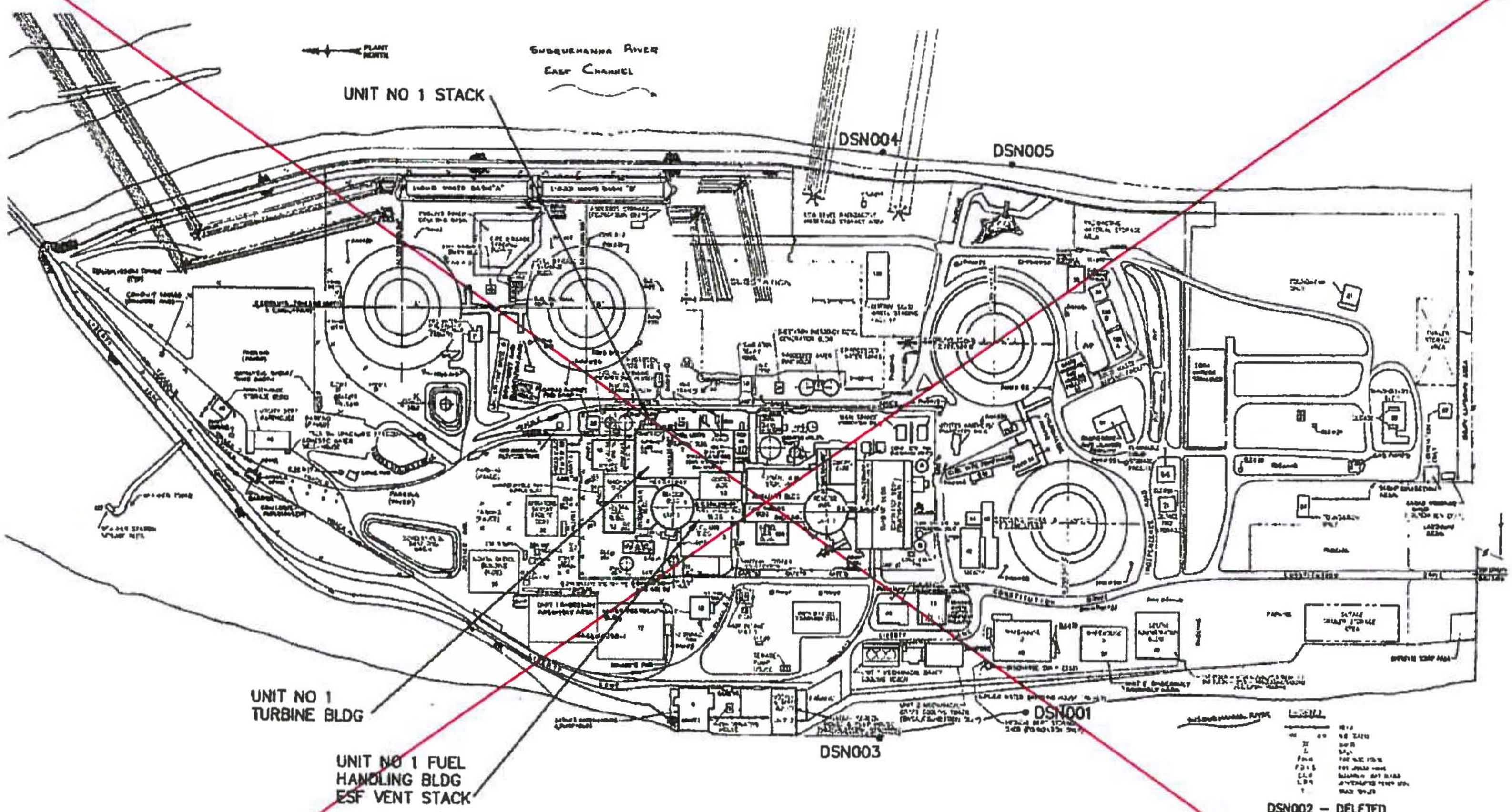
Insert 3



Exelon
Three Mile Island Nuclear Station

Site Exclusion Area and Low Population Zone

Fig. 5-2



Insert 4

AmerGen
Three Mile Island Nuclear Station

Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

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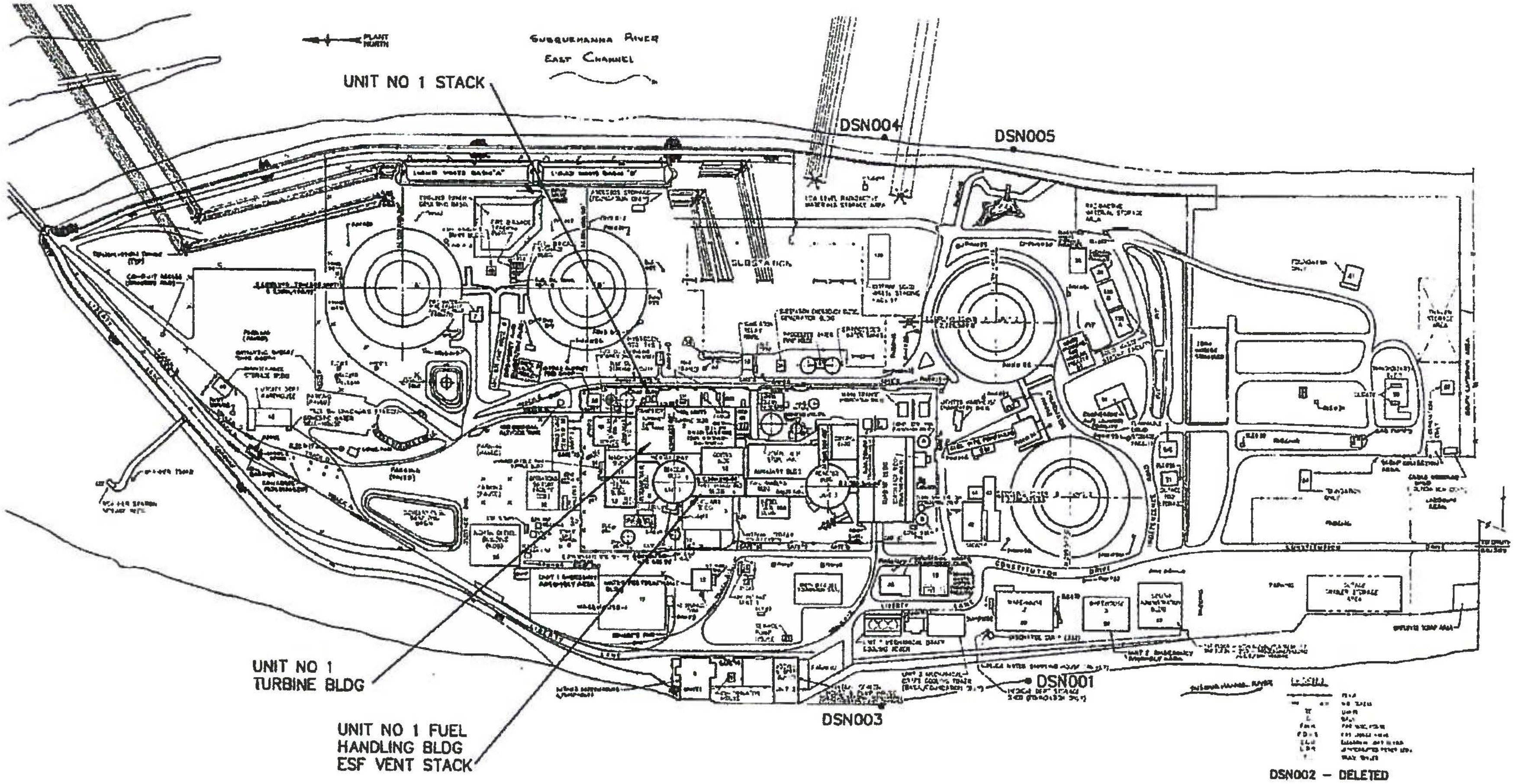
Fig. 5–3

Amendment No. ~~149, 218, 246~~, 269

Insert 4



Exelon
Three Mile Island Nuclear Station

Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

FIG 5-3

Amendment No. ~~140, 218, 246, 269~~

ATTACHMENT 3

Revised Technical Specifications Pages

Three Mile Island Generating Station, Unit 1
Renewed Facility Operating License No. DPR-50

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~~137, 146, 197, 242, 245, 246, 248, 261, 264~~

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1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is a steady state reactor core output of 2568 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 COLD SHUTDOWN

The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and T_{ave} is no more than 200°F. Pressure is defined by Specification 3.1.2.

1.2.2 HOT SHUTDOWN

The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and T_{ave} is at or greater than 525°F.

1.2.3 REACTOR CRITICAL

The reactor is critical when the neutron chain reaction is self-sustaining and Keff = 1.0.

1.2.4 HOT STANDBY

The reactor is in the hot standby condition when all of the following conditions exist:

a. T_{ave} is greater than 525°F

b. The reactor is critical

c. Indicated neutron power on the power range channels is less than two percent of rated power

1.2.5 POWER OPERATION

The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.

1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is described in Section 7.1 of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as described in Section 7.1 of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as described in Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 <u>INSTRUMENTATION SURVEILLANCE</u>

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

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3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator (SG) Tube Integrity

a. Whenever the reactor coolant average temperature is above 200°F, the following conditions are required:

(1.) SG tube integrity shall be maintained.

AND

(2.) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program. (The Steam Generator Program is described in Section 6.19.)

ACTIONS:

---NOTE---

Entry into Sections 3.1.1.2.a.(3.) and (4.), below, is allowed for each SG tube.

(3.) If the requirements of Section 3.1.1.2.a.(2.) are not met for one or more tubes then perform the following:

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until 29 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding 29 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specification submitted in accordance with 10 CFR 50, Appendix G.

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

c. To assure sufficient pressurizer heater capacity to maintain natural circulation conditions during a loss of offsite power.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 Pressurizer

3.1.3.4.1 The reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.

3.1.3.4.2 A minimum of 107 kw of pressurizer heaters, from each of two pressurizer heater groups shall be OPERABLE. Each OPERABLE 107 kw of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the established manual transfer scheme.

3.1.9 LOW POWER PHYSICS TESTING RESTRICTIONS

Applicability

Applies to Reactor Protection System requirements for low power physics testing.

Objective

To assure an additional margin of safety during low power physics testing.

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protection System Requirements

a. Below 1720 psig Shutdown Bypass trip setting limits shall apply in accordance with Table 2.3-1.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.2 Startup Rate Rod Withdrawal Hold (Reference 1) Shall be operable At All Times.

3.1.9.3 Shutdown margin may not be reduced below 1% delta k/k per 3.5.2.1.

Bases

The above specification provides additional safety margins during low power physics testing, as is also provided for startup (Reference 2.)

REFERENCES

(1) UFSAR, Section 7.2.2.1.b - "Reactivity Rate Limits"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

3.1.13 REACTOR COOLANT SYSTEM VENTS

Applicability

Provides the limiting conditions for operation of the Reactor Coolant System Vents. These limiting conditions for operation (LCO) are applicable only when Reactor is critical.

Objective

To ensure that sufficient vent flow paths are operable during the plant operating modes mentioned above.

Specification

3.1.13.1 At least one reactor coolant system vent path consisting of at least two power operated valves in series, powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head (RC-V42 & RC-V43)
b Pressurizer steam space (RC-V28 & RC-V44)
c. Reactor coolant system high point (either RC-V40A and 41A) or (RC-40B and 41B)

Action

3.1.13.2 a. With one of the above reactor coolant system vent paths inoperable, the inoperable vent path shall be maintained closed, with power removed from the valve actuators in the inoperable vent path. The inoperable vent path shall be restored to OPERABLE status within 30 days, or the plant shall be in OT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above reactor coolant system vent paths inoperable, maintain the inoperable vent path closed, with power removed from the valve actuators in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the operating status of the emergency core cooling, reactor building emergency cooling, and reactor building spray systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The reactor shall not be made critical unless the following conditions are met:

3.3.1.1 Injection Systems

a. The borated water storage tank (BWST) shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

b. Two Makeup and Purification (MU)/High Pressure Injection (HPI) pumps are OPERABLE in the engineered safeguards mode powered from independent essential buses. Specification 3.0.1 applies.

c. Two decay heat removal pumps are OPERABLE. Specification 3.0.1 applies.

d. Two decay heat removal coolers and their cooling water supplies are OPERABLE. (See Specification 3.3.1.4) Specification 3.0.1 applies.

e. Two BWST level instrument channels are OPERABLE.

f. The two reactor building sump isolation valves (DH-V-6A/B) shall be remote-manually OPERABLE. Specification 3.0.1 applies.

g. MU Tank (MUT) pressure and level shall be maintained within the Unrestricted Operating Region of Figure 3.3-1.

   1) With MUT conditions outside of the Unrestricted Operating Region of Figure 3.3-1, restore MUT pressure and level to within the Unrestricted Operating Region within 72 hrs. Specification 3.0.1 applies.

   2) Operation with MUT conditions within the Prohibited Region of Figure 3.3-1 is prohibited. Specification 3.0.1 applies.

3.3.1.2 Core Flooding System

a. Two core flooding tanks (CFTs) each containing 940 ± 30 ft^3 of borated water at 600 ± 25 psig shall be available. Specification 3.0.1 applies.

b. CFT boron concentration shall not be less than 2,270 ppm boron. Specification 3.3.2.1 applies.

c. The electrically operated discharge valves from the CFT will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

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e. CFT vent valves CF-V-3A and CF-V-3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The Reactor Building emergency sump pH control system shall be maintained with ≥ 18,815 lbs and ≤ 28,840 lbs of trisodium phosphate dodecahydrate (TSP). Specification 3.3.2.1 applies.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3.3 <u>EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)</u>

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.3.2.1 If the CFT boron concentration is outside of limits, or if the TSP baskets contain amounts of TSP outside the limits specified in 3.3.1.3.b, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both CFTs shall be OPERABLE at all times.

b. Both the motor operated valves associated with the CFTs shall be fully open at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be verified to be OPERABLE.

<u>Bases</u>

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location and sump recirculation switchover setpoint. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

The Reactor Building emergency sump pH control system ensures a sump pH between 7.3 and 8.0 during the recirculation phase of a postulated LOCA. A minimum pH level of 7.3 is required to reduce the potential for chloride induced stress corrosion cracking of austenitic stainless steel and assure the retention of elemental iodine in the recirculating fluid. A maximum pH value of 8.0 minimizes the

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall not be less than one percent delta K/K with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 in the safety or regulating rod banks shall not be permitted. Verify SDM ≥ 1% delta k/k or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent delta k/k hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent delta k/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, and cannot be aligned per 3.5.2.2.f, power shall be reduced to ≤ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to ≤ 70% of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F^N_{\Delta H}$) limits per the COLR have not been exceeded within 72 hours.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ± 5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F_{\Delta H}^{N}$ shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

2. If regulating rods are inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% delta K/K, and restore regulating rods to within restricted region within 2 hours or reduce power to ≤ power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Plant Manager.

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F_{\Delta H}^{N}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

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3.6 REACTOR BUILDING

Applicability

Applies to the CONTAINMENT INTEGRITY of the reactor building as specified below.

Objective

To assure CONTAINMENT INTEGRITY.

Specification

3.6.1 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200 degrees F or greater.

c. Nuclear fuel is in the core.

3.6.2 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% delta k/k shall not be made by control rod motion or boron dilution unless CONTAINMENT INTEGRITY is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual Containment Isolation Valves (CIVs) which should be closed are closed and are conspicuously marked.

3.6.6 When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV* in the line will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

b. For lines isolable by one CIV, where the other barrier is a closed system, the line shall be isolated by at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 72 hours or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

* All CIVs required to isolate the penetration.

c. Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days provided that the redundant diesel generator is:

1. verified to be operable immediately;

2. within 24 hours, either:

    a. determine the redundant diesel generator is not inoperable due to a common mode failure; or,

    b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

With one diesel generator inoperable, in addition to the above, verify that: All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

d. If one Unit Auxiliary Transformer is inoperable and a diesel generator becomes inoperable, the unit will be placed in HOT SHUTDOWN within 12 hours. If one of the above sources of power is not made operable within an additional 24 hours the unit shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

e. If Unit 1 is separated from the system while carrying its own auxiliaries, or if only one 230 kV line is in service, continued reactor operation is permissible provided one emergency diesel generator shall be started and run continuously until two transmission lines are restored.

f. The engineered safeguards electrical bus, switchgear, load shedding, and automatic diesel start systems shall be operable except as provided in Specification 3.7.2c above and as required for testing.

g. One station battery may be removed from service for not more than eight hours.

h. If it is determined that a trip of the Unit 1 generator, in conjunction with LOCA loading, will result in a loss of offsite power to Engineered Safeguards buses, the plant shall begin a power reduction within 24 hours and be in HOT SHUTDOWN in an additional 6 hours, except as provided in Specification 3.7.2.e above.

3.15.4 Fuel Handling Building ESF Air Treatment System

Applicability

Applies to the Fuel Handling Building (FHB) ESF Air Treatment System and its associated filters.

Objective

To specify minimum availability and efficiency for the FHB ESF Air Treatment System and its associated filters for irradiated fuel handling operations.

Specifications

3.15.4.1 Prior to fuel movement each refueling outage, two trains shall be operable. One train shall be operating continuously whenever TMI-1 irradiated fuel handling operations in the FHB are in progress.

a. With one train inoperable, irradiated fuel handling operations in the Fuel Handling Building may continue provided the redundant train is operating.

b. With both trains inoperable, handling of irradiated fuel in the Fuel Handling Building shall be suspended until such time that at least one train is operable and operating. Any fuel assembly movement in progress may be completed.

3.15.4.2 A FHB ESF Air Treatment System train is operable when its surveillance requirements are met and:

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and carbon absorber banks shall show < 0.05% DOP penetration and < 0.05% halogenated hydrocarbon penetration.

b. The results of laboratory carbon sample analysis shall show ≥ 95% radioactive methyl iodide decontamination efficiency when tested in accordance with ASTM D3803-1989 at 30°C, 95% R.H.

c. The fans AH-E-137A and B shall each be shown to operate within ± 10% of design flow (6,000 SCFM).

Bases

Compliance with these specifications satisfies the condition of operation imposed by the Licensing Board as described in NRC's letter dated October 2, 1985, item 1.c.

The FHB ESF Air Treatment System contains, controls, mitigates, monitors and records radiation release resulting from a TMI-1 postulated spent fuel accident in the Fuel Handling Building as described in the FSAR. Offsite doses will be less than the 10 CFR 100 guidelines for accidents analyzed in Chapter 14 (Reference 1).

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
b. RCS Pressure 1600 psig	(1)	(1)	NA	(1) When RCS Pressure > 1800 psig.
c. Deleted				
d. Reactor Bldg. 30 psi pressure switches	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
f. Line Break Isolation Signal (ICCW & NSCCW)	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
20. Reactor Building Spray System Logic Channel	NA		NA	
21. Reactor Building Spray 30 psig pressure switches	NA			
22. Pressurizer Temperature Channels		NA		
23. Control Rod Absolute Position	(1)	NA		(1) Check with Relative Position Indicator
24. Control Rod Relative Position	(1)	NA		(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels	NA	NA		
b. Level Channels	NA	NA		
26. Pressurizer Level Channels		NA		

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testig of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable.

Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At the frequency specified in the Surveillance Frequency Control Program and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the R. B. Cooling and Isolation System.

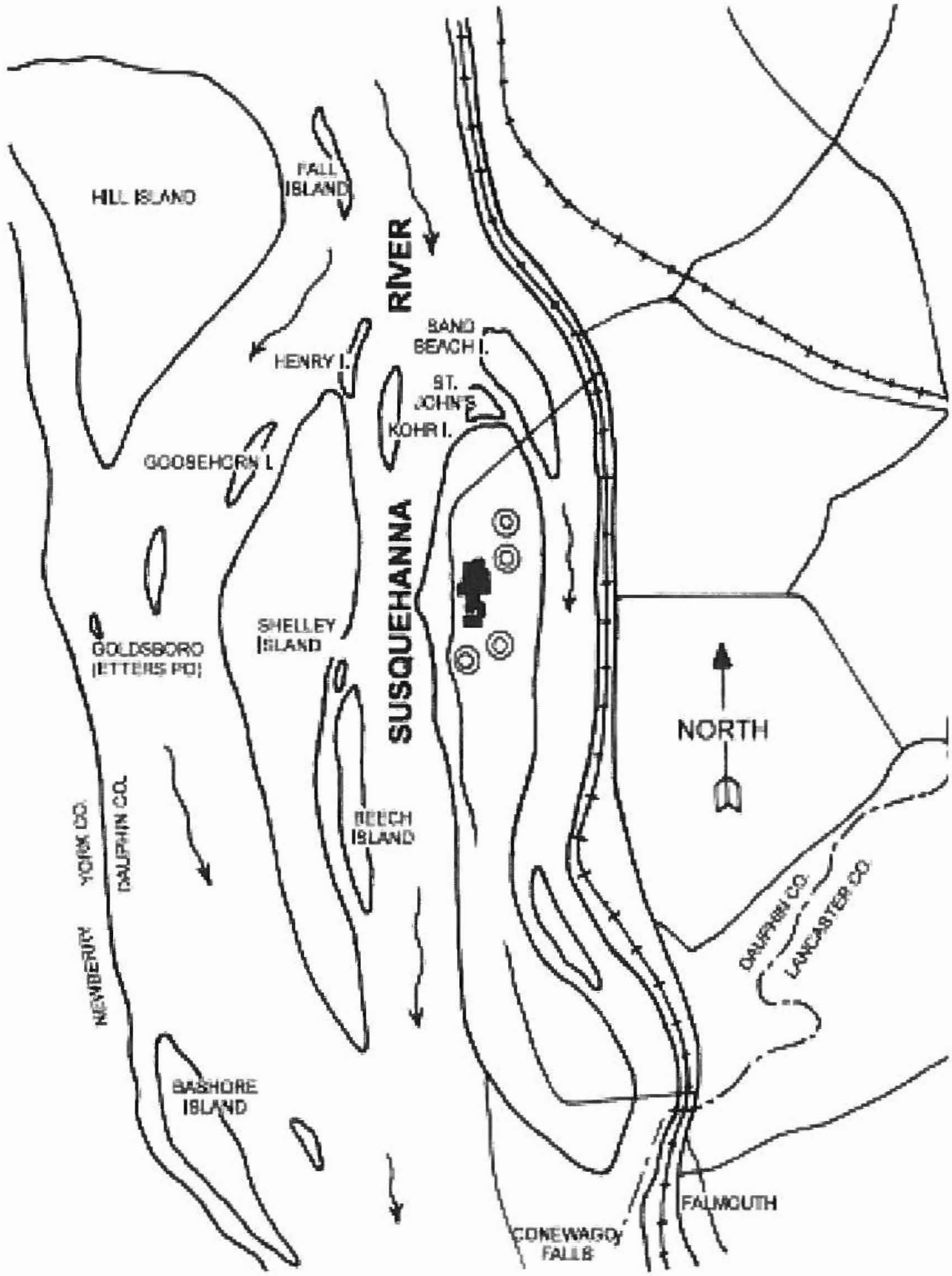
   The test will be considered satisfactory if the spray pumps have been successfully started.

2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at the frequency specified in the Surveillance Frequency Control Program.

b. Reactor Building Cooling and Isolation Systems

1. At the frequency specified in the Surveillance Frequency Control Program, a system test shall be conducted to demonstrate proper operation of the system.

2. The test will be considered satisfactory if measured system flow is greater than accident design flow rate.



Amendment No. ~~140, 216, 246~~

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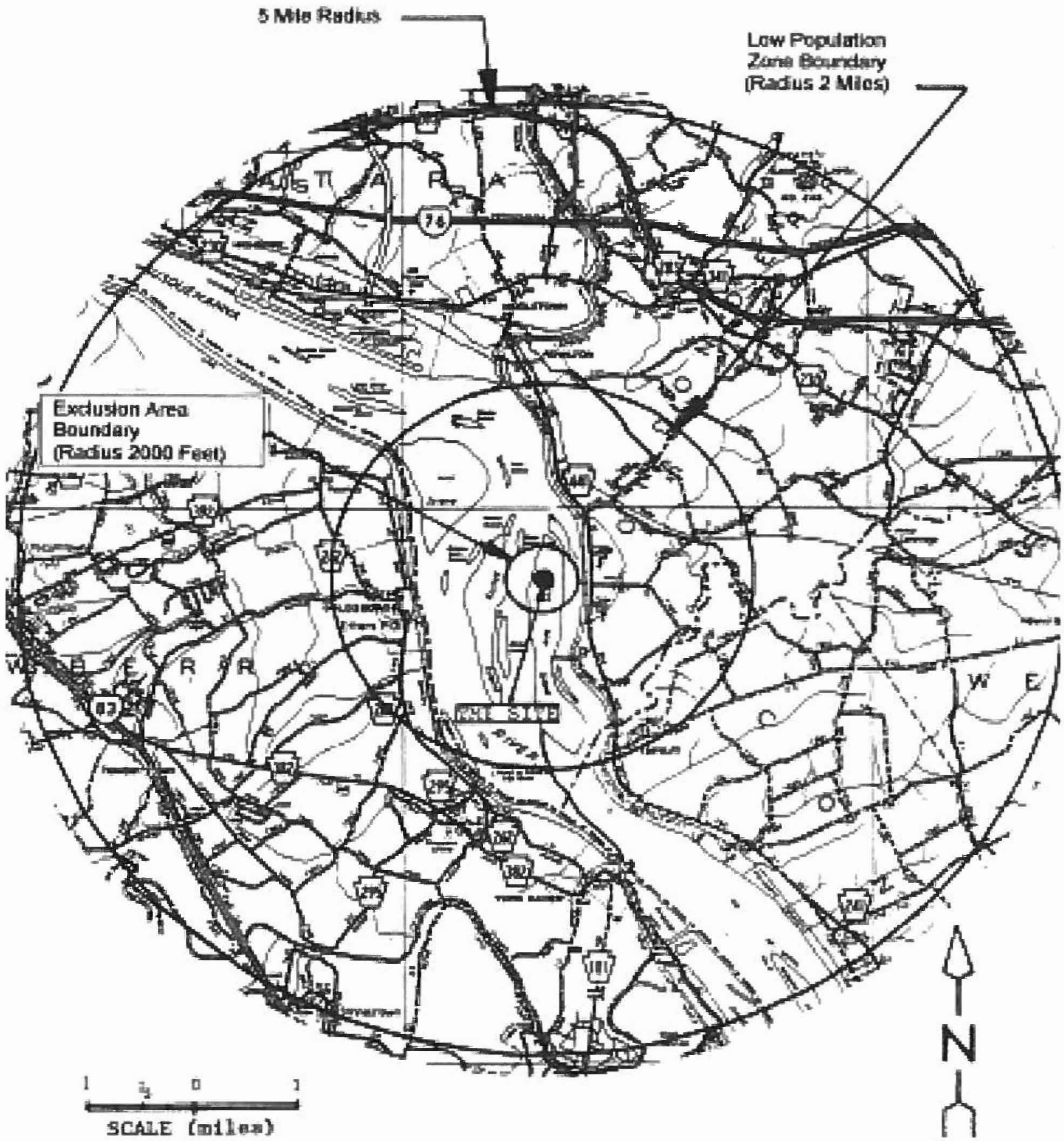
Relative Location of TMI Site

FIG 5-1



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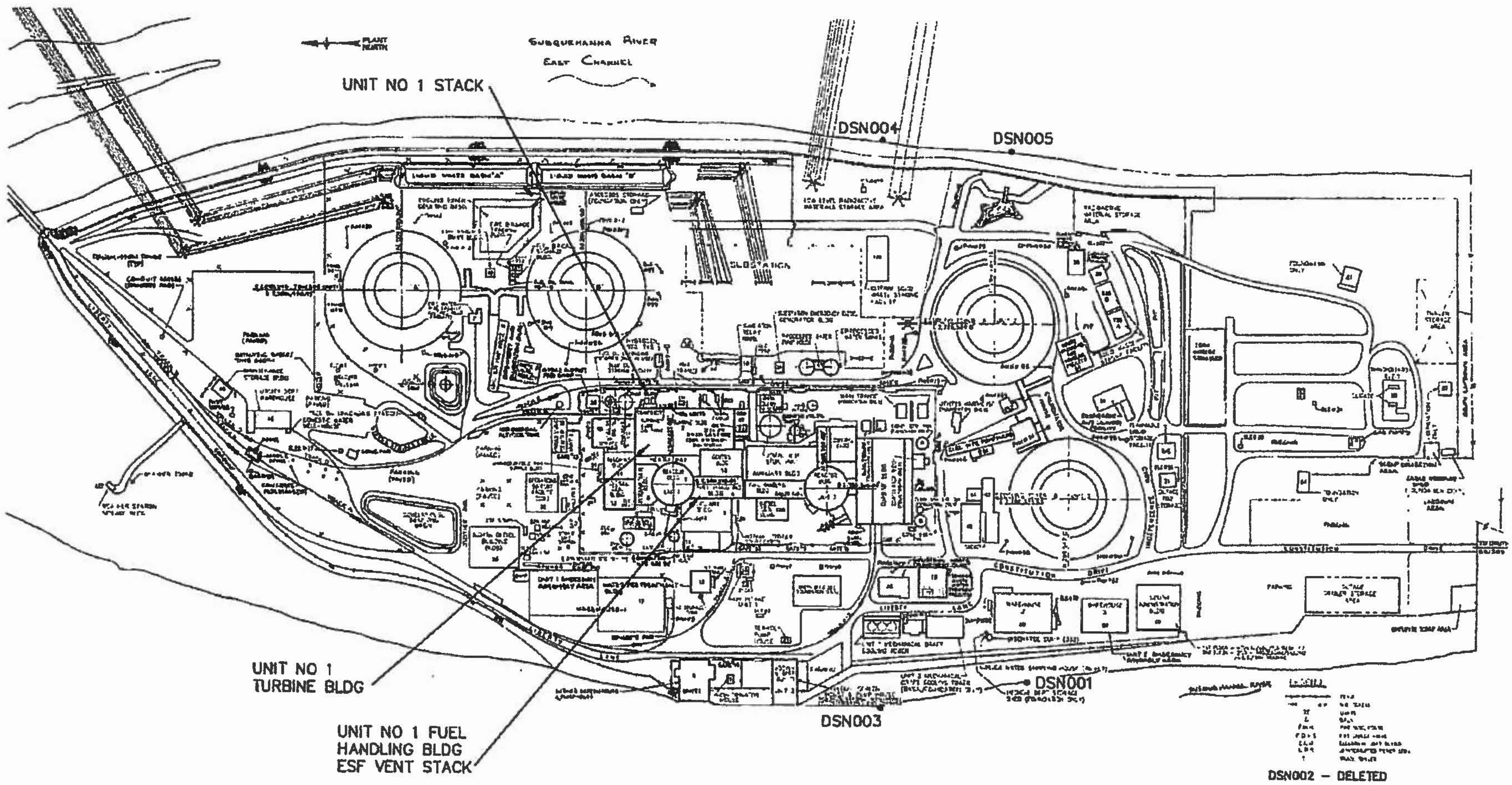
Site Exclusion Area and Low Population Zone

Fig. 5-2



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Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

FIG 5-3

Amendment No. ~~140, 218, 246, 269~~