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An Exelon/British Energy Company

10 CFR 50.90

5928-02-20074
September 30, 2002

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

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

SUBJECT: License Amendment Request (LAR) No. 300 – Containment Isolation Valve Technical Specifications and Administrative Changes

Pursuant to 10 CFR 50.90, AmerGen Energy Company, LLC (AmerGen) is requesting changes to the Technical Specifications (TS), Appendix A of the facility operating license listed above.

The purpose of this LAR is to provide clarification of the Technical Specifications (TS) related to containment isolation valves (CIVs). In discussions with NRC management related to the failure of a motor operator for a Main Steam Isolation Valve (MSIV)[1], the NRC noted that TMI Unit 1 TS 4.8 is incomplete in that the TS require a surveillance for stroke timing the MSIVs but contain no Limiting Conditions of Operation (LCOs) or required action times for these components. This LAR provides additional guidance on the handling of operability and reportability issues related to the Type III[2] CIVs that resulted from our review of the appropriateness of TS 4.8.

Other administrative changes included with this LAR include an update to Figure 5-1, "Extended Plot Plan;" an update to Figure 5-3, "Gaseous Effluent Release Points and Liquid Effluent Outfall Locations," and the list of "Locations of Liquid Effluent Outfalls Pursuant to NPDES," that refers to Figure 5-3; deletion of references to TS sections that have been removed by previous amendments; minor editorial changes; and appropriate changes to the Bases.

Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this letter and its attachments are being provided to the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection, as well as the chief executives of the township and county in which the facility is located.

A001

[1] TMI Unit 1 Licensee Event Report (LER) 99-012-00, dated November 17, 1999.
[2] As defined in Updated Safety Analysis Report (UFSAR) Section 5.3.2, a Type III CIV is a check valve or remotely operated valve located external to the Reactor Building, not connected directly to the Reactor Coolant System (RCS), and not open to the Reactor Building atmosphere, where the second isolation barrier for the penetration may be a closed loop that has a low probability of rupture during an accident.

Additionally, there are no commitments contained within this letter.

These proposed changes have been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the AmerGen Quality Assurance Program. AmerGen requests approval of this change by October 1, 2003, the amendment authorizing this change become effective upon issuance, and implemented within 60 days.

. If you have any questions or require additional information, please contact us.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on 9-30-02

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosures:
Attachment 1 - TMI Unit 1 LAR No. 300, Safety Evaluation and No Significant Hazards Consideration
Attachment 2 - TMI Unit 1 LAR No. 300, Proposed Changes to the TMI Unit 1 Technical Specification and Bases Pages, Markup and Typed Draft Pages

cc: USNRC, Regional Administrator, Region I
USNRC, Senior 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 of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
File No. 00091

Attachment 1

TMI Unit 1 License Amendment Request (LAR) No. 300

Safety Evaluation and No Significant Hazards Consideration

1.0 License Amendment Request (LAR) No. 300

AmerGen requests changes to the following Technical Specification (TS) pages:

Pages iii, iv, 1-5, 3-12, 3-14, 3-15a, 3-33, 3-41, 3-41a, 3-41c, 3-41d, 4-8, 4-38, 4-51, Figure 5-1, Figure 5-3, and 5-10.

A hand markup of the current TS and Bases pages is enclosed as Attachment 2. Camera ready pages are to be provided later when requested by the NRC.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The purpose of this LAR is to accomplish the following:

1) Revise the definition of containment integrity to ensure that all power-operated valves, relief valves, and check valves are included. This change provides operability requirements to include the Type III Containment Isolation Valves (CIVs), those valves that are in line with a containment isolation barrier consisting of a closed system within containment (e.g., Main Steam Isolation Valves).

2) Revise the applicability of CIV operability requirements for the plant conditions when containment integrity applies and the reactor is not critical.

3) Clarify that the exceptions to containment integrity provided in TS 3.6.1 apply equally to TS 3.6.2, whenever containment integrity is required.

4) Incorporate provisions for intermittent manual operation of CIVs under appropriate administrative control.

5) Delete TS 4.8, "Main Steam Isolation Valves," along with the reference to TS 4.8 in Table 4.1-2, Item No. 6. This change deletes a monthly requirement for a partial stroke test, but does not affect testing performed in accordance with the ASME Code, which will continue to assure the operability of the MSIVs.

6) Revise Figure 5-1, "Extended Plot Plan," to delete inaccurate information regarding the electrical transmission lines serving TMI Unit 1.

7) Revise the Figure 5-3, "Gaseous Effluent Release Points and Liquid Effluent Outfall Locations," and the accompanying table listing the "Locations of Liquid Effluent Outfalls Pursuant to NPDES" that refers to Figure 5-3 to reflect the modification which permanently isolated the liquid outfall associated with emergency discharge from TMI Unit 2.

8) Delete references to TS sections that have been deleted by prior amendments.

9) Delete the setpoint range provided for ECCS cubicle leak detection from the Bases of TS 3.1.6. Since calculations have shown that the alarm setpoint is sufficiently conservative, the range is not needed. Therefore, the bases are being revised to reflect 13 gpm as the basis flow rate for leakage detection rather than 13$\pm$ 2 gpm.

10) Make minor administrative and editorial changes to improve the consistency and clarity of the technical specifications.

The following lists the changes proposed by LAR No. 300, addressing each of the affected pages (referring to the existing TS page numbers):

Page iii
This Table of Contents page is revised to reflect the deletion of Specification 4.4.4, "Hydrogen Recombiner System," by TMI Unit 1 License Amendment No. 240.

Page iv
This Table of Contents page is revised to reflect the deletion of Surveillance Specification 4.8, "Main Steam Isolation valves."

Editorial Changes on this page include:
The degree symbol in TS 4.9.1 and 4.9.2 is revised to spell out the word "degrees."

Page 1-5
1) To ensure that all CIVs are included in the definition of containment integrity, TS 1.7.b and 1.7.c are revised to classify the CIVs in terms of 1) passive CIVs and isolation devices, including manual valves and blind flanges rather than nonautomatic CIVs and 2) active CIVs, including power-operated valves, check valves, and relief valves, rather than automatic CIVs.
2) TS 1.7.b is revised to provide that normally closed passive CIVs may be unisolated intermittently under administrative control. These administrative controls are defined in the revised Bases for TS 3.6.
3) TS 1.7.c is revised to provide that normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control. The administrative controls are defined in the revised Bases for TS 3.6.

Editorial Changes on this page include:
1) The terms "Containment integrity" and "Operable" are changed to all capital letters consistent with the convention throughout the TS for terms that are defined in TS Section 1.
2) TS 1.7.b is revised to introduce the acronym "CIVs" for Containment Isolation Valves and the first letter of the words for these components is capitalized as is the convention throughout the TS.
3) TS 1.7.b is revised to add a comma between the word procedure and the title of the procedure to improve the grammar.
4) Commas are also included in TS 1.7.b and 1.7.c to separate the phrase that begins with "including...".
5) TS 1.7.b is revised to move the quotation marks outside of the period at the end of the sentence, as a grammatical correction.
6) TS 1.7.c is revised to make use of the acronym "CIVs."

Page 3-12
1) Specification 3.1.6.6 is revised to delete the reference to Specification 3.22.2.1 and refer to the dose rate limit for gaseous radiological effluent, which was relocated from the TS to the ODCM in License Amendment No. 197.

Editorial Changes on this page include:

1) The word "shutdown" is revised to "shut down" as a grammatical correction.
2) The word "cool-down" is revised to "a cooldown" to delete the hyphen, consistent with use of the word in plant procedure titles and clarify use of the word "cooldown" as a noun.

Page 3-14

This Bases for Specification 3.1.6, "Leakage," is revised to delete the reference to TS 3.22.2.1 and the discussion of calculations relating the reactor coolant leakage limits to the effluent dose rate limits and refer instead to the dose rate limits of the Offsite Dose Calculation Manual (ODCM). This reference to TS 3.22.2.1 along with the discussion of a calculation should have been deleted as part of the change approved by the NRC in Amendment No. 197, which transferred the procedural details of the Radiological Effluent Technical Specifications (RETS) from the TS to the ODCM.

Page 3-15a

This Bases page for Specification 3.1.6, "Leakage," is being revised to delete the alarm setpoint range (13 $\pm$ 2 gpm) for the leakage detectors in the 7 individual Safeguards Cubicles (Makeup, Decay Heat, and Building Spray Pump cubicles) of the Auxiliary Building (AB) since the detectors' alarms are set conservatively and sufficiently below the design basis flow of 13 gpm and that a setpoint range is no longer necessary or relevant.

A modification was performed to provide loop seals for the safeguards cubical drains to prevent an operational problem of airborne radioactivity migration through the AB drains system. The modification caused a reduction in the flow rate through the drain before the liquid level backs up and energizes the excessive leak alarm. The alarm at the loop seal of the leak detectors will be activated conservatively below 13 gpm.

UFSAR Section 6.4.5, "Safeguards Cubicle Leak Detection," states: "Although it is anticipated that the maximum leak that will occur in any safeguards pump cubicle is much less than 300 gpm, the system is analyzed for a 600 gpm leak to demonstrate that even with twice the design leakage, the operator still has adequate time to isolate the leaking component." Further, the UFSAR states that 13 gpm is the flow rate above which leakage within these safeguards cubicles would be indicative of a significant failure, which may require operator action to isolate the leaking component.

Page 3-33

Specification 3.5.2.2.a is being revised to delete a reference to Specification 4.7.2.3, which was deleted by TMI Unit 1 License Amendment No. 211.

Page 3-41

1) TS 3.6.2 is revised to incorporate the same exceptions that are included in TS 3.6.1, which are valid exceptions that apply whenever containment integrity is required.

2) The revised text for TS 3.6.6 is revised to read as follows:

   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.**

The revised TS 3.6.6 incorporates the following changes.

a. CIV operability requirements are revised for applicability to the plant conditions when containment integrity applies even if the reactor is not critical.
b. The action for TS 3.6.6 is broken out into two subsections 3.6.6.a and 3.6.6.b for those containment penetrations isolable by two or more CIVs and those that may be isolable by only one CIV. This broadens 3.6.6 to provide action for the inoperability of those valves located in a containment penetration that takes credit for a closed system within containment as the second penetration barrier. These CIVs are included with a 72 hour action time within which that one CIV in the line must be closed and deactivated or a plant shutdown is required.
c. The action statement in TS 3.6.6.a is broadened to refer to containment penetrations with "two or more CIVs" since there may be more than two valves, only one of which may be closed to completely isolate the penetration, or closure more than one CIV may be required to isolate the penetration.

<u>Editorial Changes on this page include</u>:

1) The term "Containment integrity" is changed to all capital letters in four locations on this page consistent with the convention throughout the TS for terms that are defined in TS Section 1.
2) TS 3.6.1 is revised to add the word "Specifications" preceding the list of sections containing exceptions to the applicability of containment integrity.
3) The degree symbol in TS 3.6.1 is revised to spell out the word "degrees."
4) The ambiguous phrase, "inoperable in a position other than the required position," is shortened to the term "inoperable" because the additional words are not needed for a proper interpretation of the specification.
5) TS 3.6.5 is revised to introduce the acronym "CIVs" for the Containment Isolation Valves and the first letter of the words for these components is capitalized as is the convention throughout the TS.
6) TS 3.6.6 is revised to incorporate the acronym "CIV" in place of the equivalent term "reactor building isolation valve."

Page 3-41a

Editorial Changes on this page include:

1) The term "Containment integrity" is changed to all capital letters in TS 3.6.8 consistent with the convention throughout the TS for terms that are defined in TS Section 1.
2) In TS 3.6.8 the abbreviated term "T.S " is revised to the term "Specifications" in the reference to TS 3.6.1.
3) The degree symbols in TS 3.6.9 are revised to spell out the word "degrees."

Page 3-41c

The Bases for Specification 3.6, "Reactor Building," is being revised to incorporate appropriate bases for the Containment Isolation Valves (CIVs) operability requirements, applicability and actions required when CIVs become inoperable. The administrative controls required when CIVs are permitted to be unisolated or manual control may be substituted for automatic control are defined in the revised Bases in accordance with guidance provide in NRC Generic Letter 91-08.

Editorial Changes on this page include:

1) The degree symbol on this page is revised to spell out the word "degree."

Page 3-41d

A new reference is provided for NRC Generic Letter 91-08, which provides the guidance for clarification of the administrative controls required when CIVs are permitted to be unisolated or manual control substituted for automatic control.

Editorial Changes on this page include:

1) The header is revised to be consistent with other TS sections.
2) The existing Reference 1 is renumbered to Reference 2 by adding the new reference to sequence the references in the order that they appear in the Bases.

Page 4-8

Table 4.1-2 Item No. 6, Main Steam Isolation Valves is being deleted since the referenced Specification 4.8 is being deleted. With the deletion of Specification 4.8, the Inservice Test requirements in accordance with the ASME Boiler and Pressure Vessel Code, Specification 4.2.2, and 10 CFR 50.55a(f) still apply. Therefore, this item is redundant and need not be listed in the table.

Editorial Changes on this page include:

The degree symbol in Table 4.1-2, Item No. 7 is replaced by the word "degrees."

Page 4-38

Editorial Changes on this page include:

1) This page is being revised to reflect the deletion of Specification 4.4.4, "Hydrogen Recombiner System," by TMI Unit 1 License Amendment No. 240.
2) The page number at the bottom of this page is being corrected since this page was not deleted.

Page 4-51
Surveillance Specification 4.8, "Main Steam Isolation Valves" is being deleted. The effect of this change is to revise the frequency of the partial stroke test of the valves from monthly to a quarterly test in accordance with the Section XI of the ASME Boiler and Pressure Vessel Code Inservice Testing Requirements. Full stroke testing each refueling in accordance with Code requirements is unaffected.

Figure 5-1
This figure is being revised to show the "transmission lines" leaving the TMI Unit 1 site and to delete the destination (name) of the transmission lines which is incorrect as shown. The current figure shows the destination of transmission lines that do not exist as labeled. Since the text of UFSAR Section 8.2.1 provides a correct detailed description of the electrical transmission network, there is no need to provide the number and destination of the transmission lines on the extended plot plan.

Figure 5-3
This figure is being updated to show that NPDES Outfall DSN 002, the emergency outfall from the TMI-2 Mechanical Draft Cooling Tower has been permanently isolated by a modification.

Page 5-10
The description of NPDES Outfall DSN 002, the emergency discharge from the TMI-2 Mechanical Draft Cooling Tower, is being updated to show that it has been deleted since it was permanently isolated by a modification and is no longer a potential liquid effluent release point. This outfall does not appear in the current TMI site NPDES permit.

Editorial Changes on this page include:
As an editorial change in the description of NPDES Outfall No. DSN 004, the acronym "NDCT" is changed to "MDCT" as a correction since it should be described as the Emergency Discharge from Unit 1, if Unit 1 Mechanical Draft Cooling Tower (MDCT) is blocked, not if Unit 1 Natural Draft Cooling Tower (NDCT) is blocked.

3.0 SAFETY EVALUATION JUSTIFYING THE CHANGE

1) BACKGROUND

In discussions with NRC management related to the failure of a motor operator for an MSIV[1], the NRC noted that TMI Unit 1 TS 4.8 is incomplete in that it requires a surveillance for stroke timing the MSIVs but contains no Limiting Conditions of Operation (LCOs) or required action times for these components. This LAR provides closure for the recommendations for additional guidance on the handling of operability and reportability issues related to the Type III[2] CIVs that resulted from our review of the appropriateness of TS 4.8.

The TMI Unit 1 Updated Final Safety Analysis Report (UFSAR), Section 5.3.2, classifies the containment isolation design requirements for fluid penetrations that require isolation after an accident as follows:

Type I: Each line connecting directly to the Reactor Coolant System has two Reactor Building isolation valves. One valve is external and the other is internal to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves, depending on the direction of normal flow.

Type II: Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is external and the other may be internal or external to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves, depending on the direction of normal flow.

Type III: Each line not directly connected to the Reactor Coolant System or not open to the Reactor Building atmosphere has at least one valve, either a check valve or a remotely operated valve. This valve is located external to the Reactor Building. A closed loop, which has a low probability of rupture during an accident, may be used as the second isolation barrier.

Type IV: Lines that penetrate the Reactor Building and are connected to either the building or the Reactor Coolant System, but which are never opened during reactor operation, have two normally closed barriers. (e.g. blind flange, closed valve).

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[1] TMI Unit 1 Licensee Event Report (LER) 99-012-00, dated November 17, 1999.

[2] As defined in Updated Safety Analysis Report (UFSAR) Section 5.3.2, a Type III CIV is a check valve or remotely operated valve located external to the Reactor Building, not connected directly to the Reactor Coolant System (RCS), and not open to the Reactor Building atmosphere, where the second isolation barrier for the penetration may be a closed loop that has a low probability of rupture during an accident.

2) Definition of Containment Integrity:

The definition of Containment Integrity is revised to ensure that all of the CIVs and closure devices are included as appropriate. This change provides additional conservatism since the current TS do not provide an action statement for remotely operated non-automatic CIVs.

Specification 1.7.b and 1.7.c are revised to classify the CIVs in terms of 1) passive CIVs and isolation devices, including manual valves and blind flanges rather than nonautomatic CIVs and 2) active CIVs, including power-operated valves, check valves, and relief valves, rather than automatic CIVs. The terms automatic or nonautomatic as used in the current TS to classify CIV are not precisely defined.

Specifications 1.7.b and 1.7c are revised to include the provisions for intermittent opening of normally closed CIVs or manual control of power-operated valves under administrative control, while excluding from these provisions the Reactor Building Purge Valves. These provisions are consistent with the Standard Technical Specifications for Babcock and Wilcox Plants, NUREG-1430, Revision 2 (STS) Specification 3.6.3 and similar to provisions for manual control of EFW valves during surveillance testing to provide an EFW flow path in TS 4.9.1.2. These administrative controls are defined in the revised Bases for TS 3.6 consistent with the NRC position stated in NRC Generic Letter 91-08. Therefore in the event of an accident, isolation of containment will be accomplished under these provisions without exceeding the calculated dose projections of the accident analyses.

3) Containment Isolation Valves (CIV) Operability Requirements

TS 3.6.6 contains the Limiting Conditions of Operation for all CIVs except the Reactor Building Purge Valves, which are covered by TS 3.6.8. The applicability of TS 3.6.6 is being revised to apply operability requirements for the CIVs "when containment integrity is required" rather than "when the reactor is critical." This provides additional conservatism by applying the operability requirements for CIVs during periods of heatup or cooldown when Containment Integrity is required in addition to the operating conditions when the reactor is not critical. This resolves an apparent discrepancy since containment integrity is required by the current TS under conditions with nuclear fuel in the core, the Reactor Coolant System (RCS) pressure above 300 psig, and temperature above 200 degrees F although current TS 3.6.6 actions do not apply if the reactor is not critical.

Clarification of the current TS 3.6.2 is needed because the same exceptions as provided in TS 3.6.1 are intended to apply whenever containment integrity is required. Therefore, the words, "Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12," are being added to TS 3.6.2.

The phrase, "inoperable in a position other than the required position," is being deleted since it is somewhat ambiguous and no longer needed for the interpretation that when a CIV that is closed it is meeting its containment isolation function. No further action is required in accordance with TS 3.6.6 if at least one CIV in a penetration line is closed to isolate the penetration and thus the containment isolation function for that penetration is accomplished.

TS 3.6.6 is broken into two subsections. The revised TS 3.6.6.a applies to Reactor Building penetration lines isolable by two or more CIVs. Whereas the current specification addresses containment penetration lines having two CIVs, the revised wording in TS 3.6.6.a addresses penetration lines with two or more CIVs. Although the meaning is unchanged, this revision clarifies the intent of the action to isolate the penetration.

TS 3.6.6.b is included to apply to the penetrations that use a closed system inside the Reactor Building as a penetration barrier. Since another containment barrier in the line with a is a closed system, there may be only one CIV that can be closed. It is unlikely that a closed system would be inoperable as a containment barrier. A required action time of 72 hours is applied consistent with the STS 3.6.3, Completion Time for Action C.1.

4) Surveillance Specification 4.8, "Main Steam Isolation Valves," and Table 4.1-2, "minimum test frequency," Item No. 6, "Main Steam Isolation Valves"

If an MSIV were to fail the partial or full stroke test, the current TS has no operability or action requirements associated with this surveillance requirement. This specification is being deleted since, except for the requirement for a partial stroke test each month, this surveillance specification is redundant to the Inservice Testing (IST) requirements of the ASME Code Section XI, TS 4.2.2, and 10 CFR 50.55a(f). Specification 4.8 was included in the original TMI Unit 1 TS, which were issued in 1974 prior to implementation of the IST requirements for Class 2 and 3 components in 1978. There is no recommendation from the manufacturer to perform this test more frequently than quarterly. Therefore, with the deletion of Specification 4.8, the partial stroke test requirement can be changed from monthly to quarterly in accordance with IST requirements for all safety related valves.

Table 4.1-2, "Minimum Equipment Test Frequency," Item No. 6, "Main Steam Isolation Valves," which refers to TS 4.8 is also being deleted. With the deletion of specification 4.8, this item will be redundant and imposes no additional test requirements beyond that of IST in accordance with the ASME Code, TS 4.2.2, and 10 CFR 50.55a(f).

In accordance with the requested amendment, the MSIVs will be partial stroke tested once per quarter and full stroke tested once per refueling interval. The IST program in conjunction with the requested change to TS 3.6.6 described above, will provide applicable TS required actions should the MSIVs fail to stroke as required.

TMI is currently committed to applicable parts of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987 Edition as required by the ASME Section XI Division I, 1989 Edition. The TMI Unit 1 submittal of the IST program to the NRC committed to testing of the MSIVs every three months in accordance with the OM Part 10. However, TMI meets the IST partial stroke testing frequency requirement due to testing the MSIVs each month per TS 4.8. Thus, by deleting T.S. 4.8, the partial stroke testing of the MSIVs can be revised from monthly to quarterly. The frequency of full stroke testing each refueling in accordance with the ASME Code remains unchanged. Changing the partial stroke test frequency form monthly to quarterly is more conservative than the STS . In accordance with the STS, the MSIVs are not required to be partial stroke tested due to the risk of valve closure with the unit generating power. AmerGen intends to continue partial stroke testing of the MSIVs

in accordance with the Code because this test has been conducted once each month during power operation since the initial startup of TMI Unit 1 in 1974 and is judged to have some benefit without subjecting the plant to an increased risk of a trip at power.

The MSIV stop check function is unaffected by this change and will continue to be tested as part of the IST program each refueling interval in accordance with surveillance procedure 1300-3Y "MS Isolation Valve Closure IST."

5) Revision of TS Figure 5-1, "Extended Plot Plan," to Deletion Inaccurate Information Regarding The Electrical Transmission Lines

TS Figure 5-1 is being revised to delete the detail regarding transmissions line destinations leaving the TMI Unit site and label them "transmission lines." The additional detail is not used and has been found to be inaccurate. Since the text of UFSAR Section 8.2.1 provides a good verbal description of the existing transmission lines and their destinations, there is no need for this detail on Figure 5-1, "Extended Plot Plan. Therefore the label on this figure is being changed to read, "Transmission Lines."

6) Revision of TS Figure 5-3, "Gaseous Effluent Release Points and Liquid Effluent Outfall Locations," and the list of Locations of Liquid Effluent Outfalls Pursuant to the National Pollutant Discharge Elimination System (NPDES), to Reflect the Isolation of a Potential Release Path

This is an administrative change to correct an error in Section 5, "Design Features." TS Figure 5-3, "Gaseous Effluent Release Points and Liquid Effluent Outfall Locations," and the list of "Locations of Liquid Effluent Outfalls Pursuant to NPDES" on TS Page 5-10 are being revised to reflect the isolation of the emergency release path from the TMI-2 Mechanical Draft Cooling Tower Outfall (DSN 002). Reference to NPDES Outfall DSN 002, is being deleted because this outfall was isolated by modification MMA-3165-89-0169, "Mechanical and Electrical Isolations for TMI-2 MDCT," which was performed in the mid-1990s when the TMI-2 Mechanical Draft Cooling Tower was dismantled. This change was not identified by the safety evaluation for the review of the modification. No other technical specifications are affected. Outfall DSN 002 is not included in the current station NPDES permit (PA 0009920), effective September 1, 1997.

7) Deletion of References to TS Sections Deleted by Previous TS Amendments

a) The Table of contents is being revised to delete the entry for TS 4.4.4, Hydrogen Recombiner System, which was deleted by TMI Unit 1 License Amendment No. 240.

b) TS 3.5.2.2.a is being revised to delete a reference to Surveillance Specification 4.7.2.3, regarding the control rod program verification, which was deleted by TMI Unit 1 License Amendment No. 211, dated June 15, 1999.

c) TS 3.1.6.6 is being revised to delete a reference to TS 3.22.2.1, regarding the LCO for a gaseous radioactive dose rate limit that was deleted from the TS in TMI Unit 1 License Amendment No. 197, dated October 2, 1995, which relocated the procedural details of the Radiological Effluent Technical Specifications

(RETS) to the Offsite Dose Calculation Manual (ODCM). TS 3.1.6.6 and the associated Bases are therefore being revised to refer to the dose rate limits of the ODCM for evaluation of the safety implication of RCS leakage. This change does not affect the intent of the Technical Specifications and does not result in any change to the offsite dose or dose rate limits or the way dose or dose rates are calculated. The numerical value of this dose rate limit remains the same.

8) Editorial changes

The editorial changes included with this LAR are intended to improve the clarity, consistency, and readability of the TS. These changes do not affect equipment configuration or operation and do not affect the meaning or interpretation of any TS LCO or surveillance specification.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

The purpose of this change is to provide clarification of the Technical Specifications (TSs) related to containment isolation valves (CIVs) and assure that all CIVs are addressed in the TS as appropriate. Deletion of separate surveillance specification for Main Steam Safety Valves tests will permit testing these valves consistent with the ASME Code for safety related valves. Other administrative changes included with this LAR include an update to description of Design Features and deletion of references to technical specification sections that have been removed by previous amendments, minor editorial changes, and the incorporation of appropriate Bases changes.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

1. Operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Changes to the definition of containment integrity and the additional operability requirements for Containment Isolation Valves (CIVs) provide additional requirements and add clarity to the Technical Specifications. The addition of a provision for permitting intermittent opening of normally closed CIVs or manual control of power-operated CIVs under administrative control is consistent with the Standard Technical Specifications or a similar provision in the current TMI Unit 1 Technical Specifications. This assures that the containment will be isolated if necessary in the event of an accident previously evaluated and offsite dose from an accident will not be significantly increased. The additional operability requirements provide additional conservatism to the technical specifications.

   None of the changes included with this License Amendment Request will result in any change to the configuration of plant components, affect any accident initiators associated with any accident previously evaluated or result in a significant increase in the offsite dose consequences of accidents previously evaluated. The administrative changes are needed to correct errors and the editorial changes will improve the clarity, consistency, and readability of the Technical Specifications and do not affect the intent or interpretation.

   Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The changes associated with this proposed amendment do not result in any additional hardware or design changes to structures, systems, or components (SSCs) of the plant; nor will any of these changes affect the ability of an SSC to perform its design function. No new failure mechanisms, malfunctions, or accident initiators will be introduced that were not considered in the design and licensing basis.

   Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   Additional operability requirements provide conservative improvements to the Technical Specifications. The addition of a provision for permitting intermittent opening of normally closed CIVs or manual control of power-operated CIVs under administrative control is consistent with the Standard Technical Specifications or with similar provisions in the current TMI Unit 1 Technical Specifications. This condition assures that the containment will be isolated if necessary in the event of an accident. Changes to the MSIV test requirements do not alter the Inservice Test requirements in according to the American Society of Mechanical Engineers (ASME) Code, which will continue to assure operability. The administrative changes are needed to correct errors and the editorial changes will improve the clarity, consistency, and readability of the Technical Specifications and do not affect the intent or interpretation.

   None of the changes included with this request have the potential to significantly reduce a margin of safety. These changes do not affect the design of a plant component or instrument setpoint so as to effect its design basis or affect the controlling numerical value for any parameter established in the updated final safety analysis report or the license.

   Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Based on the negative responses to these three criteria, AmerGen has concluded that the requested change does not involve a significant hazards consideration.

5.0 ENVIRONMENTAL CONSIDERATIONS

10 CFR51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

AmerGen has reviewed this LAR and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These changes have been demonstrated to involve no significant hazards consideration. None of the changes included in this request have the potential to result in increased radioactive effluents released offsite or result in increased occupational radiation exposure. Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment.

Attachment 2

TMI Unit 1 License Amendment Request (LAR) No. 300

Proposed Changes to the TMI Unit 1

Technical Specification and Bases Pages

Markup and Typed Draft Pages

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1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left[\frac{\textit{Power in Any Core Quadrant}}{\textit{Average Power of All Quadrants}} - 1 \right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Caps

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.

b. All ~~nonautomatic~~ passive containment isolation valves (CIVs) and isolation devices, including manual valves, and blind flanges, are closed as required by the "Containment Integrity Check List" attached to the operating procedure, "Containment Integrity and Access Limits".

c. All ~~automatic containment isolation valves~~ active CIVs are operable (caps) or locked closed, including power-operated valves, check valves and relief valves,

d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

Normally closed passive CIVs may be unisolated intermittently under administrative control.

Normally closed active CIVs (other than the purge valves) may be unisolated intermittently and manual control of power-operated valves may be substituted for automatic control under administrative control.

3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the dose rate limits of the ODCM ~~Specification 3.22.2.1.~~

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

Amendment No. 47, 129, 180
(12-22-78)

When reactor coolant leakage occurs to the intermediate cooling closed cooling water system, the leakage is indicated by both the intermediate cooling water monitor (RM-L9) and the intermediate cooling closed cooling water surge tank liquid level indicator, both of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank.

When reactor coolant leakage occurs to either of the decay heat closed cooling water systems, the leakage is indicated by the affected system's radiation monitor (RM-L2 or RM-L3 for system A and B, respectively) and surge tank liquid level indicator, all four of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank of the affected system.

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the dose rate limits of ~~Specification 3.22.2.1~~ the ODCM would be exceeded. ~~This is shown as follows: if the specific activity of the reactor coolant is 130/E uCi/ml and the gaseous portion of it (as identified by UFSAR Table 11.1-2) is discharged to the environment via the unit's auxiliary and fuel handling building vent, the whole body dose rate resulting from this activity at the site boundary, using an annual average X/Q = 4.5 x 10^{-6} sec/m^3, is 0.34 rem/year. This may be compared with the gaseous effluent dose rate specified in Specification 3.22.2.1 of 0.5 rem/year.~~

When the reactor coolant leaks to the secondary sides of either steam generator, all the gaseous components and a very small fraction of the ionic components are carried by the steam to the main condenser. The gaseous components exit the main condenser via the unit's vacuum pump which discharges to the condenser vent past the condenser off-gas monitor. The condenser off-gas monitor will detect any radiation, above background, within the condenser vent.

However, buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

Bases (Continued)

The unidentified leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Total reactor coolant leakage is limited by this specification to 10 gpm. This limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage.

The primary to secondary leakage through the steam generator tubes is limited to 1 gpm total. This limit ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of a steam line break. Steam generator leakage is quantified by analysis of secondary plant activity.

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. Water inventories around the auxiliary building sump.

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 (± 2) gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by TMI-1 Plant Operations.

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 ~~and 4.7.2.3~~ in the safety or regulating rod banks shall not be permitted. Verify SDM $\geq$ 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 $\leq$ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to $\leq$ 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.

Amendment No. ~~17~~, ~~150~~, 211
(5-18-76)

3.6 REACTOR BUILDING

Applicability

Applies to the containment integrity (caps) of the reactor building as specified below.

Objective

To assure containment integrity (caps)

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°F (degrees) or greater.

c. Nuclear fuel is in the core.

Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12

3.6.2 Containment integrity (caps) 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% Δk/k shall not be made by control rod motion or boron dilution unless containment integrity (caps) 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, ~~While the reactor is critical~~, if a ~~reactor building isolation valve~~ CIV (other than a purge valve) is determined to be inoperable: ~~in a position other than the required position, the other reactor building isolation valve in the line~~ at least one CIV in the line shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, ~~the OPERABLE valve~~ will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within additional 30 hours.

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration

3.6.7 DELETED

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.

Amendment No. ~~87~~, ~~102~~, ~~108~~, ~~190~~, ~~198~~
240

* all CIVs required to isolate the penetration.

3.6 REACTOR BUILDING (Continued)

3.6.8 While containment integrity (CAPS) is required (see ~~T.S.~~ Specification 3.6.1), if a 48" reactor building purge valve is found to be inoperable perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (see Specification 6.8.5), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below:

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 6.8.5 and perform either (1) or (2) below.

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test in accordance with the Reactor Building Leakage Rate Testing Program. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31° degrees and (AH-V-1B&C) shall be limited to less than 33° degrees open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons:

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

Bases

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

see INSERT next page

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30~~0~~ degree open position.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

INSERT to Page 3-41c

The primary Containment Isolation Valves (CIVs) are identified in UFSAR Table 5.3-2. Additional vent, drain, test and other manually operated valves which complete the containment boundary are identified in the containment integrity checklist. For the purpose of this specification, check valves and relief valves identified in the containment integrity checklist are defined to be active valves.

The loss of redundant capability for containment isolation is limited for all penetrations, after which the containment penetration must be isolated. Isolation of certain penetrations may require the closure of multiple CIVs due to piping branches.

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.
2. For those CIVs where the second barrier is a closed system within the Reactor Building, there is no other CIV to isolate the penetration. If operability cannot be regained, the valve must be closed within 72 hours or the plant must commence shut down. An action time of 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a containment isolation boundary and the relative importance of supporting containment integrity.

The definition of Containment Integrity permits normally closed CIVs, except for the 48 inch purge valves, to be unisolated intermittently or manual control to be substituted for automatic control under administrative control. Administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment (Reference 1). The dedicated individual can be responsible for closing more than one valve provided that the valves are in close vicinity and can be closed in a timely manner. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the containment penetrations containing these valves may not be opened under administrative control.

BASES (Continued)

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Reference 1),[2] and the Reactor Building Leakage Rate Testing Program. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

Entry and exit is allowed to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock. With both air locks inoperable due to inoperability of one door in each of the two air locks, entry and exit is allowed for use of the air locks for 7 days under administrative controls. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

With one or more air locks inoperable for reasons other than those described in 3.6.12."b" or "c," Section 3.6.12.d requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour would otherwise be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Section 3.6.12.d requires that one door in the affected containment air locks(s) must be verified to be closed within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. 24 hours is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

References

(1) NRC Generic Letter 91-08.

(1)[2] 10 CFR 50, Appendix J.

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2.	Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3.	Pressurizer Safety Valves	Setpoint	in accordance with the Inservice Testing Program
4.	Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5.	Refueling System Interlocks	Functional	Start of each refueling period
6.	~~Main Steam Isolation Valves~~ (Deleted)	~~(See Section 4.8)~~ –	–
7.	Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F degrees
8.	(Deleted)	–	–
9	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10.	Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visualisual inspection of Intake Pump House Floor	Not to exceed 24 months
		(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11.	Pressurizer Block Valve (RC-V2)	Functional*	Quarterly

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

4.4.4 DELETED

~~(Page~~ 4-38 ~~deleted)~~
(Page 4-38a deleted)

4.8 ~~MAIN STEAM ISOLATION VALVES~~ DELETED

Applicability

Applies to the periodic testing of the main steam isolation valves.

Objective

To specify the minimum frequency and type of tests to be applied to the main steam isolation valves.

Specification

4.8.1 A check of valves stem movement, up to 10 percent, shall be performed on a monthly basis when the unit is operational and under normal flow and load conditions.

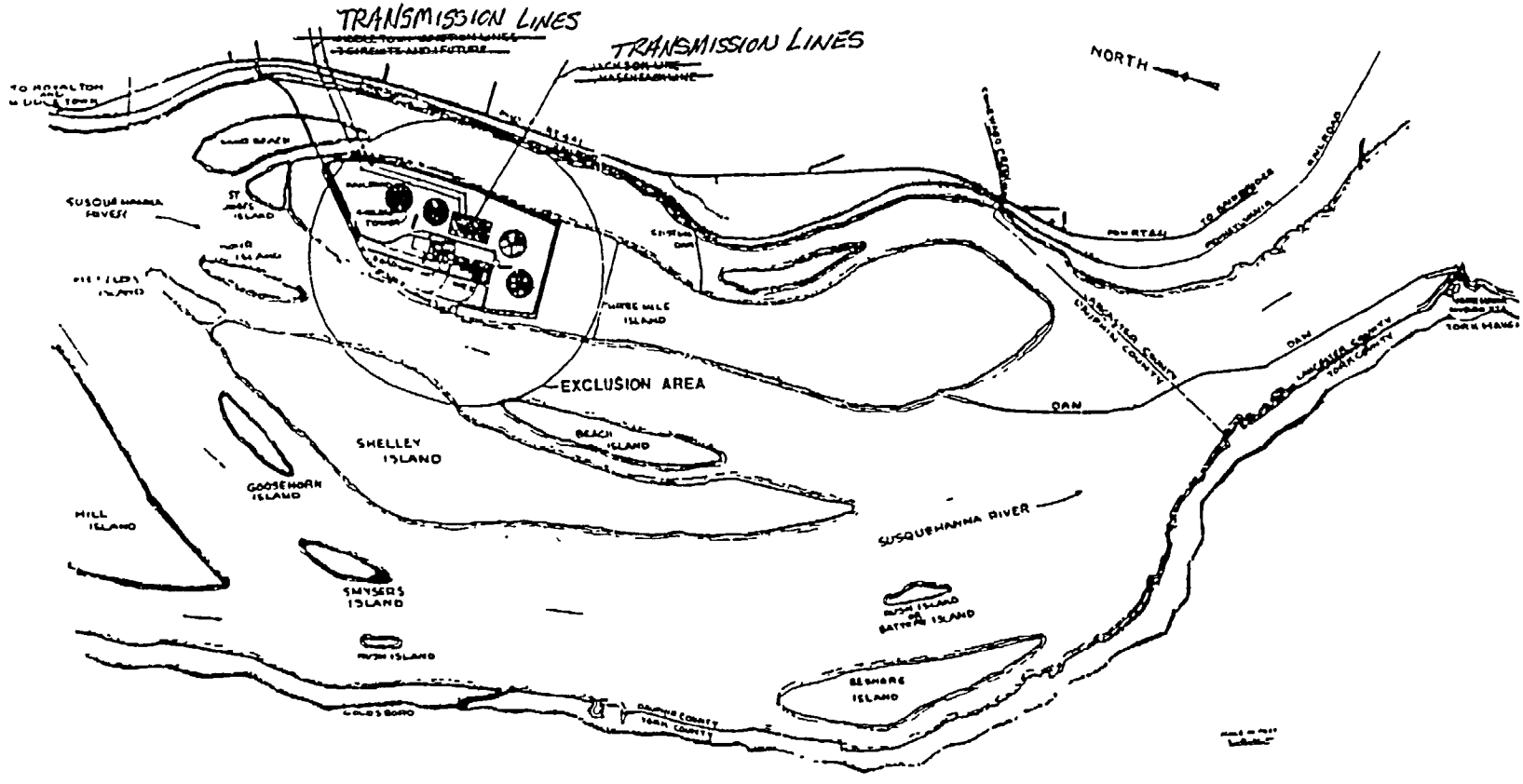
4.8.2 The main steam isolation valves shall be tested at intervals not to exceed the normal refueling outage. Closure time of <120 seconds shall be verified. This test will be performed under no flow and no load conditions.

Bases

Since a portion of the main steam lines and the steam lines to the main feed pump turbines are located in the turbine hall which is not protected against hypothetical tornado, missile, or aircraft incident; main steam isolation stop check valves are provided and located in the hardened portion of the intermediate building. These stop check valves are remotely closed by the operator from the control room, close in less than two minutes, and are tight closing for long term containment isolation (Reference 1). Their ability to close upon signal should be verified at intervals not to exceed each scheduled refueling shutdown, and valve stem freedom should be checked on a monthly basis.

References

(1) UFSAR, Section 10.3.1 - "Main Steam System" and Table 10.3-1 - "Main Steam Component Data"



AmerGen

Extended Plot Plan

Three Mile Island Nuclear Station

Fig 5.1

Amendment no ~~to~~ 218

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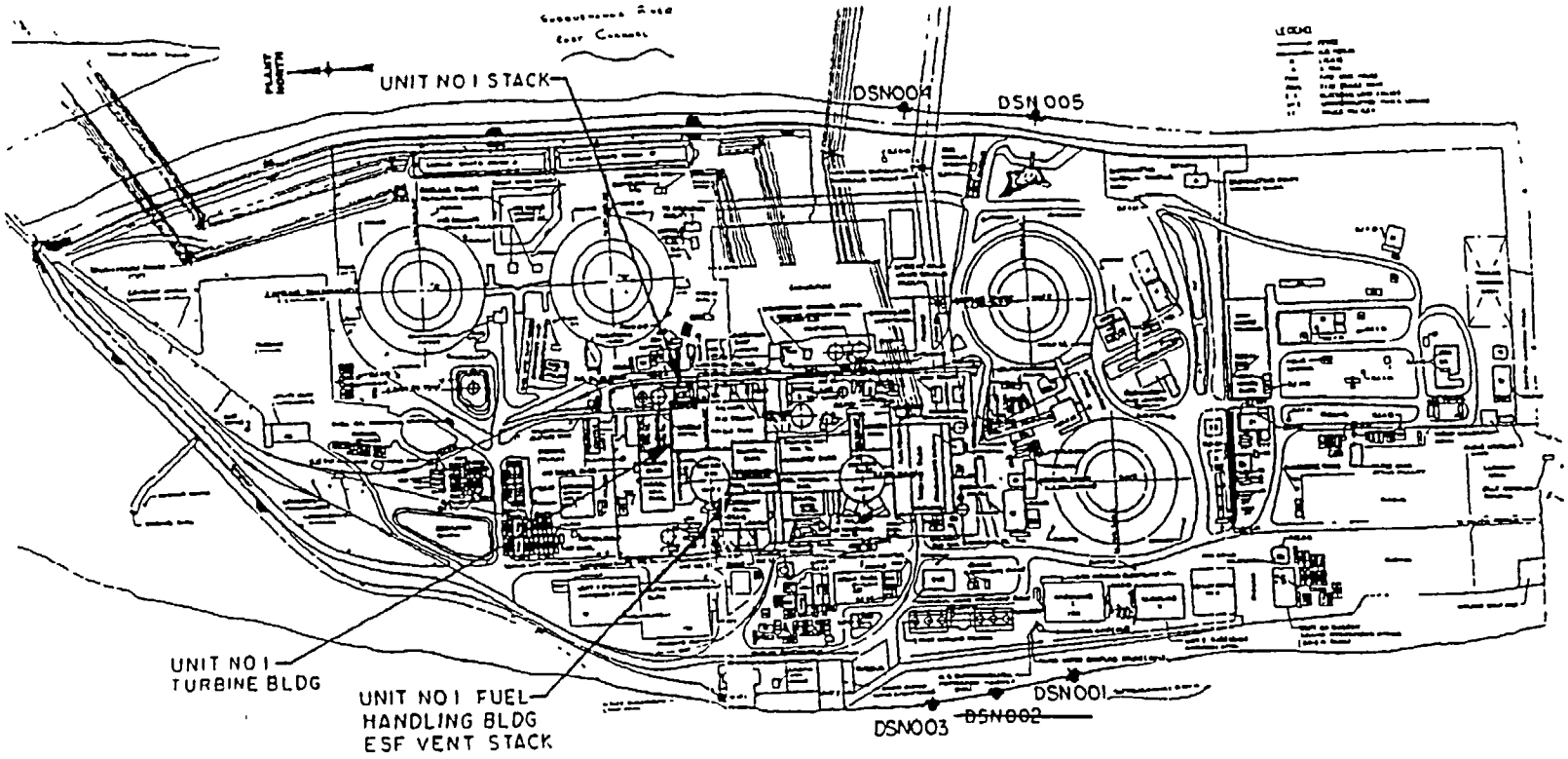

AmerGen

Gaseous Effluent Release Points and Liquid Effluent Outfall Locations

Three Mile Island Nuclear Station

Fig 5-3

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ELEVATIONS FOR GASEOUS EFFLUENT RELEASE POINTS
(See Figure 5-3)

Unit 1 Stack	483' 7"
Unit 1 Turbine Building	425' 4"
Unit 1 Fuel Handling Building ESF Vent Stack	348'

LOCATIONS OF LIQUID EFFLUENT OUTFALLS PURSUANT TO NPDES
(See Figure 5-3)

Outfall No.	Description
DSN 001	Main Station Discharge
DSN 002	~~Emergency Discharge from Unit 2 (if DSN 001 is blocked)~~ (Deleted)
DSN 003	Emergency Discharge from Unit 1 (if DSN 001 is blocked)
DSN 004	Emergency Discharge from Unit 1 (if Unit 1 NDCT blocked) CM
DSN 005	Stormwater and yard drainage and dewatering of natural draft cooling towers, maintenance dredging desiltation and basin dewatering, fire brigade training facility runoff, fire service water runoff.

mendment No. ~~72~~, ~~122~~, 149

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Typed Pages for License Amendment Request (LAR) No. 300

Technical Specification Changes and Bases Changes

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1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100\left[\frac{\text{Power in Any Core Quadrant}}{\text{Average Power of All Quadrants}} - 1\right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial Power Imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.

b. All **passive** Containment Isolation Valves **(CIVs) and isolation devices, including manual valves** and blind flanges, are closed as required by the "Containment Integrity Check List" attached to the operating procedure, "Containment Integrity and Access Limits." **Normally closed passive CIVs may be unisolated intermittently under administrative control.**

c. All **active CIVs, including power-operated valves, check valves, and relief valves,** are **OPERABLE** or locked closed. **Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.**

d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be **shut down**, and **a cooldown** to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the **dose rate** limits of **the ODCM**.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

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Bases (Continued)

When reactor coolant leakage occurs to the intermediate cooling closed cooling water system, the leakage is indicated by both the intermediate cooling water monitor (RM-L9) and the intermediate cooling closed cooling water surge tank liquid level indicator, both of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank.

When reactor coolant leakage occurs to either of the decay heat closed cooling water systems, the leakage is indicated by the affected system's radiation monitor (RM-L2 or RM-L3 for system A and B, respectively) and surge tank liquid level indicator, all four of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank of the affected system.

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the **dose rate** limits of **the ODCM** would be exceeded.

When the reactor coolant leaks to the secondary sides of either steam generator, all the gaseous components and a very small fraction of the ionic components are carried by the steam to the main condenser. The gaseous components exit the main condenser via the unit's vacuum pump which discharges to the condenser vent past the condenser off-gas monitor. The condenser off-gas monitor will detect any radiation, above background, within the condenser vent.

However, buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

Bases (Continued)

The unidentified leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Total reactor coolant leakage is limited by this specification to 10 gpm. This limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage.

The primary to secondary leakage through the steam generator tubes is limited to 1 gpm total. This limit ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of a steam line break. Steam generator leakage is quantified by analysis of secondary plant activity.

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. Water inventories around the auxiliary building sump.

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by TMI-1 Plant Operations.

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 ΔK/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% 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 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 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 $F^N_{\Delta H}$) limits per the COLR have not been exceeded within 72 hours.

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(5-18-76)

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 exit:

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.**

3.6 REACTOR BUILDING (Continued)

3.6.7 DELETED

3.6.8 While **CONTAINMENT INTEGRITY** is required (see **Specification** 3.6.1), if a 48" reactor building purge valve is found to be inoperable, perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (see Specification 6.8.5), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below.

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 6.8.5 and perform either (1) or (2) below:

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test in accordance with the Reactor Building Leakage Rate Testing Program. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31**degrees** and (AH-V-1B&C) shall be limited to less than 33 **degrees** open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons.

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

(3) Performance of Technical Specification required surveillances.

(4) Radiation Surveys.

BASES

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

The primary Containment Isolation Valves (CIVs) are identified in UFSAR Table 5.3-2. Additional vent, drain, test and other manually operated valves which complete the containment boundary are identified in the containment integrity checklist. For the purpose of this specification, check valves and relief valves identified in the containment integrity checklist are defined to be active valves.

The loss of redundant capability for containment isolation is limited for all penetrations, after which the containment penetration must be isolated. Isolation of certain penetrations may require the closure of multiple CIVs due to piping branches.

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.
2. For those CIVs where the second barrier is a closed system within the Reactor Building, there is no other CIV to isolate the penetration. If operability cannot be regained, the valve must be closed within 72 hours or the plant must commence shut down. An action time of 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a containment isolation boundary and the relative importance of supporting containment integrity.

The definition of Containment Integrity permits normally closed CIVs, except for the 48 inch purge valves, to be unisolated intermittently or manual control to be substituted for automatic control under administrative control. Administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment (Reference 1). The dedicated individual can be responsible for closing more than one valve provided that the valves are in close vicinity and can be closed in a timely manner. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the containment penetrations containing these valves may not be opened under administrative control.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable

of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30 **degree** open position.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Reference **2**), and the Reactor Building Leakage Rate Testing Program. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

Entry and exit is allowed to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock. With both air locks inoperable due to inoperability of one door in each of the two air locks, entry and exit is allowed for use of the air locks for 7 days under administrative Specifications (TS) Surveillance and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment and support TS-required equipment. This is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

With one or more air locks inoperable for reasons other than those described in 3.6.12 "b" or "c," Section 3.6.12.d requires action to be immediately initiated to evaluate previous combined leakage rates in using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door failed), containment remains OPERABLE, yet only 1 hour would otherwise be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

BASES (Continued)

Section 3.6.12.d requires that one door in the affected containment air locks(s) must be verified to be closed within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. 24 hours is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

References

(1) NRC Generic Letter 91-08
(2) 10 CFR 50, Appendix J.

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2. Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3. Pressurizer Valves	Setpoint	in accordance with the Safety Inservice Testing Program
4. Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5. Refueling System Interlocks	Functional	Start of each refueling period
6. (Deleted)	--	--
7. Reactor Coolant ystem Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525 **degrees** F
8. (Deleted)	--	--
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visual inspection of Intake Pump House Floor	Not to exceed 24 months
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer Block Valve (RC-V2)	Functional*	Quarterly

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

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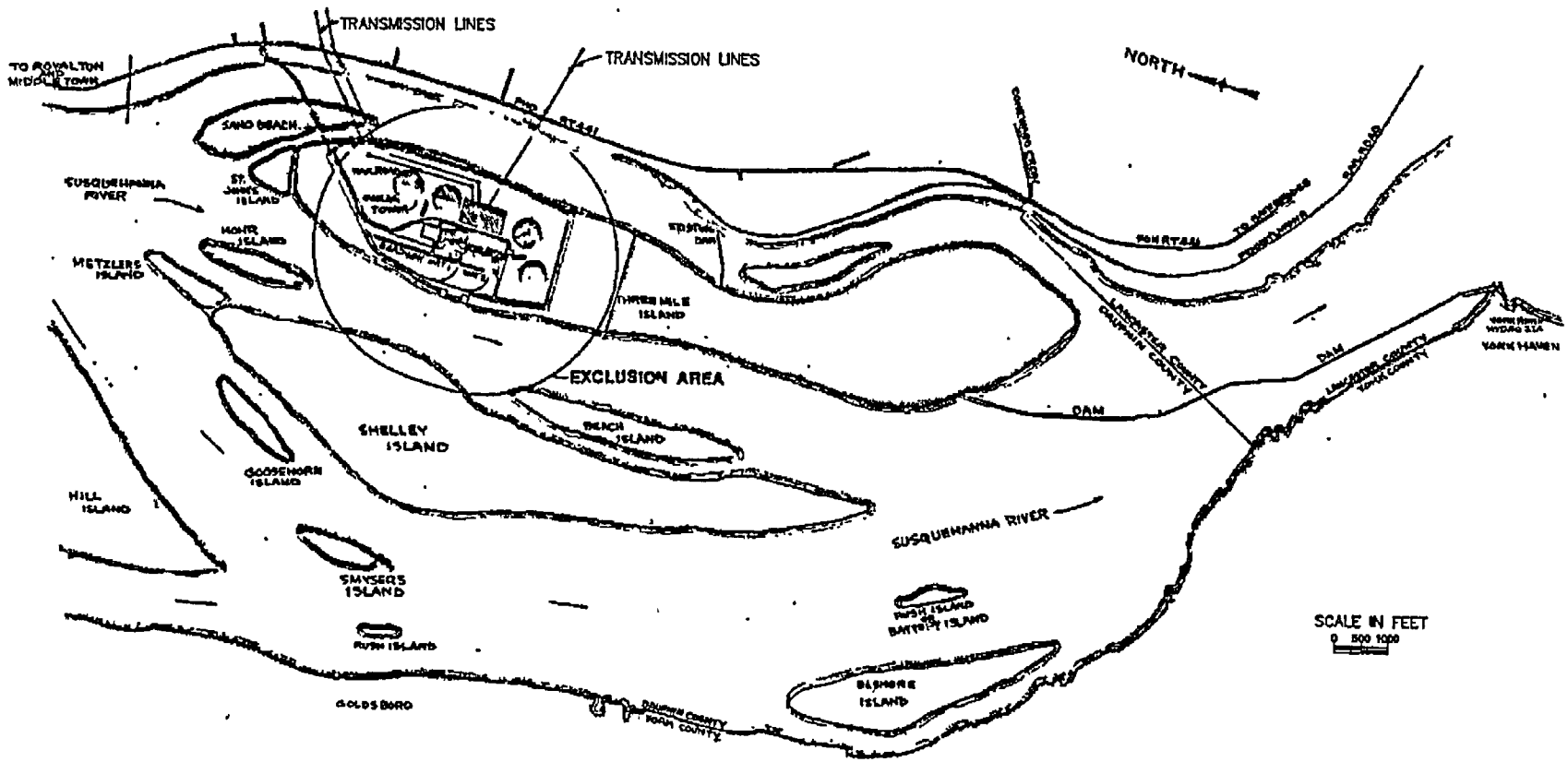
4.4.4 DELETED

4.8 **DELETED**



AmerGen
Three Mile Island Nuclear Station

EXTENDED PLOT PLAN

CAD FILE: 6717R1.DWG FIG 5-1

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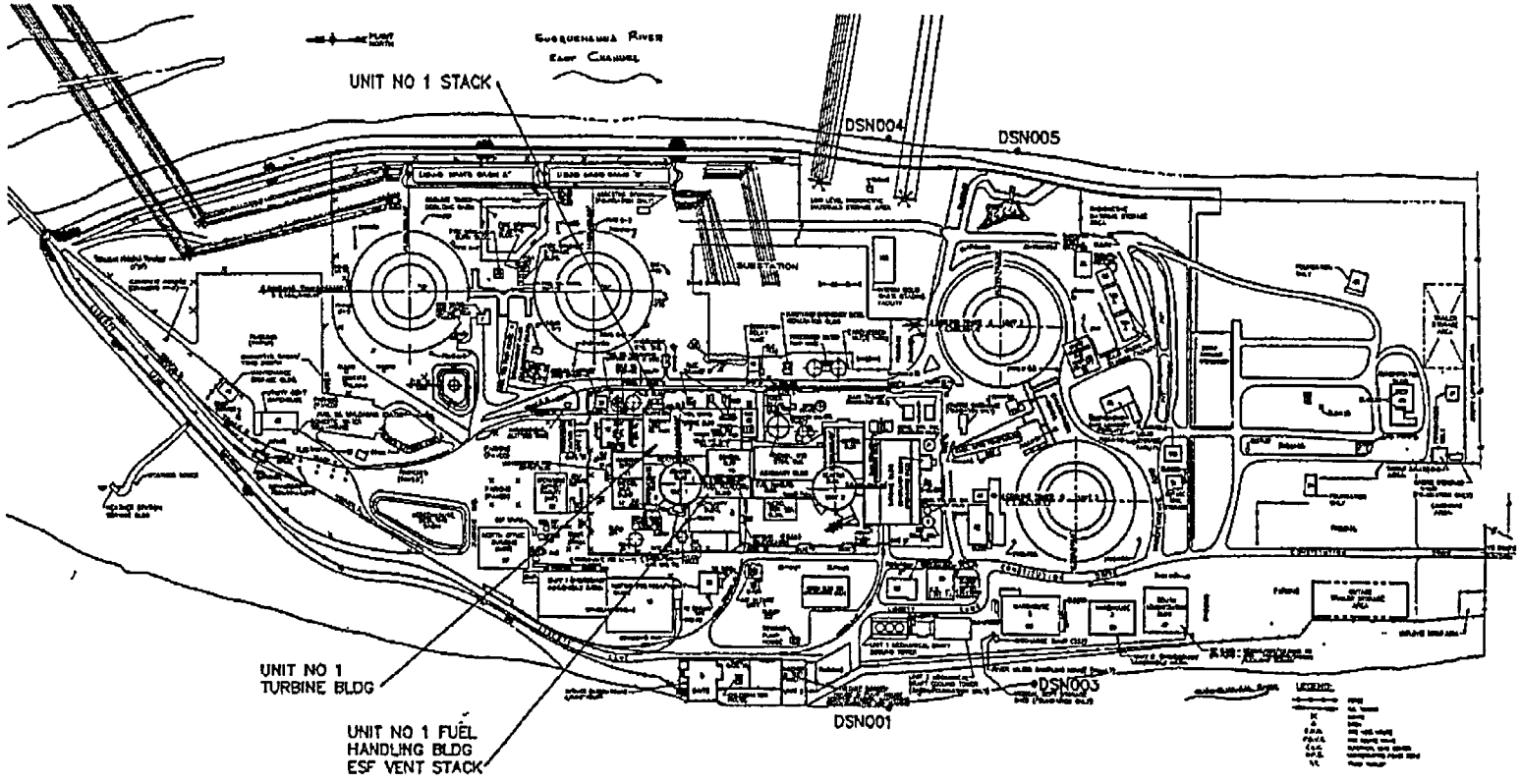

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AmerGen
Three Mile Island Nuclear Station

Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

CAD FILE: 6716R15.DWG

Fig. 5-3

ELEVATIONS FOR GASEOUS EFFLUENT RELEASE POINTS
(See Figure 5-3)

Unit 1 Stack	483' 7"
Unit 1 Turbine Building	425' 4"
Unit 1 Fuel Handling Building ESF Vent Stack	348'

LOCATIONS OF LIQUID EFFLUENT OUTFALLS PURSUANT TO NPDES
(See Figure 5-3)

Outfall No.	Description
DSN 001	Main Station Discharge
DSN 002	**(Deleted)**
DSN 003	Emergency Discharge from Unit 1 (if DSN 001 is blocked)
DSN 004	Emergency Discharge from Unit 1 (if Unit 1 MDCT blocked)
DSN 005	Stormwater and yard drainage and dewatering of natural draft cooling towers, maintenance dredging desiltation and basin dewatering, fire brigade training facility runoff, fire service water runoff.