Exelon Nuclear

THREE MILE ISLAND NUCLEAR STATION
UNIT 1

ISI PROGRAM PLAN
FOURTH TEN-YEAR INSPECTION INTERVAL

Commercial Service Date:

Unit 1 - 09/02/74

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Fourth Ten-Year Inspection Interval
Three Mile Island Nuclear Station, Unit 1

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Each time this document is revised, the Revision Approval Sheet will be signed and the following Revision Control Sheet should be completed to provide a detailed record of the revision history. The signatures above apply only to the changes made in the revision noted. If historical signatures are required, Three Mile Island Nuclear Station archives should be retrieved.

REVISION CONTROL SHEET

Major changes should be outlined within the table below. Minor editorial and formatting revisions are not required to be logged.

REVISION	DATE	REVISION SUMMARY
0	03/11/11	Initial issuance. (This ISI Program Plan was developed by Alion Science and Technology Corporation as part of the Fourth Interval ISI Program update.) Prepared: S. Coleman Reviewed: K. Johnson Approved: D. Lamond

Notes:

1. This ISI Program Plan (Sections 1 - 9 inclusive) is controlled by the Three Mile Island Nuclear Station Programs Engineering Group.
2. Revision 0 of this document was issued as the Fourth Interval ISI Program Plan and was submitted to the USNRC. Future revisions of this document made within the Fourth ISI Interval will be maintained and controlled at the Three Mile Island Nuclear Station; however, they are not required to be and will not be submitted to the USNRC. The exception to this is that new or revised Relief Requests shall be submitted to the USNRC for safety evaluation and approval.

REVISION SUMMARY

SECTION	EFFECTIVE PAGES	REVISION	DATE
Preface	i to vii	0	03/11/11
1.0	1-1 to 1-18	0	03/11/11
2.0	2-1 to 2-32	0	03/11/11
3.0	3-1 to 3-2	0	03/11/11
4.0	4-1 to 4-2	0	03/11/11
5.0	5-1 to 5-2	0	03/11/11
6.0	6-1 to 6-3	0	03/11/11
7.0	7-1 to 7-21	0	03/11/11
8.0	8-1 to 8-3	0	03/11/11
9.0	9-1 to 9-3	0	03/11/11

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1.0 INTRODUCTION AND BACKGROUND

1.1 Introduction

This Inservice Inspection (ISI) Program Plan details the requirements for the examination and testing of ISI Class 1, 2, 3, MC, and CC pressure retaining components, supports, containment structures, and post-tensioning systems at Three Mile Island (TMI) Nuclear Station, Unit 1. This ISI Program Plan also includes Containment Inservice Inspection (CISI), Risk-Informed Inservice Inspection (RISI), Augmented Inservice Inspections (AUG), Snubber Visual Examination and Functional Testing (SNUB), and System Pressure Testing (SPT) requirements imposed on or committed to by TMI. At TMI, the Inservice Testing (IST) Program is maintained and implemented separately from the ISI Program. The IST Program Plan contains all of the applicable IST program requirements. (CM-1), (CM-2), (CM-3), (CM-4), (CM-5), (CM-6), (CM-7), (CM-8)

The Fourth ISI and Second CISI Intervals are effective from April 20, 2011 through April 19, 2022 for TMI. (See Tables 1.1-1, 1.1-2, and 1.1-3 for detailed notes regarding current extensions being taken.) With the update to the ISI Program for the Fourth ISI Interval for Class 1, 2, and 3 components, including their supports, the CISI Program is also being updated to its Second CISI Interval for Class MC and CC components. (Note that the Second CISI Interval will be designated as the Fourth CISI Interval in the TMI ISI Database only for ease of reporting.) This update will enable all of the ISI and CISI Program components / elements to be based on the same effective Edition and Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Division 1, as well as share a common interval start and end date. The common ASME Code of Record for the Fourth ISI Interval and the Second CISI Interval is the 2004 Edition, No Addenda. This ISI Program Plan is controlled and revised in accordance with the requirements of procedure ER-AA-330, "Conduct of Inservice Inspection Activities," which implements the ASME Section XI ISI Program. (Note that the IST Program is in the Fourth IST Interval that is applicable from September 23, 2004 through September 22, 2014. See the IST Program Plan for further details.)

Paragraph IWA-2430(d)(1) of ASME Section XI allows an inspection interval to be extended or decreased by as much as one year, and Paragraph IWA-2430(e) allows an inspection interval to be extended when a unit is out of service continuously for six months or more. The extension may be taken for a period of time not to exceed the duration of the outage. See Tables 1.1-1, 1.1-2, and 1.1-3 for intervals, periods, and extensions that apply to the TMI Fourth ISI Interval and Second CISI Interval.

The Fourth ISI Interval and Second CISI Interval are divided into two or three inspection periods as determined by calendar years within the intervals. Tables 1.1-1, 1.1-2, and 1.1-3 identify the period start and end dates for the Fourth ISI

Interval and the Second CISI Interval as defined by Inspection Program B. In accordance with Paragraph IWA-2430(d)(3), the inspection periods specified in these Tables may be decreased or extended by as much as 1 year to enable inspections to coincide with the TMI refueling outages.

TABLE 1.1-1

FOURTH ISI INTERVAL/PERIOD/OUTAGE MATRIX
(FOR ISI CLASS 1, 2, AND 3 COMPONENT EXAMINATIONS)

Interval	Period	Outages	
Start Date to End Date	**Start Date to End Date**	**Projected Outage Start Date or Outage Duration**	**Outage Number**
4th 04/20/11 to 04/19/22[1]	1st 04/20/11 to 04/19/14	Scheduled 10/11	T1R19
		Scheduled 10/13	T1R20
	2nd 04/20/14 to 04/19/18	Scheduled 10/15	T1R21
		Scheduled 10/17	T1R22
	3rd 04/20/18 to 04/19/22	Scheduled 10/19	T1R23
		Scheduled 10/21	T1R24

Note 1: The end of the TMI Fourth ISI Interval was extended for one year from April 19, 2021 to April 19, 2022 per Paragraph IWA-2430(d)(1) of ASME Section XI. The interval date was extended to allow Refueling outage T1R24 to fall within the Fourth ISI Interval. The start date of the Fifth ISI Interval is not revised, thus creating an overlap of the two intervals. Examinations can be performed during this time for either interval; however, no single exam can be credited to both intervals.

TABLE 1.1-2

SECOND CISI INTERVAL/PERIOD/OUTAGE MATRIX
(FOR CISI CLASS MC COMPONENT EXAMINATIONS)

Interval	Period	Outages	
Start Date to End Date	**Start Date to End Date**	**Projected Outage Start Date or Outage Duration**	**Outage Number**
2nd 04/20/11 to 04/19/22[1]	1st 04/20/11 to 04/19/14	Scheduled 10/11	T1R19
		Scheduled 10/13	T1R20
	2nd 04/20/14 to 04/19/18	Scheduled 10/15	T1R21
		Scheduled 10/17	T1R22
	3rd 04/20/18 to 04/19/22	Scheduled 10/19	T1R23
		Scheduled 10/21	T1R24

Note 1: The end of the TMI Second CISI Interval was extended for one year from April 19, 2021 to April 19, 2022 per Paragraph IWA-2430(d)(1) of ASME Section XI. The interval date was extended to allow Refueling outage T1R24 to fall within the Second CISI Interval. The start date of the Third CISI Interval is not revised, thus creating an overlap of the two intervals. Examinations can be performed during this time for either interval; however, no single exam can be credited to both intervals.

TABLE 1.1-3

SECOND CISI INTERVAL/PERIOD/OUTAGE MATRIX
(FOR CISI CLASS CC COMPONENT EXAMINATIONS)

Interval	5-Year Period	Outages	
Start Date to End Date	**Rolling Exam # - Date (2 Year Window)**	**Projected Outage Start Date or Outage Duration**	**Outage Number**
2nd 04/20/11 to 04/19/22[2]	No Section XI Exams	Scheduled 10/11	T1R19
	10th Period (40th Year) - 03/09/14 (03/09/13 to 03/08/15)[1]	Scheduled 10/13	T1R20
	No Section XI Exams	Scheduled 10/15	T1R21
	No Section XI Exams	Scheduled 10/17	T1R22
	11th Period (45th Year) - 03/09/19 (03/09/18 to 03/08/20)[1]	Scheduled 10/19	T1R23
	No Section XI Exams	Scheduled 10/21	T1R24

Note 1: The CISI Interval for Class CC components parallels the CISI Interval for Class MC components, as well as the ISI Interval, for the purpose of establishing the ten-year start and end dates and the applicable Code of Record. The actual inspection schedule however is based on a rolling 5 year frequency (+/- 1 year) from the date of completion of the Containment Structural Integrity Test (SIT). The initial Containment SIT was performed on March 9, 1974 per Gilbert Associates Inc. Report 1838. The rolling 5 year inspection schedule for containment concrete and post-tensioning systems is in accordance with the Inservice Inspection Schedule specified in Subarticle IWL-2400.

Note 2: The end of the TMI Second CISI Interval was extended for one year from April 19, 2021 to April 19, 2022 per Paragraph IWA-2430(d)(1) of ASME Section XI. The interval date was extended to allow Refueling outage T1R24 to fall within the Second CISI Interval. The start date of the Third CISI Interval is not revised, thus creating an overlap of the two intervals. Examinations can be performed during this time for either interval; however, no single exam can be credited to both intervals.

1.2 Background

Metropolitan Edison Company, a subsidiary of General Public Utilities (GPU) obtained a Construction Permit to build TMI, on April 19, 1967. The Docket Number assigned to TMI is 50-289. After satisfactory plant construction and preoperational testing was completed, GPU was granted a full power operating license April 19, 1974, DPR-50, for TMI, and subsequently commenced commercial operation on September 2, 1974. (Note that TMI was purchased by AmerGen Energy Company, LLC on December 20, 1999. AmerGen Energy Company, LLC was then absorbed by Exelon Nuclear in 2009 and the current licensee for TMI is Exelon Nuclear.)

The TMI non-nuclear power piping was designed, fabricated, tested, and inspected in accordance with USAS B31.1.0, 1967 Edition. The TMI nuclear power piping was designed in accordance with USAS B31.1.0, 1967 Edition, but it was fabricated, tested, and inspected to USAS B31.7, February 1968 Draft, including the June 1968 Errata as documented in UFSAR Section 1.3.2.31. B&W supplied Reactor Coolant System (RCS) piping was designed, fabricated, tested, and inspected in accordance with USAS B31.7, February 1968 Draft, including the June 1968 Errata as documented in Engineering Document ES-001T. The reactor pressure vessel, steam generators, and pressurizer were designed to meet the requirements of ASME Section III. The Nuclear Steam Supply System (NSSS) was furnished by Babcock & Wilcox (B&W) and has two Reactor Coolant loops. The Architect Engineer for TMI during construction was Gilbert Commonwealth.

1.3 First Interval ISI Program

TMI started commercial operation on September 2, 1974, which marked the beginning of the First Interval ISI Program. The First Period of the First ISI Interval was based on ASME Section XI, 1970 Edition with Addenda through Winter 1970. Selection of components subject to examination for the First Period was based on an owner established "Focused Approach" program. Federal regulations later required an update of the ASME Section XI ISI Program to comply with ASME Section XI, 1974 Edition with Addenda through the Summer 1975 for the remaining two periods of the First ISI Interval beginning on January 1, 1978.

ASME Section XI stated that plants which are out of service continuously for six months or more may extend the inspection interval for an equivalent period. TMI was out of service between February 17, 1979 and October 3, 1985 due to a scheduled refueling outage and an Order of the United States Nuclear Regulatory Commission (USNRC) resulting from the accident at TMI Unit 2. As such, GPU chose to extend the First ISI Interval by an equivalent time period. Therefore, the TMI First ISI Interval started September 2, 1974 and ended on April 19, 1991.

As allowed by ASME Section XI, 1974 Edition with Addenda through Summer 1975, Paragraph IWA-2400(a), certain First ISI Interval examinations were deferred to the Cycle 9 Refueling (9R) Outage, which was completed in October, 1991.

1.4 Second Interval ISI Program

Pursuant to 10 CFR 50.55a(g), TMI was required to update the ISI Program at the end of the First ISI Interval. The ISI Program was required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in Code of Federal Regulations, Title 10, Part 50, Section 55a (10 CFR 50.55a) twelve (12) months prior to the start of the Second ISI Interval per 10 CFR 50.55a(g)(4)(ii).

The TMI Second ISI Interval started on April 20, 1991 and ended on December 6, 2001. The Second Interval ISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a and the 1986 Edition, No Addenda of ASME Section XI. The Second Interval ISI Program Plan addressed Subsections IWA, IWB, IWC, IWD, and IWF of ASME Section XI, and utilized Inspection Program B as defined therein.

The Pressure Test Program was not applicable to the ISI Class 2 and Class 3 systems until 10 CFR 50 required inclusion in the program in January 1978. Therefore, pressure tests of ISI Class 2 and Class 3 systems did not begin until the Second Period of the First ISI Interval. This resulted in only two-thirds of all hydrostatic tests having been completed by the end of the First Ten-Year (120 month) ISI Interval. Components not hydrostatically tested during the First 120 month ISI Interval (the remaining one-third of the total number of such tests) were tested during the First Period of the Second ISI Interval to the requirements of the 1986 Edition of the ASME Section XI. One hundred percent (100%) of the total number of hydrostatic tests were completed by the end of the First Period of the Second ISI Interval.

As allowed by ASME Section XI, 1986 Edition, No Addenda, Paragraph IWA-2430(d) certain Second ISI Interval examinations were deferred to the Cycle 14 Refueling (1R14) Outage which was completed in December, 2001.

1.5 Third Interval ISI Program

Pursuant to 10 CFR 50.55a(g), TMI was required to update the ISI Program at the end of the Second ISI Interval. The ISI Program was required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the Third ISI Interval per 10 CFR 50.55a(g)(4)(ii).

The TMI Third Interval ISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a and the 1995 Edition, 1996 Addenda of ASME Section XI. The TMI Third Interval ISI Program Plan addressed Subsections IWA, IWB, IWC, IWD, IWF, Mandatory Appendices, approved ASME Code Cases, approved alternatives through relief requests and SER's, and utilized Inspection Program B as defined therein.

The Third ISI Interval was effective from April 20, 2001 through April 19, 2011. However, at the end of the Third ISI Interval and in preparation for the Fourth ISI Interval, a one year extension was taken per Paragraph IWA-2430(d)(1) of ASME Section XI, which allowed an inspection interval to be extended or decreased by as much as one year. As permitted by this allowance, the TMI Third ISI Interval was extended by one year from April 20, 2011 through April 19, 2012 to allow Refueling outage T1R19 to fall within the Third ISI Interval. The start date of the Fourth ISI Interval was not revised, thus creating an overlap of the two intervals. Examinations could have been performed during this time for either interval, however, no single examination could be credited to both intervals.

Certain Third ISI Interval examinations were deferred to Cycle 19 Refueling outage T1R19 in the Fall of 2011 as allowed by Paragraph IWA-2430(d).

1.6 Fourth Interval ISI Program

Pursuant to 10 CFR 50.55a(g), licensees are required to update their ISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program is required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per 10 CFR 50.55a(g)(4)(ii). Based on this date, the latest Edition and Addenda of ASME Section XI referenced in 10 CFR 50.55a(b)(2) twelve months prior to the start of the Fourth ISI Interval was the 2004 Edition, No Addenda.

The TMI Fourth Interval ISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a including all published changes through October 10, 2008, and the 2004 Edition, No Addenda of ASME Section XI, subject to the limitations and modifications contained within Paragraph (b) of the regulation. These limitations and modifications are detailed in Table 1.9-1 of this section. This Fourth Interval ISI Program Plan addresses Subsections IWA, IWB, IWC, IWD, IWF, Mandatory Appendices, approved ASME Code Cases, approved alternatives through relief requests and SER's, and utilizes Inspection Program B as defined therein.

The TMI Fourth ISI Interval is effective from April 20, 2011 through April 19, 2021. However, the end of the TMI Fourth ISI Interval is extended for one year from April 20, 2021 to April 19, 2022 per Paragraph IWA-2430(d)(1) of ASME Section XI. The interval date will be extended to allow Refueling outage T1R24 to fall within the Fourth ISI Interval. The start date of the Fifth ISI Interval is not revised, thus creating an overlap of the two intervals. Examinations can be performed during this time for either interval; however, no single examination can be credited to both intervals.

TMI adopted the EPRI Topical Report TR-112657, Rev. B-A methodology, which was supplemented by ASME Code Case N-578-1, for implementing risk-informed inservice inspections during the Third ISI Interval. The RISI Program will continue for the Fourth ISI Interval. Implementation of the RISI Program is in accordance with Relief Request I4R-02.

1.7 First Interval CISI Program

CISI examinations were originally invoked by amended regulations contained within a Final Rule issued by the USNRC. The amended regulation incorporated the requirements of the 1992 Edition through the 1992 Addenda of ASME Section XI, Subsections IWE and IWL, subject to specific modifications that were included in Paragraphs 10 CFR 50.55a(b)(2)(ix) and 10 CFR 50.55a(b)(2)(x).

The final rulemaking was published in the Federal Register on August 8, 1996 and specified an effective date of September 9, 1996. Implementation of the Subsection IWE and IWL Program from a scheduling standpoint was driven by the five year expedited implementation period per 10 CFR 50.55a(g)(6)(ii)(B), which specified that the examinations required to be completed by the end of the First Period of the First CISI Interval (per Table IWE-2412-1) be completed by the effective date (by September 9, 2001).

The TMI First Interval CISI Program was aligned with the Third Interval ISI Program and was effective from April 20, 2001 through April 19, 2011. However, at the end of the First CISI Interval and in preparation for the Second CISI Interval, a one year extension was taken per Paragraph IWA-2430(d)(1) of ASME Section XI, which allowed an inspection interval to be extended or decreased by as much as one year. As permitted by this allowance, the TMI Third ISI Interval was extended by one year from April 20, 2011 through April 19, 2012 to allow Refueling outage T1R19 to fall within the First ISI Interval. The start date of the Second ISI Interval was not revised, thus creating an overlap of the two intervals. Examinations could have been performed during this time for either interval, however, no single examination could be credited to both intervals.

1.8 Second Interval CISI Program

Pursuant to 10 CFR 50.55a(g), licensees are required to update their CISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The CISI Program is required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per

10 CFR 50.55a(g)(4)(ii). Based on this date, the latest Edition and Addenda of ASME Section XI referenced in 10 CFR 50.55a(b)(2) twelve months prior to the start of the Second CISI Interval was the 2004 Edition, No Addenda.

The TMI Second Interval CISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a including all published changes through October 10, 2008, and the 2004 Edition, No Addenda of ASME Section XI, subject to the limitations and modifications contained within Paragraph (b) of the regulation. These limitations and modifications are detailed in Table 1.9-1 of this section. The Second Interval CISI Program Plan addresses Subsections IWE, IWL, approved ASME IWE/IWL Code Cases, approved alternatives through related relief requests and SER's, and utilizes Inspection Program B as defined therein.

The TMI Second CISI Interval is effective from April 20, 2011 through April 19, 2021. However, the end of the TMI Second CISI Interval is extended for one year from April 20, 2021 to April 19, 2022 per Paragraph IWA-2430(d)(1) of ASME Section XI. The interval date will be extended to allow Refueling outage T1R24 to fall within the Second CISI Interval. The start date of the Third CISI Interval is not revised, thus creating an overlap of the two intervals. Examinations can be performed during this time for either interval; however, no single examination can be credited to both intervals.

1.9 Code of Federal Regulations 10 CFR 50.55a Requirements

There are certain Paragraphs in 10 CFR 50.55a that list the limitations, modifications, and/or clarifications to the implementation requirements of ASME Section XI. These Paragraphs in 10 CFR 50.55a that are applicable to TMI are detailed in Table 1.9-1.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(viii)(E)	**(CISI)** *Examination of concrete containments:* For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report required by IWA-6000: (*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation; (*2*) An evaluation of each area, and the result of the evaluation, and; (*3*) A description of necessary corrective actions.
10 CFR 50.55a(b)(2)(viii)(F)	**(CISI)** *Examination of concrete containments:* Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA-2300. The "owner-defined" personnel qualification provisions in IWL-2310(d) are not approved for use.
10 CFR 50.55a(b)(2)(viii)(G)	**(CISI)** *Examination of concrete containments:* Corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA-1400.
10 CFR 50.55a(b)(2)(ix)(A)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA-6000: (*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation; (*2*) An evaluation of each area, and the result of the evaluation, and; (*3*) A description of necessary corrective actions.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(ix)(B)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.
10 CFR 50.55a(b)(2)(ix)(F)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* VT-1 and VT-3 visual examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 visual examination method shall be qualified in accordance with IWA-2300. The "owner-defined" personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 visual examinations are not approved for use.
10 CFR 50.55a(b)(2)(ix)(G)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* The VT-3 visual examination method must be used to conduct the examinations in Item E1.12 of Table IWE-2500-1 and the VT-1 visual examination method must be used to conduct the examination in Item E4.11 of Table IWE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 visual examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 visual examinations.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(ix)(H)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 visual examination method. Flaws or degradation identified during the performance of a VT-3 visual examination must be examined in accordance with the VT-1 visual examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 visual examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 visual examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.
10 CFR 50.55a(b)(2)(ix)(I)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* The ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.
10 CFR 50.55a(b)(2)(xii)	**(ISI)** *Underwater Welding:* The provisions in IWA-4660, "Underwater Welding," of Section XI, 1997 Addenda through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section, are not approved for use on irradiated material.
10 CFR 50.55a(b)(2)(xviii)(A)	**(ISI)** *Certification of NDE personnel:* Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA-2314, and IWA-2314(a) and IWA-2314(b) of the 1999 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(xviii)(B)	**(ISI)** *Certification of NDE personnel:* Paragraph IWA-2316 of the 1998 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA-5211(a) and (b), 1998 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.
10 CFR 50.55a(b)(2)(xviii)(C)	**(ISI)** *Certification of NDE personnel:* When qualifying visual examination personnel for VT-3 visual examinations under paragraph IWA-2317 of the 1998 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.
10 CFR 50.55a(b)(2)(xix)	**(ISI)** *Substitution of alternative methods:* The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-2240, 1998 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use. The provisions in IWA-4520(c), 1997 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, allowing the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code are not approved for use.
10 CFR 50.55a(b)(2)(xx)(B)	**(ISI)** *System leakage tests:* The NDE provision in IWA-4540(a)(2) of the 2002 Addenda of Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(xxi)(A)	**(ISI)** *Table IWB-2500-1 examination requirements:* The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.120 and B3.140 (Inspection Program B) in the 1998 Edition must be applied when using the 1999 Addenda through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section. A visual examination with magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, 1997 Addenda through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section, with a limiting assumption on the flaw aspect ratio (i.e., a/l=0.5), may be performed instead of an ultrasonic examination.
10 CFR 50.55a(b)(2)(xxi)(B)	**(ISI)** The provisions of Table IWB-2500-1, Examination Category B-G-2, Item B7.80, that are in the 1995 Edition are applicable only to reused bolting when using the 1997 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.
10 CFR 50.55a(b)(2)(xxii)	**(ISI)** *Surface Examination:* The use of the provision in IWA-2220, "Surface Examination," of Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, that allow use of an ultrasonic examination method is prohibited.
10 CFR 50.55a(b)(2)(xxiii)	**(ISI)** *Evaluation of Thermally Cut Surfaces:* The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA-4461.4.2 of Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section are prohibited.
10 CFR 50.55a(b)(2)(xxiv)	**(PDI)** *Incorporation of the Performance Demonstration Initiative and Addition of Ultrasonic Examination Criteria:* The use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 of Section XI of the ASME BPV Code, 2002 Addenda through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section, is prohibited.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(xxv)	**(ISI)** *Mitigation of Defects by Modification:* The use of the provisions in IWA-4340, "Mitigation of Defects by Modification," Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section are prohibited.
10 CFR 50.55a(b)(2)(xxvi)	**(SPT)** *Pressure Testing Class 1, 2, and 3 Mechanical Joints:* The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section.
10 CFR 50.55a(b)(2)(xxvii)	**(ISI)** *Removal of Insulation:* When performing visual examinations in accordance with IWA-5242 of Section XI, 2003 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of the section, insulation must be removed from 17-4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A-286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

TABLE 1.9-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a LIMITATIONS

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(5)	(ISI) *Inservice Inspection Code Cases:* Licensees may apply the ASME Boiler and Pressure Vessel Code Cases listed in Regulatory Guide 1.147 without prior USNRC approval subject to the following: (i) When a licensee initially applies a listed Code Case, the licensee shall apply the most recent version of that Code Case incorporated by reference in this paragraph. (ii) If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case as authorized or may apply the later version of the Code Case, including any USNRC-specified conditions placed on its use. (iii) Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.147. Any Code Case listed as annulled in any Revision of Regulatory Guide 1.147 which a licensee has applied prior to it being listed as annulled, may continue to be applied by that licensee to the end of the 120-month interval in which the Code Case was implemented.

1.10 Code Cases

Per 10 CFR 50.55a(b)(5), ASME Code Cases that have been determined to be suitable for use in ISI Program Plans by the USNRC are listed in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1". The approved Code Cases in Regulatory Guide 1.147, which are being utilized by TMI, are included in Section 2.1.1. The most recent version of a given Code Case incorporated in the revision of Regulatory Guide 1.147 referenced in 10 CFR 50.55a(b)(5)(i) at the time it is applied within the ISI Program shall be used. The latest version of Regulatory Guide 1.147 incorporated into this document is Revision 16. As this guide is revised, newly approved Code Cases should be assessed for plan implementation at TMI per Paragraph IWA-2441(d) and proposed for use in revisions to the ISI Program Plan.

The use of other Code Cases (than those listed in Regulatory Guide 1.147) may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to 10 CFR 50.55a(a)(3). Code Cases not approved for use in Regulatory Guide 1.147, which are being utilized by TMI through associated relief requests, will be included in Section 8.0.

1.11 Relief Requests

In accordance with 10 CFR 50.55a, when a licensee either proposes alternatives to ASME Section XI requirements which provide an acceptable level of quality and safety, determines compliance with ASME Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or determines that specific ASME Section XI requirements for inservice inspection are impractical, the licensee shall notify the USNRC and submit information to support the determination.

The submittal of this information will be referred to in this document as a "Relief Request". Relief Requests for the Fourth ISI Interval and the Second CISI Interval are included in Section 8.0 of this document. The text of the Relief Requests contained in Section 8.0 will demonstrate one of the following: the proposed alternatives provide an acceptable level of quality and safety per 10 CFR 50.55a(a)(3)(i), compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety per 10 CFR 50.55a(a)(3)(ii), or the code requirements are considered impractical per 10 CFR 50.55a(g)(5)(iii).

Per 10 CFR 50.55a Paragraphs (a)(3) and (g)(6)(i), the Director of the Office of Nuclear Reactor Regulation will evaluate relief requests and "may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility".

2.0 BASIS FOR INSERVICE INSPECTION PROGRAM

2.1 ASME Section XI Examination Requirements

As required by the 10 CFR 50.55a, this Program was developed in accordance with the requirements detailed in the 2004 Edition, No Addenda, of the ASME Section XI, Subsections IWA, IWB, IWC, IWD, IWE, IWF, IWL, Mandatory Appendices, Inspection Program B of Paragraph IWA-2432, approved ASME Code Cases, and approved alternatives through relief requests and Safety Evaluation Reports (SER's).

The Performance Demonstration Initiative (PDI) is an organization comprised of all US nuclear utilities that was formed to provide an efficient implementation of Appendix VIII performance demonstration requirements. The Electric Power Research Institute (EPRI) NDE Center was selected as the administrator of this program. The PDI program is administered according to the "PDI Program Description". The ISI Program implements Appendix VIII "Performance Demonstration for Ultrasonic Examination Systems," ASME Section XI 2001 Edition, No Addenda as required by 10 CFR 50.55a(b)(2)(xxiv) and with modifications as identified in 10 CFR 50.55a(b)(2)(xiv), (xv), and (xvi). Appendix VIII requires qualification of the procedures, personnel, and equipment used to detect and size flaws in piping, bolting, and the reactor pressure vessel. Each organization (e.g., owner or vendor) will be required to have a written program to ensure compliance with the requirements. These requirements were initially implemented through the Performance Demonstration Initiative (PDI) Program according to the schedule defined in 10 CFR 50.55a(g)(6)(ii)(C). TMI maintains has responsibility to ensure that Appendix VIII requirements are properly implemented.

For the Fourth ISI Interval, the TMI inspection program for ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 will be governed by risk-informed regulations. The RISI Program methodology is described in the EPRI Topical Report TR-112657, Rev. B-A. To supplement the EPRI Topical Report, ASME Code Case N-578-1 (as applicable per Relief Request I4R-02) is also being used for the classification of piping structural elements under the RISI Program. The RISI Program scope has been implemented as an alternative to the 2004 Edition, No Addenda of the ASME Section XI examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-F-1 and C-F-2 welds in accordance with 10 CFR 50.55a(a)(3)(i). The basis for the resulting Risk Categorizations of the nonexempt ISI Class 1 and 2 piping systems at TMI is defined and maintained in the Final Report "Risk Informed Inservice Inspection Evaluation" as referenced in Section 9.0 of this document.

The CISI Program per Subsection IWE and IWL is included in Section 6.0, "Containment ISI Plan". The CISI relief requests are included in Section 8.0 of this document.

2.1.1 ASME Section XI Code Cases

As referenced by 10 CFR 50.55a(b)(5) and allowed by USNRC Regulatory Guide 1.147, Revision 16, the following Code Cases are being incorporated into the TMI ISI Program. Several of these Code Cases are included as contingencies, to ensure that they are available for future repair/replacement activities.

N-432-1 Repair Welding Using Automatic or Machine Gas Tungsten-Arc Welding (GTAW) Temper Bead Technique. Regulatory Guide 1.147, Revision 16.

N-460 Alternative Examination Coverage for Class 1 and Class 2 Welds. Regulatory Guide 1.147, Revision 16.

N-504-4 Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping.

Code Case N-504-4 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- The provisions of Section XI, Nonmandatory Appendix Q, "Weld Overlay Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Weldments," must also be met. In addition, the following conditions shall be met: (a) the total laminar flaw area shall not exceed 10% of the weld surface area, and no linear dimension of the laminar flaw area shall exceed the greater of 3 inches or 10% of the pipe circumference; (b) the finished overlay surface shall be 250 micro-in (6.3 micrometers) root mean square or smoother; (c) the surface flatness shall be adequate for ultrasonic examination; and (d) radiography shall not be used to detect planar flaws under or masked by laminar flaws.

(*Note that Relief Request RR-10-02 was previously submitted utilizing Code Case N-504-3, Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, as approved conditionally in Regulatory Guide 1.147, Revision 15.)

N-513-3 Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping.

Code Case N-513-3 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- The repair or replacement activity temporarily deferred under the provisions of this Code Case shall be performed during the next scheduled outage.

N-516-3 Underwater Welding

Code Case N-516-3 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- Licensee must obtain USNRC approval in accordance with 10 CFR 50.55a(a)(3) regarding the technique to be used in the weld repair or replacement of irradiated material underwater.

N-517-1 Quality Assurance Program Requirements for Owner. Regulatory Guide 1.147, Revision 16.

N-526 Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels. Regulatory Guide 1.147, Revision 16.

N-528-1 Purchase, Exchange, or Transfer of Material Between Nuclear Plant Sites

Code Case N-528-1 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- The requirements of 10 CFR Part 21, "Reporting of Defects and Noncompliance" (Ref 5), are to be applied to the nuclear plant site supplying the material as well as to the nuclear plant sire receiving the material that has been purchased, exchanged, or transferred between sites.

N-532-4 Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000. Regulatory Guide 1.147, Revision 16.

N-552 Alternative Methods – Qualification for Nozzle Inside Radius Section from the Outside Surface

Code Case N-552 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- To achieve consistency with the 10 CFR 50.55a rule change published September 22, 1999 (64 FR 51370), incorporative Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to ASME Section XI, add the following to the specimen requirements:
- "At least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation must be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less that 4 inches may be notches."
- Add to detection criteria, "The number of false calls must not exceed three."

N-557-1 In-Place Dry Annealing of a PWR Nuclear Reactor Vessel.

Code Case N-557-1 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- The secondary stress allowable of 3Sm, shown in Figure 1 of the Code Case, must be applied to the entire primary plus secondary stress range during the anneal.

N-566-2 Corrective Action for Leakage Identified at Bolted Connections. Regulatory Guide 1.147, Revision 16.

N-569-1 Alternative Rules for Repair by Electrochemical Deposition of Class 1 and 2 Steam Generator Tubing

Code Case N-569-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- Note: Steam Generator tube repair methods require prior USNRC approval through the Technical Specifications. This Code Case does not address certain aspects of this repair, e.g., the qualification of the inspection and plugging criteria necessary for staff approval of the repair method. In addition, if the user plans to "reconcile," as described in Footnote 2, the reconciliation is to be performed in accordance with Subarticle IWA-4200 in the 1995 Edition through the 1996 Addenda of ASME Section XI.

N-586-1 Alternative Additional Examination Requirements for Class 1, 2, and 3 Piping, Components, and Supports. Regulatory Guide 1.147, Revision 16.

NOTE - RISI Program Relief Request I4R-02 requires that scope expansion for RISI elements will be determined using Paragraph -2430 of Code Case N-578-1.

N-593 Alternative Examination Requirements for Steam Generator Nozzle to Vessel Welds.

Code Case N-593 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- Essentially 100 percent (not less than 90 percent) of the examination volume A-B-C-D-E-F-G-H must be inspected.

N-597-2 Requirements for Analytical Evaluation of Pipe Wall Thinning

Code Case N-597-2 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- (1) Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, April 1999, "Recommendations for an Effective Flow Accelerated Corrosion Program," for developing the inspection requirements, the method of predicting the rate of wall thickness loss, and the value of the predicted remaining wall thickness. As used in NSAC-202L-R2, the term "should" is to be applied as "shall" (i.e., a requirement).

- (2) Components affected by flow-accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction Code of Record and Owner's requirements or a later USNRC approved edition of Section III, "Rules for Construction of Nuclear Plant Components," of the ASME Code prior to the value of tp reaching the allowable minimum wall thickness, tmin, as specified in -3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to USNRC review and approval per 10 CFR 50.55a(a)(3).

- (3) For Class 1 piping not meeting the criteria of -3221, the use of evaluation methods and criteria is subject to USNRC review and approval per 10 CFR 50.55a(a)(3).

- (4) For those components that do not require immediate repair or replacement, the rate of wall thickness loss is to be used to determine a suitable inspection frequency so that repair or replacement occurs prior to reaching allowable minimum wall thickness, tmin.

- (5) For corrosion phenomenon other than flow accelerated corrosion, use of the Code Case is subject to USNRC review and approval per 10 CFR 50.55a(a)(3). Inspection plans and wall thinning rates may be difficult to justify for certain degradation mechanisms such as MIC and pitting.

N-600	Transfer of Welder, Welding Operator, Brazer, and Brazing Operator Qualifications Between Owners. Regulatory Guide 1.147, Revision 16.
N-613-1	Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Item No's. B3.10 and B3.90, Reactor Nozzle-To-Vessel Welds, Figs. IWB-2500-7(a), (b), and (c). Regulatory Guide 1.147, Revision 16.
N-624	Successive Inspections. Regulatory Guide 1.147, Revision 16.
N-629	Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials. Regulatory Guide 1.147, Revision 16.

N-638-4 Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique.

Code Case N-638-4 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- (1) Demonstration for ultrasonic examination of the repaired volume is required using representative samples which contain construction type flaws.
- (2) The provisions of 3(e)(2) or 3(e)(3) may only be used when it is impractical to use the interpass temperature measurement methods described in 3(e)(1), such as in situations where the weldment area is inaccessible (e.g., internal bore welding) or when there are extenuating radiological conditions.

(*Note that Relief Request RR-10-02 was previously submitted utilizing Code Case N-638-1, Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique, as approved conditionally in Regulatory Guide 1.147, Revision 15.)

N-639 Alternative Calibration Block Material

Code Case N-639 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- Chemical ranges of the calibration block may vary from the materials specification if (1) it is within the chemical range of the component specification to be inspected, and (2) the phase and grain shape are maintained in the same ranges produced by the thermal process required by the material specification.

N-641 Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements. Regulatory Guide 1.147, Revision 16.

N-643-2 Fatigue Crack Growth Rate Curves for Ferritic Steels in PWR Water Environment. Regulatory Guide 1.147, Revision 16.

N-648-1 Alternative Requirements for Inner Radius Examination of Class 1 Reactor Vessel Nozzles.

Code Case N-648-1 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- In place of a UT examination, licensees may perform a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria of Table IWB-3512-1 with limiting assumption on the flaw aspect ratio. The provisions of Table IWB-2500-1, Examination Category B-D, continue to apply except that, in place of examination volumes, the surfaces to be examined are the external surfaces shown in the figures applicable to this table (the external surface is from point M to point N in the figure).

N-651 Ferritic and Dissimilar Metal Welding Using SMAW Temper Bead Technique Without Removing the Weld Bead Crown for the First Layer. Regulatory Guide 1.147, Revision 16.

N-661-1 Alternative Requirements for Wall Thickness Restoration of Classes 2 and 3 Carbon Steel Piping for Raw Water Service.

Code Case N-661-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

- (1) If the cause of the degradation has not been determined, the repair is acceptable until the next refueling outage.
- (2) When through-wall repairs are made by welding on surfaces that are wet or exposed to water, the weld overlay repair is only acceptable until the next refueling outage.

N-665 Alternative Requirements for Beam Angle Measurements Using Refracted Longitudinal Wave Search Units. Regulatory Guide 1.147, Revision 16.

N-666 Weld Overlay of Class 1, 2, and 3 Socket Welded Connections. Regulatory Guide 1.147, Revision 16.

N-686-1 Alternative Requirements for Visual Examinations, VT-1, VT-2, and VT-3. Regulatory Guide 1.147, Revision 16.

N-705 Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks. Regulatory Guide 1.147, Revision 16.

N-706-1 Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchangers. Regulatory Guide 1.147, Revision 16.

N-722 Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials.

- Apply USNRC modifications to this Code Case in 10 CFR 50.55a(g)(6)(ii)(E). (This case is not approved through relief request, but rather is required by the regulation effective 10/10/2008.)

The conditions listed below were imposed by the USNRC within the regulation.

- Condition (1) requires that PWR licensees implement N-722 except for those welds that have been mitigated by weld overlay or stress improvements.
- Condition (2) requires that if leakage occurs from a component, licensees take additional actions to characterize the orientation of the crack that caused the leakage.
- Condition (3) requires that if the crack that leads to leakage is circumferentially oriented and potentially the result of primary water stress-corrosion cracking, licensees perform non-visual sample inspections of the population of the components.
- Condition (4) requires that the ultrasonic examinations of the butt welds as required by Condition (2) and (3) follow the appropriate supplement of Appendix VIII of the ASME Code, Section XI.

N-729-1 Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetrations Welds.

- Apply USNRC modifications to this Code Case in 10 CFR 50.55a(g)(6)(ii)(D). (This case is not approved through relief request, but rather is required by the regulation effective 10/10/2008.) (CM-8).

The conditions listed below were imposed by the USNRC within the regulation.

- Condition (1) requires that PWR licensees implement N-729-1 to replace USNRC Order EA-03-009.
- Condition (2) prohibits use of Note 9 in the Case.
- Condition (3) modifies the examination method contained in Note 6 of Table 1 in the Case.
- Condition (4) requires alternative personnel, procedures, and equipment qualifications to those provided in Paragraph -2500 of the Case.
- Condition (5) requires that flaws attributed to PWSCC be re-inspected each refueling outage.
- Condition (6) prohibits use of Appendix I in the Case without prior USNRC approval.

N-731 Alternative Class 1 System Leakage Test Pressure Requirements. Regulatory Guide 1.147, Revision 16.

N-733 Mitigation of Flaws in NPS 2 (DN 50) and Smaller Nozzles and Nozzle Partial Penetration Welds in Vessels and Piping by Use of a Mechanical Connection Modification. Regulatory Guide 1.147, Revision 16.

N-735 Successive Inspection of Class 1 and 2 Piping Welds. Regulatory Guide 1.147, Revision 16.

N-739 Alternative Qualification Requirements for Personnel Performing Class CC Concrete and Post-Tensioning System Visual Examinations. Regulatory Guide 1.147, Revision 16.

N-751 Pressure Testing of Containment Penetration Piping.

Code Case N-751 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

- When a 10 CFR 50, Appendix J, Type C test is performed as an alternative to the requirements of IWA-4540 during repair and replacement activities, nondestructive examination must be performed in accordance with IWA-4540(a)(2) of the 2002 Addenda of Section XI.

N-753 Vision Tests. Regulatory Guide 1.147, Revision 16.

Additional Code Cases invoked in the future shall be in accordance with those approved for use in the latest published revision of Regulatory Guide 1.147 or 10 CFR 50.55a at that time.

2.2 Augmented Examination Requirements

Augmented examination requirements are those examinations that are performed above and beyond the requirements of ASME Section XI. Below is a summary of those examinations performed by TMI that are not specifically addressed by ASME Section XI, or the examinations that will be performed in addition to the requirements of ASME Section XI on a routine basis during the Fourth ISI Interval and Second CISI Interval.

2.2.1 USNRC Mechanical Engineering Branch (MEB) Technical Position 3-1 (MEB 3-1), "High Energy Fluid Systems, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," dated November 24, 1975

The USNRC MEB Technical Position 3-1, "High Energy Fluid Systems, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," discusses protection against postulated piping failures in fluid systems outside containment, and includes requirements for licensees to perform 100% volumetric examination of circumferential and longitudinal pipe welds within the pipe break exclusion areas associated with high energy piping in containment penetration areas.

TMI has committed to requirements of the MEB 3-1 to examine welds as a result of the postulated break analysis performed for these locations. Welds FW-039, FW-038, FW-037, FW-036, FW-035, and FW-034 are required to be examined once per interval. (See GPU Nuclear Memo MSS-85-485, September 20, 1985 From M. O. Sanford to J. S. Jandovitz).

Implementation of the examination requirement is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.2 Reactor Coolant Pump Flywheels, Technical Specifications, Section 4.2.4, requires ultrasonic examination on the accessible areas of all reactor coolant pump motor flywheel assemblies. One flywheel assembly must be examined by the end of the First Period, two flywheel assemblies by the end of the Second Period and all four by the end of the Fourth Inspection Interval. (Reference USNRC Regulatory Guide 1.14.)

Implementation of the examination requirement is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.3 Main Steam System Welds, Technical Specifications, Section 4.15, requires the volumetric examination of welds at 3½-year intervals or the nearest refueling outage. This augmented examination is based on calculations in which postulated failure would produce pressures in excess of compartment wall and/or slab capabilities in the Intermediate Building.

Implementation of the examination requirement is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.4 Reactor Pressure Vessel Head and Internals Handling Fixture (tripod) and RV Closure Head Lifting Lugs - Exelon commitment to NUREG 0612 "Control of Heavy Loads" requires visual and/or magnetic particle examination of load bearing welds.

Implementation of the examination requirement is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.5 Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants

Various industry and USNRC documents provide requirements for controlling boric acid corrosion of carbon steel reactor coolant pressure boundary components in PWR plants. TMI is implementing a boric acid control program via procedure ER-AP-331 to control boric acid leaks at TMI.

2.2.6 Augmented examinations of Alloy 600 material installed at TMI is controlled via the Exelon Alloy 600 Management Plan, ER-AP-330-1001. The examinations are based upon MRP-126, MRP-139, MRP-139 interim guidance, USNRC Bulletin 2004-01, USNRC Bulletin 2003-02, and Code Cases N-722 and N-729-1 as modified by 10 CFR 50.55a(g)(6)(ii)(E) and 10 CFR 50.55a(g)(6)(ii)(D) respectively. Specific examination and mitigation plans/schedules are included in the ISI Selection Document.

Locations potentially susceptible to PWSCC, whether repaired, mitigated, or unmitigated, that fall within the TMI Alloy 600 Augmented Examination Program (MRP-139, ASME Code Case N-722, ASME Code Case N-729-1, ASME Section XI Appendix Q, etc.) under which they are tracked, categorized, and subject to augmented examination selection criteria and requirements are also evaluated under the RISI Program for appropriate degradation mechanism (DM) assignment. Where PWSCC is the only DM assigned, the locations are removed from the RISI scope and deferred to the Alloy 600 Augmented Examination Program for examinations. These PWSCC examination programs are based upon criteria that is generally more stringent than the RISI Program because they are generally based upon deterministic criteria that accounts for flaw growth rates and time to structural instability. For locations other than those which have full structural weld overlay applied where the degradation mechanism assessment identifies PWSCC and another DM under the RISI Program, the elements remain in the RISI Program and are subject to the normal RISI element selection process solely for the additional DMs assigned. Locations with full structural weld overlay will be removed from the RISI Program and treated under the Alloy 600 augmented examination program.

Implementation of the examination requirements is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.7 Augmented examinations of the reactor pressure vessel closure head shielded work platform will be performed prior to use each refueling outage. The examinations will be a visual examination for obvious signs of deformation, cracked paint at welds, bent or deformed members, etc. MT examination of the accessible swivel eye attachment welds (see drawing 6022549D, sheet 2) will be performed with the acceptance criteria to be no linear indications or service induced degradation. Reference PIMS AR#2131302E01 and AR#2076020E03, Areva Document 01-5031556-01, and TMI Procedure MA-TM-134-903. These inspections also satisfy requirements for NUREG 0612 heavy loads control.

Implementation of the examination requirement is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.8 Augmented UT examination of six Decay Heat system welds (not counting adjacent reducing elbow long seam welds) will be performed in accordance with MRP-192. These examinations are being performed to address high cycle thermal fatigue issues identified in MRP-192. Reference IR 618173 for further guidance.

Implementation of the examination requirements is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.9 Augmented UT examination and bare metal visual examination of three reactor coolant drain nozzles and three HPI nozzles will be performed in accordance with MRP-146. IR number 443140-25 is tracking completion of additional evaluations for MRP-146.

Implementation of the examination requirements is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.10 Future PWR Internals Program augmented examinations will be implemented in accordance with MRP-227. Note that the NRC has not yet endorsed MRP-227 so specific examination scope and schedule is not yet developed. TMI has committed to submitting a reactor vessel internals inspection plan to the NRC no later than 24 months prior to entering the period of extended operation. This plan will provide detailed information on components subject to examination and the schedule for examination.

Implementation of the examination requirements will be included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.11 Augmented UT examination of the reactor pressure vessel internals core barrel bolts are being performed in accordance with B&W Owners Group letter OG-06-1880.

Implementation of the examination requirements is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.3 System Classifications and ISI Boundary Drawings

The ISI Classification Basis Document details those systems that are ISI Class 1, 2, or 3 that fall within the ISI scope of examinations including the containment structures (metal and concrete). Below is a summary of the classification criteria used within the ISI Classification Basis Document.

Each safety related, fluid system containing water, steam, air, oil, etc. included in the TMI UFSAR was reviewed to determine which safety functions they perform during all modes of system and plant operation. Based on these safety functions, the systems and components were evaluated per classification documents. The systems were then designated as ISI Class 1, 2, 3, or non-classed accordingly.

When a particular group of components is identified as performing an ISI Class 1, 2, or 3 safety function, the components are further reviewed to assure the interfaces (boundary valves and boundary barriers) meet the criteria set by 10 CFR 50.2, 10 CFR 50.55a(c)(1), 10 CFR 50.55a(c)(2), and Regulatory Guide 1.26, Revision 3. SRP 3.2.2 and ANSI/ANS-58.14-1993 (TMI is not committed to or licensed in accordance with these documents) were also used for guidance in evaluating the classification boundaries where 10 CFR and the Regulatory Guide did not address a given situation.

Components within the reactor coolant pressure boundary, as defined by 10 CFR 50.2, are typically designated as ISI Class 1 while the other safety related components are evaluated for ISI Class 2 or 3 designation in accordance with the guidelines of Regulatory Guide 1.26. Per Regulatory Guide 1.26 Paragraphs A and B, the ISI Class 2 and 3 boundaries are limited to safety related systems and components. Where sufficient classification criteria is not provided within 10 CFR 50 or Regulatory Guide 1.26, other industry documents such as NUREG-0800 and ANSI/ANS standards are consulted "for guidance".

According to 10 CFR 50.55a, Paragraph (g)(4), the ISI requirements of ASME Section XI are assigned to these components, within the constraints of existing plant design. The TMI ISI Class 1, 2, and 3 components that are exempt from examination are those which meet the criteria of ASME Section XI, Paragraphs IWB-1220, IWC-1220, and IWD-1220. Supports which meet the criteria of Paragraph IWF-1230 of ASME Section XI are also exempt from examination. For Containment, CISI Class MC components which meet Paragraph IWE-1220 are exempt from examination, and CISI Class CC components which meet Paragraph IWL-1220 are exempt from examination.

The systems and components (piping, pumps, valves, vessels, etc.), which are subject to the examinations of Articles IWB-2000, IWC-2000, IWD-2000, and IWF-2000, and pressure tests of Articles IWB-5000, IWC-5000, and IWD-5000 are identified on the ISI Boundary Drawings.

Table 2.3-1 provides a listing of the ISI Boundary Drawings that depict the ISI Class 1, 2, and 3 components subject to the requirements of ASME Section XI during the Fourth ISI Interval at TMI. These color-coded ISI Boundary Drawings are derived from the TMI Piping and Instrumentation Diagrams (P&IDs)/Flow Diagrams. A summary of the color codings utilized for ISI Class 1, 2, and 3 components is included on each ISI Boundary Drawing.

TABLE 2.3-1
ISI BOUNDARY DRAWINGS

DRAWING NUMBER	TITLE
1D-ISI-SS-001	Symbols for ISI Boundary and Configuration Sketches
1D-ISI-FD-001	Main Steam System
1D-ISI-FD-002	River Water System
1D-ISI-FD-003	Decay Heat Closed Cycle Cooling System
1D-ISI-FD-004	Core Flooding System
1D-ISI-FD-005	Decay Heat Removal System
1D-ISI-FD-008	Condensate System
1D-ISI-FD-009	Emergency Feedwater and Feedwater Systems
1D-ISI-FD-010	Nuclear Services Closed Cycle Cooling Water System
1D-ISI-FD-011	Control Building Chilled Water System
1D-ISI-FD-012	Reactor Building Spray System
1D-ISI-FD-015	Penetration Pressurization and Hydrogen Recombiner Systems
1D-ISI-FD-016	Makeup and Purification System (Letdown Portion)
1D-ISI-FD-017	Makeup and Purification System (Makeup Portion)
1D-ISI-FD-018	Spent Fuel Cooling System
1D-ISI-FD-019	Reactor Coolant System
1D-ISI-FD-020	Chemical Sampling and OTSG Chemical Cleaning Systems
1D-ISI-FD-021	Chemical Addition and Liquid Waste Disposal Systems
1D-ISI-FD-022	Intermediate Cooling System
1D-ISI-FD-023	Hydrogen Purge System and Miscellaneous Penetrations

2.4 ISI Isometric and Component Sketches for Nonexempt ISI Class Components and Supports

ISI Isometric and Component Sketches were developed to identify the ISI Class 1, 2, and 3 components (welds, bolting, etc.) and support locations at TMI. These ISI component locations are identified on the ISI Isometric and Component Sketches listed in Table 2.4-1. The CISI Class MC and CC components are identified on the CISI Reference Drawings listed in Table 2.4-2.

The TMI ISI Program, including the ISI Database, ISI Classification Basis Document, and ISI Selection Document and schedule, addresses the nonexempt components, which require examination and testing.

A summary of TMI ASME Section XI nonexempt components and supports is included in Section 7.0.

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
MAIN STEAM SYSTEM (MS)	
1D-ISI-MS-001	MS Piping From Steam Generator RC-H-1A to Penetrations 112 & 113
1D-ISI-MS-002	MS Piping From Steam Generator RC-H-1B to Penetrations 114 & 419
1D-ISI-MS-003	MS Piping Intermediate Building From Penetration Nos. 419 & 114
1D-ISI-MS-004	MS Piping Intermediate Building From Penetration Nos. 112 & 113
1D-ISI-MS-005	MS Piping to Emergency Feedwater Pump Turbine
1D-ISI-MS-006	MS Piping to Emergency Feedwater Pump Turbine
1D-ISI-MS-007	MS System Details
1D-411-23 Series	MS Piping Analysis (Class 3 Supports)
1D-ISI-XX-001	Penetration Details
RIVER WATER SYSTEM (RW)	
1D-531-23 Series	RW Piping Analysis (Class 3 Supports)
1D-533-23 Series	RW Piping Analysis (Class 3 Supports)
1D-534-23 Series	RW Piping Analysis (Class 3 Supports)
1D-ISI-XX 001	Penetration Details – Note that system designator is "RR" on drawing.
DECAY HEAT CLOSED CYCLE COOLING SYSTEM (DC)	
1D-543-23 Series	DC Piping Analysis (Class 3 Supports)
CORE FLOODING SYSTEM (CF)	
1D-ISI-CF-001	Core Flood System From Tank CF-T1A
1D-ISI-CF-002	Core Flood System From Tank CF-T1B
1D-ISI-CF-003	Core Flood Tanks CF-T1A & CF-T1B
DECAY HEAT REMOVAL SYSTEM (DH)	
1D-ISI-DH-001	DH System – Reactor Building From RC Loop "B"
1D-ISI-DH-002	DH System – 10" Line From Core Flood 14" Pipe
1D-ISI-DH-003	DH System – Reactor Building From 14" Core Flood Pipe
1D-ISI-DH-004	DH System – Auxiliary Building From Penetrations 303, 306, & 310

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
1D-ISI-DH-005	Decay Heat Systems Details
1D-ISI-DH-006	Decay Heat Removal Auxiliary Building Elevation 261'-0"
1D-ISI-DH-007	Decay Heat Removal Auxiliary Building Elevation 261'-0"
1D-ISI-DH-008	Decay Heat Pump DH-P1B Suction & Cooler DH-C1B Outlet
1D-ISI-DH-009	Decay Heat Cooler DH-C1B Outlet
1D-ISI-DH-010	Decay Heat Pump Discharge & Cooler DH-C1B Inlet
1D-ISI-DH-011	Decay Heat Removal 14" DH Pump Suction From BWST
1D-ISI-DH-012	Decay Heat Removal 10" Line From BWST
1D-ISI-DH-013	Decay Heat Removal 24" Line From BWST
1D-ISI-DH-014	Decay Heat Removal 8" Line From BWST
1D-ISI-DH-015	DH 8" Line to SF Cooling – to RB Spray
1D-ISI-DH-016	Decay Heat Removal 3" Line to SF Cooling System
1D-ISI-DH-017	Decay Heat Removal 2" Pressurizer Spray Line
1D-ISI-DH-018	Decay Heat Removal 2" Pressurizer Spray Line
1D-ISI-DH-019	Decay Heat Removal Cooler DH-C1A Outlet
1D-ISI-DH-020	Decay Heat Removal 6" DH-C1A & DH-C1B Crossconnect
1D-ISI-DH-021	DH Pump DH-P1A Discharge & Cooler DH-C1A Inlet
1D-ISI-DH-022	Decay Heat Removal Pump DH-P1A Suction
1D-ISI-DH-023	Decay Heat Removal Auxiliary Building Elevation 261'-0"
1D-ISI-DH-024	Decay Heat Removal – Hanger Details
1D-ISI-DH-025	Decay Heat Coolers DH-C1A & DH-C1B
1D-ISI-XX-001	Penetration Details
CONDENSATE SYSTEM (CO)	
1D-421-23 Series	CO Piping Analysis (Class 3 Supports)
EMERGENCY FEEDWATER SYSTEM (EF)	
1D-ISI-EF-001	EF System Inside Reactor Building to Steam Generator RC-H-1B
1D-ISI-EF-002	EF System Inside Reactor Building to Steam Generator RC-H-1A

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
1D-ISI-EF-003	EF System Emergency Feedwater Heaters
1D-ISI-EF-004	EF System Outside Reactor Building to Steam Generator RC-H-1A
1D-ISI-EF-005	EF System Outside Reactor Building to Steam Generator RC-H-1B
1D-524-23 Series	EF Piping Analysis (Class 3 Supports)
1D-ISI-XX-002	Penetration Details
FEEDWATER SYSTEM (FW)	
1D-ISI-FW-001	FW System to Steam Generator RC-H-1B
1D-ISI-FW-002	FW System to Steam Generator RC-H-1A
1D-ISI-FW-003	FW Header to Steam Generators RC-H-1A & RC-H-1B
1D-ISI-FW-004	FW System Hanger Details
1D-ISI-XX-001	Penetration Details
NUCLEAR SERVICES CLOSED CYCLE COOLING WATER SYSTEM (NS)	
1D-ISI-RR-001	Emergency Cooling Water to RB Emergency Coolers
1D-ISI-RR-002	Emergency Cooling Water to RB Emergency Coolers
1D-ISI-RR-003	Emergency Cooling Water to Inside Reactor Building
1D-ISI-RR-004	Emergency Cooling Water From RB Emergency Coolers
1D-541-23 Series	NS Piping Analysis (Class 3 Supports)
1D-ISI-RB-001	Reactor Building Normal Cooling
1D-ISI-XX-002	Penetration Details
REACTOR BUILDING SPRAY SYSTEM (BS)	
1D-ISI-BS-001	Reactor Building Spray System Spray Pump Suction
1D-ISI-BS-002	Reactor Building Spray System Spray Pump Discharge
MAKEUP AND PURIFICATION SYSTEM (LETDOWN PORTION) (MU)	
1D-ISI-MU-001	Makeup & Purification System Letdown Cooler Room
1D-ISI-MU-002	Makeup & Purification System Details

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
1D-ISI-MU-003	Makeup & Purification System Letdown Cooler Discharge
1D-ISI-MU-004	Makeup & Purification System Letdown Cooler Inlet
1D-ISI-MU-005	MU System Seal No. 1 By-Pass & Leak-Off
1D-ISI-MU-006	MU System Seal No. 1 By-Pass From Pump RC-P1D
1D-ISI-MU-007	MU System Seal No. 1 By-Pass From Pump RC-P1C
1D-ISI-MU-008	MU System Seal No. 1 By-Pass From Pump RC-P1A
1D-ISI-MU-009	MU System Seal No. 1 By-Pass From Pump RC-P1B
1D-ISI-MU-010	MU System Seal No. 1 Leak-Off From Pump RC-P1D
1D-ISI-MU-011	MU System Seal No. 1 Leak-Off From Pump RC-P1C
1D-ISI-MU-012	MU System Seal No. 1 Leak-Off From Pump RC-P1B
1D-ISI-MU-013	MU System Seal No. 1 Leak-Off From Pump RC-P1A
1D-ISI-MU-014	MU System Seal No. 1 Leak-Off From Pump RC-P1A
1D-ISI-MU-015	MU System Seal No. 1 Leak-Off From Pump RC-P1B
1D-ISI-MU-016	MU System Seal No. 1 Leak-Off From Pump RC-P1C
1D-ISI-MU-017	MU System Seal No. 1 Leak-Off From Pump RC-P1D
1D-ISI-MU-018	MU System Seal No. 1 By-Pass From Pump RC-P1A
1D-ISI-MU-019	MU System Seal No. 1 By-Pass From Pump RC-P1B
1D-ISI-MU-020	MU System Seal No. 1 By-Pass From Pump RC-P1C
1D-ISI-MU-021	MU System Seal No. 1 By-Pass From Pump RC-P1D
1D-ISI-MU-036	Letdown Cooler Detail
MAKEUP AND PURIFICATION SYSTEM (MAKEUP PORTION) (MU)	
1D-ISI-MU-022	MU System Seal Injection to RC Pump RC-P1B
1D-ISI-MU-023	MU System Seal Injection to RC Pump RC-P1A
1D-ISI-MU-024	MU System Seal Injection to RC Pump RC-P1C
1D-ISI-MU-025	MU System Seal Injection to RC Pump RC-P1D

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
1D-ISI-MU-026	MU System Seal Injection to RC Pump RC-P1B
1D-ISI-MU-027	MU System Seal Injection to RC Pump RC-P1A
1D-ISI-MU-028	MU System Seal Injection to RC Pump RC-P1C
1D-ISI-MU-029	MU System Seal Injection to RC Pump RC-P1D
1D-ISI-MU-030	MU System Seal Injection to Pumps RC-P1A, B, C, & D
1D-ISI-MU-031	MU System High Pressure Injection From Penetrations 321, 322, & 323
1D-ISI-MU-032	MU System Hanger Details
1D-ISI-MU-033	MU System High Pressure Injection to "B" Loop
1D-ISI-MU-034	MU System High Pressure Injection to "A" Loop
1D-ISI-MU-035	Make Up Pump Suction
1D-ISI-MU-037	MU Hanger Details
1D-ISI-MU-038	Make Up Pumps Discharge
1D-ISI-MU-039	MU System Normal Make-up and Seal Injection Lines
1D-ISI-MU-040	High Pressure Injection to Reactor Coolant Loop A
1D-ISI-MU-041	High Pressure Injection to Reactor Coolant Loop B
1D-ISI-MU-042	Make Up Pumps Discharge Bypass Lines
1D-ISI-XX-002	Penetration Details
REACTOR COOLANT SYSTEM (RC)	
1D-ISI-RC-001	Reactor Coolant System OTSG "A" Loop
1D-ISI-RC-002	Reactor Coolant System OTSG "A" Loop Details
1D-ISI-RC-002A	Reactor Coolant System OTSG A Loop Details Pressurizer Surge Line To Hot Leg Weld Overlay Weld SR-010BM
1D-ISI-RC-003	Reactor Coolant System OTSG "B" Loop
1D-ISI-RC-004	Reactor Coolant System OTSG "B" Loop Details
1D-ISI-RC-005	Reactor Coolant System Pressurizer Piping
1D-ISI-RC-006	Reactor Coolant Pumps RC-P1A & RC-P1B

TABLE 2.4-1
ISI ISOMETRIC AND COMPONENT SKETCHES

DRAWING NUMBER	TITLE
1D-ISI-RC-007	Reactor Coolant Pumps RC-P1C & RC-P1D
1D-ISI-RC-008	Steam Generator RC-H1A Details
1D-ISI-RC-009	Steam Generator RC-H1B Details
1D-ISI-RC-010	Reactor Vessel RC-T1 Details
1D-ISI-RC-011	Control Rod Drive
1D-ISI-RC-012	Pressurizer RC-T2 Details
INTERMEDIATE COOLING SYSTEM (IC)	
1D-ISI-IC-001	IC System Inside Reactor Building Elevation 308'-0"
1D-ISI-IC-002	Intermediate Cooling From Penetration No. 302 Elevation 281'-0"
1D-ISI-IC-003	IC System Auxiliary Building From Penetration Nos. 333 & 334
1D-ISI-IC-004	IC Auxiliary Building From Penetration 302
1D-ISI-XX-002	Penetration Details
HYDROGEN PURGE SYSTEM AND MISCELLANEOUS PENETRATIONS (HP)	
1D-ISI-HP-001	Hydrogen Purge
1D-ISI-AH-001	Air Handling System
1D-ISI-AH-002	Air Handling System
1D-ISI-XX-002	Penetration Details

TABLE 2.4-2
CONTAINMENT REFERENCE DRAWINGS

DRAWING NUMBER	TITLE
CONTAINMENT	
TMI1-0011	IWE Component Rollout Inside Containment Liner Looking Out 0 Degrees to 180 Degrees Azimuth
TMI1-0012	IWE Component Rollout Inside Containment Liner Looking Out 180 Degrees to 0 Degrees Azimuth
TMI1-0013	IWE Component Rollout Inside Containment Liner Looking Up at Inside of Dome
TMI1-0014	IWE Component Rollout Outside Containment Concrete Looking In 0 Degrees to 180 Degrees Azimuth
TMI1-0015	IWE Component Rollout Outside Containment Concrete Looking In 180 Degrees to 0 Degrees Azimuth
TMI1-0016	IWE Component Rollout Outside Containment Plan of Dome and Ring Girder at EL 471 FT-4 ½ IN.
E-421-006	Reactor Building Foundation Mat Anchor Bolt and Dowel Plan

2.5 Technical Approach and Positions

When the requirements of ASME Section XI are not easily interpreted, TMI has reviewed general licensing/regulatory requirements and industry practice to determine a practical method of implementing the Code requirements. The Technical Approach and Position (TAP) documents contained in this section have been provided to clarify the TMI implementation of ASME Section XI requirements. An index which summarizes each technical approach and position is included in Table 2.5-1.

TABLE 2.5-1
TECHNICAL APPROACH AND POSITIONS INDEX

Position Number	Revision Date[2]	Status[1]	(Program) Description of Technical Approach and Position
I4T-01	0 03/11/11	Active	**(ISI)** Repair Requirements for ISI Class 1 Repairs in Piping > 3/8" Nominal Pipe Size and Tubing Size > 1/2" in Diameter.
I4T-02	0 03/11/11	Active	**(SPT)** System Leakage Testing of Non-Isolable Buried Components.
I4T-03	0 03/11/11	Active	**(SPT)** Valve Seats/Disks as Pressurization Boundaries.
I4T-04	0 03/11/11	Active	**(ISI)** RISI Examination Information.
I4T-05	0 03/11/11	Active	**(ISI)** Inside Access for Class 2 Item Number C2.32 Nozzle Welds

Note 1: ISI Program Technical Approach and Position Status Options: Active - Current Technical Approach and Position is being utilized at TMI; Deleted - Technical Approach and Position is no longer being utilized at TMI.

Note 2: The revision listed is the latest revision of the subject Technical Approach and Position. The date noted in the second column is the date of the ISI Program Plan revision when the Technical Approach and Position was incorporated into the document.

TECHNICAL APPROACH AND POSITION NUMBER I4T-01
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	1
References:	IWA-4131(a)(2)
Examination Category:	NA
Item Number:	NA
Description:	Repair Requirements for ISI Class 1 Repairs in Piping > 3/8" Nominal Pipe Size and Tubing Size > 1/2" in Diameter
Component Number:	Not Applicable

CODE REQUIREMENT:

IWA-4131.1(a)(2) requires that "the size and design such that, in the event of postulated failure during normal plant operating conditions, the reactor can be shut down and cooled in an orderly manner, assuming makeup is provided by normal reactor coolant makeup systems operable from on-site emergency power.

POSITION:

ISI Class 1 Repairs and Replacements on NPS 3/8 (3/8" nominal pipe size) and less diameter, and tubing ½" diameter and less may apply the small items alternative requirements of IWA-4131.1(a)(2). Reference IR# 350992 for additional basis.

TECHNICAL APPROACH AND POSITION NUMBER I4T-02
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	2 and 3
Reference:	IWA-5244(b)(2)
Examination Category:	C-H, D-B
Item Number:	C7.10, D2.10
Description:	System Leakage Testing of Non-Isolable Buried Components
Component Number:	Non-Isolable Buried Pressure Retaining Components

CODE REQUIREMENT:

Paragraph IWA-5244(b)(2) requires non-isolable buried components be tested to confirm that flow during operation is not impaired.

POSITION:

Article IWA-5000 provides no guidance in setting acceptance criteria for what can be considered "adequate flow". In lieu of any formal guidance provided by the Code, TMI has established the following acceptance criteria:

- For opened ended lines on systems that require Inservice Testing (IST) of pumps, adherence to IST acceptance criteria is considered as reasonable proof of adequate flow through the lines.

- For lines in which the open end is accessible to visual examination while the system is in operation, visual evidence of flow discharging the line is considered as reasonable proof of adequate flow through the open ended line.

- For open ended portions of systems where the process fluid is pneumatic, evidence of gaseous discharge shall be considered reasonable proof of adequate flow through the open ended line. Such test may include passing smoke through the line, hanging balloons or streamers, using a remotely operated blimp, using thermography to detect hot air, etc.

This acceptance criteria will be utilized in order to meet the requirements of Paragraph IWA-5244(b)(2).

The TMI position is that proof of adequate flow is all that is required for testing the buried pipe segments of these open ended lines and that no further visual examination is necessary. This is consistent with the requirements for buried piping, which is not subject to visual examination.

TECHNICAL APPROACH AND POSITION NUMBER I4T-03
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	1, 2, and 3
Reference:	IWA-5221 IWA-5222
Examination Category:	B-P, C-H, D-B
Item Number:	B15.10, C7.10, D2.10
Description:	Valve Seats/Disks as Pressurization Boundaries
Component Number:	All Pressure Testing Boundary Valves

CODE REQUIREMENT:

Paragraph IWA-5221 requires the pressurization boundary for system leakage testing extend to those pressure retaining components under operating pressures during normal system service.

POSITION:

The TMI position is that the pressurization boundary extends up to the valve seat/disk of the valve utilized for isolation. For example, in order to pressure test the ISI Class 1 components, the valve that provides the ISI Class break would be utilized as the isolation point. In this case the true pressurization boundary, and ISI Class break, is actually at the valve seat/disk.

Any requirement to test beyond the valve seat/disk is dependent only on whether or not the piping on the other side of the valve seat/disk is ISI Class 1, 2, or 3.

In order to simplify examination of classed components, TMI will perform a VT-2 visual examination of the entire boundary valve body and bonnet (during pressurization up to the valve seat/disk).

TECHNICAL APPROACH AND POSITION NUMBER I4T-04
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	1 and 2
Reference:	TMI Relief Request I4R-02, Alternative to the ASME Section XI Requirements for ISI Class 1 and Class 2 Piping Welds TMI, Risk Informed Inservice Inspection Evaluation (Final Report) ASME Code Case N-578-1: Risk-Informed Requirements for ISI Class 1, 2, or 3 Piping, Method B Section XI, Division 1 Electric Power Research Institute (EPRI) Topical Report (TR) 112657, Rev. B-A, Revised Risk-Informed Inservice Inspection Evaluation
Examination Category:	Previously Examination Category B-F, B-J, C-F-1, and C-F-2 now incorporated into R-A
Description:	RISI Examination Information

CODE REQUIREMENT:

ASME Code Case N-578-1 Table 1, "Examination Category R-A, "Risk-Informed Piping Examinations" will also be used in conjunction with Table 4-1 of EPRI TR-112657 to categorize the parts examined under the RISI Program. Code Case N-578-1 Table 1 provides examination requirements, examination method, acceptance standards, examination extent and frequency for piping structural elements not subject to a damage mechanism.

Examination programs developed in accordance with this Code Case shall use NDE techniques suitable for specific degradation mechanisms and examination locations. The examination volumes and methods that are appropriate for each degradation mechanism are provided in Table 1 of this Code Case. The methods and procedures used for the examinations shall be qualified to reliably detect and size the relevant degradation mechanisms identified for each elements.

The requirements for examination methods and areas/volumes are located in several sources other than the station's base edition of ASME Section XI. The guidance for the examination volume for a given degradation mechanism is provided by the EPRI Topical Report TR-112657, Rev. B-A while the guidance for the examination method is provided by ASME Code Case N-578-1.

TECHNICAL APPROACH AND POSITION NUMBER I4T-04
Revision 0

POSITION:

Table I4T-04-1: ASME Code Case N-578-1 Table 1

Section XI Item No.	Components and Parts To Be Examined	Exam Method	Examination Requirements For 10-Year Interval	Notes
R1.10	High-Safety-Significant Piping Structural Elements			
R1.11	Elements Subject to Thermal Fatigue (TT & TASCS)	Volumetric (UT-E)	Figures 4-1 through 4-4 of EPRI Report TR-112657, Revision B-A.	
R1.12	Elements Subject to High Cycle Mechanical Fatigue (VF)	Visual (VT-2)	Not currently applicable to TMI.	
R1.13	Elements Subject to Erosion Cavitation (EC)	Volumetric (UT-WT)	Wall thickness measurement extent and grid size to be performed in accordance with the Site Flow Accelerated Corrosion Program.	Table 4-1 of EPRI Report TR-112657, Revision B-A applies (including reference to Figures 4-16 through 4-22 of the report).
R1.14	Elements Subject to Crevice Corrosion Cracking (CC)	Volumetric (UT)	Not currently applicable to TMI.	
R1.15	Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC)	Volumetric (UT-E) and Visual (VT-2)	Per Relief Request I4R-02	Examinations will be performed per applicable augmented examination basis (e.g.; MRP-139, Q-4300, Code Case N-770, etc.)
R1.16	Elements Subject to Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC)	Volumetric (UT)	Per figures 4-10 through 4-14 of EPRI Report TR-112657, Revision B-A	
R1.17	Elements Subject to Microbiologically Corrosion (MIC & PIT)	Visual (VT-3) Internal Surfaces or Volumetric (UT)	Not currently applicable to TMI.	
R1.18	Elements Subject to Flow Accelerated Corrosion (FAC)	FAC Program	Per FAC program.	In accordance with the Owner's FAC program.
R1.19	Elements Subject to External Chloride Stress Corrosion Cracking (ECSCC)	Surface	Not currently applicable to TMI.	

TECHNICAL APPROACH AND POSITION NUMBER I4T-04
Revision 0

Section XI Item No.	Components and Parts To Be Examined	Exam Method	Examination Requirements For 10-Year Interval	Notes
R1.20	No Damage Mechanism (NONE)	Volumetric (UT-E)	IWB-2500-8c, IWB-2500-9,10,11, IWC-2500-7a	The length for the examination volume shall be increased to include 1/2 inch beyond each side of the base metal thickness transition or counterbore or ½" beyond the edge of the weld if no base metal thickness transition or counterbore is detected.
	Socket Welds (All DM)	Visual (VT-2)		Socket welds require only a VT-2 visual examination each refueling outage.

TECHNICAL APPROACH AND POSITION NUMBER I4T-05
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	2
References:	Table IWC-2500-1
Examination Category:	C-B
Item Number:	C2.32
Description:	Inside Access for Class 2 Item Number C2.32 Nozzle Welds
Component Number:	All Item Number C2.32 Components

CODE REQUIREMENT:

Table IWC-2500-1 requires volumetric examination of nozzle welds with reinforcing plates "When Inside of Vessel is Accessible".

POSITION:

Except for reactor vessel internals examinations, ASME Section XI does not require disassembly of components for inspection only. TMI will complete this examination on the Item Number C2.32 nozzle welds if the welds become accessible during required maintenance activities and will not disassemble the component for the sole purpose of nozzle volumetric examination.

3.0 COMPONENT ISI PLAN

The TMI Component ISI Plan includes ASME Section XI nonexempt pressure retaining welds, piping structural elements, pressure retaining bolting, welded attachments, pump casings, valve bodies, reactor pressure vessel interior, reactor pressure vessel interior attachments, reactor pressure vessel welded core support structures, and steam generator tubing of ISI Class 1, 2, and 3 components that meet the criteria of Subarticle IWA-1300. These components are identified on the ISI Boundary Drawings listed in Table 2.3-1. Procedure ER-AA-330-002 "Inservice Inspection of Section XI Welds and Components," implements the ASME Section XI welds and components program. This Component ISI Plan also includes Augmented Examination Program examination requirements specified by documents other than ASME Section XI. For a detailed discussion of these examination requirements, see Section 2.2 of this document.

3.1 Nonexempt ISI Class Components

The TMI ISI Class 1, 2, and 3 components subject to examination are those which are not exempted under the criteria of Paragraphs IWB-1220, IWC-1220, and IWD-1220, respectively. A summary of TMI ASME Section XI nonexempt components is included in Section 7.0.

3.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Components

ISI Class 1, 2, and 3 nonexempt components are identified on the ISI Isometrics and Component Sketches listed in Section 2.4, Table 2.4-1. ISI Class 3 welded attachments are also identified by controlled individual support detail drawings.

3.2 Risk-Informed Examination Requirements

Piping structural elements that fall under RISI Examination Category R-A are risk ranked as High (1, 2, and 3), Medium (4 and 5), and Low (6 and 7). Per the EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1, piping structural elements ranked as High or Medium Risk are subject to examination while piping structural elements ranked as Low Risk are not subject to examinations (except for pressure testing). Thin wall welds that were excluded from volumetric examination under ASME Section XI rules per Table IWC-2500-1 are included in the element scope that is potentially subject to RISI examination at TMI.

Piping structural elements may be excluded from examination (other than pressure testing) under the RISI Program if the only degradation mechanism present for a given location is inspected for cause under certain other TMI programs such as the Flow Accelerated Corrosion (FAC) Program or Alloy 600 Program. These piping structural elements will remain part of the assigned programs, which already perform "for cause" inspections to detect these degradation mechanisms. Piping structural elements susceptible to FAC or PWSCC along with another degradation mechanism (e.g., thermal fatigue) are retained as part of the RISI scope and are included in the element selection for the purpose of performing examinations to detect the additional degradation mechanism. The RISI Program element examinations are performed in accordance with Relief Request I4R-02.

4.0 SUPPORT ISI PLAN

The TMI Support ISI Plan includes the supports of ASME Section XI nonexempt ISI Class 1, 2, and 3 components as described in Section 3.0. Procedure ER-AA-330-003 "Inservice Inspection of Section XI Component Supports," implements the ASME Section XI Support ISI Plan.

4.1 Nonexempt ISI Class Supports

The TMI ISI Class 1, 2, and 3 nonexempt supports are those which do not meet the exemption criteria of Paragraph IWF-1230 of ASME Section XI. A summary of TMI ASME Section XI nonexempt supports is included in Section 7.0.

4.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Supports

ISI Class 1, 2, and 3 nonexempt supports are identified on TMI ISI Isometric and Component Sketches listed in Section 2.4, Table 2.4-1. Supports are identified by controlled individual support detail drawings.

4.2 Snubber Examination and Testing Requirements

4.2.1 ASME Section XI Paragraphs IWF-5200(a) and (b) and IWF-5300(a) and (b) require VT-3 Visual Examinations and Inservice Tests (functional tests) of snubbers to be performed in accordance with the Operation and Maintenance of Nuclear Power Plants (OM), Standard ASME/ANSI OM, Part 4. Relief Request I4R-04 was submitted to propose an alternative to the visual examination and functional testing requirements of Paragraphs IWF-5200(a), IWF-5300(a), IWF-5200(b), and IWF-5300(b). The purpose of this relief request is to justify replacing the visual examination and functional testing requirements of Paragraphs IWF-5200(a) and (b) and IWF-5300(a) and (b) and the frequency required by OMa-1988 Addenda to the ASME/ANSI OM-1987 Edition, Part 4, with the requirements of TMI Technical Specifications, Section 4.17, Shock Suppressors (Snubbers). TMI site specific procedures implement and control the visual examination and functional testing program for safety related snubbers.

Procedures ER-AA-330-004 "Visual Examination of Snubbers," ER-AA-330-010 "Snubber Functional Testing," ER-AA-330-011 "Snubber Service Life Monitoring Program," and TMI surveillance test procedures are also used to implement the visual examination, functional testing, and service life monitoring requirements for safety related snubbers.

The ASME Section XI ISI Program uses Subsection IWF to define support inspection requirements. The ISI Program maintains the Code Class snubbers in the populations subject to inspection per Article IWF-2000. This is done to accommodate scheduling and inspection requirements of the related attachment hardware per Paragraphs IWF-5200(c) and IWF-5300(c). (See Section 4.2.2 below.)

4.2.2 ASME Section XI Paragraphs IWF-5200(c) and IWF-5300(c) require integral and non-integral attachments for snubbers to be examined in accordance with Subsection IWF of ASME Section XI. This results in VT-3 visual examination of the snubber attachment hardware including the bolting, pins, and their interface to the clamp, but does not include the component-to-clamp interface.

The ASME Section XI ISI Program uses Subsection IWF to define the inspection requirements for all ISI Class 1, 2, and 3 supports, regardless of type. The ISI Program maintains the Code Class snubbers in the support populations subject to inspection per Article IWF-2000. This is done to facilitate scheduling and inspection requirements of the snubber attachment hardware (e.g., bolting and pins) per Paragraphs IWF-5200(c) and IWF-5300(c).

It should be noted that the examination of snubber welded attachments will be performed in accordance with the ASME Section XI Subsections IWB, IWC, and IWD welded attachment examination requirements (e.g., Examination Categories B-K, C-C, and D-A).

5.0 SYSTEM PRESSURE TESTING ISI PLAN

The TMI System Pressure Testing (SPT) ISI Plan includes pressure retaining ASME Section XI, ISI Class 1, 2, and 3 components, with the exception of those specifically exempted by Paragraphs IWA-5110(c), IWC-5222(b), and IWD-5222(b). RISI piping structural elements, regardless of risk classification, remain subject to pressure testing as part of the current ASME Section XI program.

The SPT ISI Plan performs system pressure tests and required VT-2 visual examinations on the ISI Class 1, 2, and 3 pressure retaining components to verify system and component structural integrity. This program conducts both Periodic and Interval (10-Year frequency) pressure tests as defined in ASME Section XI Inspection Program B. Procedure ER-AA-330-001, "Section XI Pressure Testing," as well as TMI site-specific test procedures, implement the ASME Section XI System Pressure Testing ISI Plan.

This SPT ISI Plan also includes Augmented Examination Program examination requirements specified by documents other than ASME Section XI. For a detailed discussion of these examination requirements, see Section 2.2 of this document.

5.1 ISI Class Systems

All ISI Class 1 pressure retaining components, typically defined as the reactor coolant pressure boundary, are required to be tested. Those portions of ISI Class 2 and 3 systems that are required to be tested include the pressure retaining boundaries of components required to operate or support the system safety functions. ISI Class 2 and 3 open ended discharge piping and components are excluded from the examination requirements per Paragraphs IWC-5222(b) and IWD-5222(b).

5.1.1 Identification of ISI Class 1, 2, and 3 Components

Components subject to ASME Section XI System Pressure Testing are shown on the ISI Boundary Drawings listed in Section 2.3, Table 2.3-1. These portions of systems will be pressure tested in accordance with ASME Section XI except for those identified in brown on the ISI Boundary Drawings because they do not require periodic pressure testing per ASME Section XI. The brown highlighted portions of systems are those ISI Class 3 portions of systems that perform functions other than those identified in ASME Section XI, Paragraph IWD-1210.

Additional information on the classification of various system boundaries is provided in the ISI Classification Basis Document.

5.1.2 Identification of System Pressure Tests

Components subject to pressure testing are identified on the ISI Boundary Drawings (1D-ISI-FD series). Individual tests are identified and maintained in the TMI ISI Database.

5.2 Risk-Informed Examinations of Socket Welds

Socket welds selected for examination under the RISI program are to be inspected with a VT-2 visual examination <u>each</u> refueling outage per ASME Code Case N-578-1 (see footnote 12 in Table 1 of the Code Case). To facilitate this, socket welds selected for inspection under the RISI program are pressurized each refueling outage during a system pressure test in accordance with Paragraph IWA-5211(a).

6.0 CONTAINMENT ISI PLAN

The TMI Containment ISI (CISI) Plan includes ASME Section XI Class MC pressure retaining components and their integral attachments (including the Class CC metal liner), and Class CC components and structures, and post-tensioning systems that meet the criteria of Subarticle IWA-1300. This Containment ISI Plan also includes information related to augmented examination areas, component accessibility, and examination review.

The inspection of containment structures and components are performed per procedures ER-AA-330-005, "Visual Examination of Section XI Class CC Concrete Containment Structures," ER-AA-330-006, "Inservice Inspection and Testing of the Pre-Stressed Concrete Containment Post Tensioning Systems," and ER-AA-330-007, "Visual Examination of Section XI Class MC Surfaces and Class CC Liners." In addition, site specific procedures are used to complete more complex surveillances such as tendon testing.

6.1 Nonexempt CISI Class Components

The TMI CISI Class MC and CC components identified on the Containment Reference Drawings are those not exempted under the criteria of Paragraphs IWE-1220 and IWL-1220 in the 2004 Edition, No Addenda of ASME Section XI. A summary of TMI ASME Section XI nonexempt CISI components is included in Section 7.0.

The process for scoping TMI components for inclusion in the Containment ISI Plan is included in the containment sections of the ISI Classification Basis Document. These sections include a listing and detailed basis for inclusion of containment components.

Components that are classified as CISI Class MC and CC must meet the requirements of ASME Section XI in accordance with 10 CFR 50.55a(g)(4).

6.1.1 Identification of CISI Class MC and CC Nonexempt Components

CISI Class MC and CC components are identified on the Containment Reference Drawings listed in Section 2.4, Table 2.4-2.

6.1.2 Identification of CISI Class MC and CC Exempt Components

Certain containment components or parts of components may be exempted from examination based on design and accessibility per the requirements of Paragraphs IWE-1220 and IWL-1220.

The process for exempting TMI components from the Containment ISI Plan per Paragraphs IWE-1220 and IWL-1220 is included in the containment sections of the ISI Classification Basis Document. These sections include discussions of exempt components and the bases for those exemptions.

6.2 Augmented Examinations Areas

The containment section of the ISI Classification Basis Document discusses the containment design and components. Metal containment surface areas subject to accelerated degradation and aging require augmented examination per Examination Category E-C and Paragraph IWE-1240.

Similarly, concrete surfaces may be subject to Detailed Visual examination in accordance with Item Number L1.12 and Paragraph IWL-2310(b), if declared to be 'Suspect Areas'.

The moisture barrier is located on the interior surface of containment where the concrete floor slab meets the metal liner at elevation 281'-0". This is a cork and silicone sealant seal between the concrete floor surface and the metallic liner. For the First CISI Interval, TMI had classified portions of the metallic liner near the moisture barrier under Examination Category E-C, Item Numbers E4.11 and E4.12. Successive examinations have since been performed and the areas no longer require the Examination Category E-C designation per ASME Section XI. The areas of corrosion identified during the 2007 and earlier examinations were examined in 2009 and determined to be unchanged from prior examinations. The areas of corrosion where the liner was reduced to <90% of nominal thickness were restored to >90% nominal thickness during the 2009 refueling outage. This corrosion was attributed to degradation of the moisture barrier which has been replaced for 360 degrees around the liner. However, for the Second CISI Interval, TMI will optionally continue to identify the area near the moisture barrier Examination Category E-C, Item Number E4.11 and will implement an owner's augmented examination program. The owner's augmented examination program will be to conduct a visual examination each refueling outage at the metallic liner to moisture barrier interface.

No other significant conditions are currently identified in the Second CISI Interval as requiring application of additional augmented examination requirements under Paragraph IWE-1240 or IWL-2310, or as additional owner's augmented examination programs.

6.3 Component Accessibility

CISI Class MC and CC components subject to examination shall remain accessible for either direct or remote visual examination from at least one side per the requirements of ASME Section XI, Paragraph IWE-1230.

Paragraph IWE-1231(a)(3) requires 80% of the pressure-retaining boundary that was accessible after construction to remain accessible for either direct or remote visual examination, from at least one side of the vessel, for the life of the plant.

Portions of components embedded in concrete or otherwise made inaccessible during construction are exempted from examination, provided that the requirements of ASME Section XI, Paragraph IWE-1232 have been fully satisfied.

In addition, inaccessible surface areas exempted from examination include those surface areas where visual access by line of sight with adequate lighting from permanent vantage points is obstructed by permanent plant structures, equipment, or components; provided these surface areas do not require examination in accordance with the inspection plan, or augmented examination in accordance with Paragraph IWE-1240.

6.4 Responsible Individual and Engineer

ASME Section XI Subsection IWE requires the Responsible Individual to be involved in the development, performance, and review of the CISI examinations. The Responsible Individual shall meet the requirements of ASME Section XI, Paragraph IWE-2320.

ASME Section XI Subsection IWL requires the Responsible Engineer to be involved in the development, approval, and review of the CISI examinations. The Responsible Engineer shall meet the requirements of ASME Section XI, Paragraph IWL-2320.

7.0 COMPONENT SUMMARY TABLES

7.1 Inservice Inspection Summary Tables

The following Table 7.1-1 provides a summary of the ASME Section XI pressure retaining components, supports, containment structures, post-tensioning systems, system pressure testing, and augmented program components for the Fourth ISI Interval and the Second CISI Interval at TMI.

The format of the Inservice Inspection Summary Tables is as depicted below and provides the following information:

Examination Category (with Examination Category Description)	Item Number (or Risk Category or Augmented Number)	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)

(1) Examination Category (with Examination Category Description):

Provides the Examination Category and description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, IWF-2500-1, and IWL-2500-1.

Examination Category "R-A" from ASME Code Case N-578-1 is used in lieu of ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 to identify ISI Class 1 and 2 piping structural elements for the RISI program. Only those Examination Categories applicable to TMI are identified.

Examination Category "NA" is used to identify Augmented Examination Programs.

(2) Item Number (or Risk Category Number or Augmented Number):

Provides the Item Number as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, IWF-2500-1, and IWL-2500-1. Only those Item Numbers applicable to TMI are identified.

For piping structural elements under the RISI program, the Risk Category Number (1 through 5) is used in place of the Item Number.

Specific abbreviations have been developed to identify Augmented Examination Programs.

(3) Item Number (or Risk Category Number or Augmented Number) Description:

Provides the description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, IWF-2500-1, and IWL-2500-1.

For Risk-Informed piping examinations, a description of the Risk Category is provided.

For Augmented Examination Program examinations, a description of the Augmented Examination Program basis is provided.

(4) Examination Requirements:

Provides the examination methods required by ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, IWF-2500-1, and IWL-2500-1.

Provides the examination requirements for piping structural elements under RISI that are in accordance with the EPRI Topical Reports TR-112657, Rev. B-A, TR-1006937, Rev. 0-A, and ASME Code Case N-578-1.

Provides the examination requirements for Augmented Examination Programs.

(5) Total Number Of Components by System:

Provides the system designator (abbreviations).

This column also provides the number of components within a particular system for that Item Number, Risk Category Number, or Augmented Examination Program.

Note that the total number of components by system are subject to change after completion of plant modifications, design changes, and ISI system classification updates.

(6) Relief Request/Technical Approach & Position Number:

Provides a listing of Relief Request/ TAP Numbers applicable to specific components, the ASME Section XI Item Number, Risk Category Number, or Augmented Number. Relief Requests and TAP Numbers that generically apply to all components, or an entire class are not listed. If a Relief Request/ TAP Number is identified, see the corresponding relief request in Section 8.0 or the TAP Number in Section 2.5.

(7) <u>Notes:</u>

Provides a listing of program notes applicable to the ASME Section XI Item Number, Risk Category Number, or Augmented Number. If a program note number is identified, see the corresponding program note in Table 7.1-2.

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
B-A Pressure Retaining Welds in Reactor Vessel	B1.11	Circumferential Shell Welds (Reactor Vessel)	Volumetric	RC (RPV): 4	RR-09-01	
	B1.12	Longitudinal Shell Welds (Reactor Vessel)	Volumetric	RC (RPV): 4	RR-09-01	
	B1.21	Circumferential Head Welds (Reactor Vessel)	Volumetric	RC (RPV): 1	RR-09-01	
	B1.30	Shell-to-Flange Weld (Reactor Vessel)	Volumetric	RC (RPV): 1	RR-09-01	
	B1.51	Beltline Region Repair Welds (Reactor Vessel)	Volumetric	RC (RPV): 1	RR-09-01	
B-B Pressure Retaining Welds in Vessels Other Than Reactor Vessels	B2.11	Circumferential Shell-To-Head Welds (Pressurizer)	Volumetric	RC (PZR): 2		
	B2.12	Longitudinal Shell-To-Head Welds (Pressurizer)	Volumetric	RC (PZR): 3		
	B2.31	Circumferential Head Weld (Steam Generator)	Volumetric	RC (SG): 6		
	B2.40	Tube Sheet-To-Head Weld (Steam Generator)	Volumetric	RC (SG): 4		
	B2.70	Longitudinal Welds (Heat Exchanger)	Volumetric or Visual, VT-2	MU: 8		13
B-D Full Penetration Welds of Nozzles in Vessels	B3.90	Nozzle-to-Vessel Welds (Reactor Vessel)	Volumetric	RC (RPV): 8	RR-09-01	11
	B3.100	Nozzle Inside Radius Section (Reactor Vessel)	Volumetric or Enhanced Visual	RC (RPV): 8	RR-09-01	12
	B3.110	Nozzle-to-Vessel Welds (Pressurizer)	Volumetric	RC (PZR): 5		
	B3.120	Nozzle Inside Radius Section (Pressurizer)	Volumetric or Enhanced Visual	RC (PZR): 5		9
	B3.140	Nozzle Inside Radius Section (Steam Generator)	Volumetric or Enhanced Visual	RC (SG): 6		9

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
B-G-1 Pressure Retaining Bolting, Greater Than 2 in. In Diameter	B6.10	Closure Head Nuts (Reactor Vessel)	Visual, VT-1	RC (RPV): 60		
	B6.20	Closure Studs (Reactor Vessel)	Volumetric	RC (RPV): 60		10
	B6.40	Threads in Flange (Reactor Vessel)	Volumetric	RC (RPV): 60		
	B6.50	Closure Washers (Reactor Vessel)	Visual, VT-1	RC (RPV): 60 (60 Sets of 2 Washers Each)		
	B6.60	Bolts & Studs (Pressurizer)	Volumetric	RC (PZR): 1 (12 Studs Only)		
	B6.70	Flange Surface, when connection disassembled (Pressurizer)	Visual, VT-1	RC (PZR): 1		
	B6.80	Nuts, Bushings, & Washers (Pressurizer)	Visual, VT-1	RC (PZR): 1 (12 Nuts Only)		
	B6.180	Bolts & Studs (Pumps)	Volumetric	RC (RCP): 96		
	B6.190	Flange Surface, when connection disassembled (Pumps)	Visual, VT-1	RC (RCP): 96		
	B6.200	Nuts, Bushings, & Washers (Pumps)	Visual, VT-1	RC (RCP): 24		
B-G-2 Pressure Retaining Bolting, 2 in. and Less (Note that the B-G-2 population Is based on the number of Connections and not individual Fastener population)	B7.20	Bolts, Studs, & Nuts (Pressurizer)	Visual, VT-1	RC (PZR): 3		
	B7.30	Bolts, Studs, & Nuts (Steam Generator)	Visual, VT-1	RC (SG): 6		
	B7.50	Bolts, Studs, & Nuts (Piping)	Visual, VT-1	MU: 4 RC (PZR): 4 RC (RCP): 4		
	B7.60	Bolts, Studs, & Nuts (Pumps)	Visual, VT-1	RC (RCP): 4		
	B7.70	Bolts, Studs, & Nuts (Valves)	Visual, VT-1	CF: 4 MU: 2 RC: 3		
	B7.80	Bolts, Studs, & Nuts in CRD Housing (Reactor Vessel)	Visual, VT-1	RC (RDU): 69		8

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
B-K Welded Attachments for Vessels, Piping, Pumps, and Valves	B10.10	Welded Attachments (Pressure Vessels)	Surface or Volumetric	MU: 2 RC (PZR): 1 RC (SG): 2		
	B10.20	Welded Attachments (Piping)	Surface	CF: 2 DH: 1 MU: 10		
	B10.30	Welded Attachments (Pumps)	Surface	RC (RCP): 4		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
B-L-2 Pump Casings	B12.20	Pump Casings (Pumps)	Visual, VT-3	RC (RCP): 4		
B-M-1 Pressure Retaining Welds In Valve Bodies	B12.30	Valve Body Welds (less than NPS 4) (Valves)	Surface	RC: 1 (Pressurizer PORV)		14
B-M-2 Valve Bodies	B12.50	Valve Bodies (Exceeding NPS 4) (Valves)	Visual, VT-3	CF: 4 DH: 4		
B-N-1 Interior of Reactor Vessel	B13.10	Vessel Interior (Reactor Vessel)	Visual, VT-3	RC (RI): 1		19
B-N-2 Welded Core Support Structures and Interior Attachments to Reactor Vessels	B13.50	Interior Attachments Within Beltline Region (Reactor Vessel)	Visual, VT-1	RC (RI): 1 (12 RPV Guide Lugs)	RR-09-02	19
B-N-3 Removable Core Support Structures	B13.70	Core Support Structure (Reactor Vessel)	Visual, VT-3	RC (RI): 1	RR-09-02	19
B-O Pressure Retaining Welds in Control Rod Housings	B14.20	Welds in CRD Housing (Reactor Vessel) (10% of peripheral CRD Housings to be inspected. 24 of the 69 CRD Housings are identified as peripheral)	Volumetric or Surface	RC (RDU): 276 (69 CRD Housings with 4 Welds Each). 24 Drive Units Are At Peripheral Locations		18

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
B-P All Pressure Retaining Components	B15.10	System Leakage Test (IWB-5220)	Visual, VT-2	CF DH MU RC	I4T-03	
B-Q Steam Generator Tubing	B16.10	Steam Generator Tubing in Straight Tube Design	Volumetric Per Tech Specs	RC (SG): 2		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
C-A Pressure Retaining Welds in Pressure Vessels	C1.10	Shell Circumferential Welds (Pressure Vessels)	Volumetric or Visual, VT-2	DH: 2 (Coolers)		13
	C1.20	Head Circumferential Welds (Pressure Vessels)	Volumetric or Visual, VT-2	DH: 2		13
	C1.30	Tubesheet-to-Shell Welds (Pressure Vessels)	Volumetric	RC (SG): 4 (EOSTG)		
C-B Pressure Retaining Nozzle Welds in Vessels	C2.21	Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Welds Without Reinforcing Plate, Greater Than ½" Nominal Thickness (Pressure Vessels)	Volumetric & Surface	RC (SG): 4 (EOSTG)		
	C2.31	Reinforcing Plate Welds to Nozzle and Vessel With Reinforcing Plate, Greater Than ½" Nominal Thickness (Pressure Vessels)	Surface or Visual, VT-2	DH: 4		
	C2.32	Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Welds When Inside of Vessel is Accessible With Reinforcing Plate, Greater Than ½" Nominal Thickness (Pressure Vessels)	Volumetric or Visual, VT-2	DH: 4	I4T-05	
	C2.33	Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Welds When Inside of Vessel Is Inaccessible With Reinforcing Plate, Greater Than ½" Nominal Thickness (Pressure Vessels)	Visual, VT-2	DH: 4		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
C-C Welded Attachments for Vessels, Piping, Pumps, and Valves	C3.20	Welded Attachments (Piping)	Surface	BS: 10 DH: 27 EF: 4 FW: 13 HP: 1 MS: 10 MU: 32 RR: 36		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
C-H All Pressure Retaining Components	C7.10	System Leakage Test (IWC-5220)	Visual, VT-2	BS CF DH EF HP IC MS MU NS RC	I4R-03 I4T-02 I4T-03	

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
D-A Welded Attachments for Vessels, Piping, Pumps, and Valves	D1.10	Welded Attachments (Pressure Vessels)	Visual, VT-1	CO: 2 DC: 2 DH: 2 DR: 2 NR: 3 NS: 4 RR: 2		
	D1.20	Welded Attachments (Piping)	Visual, VT-1	CO: 4 DC: 19 DR: 30 EF: 20 MS: 6 NS: 27		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
D-B All Pressure Retaining Components	D2.10	System Leakage Test (IWD-5221)	Visual, VT-2	CA CH CO DH EF MS MU NS RW SF	I4T-02 I4T-03	

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components	Relief Request/ TAP Number	Notes
E-A Containment Surfaces	E1.11	Containment Vessel Pressure Retaining Boundary - Accessible Surface Areas	General Visual	5 (5 Areas/Zones) Liner Dome, Walls, Penetrations, Hatches, Transfer Tubes (Transfer Tubes are Class 2 (Not Requiring Examinations per IWE)) and Attachments		
	E1.11	Containment Vessel Pressure Retaining Boundary - Bolted Connections, Surfaces	Visual, VT-3	33		4
	E1.12	Containment Vessel Pressure Retaining Boundary - Wetted Surfaces of Submerged Areas	Visual, VT-3	2 (2 Penetrations - Fuel Transfer Tubes)		5
	E1.30	Moisture Barriers	General Visual	1 (1 Seal, 360° at the 281' Elevation)		
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Containment Surface Areas - Visible Surfaces	Visual, VT-1	1 (1 Area (360°) Adjacent to the 281' Elevation)		6
	E4.12	Containment Surface Areas - Surface Area Grid Minimum Wall Thickness Locations	Ultrasonic Thickness	0		7

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
F-A Supports	F1.10	Class 1 Piping Supports	Visual, VT-3	CF: 12 DH: 12 MU: 196 RC: 26	I4R-04	1
	F1.20	Class 2 Piping Supports	Visual, VT-3	BS: 32 DH: 123 EF: 68 FW: 29 HP: 2 MS: 114 MU: 106 RR: 93	I4R-04	1
	F1.30	Class 3 Piping Supports	Visual, VT-3	CO: 14 DC: 60 DR: 77 EF: 61 MS: 14 NR: 22 NS: 165	I4R-04	1

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
F-A Supports (Continued)	F1.40	Supports Other Than Piping Supports (Class 1, 2, and 3)	Visual, VT-3	CO: 2 DC: 4 DH: 5 DR: 4 EF: 9 MS: 8 MU: 3 NR: 7 NS: 7 RC: 5 RC (PZR): 1 RC (RCP): 48 RC (RPV): 1 RC (SG): 10 RR: 4	I4R-04	1

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components	Relief Request/ TAP Number	Notes
L-A Concrete Surfaces	L1.11	Concrete Surfaces - All Accessible Surface Areas	General Visual	3 Areas/Zones Containment Dome, Walls and Basemat		
	L1.12	Concrete Surfaces - Suspect Areas (No Suspect Areas Identified)	Detailed Visual	Areas Identified From Above		
L-B Unbonded Post-Tensioning System	L2.10	Tendon	Physical IWL-2522	643		
	L2.20	Tendon - Wire or Strand	Visual and Physical IWL-2523.2	1 of Each Type Per Selection		
	L2.30	Tendon - Anchorage Hardware and Surrounding Concrete	Detailed Visual	643 X 2 = 1286		
	L2.40	Tendon - Corrosion Protection Medium	Physical IWL-2525.2(a)	643 X 2 = 1286		
	L2.50	Tendon - Free Water	Physical IWL-2525.2(b) (Note IWL-2524.2)	If Present		

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Risk Category Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
R-A Risk-Informed Piping Examinations	1	Risk Category 1 Elements	See Notes	MS: 60	I4R-02 I4T-04	2 3
	2	Risk Category 2 Elements	See Notes	DH: 4 MU: 114 RC: 27	I4R-02 I4R-05 I4T-04 RR-10-02	2 3
	4	Risk Category 4 Elements	See Notes	BS: 111 CF: 15 DH: 121 EF: 4 MS: 71 MU: 653 RC: 201 RC (PZR): 3 RC (RCP): 8 RC (RPV): 1	I4R-02 I4R-05 I4T-04	2 3
	5	Risk Category 5 Elements	See Notes	DH: 5	I4R-02 I4T-04	2 3

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY

Examination Category (with Examination Category Description)	Aug Number	Description	Exam Requirements	Total Number of Components by System	Relief Request/ TAP Number	Notes
NA Augmented Components	3.6.2	USNRC MEB Technical Position 3-1, Examination of High Energy Circumferential and Longitudinal Piping Welds (MEB 3-1, See ISI Program Plan Section 2.2.1)	Volumetric and Surface	FW: 6		
	TS4.2.4	Augmented Examination of Reactor Coolant Pump Flywheel per Regulatory Guide 1.14	Volumetric, Surface, and Visual	RC (RCP): 4		15
	TS4.15	Technical Specifications Section 4.15 Examination	Volumetric	MS: 4		16
	TS4.17	Snubber Visual Examination and Functional Testing Program	Visual, VT-3 & Functional Test	In accordance with Snubber Program		
	NUREG 0612	Examination of Load Bearing Welds per NUREG 0612 Control of Heavy Loads	Surface, MT and Visual	RC (RPV): 4 (Tripod) RC (RPV): 1 (3 RPV Head Lifting Lugs) RC (RPV): 1 (Shielded Work Platform)		
	Alloy 600/82/182	MRP-126 and MRP-139 - Alloy 600 Program (Code Cases N-722 and N-729-1 per 10 CFR 50.55a)	BMV or Volumetric	DH: 1 MU: 2 RC: 61 RC (PZR): 5 RC (RPV): 3		17
	MRP-146	Stagnant Water Branch Connection Thermal Fatigue	Visual, Volumetric	MU: 6 RC: 3		
	MRP-192	Decay Heat Mixing Tee Thermal Fatigue (MRP-192)	Volumetric	DH: 6		

TABLE 7.1-2
INSERVICE INSPECTION SUMMARY TABLE PROGRAM NOTES

Note #	Note Summary
1	ISI Snubber visual examinations and functional testing are performed in accordance with Relief Request I4R-04 and the TMI Technical Specifications, Section 4.17, Shock Suppressors (Snubbers). The number of TMI supports identified, includes snubbers for the visual examination and functional testing of the snubber per Relief Request I4R-04, and include the integral and nonintegral attachments per Paragraphs IWF-5200(c), IWF-5300(c), and IWF-2500(a). The snubbers are scheduled and administratively tracked in the ISI Program; however, the Technical Specifications, Section 4.17, snubber program will be the mechanism for actually performing the visual examinations and functional testing scheduled within the ISI Program. It should be noted that snubber welded attachments will be performed in accordance with the ASME Section XI Subsections IWB, IWC, and IWD welded attachment examination requirements (e.g.; Examination Categories B-K, C-C, and D-A.) For a detailed discussion of the snubber program, see Section 4.2 of this document.
2	For the Fourth Inspection Interval, the TMI ISI Class 1 and 2 piping inspection program will be governed by risk-informed regulations. The RISI Program methodology is described in the EPRI Topical Reports TR-112657, Rev. B-A and ASME Code Case N-578-1. The RISI Program scope has been implemented as an alternative to the 2004 Edition, No Addenda of the ASME Section XI examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-F-1 and C-F-2 welds in accordance with 10 CFR 50.55a(a)(3)(i).
3	Examination requirements within the RISI Program are determined by the various degradation mechanisms present at each individual piping structural element. See EPRI Topical Reports TR-112657, Rev. B-A and ASME Code Case N-578-1 for specific examination method requirements.
4	Bolted connections examined per Item Number E1.11 require a General Visual examination each period and a VT-3 visual examination once per interval and each time the connection is disassembled during a scheduled Item Number E1.11 examination. Additionally, a VT-1 visual examination shall be performed if degradation or flaws are identified during the VT-3 visual examination. These modifications are required by 10 CFR 50.55a(b)(2)(ix)(G) and 10 CFR 50.55a(b)(2)(ix)(H).
5	Item Number E1.12 requires VT-3 visual examination in lieu of General Visual examination, as modified by 10 CFR 50.55a(b)(2)(ix)(G).
6	Item Number E4.11 requires VT-1 visual examination in lieu of Detailed Visual examination, as modified by 10 CFR 50.55a(b)(2)(ix)(G).
7	The ultrasonic examination acceptance standard specified in Paragraph IWE-3511.3 for CISI Class MC pressure-retaining components must also be applied to metallic liners of CISI Class CC pressure-retaining components, as modified by 10 CFR 50.55a(b)(2)(ix)(I).
8	Per 10 CFR 50.55a(b)(2)(xxi)(B), *Table IWB-2500-1 examination requirements*, the provisions of Table IWB-2500-1, Examination Category B-G-2, Item Number B7.80, that are in the 1995 Edition are applicable only to reused bolting when using the 1997 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.
9	Per 10 CFR 50.55a(b)(2)(xxi)(A), Table IWB-2500-1 examination requirements, the provisions of Table IWB-2500-1, Examination Category B-D, Items B3.120 and B3.140 in the 1998 Edition must be applied when using the 1999 Addenda through the latest Edition and Addenda, and requires that a visual examination with magnification may be performed on the inside radius section in lieu of an ultrasonic examination.
10	Examination Category B-G-1, Item Numbers B6.20 "Closure Studs, In Place" and B6.30 "Closure Studs, When Removed" have been combined into and renamed as Item Number B6.20 "Closure Studs," in Table IWB-2500-1 of ASME Section XI, 2004 Edition, No Addenda.
11	As allowed by ASME Code Case N-613-1, TMI will perform a volumetric examination using a reduced examination volume (A-B-C-D-E-F-G-H) of Figures 1, 2, and 3 of the Code Case in lieu of the previous examination volumes of ASME Section XI, Figures IWB-2500-7(a), (b), and (c).

TABLE 7.1-2
INSERVICE INSPECTION SUMMARY TABLE PROGRAM NOTES

Note #	Note Summary
12	As allowed by ASME Code Case N-648-1, TMI will perform a visual examination with enhanced magnification (EVT-1) in lieu of a volumetric examination in ASME Section XI.
13	As allowed by ASME Code Case N-706-1, TMI will perform a VT-2 visual examination each period in lieu of the volumetric and/or volumetric and surface examinations of ASME Section XI. Note that the alternative requirements detailed in Table 1 of the Code Case apply only to the letdown cooler components.
14	There are two (2) PORV valves installed on a rotating basis at TMI; therefore, one of the PORV valves is examined each ten (10) years.
15	TMI Technical Specifications, Section 4.2.4, requires that each Reactor Coolant Pump Motor Flywheel be inspected per the recommendations of USNRC Regulatory Guide 1.14, Reactor Coolant Pump Flywheel Integrity.
16	TMI Technical Specifications, Section 4.15, requires that welds be ultrasonically examined at 3-1/2 year intervals. Technical Specification, Section 4.15.1 (Basis), states that these examinations have revealed no degradation of the welds has occurred during the inspection cycles up to and including the Outage T1R15 inspection.
17	Augmented examinations of Alloy 600 material installed at TMI is controlled via the Exelon Alloy 600 Management Plan ER-AP-330-1001. The examinations are based upon MRP-126, MRP-139, MRP-139 interim guidance, USNRC Bulletin 2004-01, USNRC Bulletin 2003-02, and Code Cases N-722 and N-729-1 as modified by 10 CFR 50.55a(g)(6)(ii)(E) and 10 CFR 50.55a(g)(6)(ii)(D) respectively.
18	Examination Category B-O (Pressure-Retaining Welds In Control Rod Housings), Item Number B14.20 (Welds in CRD Housing) - the scope of examination is for pressure retaining welds in 10% of the peripheral CRD Housings. A total of 24 out of the 69 CRD Housings are classified as peripheral components. TMI is required to select the welds on 3 Peripheral CRD Housings (four welds per housing) to be examined during the interval (10% of 24).
19	The RPV interior requires examination per ASME Section XI requirements for Examination Categories B-N-1, B-N-2, and B-N-3. In addition to the ASME Section XI requirements, the TMI PWR Internals Program for the inspection, repair, replacement, degradation evaluation, and mitigation of the PWR Reactor Internals will ensure that Materials Reliability Program (MRP) and PWR Owners' Group (PWROG) Guidelines are met. Augmented requirements associated with the TMI PWR Internals Program are maintained and controlled in procedures ER-AP-333, "Pressurized Water Reactor Internals Management Program" and ER-AP-333-1001, "Pressurized Water Reactor Internals Program".

8.0 RELIEF REQUESTS FROM ASME SECTION XI

This section contains relief requests written per 10 CFR 50.55a(a)(3)(i) for situations where alternatives to ASME Section XI requirements provide an acceptable level of quality and safety; per 10 CFR 50.55a(a)(3)(ii) for situations where compliance with ASME Section XI requirements results in a hardship or an unusual difficulty without a compensating increase in the level of quality and safety; and per 10 CFR 50.55a(g)(5)(iii) for situations where ASME Section XI requirements are considered impractical.

The following USNRC guidance was utilized to determine the correct 10 CFR 50.55a paragraph citing for TMI relief requests. 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii) provide alternatives to the requirements of ASME Section XI, while 10 CFR 50.55a(g)(5)(iii) recognizes situational impracticalities.

<u>10 CFR 50.55a(a)(3)(i):</u> Cited in relief requests when alternatives to the ASME Section XI requirements which provide an acceptable level of quality and safety are proposed. Examples are relief requests which propose alternative NDE methods and/or examination frequency.

<u>10 CFR 50.55a(a)(3)(ii):</u> Cited in relief requests when compliance with the ASME Section XI requirements is deemed to be a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardship and/or unusual difficulty include, but are not limited to, excessive radiation exposure, disassembly of components solely to provide access for examinations, and development of sophisticated tooling that would result in only minimal increases in examination coverage.

<u>10 CFR 50.55a(g)(5)(iii):</u> Cited in relief requests when conformance with ASME Section XI requirements is deemed impractical. Examples of impractical requirements are situations where the component would have to be redesigned, or replaced to enable the required inspection to be performed.

An index for TMI relief requests is included in Table 8.0-1. The relief requests are applicable to ISI, CISI, SPT, and PDI.

The following relief requests are subject to change throughout the inspection interval (e.g., USNRC approval, withdrawal). Changes to USNRC approved alternatives (other than withdrawal) require USNRC approval.

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[3]	Status[2]	(Program) Description/ Approval Summary[1]
I4R-01	0 03/11/11	Withdrawn	(ISI) Expanded Applicability for use of ASME Code Case N-513-2, Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping. **Withdrawn.**
I4R-02	0 03/11/11	Submitted	(ISI) Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds. **Revision 0 Submitted.**
I4R-03	0 03/11/11	Submitted	(SPT) Pressure Testing the Reactor Pressure Vessel Head Flange Connection Lines. **Revision 0 Submitted.**
I4R-04	0 03/11/11	Submitted	(ISI) ISI Snubbers Included in the Technical Specifications Snubber Visual Examination and Functional Testing Program. **Revision 0 Submitted.**
I4R-05	0 03/11/11	Submitted	(ISI) Qualification Requirements of ASME Section XI Appendix VIII, Supplement 11 for Examination of Structural Weld Overlays (SWOLs). **Revision 0 Submitted.**
I4R-06	0 03/11/11	Submitted	Applicability of ASME Code Case N-649, Alternative Requirements for IWE-5240 Visual Examination. **Revision 0 Submitted.**
RR-09-01	0 03/11/11	Authorized	(ISI) Deferral of Reactor Vessel Weld Examinations in Accordance with 10 CFR 50.55a(a)(3)(i). **Relief Request (Third ISI Interval Relief Request RR-09-01) for the deferral of Reactor Vessel weld volumetric examinations was authorized by the USNRC per SER dated 09/21/10. The deferral of these examinations applies until the Fall 2015 refueling outage T1R21.**

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[3]	Status[2]	(Program) Description/ Approval Summary[1]
RR-09-02	0 03/11/11	Authorized	(ISI) Deferral of Reactor Vessel Internal Visual Examinations (Examination Categories B-N-2 and B-N-3). **Relief Request (Third ISI Interval Relief Request RR-09-02) for the deferral of Reactor Vessel internal visual examinations was authorized by the USNRC per SER dated 09/21/10. The deferral of these examinations applies until the Fall 2015 refueling outage T1R21.**
RR-10-02	0 03/11/11	Submitted	(ISI) Alternative Requirements of Full Structural Weld Overlays (SWOLs) of Pressurizer Spray Dissimilar Metal Welds. **Relief Request RR-10-02 was submitted to the USNRC per letter dated 09/30/10.**

Note 1: The USNRC grants relief requests pursuant to 10 CFR 50.55a(g)(6)(i) when Code requirements cannot be met and proposed alternatives do not meet the criteria of 10 CFR 50.55(a)(3). The USNRC authorizes relief requests pursuant to 10 CFR 50.55a(a)(3)(i) if the proposed alternatives would provide an acceptable level of quality and safety or under 10 CFR 50.55a(3)(ii) if compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of safety.

Note 2: This column represents the status of the latest revision. Relief Request Status Options: Authorized - Approved for use in an USNRC SER (See Note 1); Granted - Approved for use in an USNRC SER (See Note 1); Authorized Conditionally - Approved for use in an USNRC SER which imposes certain conditions; Denied - Use denied in an USNRC SER; Expired - Approval for relief has expired; Withdrawn - Relief has been withdrawn by TMI; Not Required - The USNRC has deemed the relief unnecessary in an SER or RAI; Cancelled - Relief has been cancelled by TMI prior to issue; Submitted - Relief has been submitted to the USNRC by the station and is awaiting approval; Pending - Relief has been awaiting station and Corporate review and submittal to the USNRC.

Note 3: The revision listed is the latest revision of the subject relief request. The date this revision became effective is the date of the approving SER which is listed in the fourth column of the table. The date noted in the second column is the date of the ISI Program Plan revision when the relief request was incorporated into the document.

10 CFR 50.55a RELIEF REQUEST: I4R-01
Revision 0

Expanded Applicability for use of ASME Code Case N-513-2, Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	2 and 3
Reference:	IWA-2441(b) and ASME Code Case N-513-2
Examination Category:	NA
Item Number:	NA
Description:	Expanded Applicability for use of ASME Code Case N-513-2, Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping
Component Number:	Moderate Energy Class 2 and 3 Piping

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The Inservice Inspection (ISI) program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda.

3.0 APPLICABLE CODE REQUIREMENT:

IWA-2441(b) requires Code Cases be applicable to the Edition and Addenda specified in the Inspection Plan.

ASME Code Case N-513-2, Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping, provides requirements that may be used to accept flaws without performing a repair/replacement activity for a limited time period.

4.0 REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

On April 20, 2011 TMI, Unit 1 will start its fourth ten-year interval ISI program under the requirements of the 2004 Edition, No Addenda of ASME Section XI. When implementing this edition of ASME Section XI, Paragraph IWA-2441(b) requires code cases be applicable to the Edition and Addenda specified in the Inspection Plan.

10 CFR 50.55a RELIEF REQUEST: I4R-01
Revision 0

ASME Code Case N-513-2 has an applicability limited up to the 2001 Edition with the 2003 Addenda, which is identified in Section 7.0 of the code case and in the latest applicability index for Section XI Code Cases. Since ASME Code Case N-513-2 only applies up to the 2001 Edition with the 2003 Addenda, Paragraph IWA-2441(b) does not allow the use of ASME Code Case N-513-2 for the TMI, Unit 1 fourth ten-year interval ISI program.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

TMI requests the applicability of ASME Code Case N-513-2 be extended to the 2004 Edition, No Addenda for use in the plant's fourth interval ISI program. The USNRC has accepted the use of ASME Code Case N-513-2 as an acceptable method for evaluating the structural integrity of flaws identified in moderate energy piping in the latest revision of Regulatory Guide 1.147, Revision 15.

No technical changes to ASME Code Case N-513-2 are being proposed in this relief request. This relief request is being submitted to correct a timing situation, which has resulted from the application of the 2004 Edition, No Addenda of ASME Section XI for TMI, Unit 1. Since no technical change is proposed in this relief request, TMI, Unit 1 considers that this alternative provides an acceptable level of quality and safety, and is consistent with provisions of 10 CFR 50.55a(a)(3)(i).

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the fourth ten-year ISI interval for TMI, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been requested for:

Letter from J. Price (Dominion Nuclear Connecticut, Inc.) to U.S. Nuclear Regulatory Commission, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3 Alternative Request IR-3-12 for the Use of ASME Code Case N-513-2," dated May 28, 2009.

Letter from J. Price (Dominion Nuclear Connecticut, Inc.) to U.S. Nuclear Regulatory Commission, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2 Relief Requests RR-04-02, Alternative VT-2 Pressure Testing Requirements for the Lower Portion of the Reactor Pressure Vessel, and RR-04-03, Alternative Evaluation Criteria for Code Case N-513-2, Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping," dated March 30, 2010.

10 CFR 50.55a RELIEF REQUEST: I4R-02
Revision 0

Request for Relief for Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1 and 2
Reference:	Table IWB-2500-1, Table IWC-2500-1
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.40, B5.50, B9.11, B9.21, B9.22, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, C5.70, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Pressure Retaining Piping

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The ISI program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda.

3.0 APPLICABLE CODE REQUIREMENT:

Table IWB-2500-1, Examination Category B-F, requires volumetric and surface examinations on welds for Item Numbers B5.10 and B5.40, and surface examinations for welds for Item Number B5.50.

Table IWB-2500-1, Examination Category B-J, requires volumetric and surface examinations on a sample of welds for Item Numbers B9.11 and B9.31, volumetric examinations on a sample of welds for Item Number B9.22, and surface examinations on a sample of welds for Item Numbers B9.21, B9.32, and B9.40. The weld population selected for inspection includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.

2. All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

10 CFR 50.55a RELIEF REQUEST: I4R-02
Revision 0

a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.

b. cumulative usage factor *U* of 0.4.

3. All dissimilar metal welds not covered under Examination Category B-F.

4. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25% of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds exempted by Paragraph IWB-1220 or welds in Item Number B9.22.

5. A 10% sample of PWR high pressure safety injection system circumferential welds in piping ≥ NPS 1½ and < NPS 4 shall be selected for examination. This sample shall be selected from locations determined by the Owner as most likely to be subject to thermal fatigue.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations on a sample of welds for Item Numbers C5.11, C5.21, and C5.51, and surface examinations on a sample of welds for Item Numbers C5.30, C5.41, C5.70, and C5.81. The weld population selected for inspection includes the following:

1. Welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or of all carbon and low alloy steel welds (Examination Category C-F-2) not exempted by Paragraph IWC-1220. (Some welds not exempted by Paragraph IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

   a. the examinations shall be distributed among the ISI Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or carbon and low alloy welds (Examination Category C-F-2) in each system;

   b. within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and

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c. within each system, examinations shall be distributed between piping sizes prorated to the degree practicable.

4.0 REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative utilizing Reference 1 along with two enhancements from Reference 4 will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (Reference 2):

> "The staff concludes that the proposed RI-ISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."

The initial TMI, Unit 1 Risk-Informed Inservice Inspection (RISI) Program was submitted at the beginning of the second period of the third inspection interval. This initial RISI Program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by ASME Code Case N-578-1. The program was approved for use by the USNRC via a Safety Evaluation as transmitted to Exelon (Reference 5).

The transition from the 1995 Edition, 1996 Addenda to the 2004 Edition, No Addenda of ASME Section XI for the TMI, Unit 1 fourth inspection interval does not impact the currently approved RISI evaluation methods and process used in the third inspection interval, and the requirements of the new Code Edition/Addenda will be implemented as detailed in the TMI ISI Program Plan. Therefore, with the exception of specific weld locations that may have changed due to maintenance or modification activities (e.g., steam generator replacement) and the addition of an Alloy 600 Augmented Examination Program, the proposed alternative RISI Program for the fourth inspection interval is the same program methodology as approved in Reference 5 for the third inspection interval.

Locations potentially susceptible to Primary Water Stress Corrosion Cracking (PWSCC), whether repaired, mitigated, or unmitigated, that fall within the TMI Alloy 600 Augmented Examination Program (MRP-139, ASME Code Case N-722, ASME Section XI Appendix Q, etc.) under which they are tracked, categorized, and subject to augmented

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examination selection criteria and requirements, are also evaluated under the RISI Program for appropriate degradation mechanism (DM) assignment. Where PWSCC is the only DM assigned, the locations are removed from the RISI scope and deferred to the Alloy 600 Augmented Examination Program for examinations. These PWSCC examination programs are based upon criteria that is generally more stringent than the RISI Program because they are generally based upon deterministic criteria that accounts for flaw growth rates and time to structural instability. For locations other than those which have full structural weld overlay applied where the degradation mechanism assessment identifies PWSCC and another DM under the RISI Program, the elements remain in the RISI Program and are subject to the normal RISI element selection process solely for the additional DMs assigned. Locations with full structural weld overlay will be removed from the RISI Program and treated under the Alloy 600 augmented examination program.

The Risk Impact Assessment completed as part of the original baseline RISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the fourth interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RISI methodology and evaluation will be maintained for the new interval. The original methodology of the evaluation has not changed, and the change in risk was simply re-assessed using the initial 1995 Edition, 1996 Addenda ASME Section XI program prior to RISI and the new element selection for the fourth interval RISI program. This same process has been maintained in each revision to the TMI, Unit 1 RISI assessment that has been performed to date.

The actual "evaluation and ranking" procedure including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 5) RISI Program remain unchanged and are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These portions of the RISI Program have been and will continue to be reevaluated and revised as major revisions of the site Probabilistic Risk Assessment (PRA) occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, Element Selection, and Risk Impact Assessment steps encompass the complete living program process applied under the TMI, Unit 1 RISI Program.

5.0 <u>PROPOSED ALTERNATIVE AND BASIS FOR USE:</u>

The proposed alternative originally implemented in RISI inspection plan for TMI, Unit 1 (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This original program along

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with these same two enhancements is currently approved for the TMI, Unit 1 third inspection interval as documented in Reference 5.

The fourth interval RISI Program will be a continuation of the current application and will continue to be a living program as described in the "Reason For Request" section of this relief request. No changes to the evaluation methodology as currently implemented under EPRI TR-112657, Revision B-A, are required as part of this interval update. The following two enhancements will continue to be implemented.

a. In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, TMI will utilize the requirements of Paragraph -2430, "Additional Examinations" contained in ASME Code Case N-578-1 (Reference 4). The alternative criteria for additional examinations contained in ASME Code Case N-578-1 provide a more refined methodology for implementing necessary additional examinations. The reason for this selection is that the guidance discussed in EPRI TR-112657 includes requirements for additional examinations at a high level, based on service conditions, degradation mechanisms, and the performance of evaluations to determine the scope of additional examinations, whereas ASME Code Case N-578-1 provides more specific and clearer guidance regarding the requirements for additional examinations that is structured similar to the guidance provided in ASME Section XI, Paragraphs IWB-2430 and IWC-2430. Additionally, similar to the current requirements of ASME Section XI, TMI intends to perform additional examinations that are required due to the identification of flaws or relevant conditions exceeding the acceptance standards, during the outage the flaws are identified.

b. To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods" of EPRI TR-112657, TMI, Unit 1 will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" contained in ASME Code Case N-578-1 (Reference 4). To implement Note 10 of this table, paragraphs and figures from the 2004 Edition, No Addenda of ASME Section XI (TMI, Unit 1 code of record for the fourth interval) will be utilized which parallel those referenced in the Code Case. Table 1 of ASME Code Case N-578-1 will be used as it provides a detailed breakdown for Examination Method and Categorization of Parts to be Examined. Based on these Methods and Categorization, the examination figures specified in Section 4 of EPRI TR-112657 will then be used to determine the examination volume/area based on the degradation mechanism and component configuration. For elements not subject to a degradation mechanism, Note 1 to Table 1 of ASME Code Case N-578-1 will be applied using

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the expanded volume. Also, as discussed in the "Reason For Request" section of this relief request, elements potentially subject to PWSCC and no other degradation mechanism will be removed from the RISI population considered during the RISI element selection process and will be treated under the TMI Alloy 600 Augmented Examination Program, and all elements with full structural weld overlay applied will be removed from the RISI Program completely and examined under the Alloy 600 augmented program.

Piping examinations under this augmented program are currently performed in accordance with the criteria below. This program is subject to change and will be maintained in accordance with the latest regulations relative to PWSCC and the Alloy 600 Augmented Examination Program. For elements evaluated under RISI to only be subject to the PWSCC degradation mechanism and all elements with full structural weld overlay applied, the requirements incorporated in the augmented examination program are:

1. Bare metal visual examinations are currently performed in accordance with 10 CFR 50.55a(g)(6)(ii)(E).
2. Ultrasonic examinations are currently performed in accordance with MRP-139, Revision 1, "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline" (Reference 8).
3. Ultrasonic examination of completed weld overlay repaired welds are currently performed in accordance with ASME Section XI Non-Mandatory Appendix Q, Weld Overlay Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Weldments, Paragraph Q-4300.

The TMI RISI Program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, and 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by ASME Code Case N-578-1.

For U.S. Nuclear Regulatory Commission staff consideration in the evaluation of this alternative Risk-Informed ISI Program, Appendix 1 to this relief request contains a summary of the Regulatory Guide 1.200, Revision 2 (Reference 6), evaluation performed on TMI-PRA-014, Quantification Notebook (Reference 7), and the impact of the identified gaps on the technical adequacy of the TMI PRA Model to support this RISI application.

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In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to receive Code-required pressure testing as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the TMI, Unit 1 pressure testing program, which remains unaffected by the RISI Program.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the fourth ten-year inspection interval for TMI, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

The TMI, Unit 1 third ISI interval Relief Request RR-21 was authorized per U.S. Nuclear Regulatory Commission (USNRC) Safety Evaluation Report (SER) dated November 7, 2003. The fourth ISI interval Relief Request utilizes the RISI methodology as was previously approved.

The Braidwood Station, Units 1 and 2 third ISI interval Relief Request I3R-01 was authorized per USNRC SER dated November 5, 2009.

The Peach Bottom Atomic Power Station, Units 2 and 3 fourth ISI interval Relief Request I4R-44 was authorized per USNRC SER dated February 26, 2009.

8.0 REFERENCES:

1. Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999.

2. Letter from W. Bateman (USNRC) to G. Vine (EPRI), "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," dated October 28, 1999.

3. Letter from M. Gallagher (AmerGen Energy Company, LLC) to U.S. Nuclear Regulatory Commission, "Third Ten-Year Interval Inservice Inspection (ISI) Program Risk-Informed Inservice Inspection Program Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds," dated October 1, 2002.

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4. American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1."

5. Letter from R. Laufer (USNRC) to J. Skolds (AmerGen Energy Company, LLC), "Three Mile Island Nuclear Station, Unit 1, RE: Third 10-Year Interval Inservice Inspection (ISI) Program Requests for Relief (TAC NOS. MB6498 and MB6499)," dated November 7, 2003.

6. Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009.

7. TMI-PRA-014, Quantification Notebook, Revision 3, June 2009, Model 2009 (TM1080).

8. MRP-139, Revision 1, "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline."

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PRA Technical Adequacy

Exelon Generation Company, LLC (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Three Mile Island (TMI) PRA.

1.0 PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.

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- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.

- Guidelines for updating the full power, internal events PRA models for EGC nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

2.0 PRA Self Assessment and Peer Review

The TM1080 version of the TMI PRA model is the most recent evaluation of the risk profile at TMI for internal event challenges, including internal flooding. The TMI PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the TMI PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

2.1 Self Assessment and Peer Review Results

Several assessments of technical capability have been made for the TMI PRA model. These assessments are as follows and further discussed in the paragraphs below.

- An independent PRA peer review was conducted under the auspices of the B&W Owners Group in 2000, following the Industry PRA Peer Review process [3]. This peer review included an assessment of the PRA model maintenance and update process.

- A limited scope gap assessment was performed in 2005 to support the Mitigating Systems Performance Indicator (MSPI) implementation. Additionally, the TMI PRA model results were evaluated in the B&W Owners Group PRA cross-comparisons study performed in support of implementation of the MSPI process.

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- A RG-1.200 Peer Review was conducted in October 2008 against the ASME PRA Standard, Addenda RA-Sb-2005 and RA-Sc-2007 [4]. The DA and IF elements (Data and Internal Flooding) were not reviewed at this time.

- A focused-scope RG-1.200 Peer Review was conducted in April 2010, for the Data (DA) element, against the ASME PRA Standard, Addenda RA-Sb-2005 and RA-Sc-2007.

A summary of the disposition of the PRA Peer Review facts and observations (F&Os) for the TMI PRA model was documented as part of the statement of PRA capability for MSPI in the TMI MSPI Basis Document [5]. As noted in that document, the one significance level A F&O and all but one significance level B F&Os from that peer review have been addressed and closed out as of the TMI 2004 Revision 1 PRA model. The remaining issue was resolved in the TMI 2004 Revision 2 PRA model (TMI1080).

As indicated above, a PRA model update was completed in 2009, resulting in the TM1080 updated model. In updating the PRA, changes were made to the PRA model to address most of the identified gaps from the 2008 peer review, as well as to address other open Updating Requirement Evaluations (UREs). Open findings from the peer review are summarized in Table 2-1.

The 2008 peer review did not cover the DA or IF elements. For DA, a separate focused-scope peer review was performed in April 2010. The preliminary results from that review and the DA element self-assessment are provided in Table 2-2.

For Internal Flooding, all the B F&Os associated with the IF technical element from the 2000 peer review have been dispositioned. A self-assessment against the Standard was performed for IF; the results are provided in Table 2-3.

All remaining gaps will be reviewed for consideration for the next periodic PRA model update, but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

2.2 Sensitivity Studies

Three gaps were identified in Tables 2-1 and 2-3 (QU-C1-01, LE-D1b-01 and IFEV-A5-01) which warranted sensitivity studies. QU-C1-01 was evaluated separately. Gaps LE-D1b-01 and IFEV-A5-01 were evaluated simultaneously using one sensitivity study.

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QU-C1-01

A review was performed of the CDF and LERF cutsets containing the recovery: REO - Operator Manually Actuates HPI Components if ESAS Fails. No cutsets contained any other failed operator actions, therefore HEP dependence is not an issue. As a bounding sensitivity, REO was set to 1.0. This resulted in no changes to any Risk-Informed Inservice Inspection (RISI) evaluations; the only calculations impacted were CCDP and CLERP for initiators %SBL, %TRIA, %TRIB, and %VSB, which are already ranked as High consequence.

LE-D1b-01 and IFEV-A5-01

Gaps LE-D1b-01 and IFEV-A5-01 were evaluated simultaneously in one sensitivity.

> LE-D1b-01: The concern is that the mechanical and/or electrical penetrations have a lower strength than assumed in the LERF analysis. It was determined that if the strength of these penetrations were indeed lower, it would impact early containment failure due to overpressurization. One split fraction in the LERF model, NOAFTSTREN2 (Probability of Containment Failing Due to Combustible Gas Burn with Low Base Pressure), has a low value (1E-3) that was adjusted to 1E-2. This value is a factor of 10 higher and consistent with generic values provided in NUREG/CR-6595 [8].

> IFEV-A5-01: It is expected that new pipe failure rates are lower than those used in the current TMI Internal Flood model and plant specific experience has been nominal. Human-induced flooding, on the other hand, would increase the flood initiating event frequencies. As a sensitivity, all flood initiating event frequencies were increased by a factor of 10.

The results of the sensitivity calculations showed that three runs changed risk rank (RI-D7, RI-D9 and RI-D13) and went from Medium to High consequence rank. However, most consequence determinations for a pipe break are composed of several possible scenarios, and the final set of consequence results is unaffected by these PRA changes. All 100 evaluations maintained the same risk rank as the base analysis.

3.0 General Conclusion Regarding PRA Capability

The TMI PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

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4.0 Assessment of PRA Capability Needed for Risk Informed Inservice Inspection

In the RISI program at TMI, the EPRI Risk-Informed ISI methodology [1] is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RISI plan development to support the consequence assessment, risk ranking, element selection and risk impact steps.

The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three fundamental components of the EPRI methodology.

First, PRA consequence results are binned into one of three conditional core damage probability (CCDP) and conditional large early release probability (CLERP) ranges before any welds are chosen for RISI inspection as illustrated below. Broad ranges are used to define these bins so that the impact of uncertainty is minimized and only substantial PRA changes would be expected to have an impact on the consequence ranking results.

Consequence Results Binning Groups		
Consequence Category	**CCDP Range**	**CLERP Range**
High	CCDP > 1E-4	CLERP > 1E-5
Medium	1E-6 < CCDP ≤ 1E-4	1E-7 < CLERP ≤ 1E-5
Low	CCDP ≤ 1E-6	CLERP ≤ 1E-7

The risk importance of a weld is therefore not tied directly to a specific PRA result. Instead, it depends only on the range in which the PRA result falls. As a consequence, any PRA modeling uncertainties would be mitigated by the wide binning provided in the methodology. Additionally, conservatism in the binning process (e.g., as would typically be introduced through PRA attributes meeting ASME PRA Standard Capability Category I versus II) will tend to result in a larger inspection population.

Secondly, the impacts of particular PRA consequence results are further dampened by the joint consideration of the weld failure potential via a non-PRA-dependent damage mechanism assessment. The results of the consequence assessment and the damage mechanism assessment are combined to determine the risk ranking of each pipe segment (and ultimately each element) according to the EPRI Risk Matrix. The Risk Matrix, which equally takes both assessments into consideration, is reproduced below.

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POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	CONSEQUENCES OF PIPE RUPTURE IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	HIGH Category 3	HIGH Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	HIGH Category 2
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

Thirdly, the EPRI RISI methodology uses an absolute risk ranking approach. As such, conservatism in either the consequence assessment or the failure potential assessment will result in a larger inspection population rather than masking other important components. That is, providing more realism into the PRA model (e.g., by meeting higher capability categories) most likely would result in a smaller inspection population.

These three facets of the methodology reduce the importance and influence of PRA on the final list of candidate welds.

The limited manner of PRA involvement in the RISI process is also reflected in the Risk-Informed license application guidance provided in Regulatory Guide 1.174 [7]. Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*

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An example is risk-informed inservice inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.

In addition to the above, it is noted that welds determined to be low risk significant are not eliminated from the ISI program on the basis of risk information. For example, the risk significance of a weld may fall from Medium Risk Ranking to Low Risk Ranking, resulting in it not being a candidate for inspection. However, it remains in the program, and if, in the future, the assessment of its ranking changes (either by damage mechanism or PRA consequence risk) then it may again become a candidate for inspection. If it is discovered during the RISI update process that a weld is now susceptible to flow-accelerated corrosion (FAC), intergranular stress corrosion cracking (IGSCC), or microbiological induced cracking (MIC) in the absence of any other damage mechanism, then it is addressed in an "augmented" program where it is monitored for those special damage mechanisms. That occurs no matter what the Risk Ranking of the weld is determined to be.

5.0 Conclusion Regarding PRA Capability for Risk-Informed ISI

The TMI PRA model continues to be suitable for use in the Risk-Informed Inservice Inspection application. This conclusion is based on:

- the PRA maintenance and update processes in place,
- the PRA technical capability evaluations that have been performed and are being planned, and
- the RISI process considerations, as noted above, that demonstrate the relatively limited sensitivity of the EPRI RISI process to PRA attribute capability beyond ASME PRA Standard Capability Category I.

In support of the PRA analyses for the TMI-1 Ten-Year Interval evaluations using the TM1080 PRA model, the remaining gaps to the PRA standard have been reviewed to determine which, if any, would merit RISI-specific sensitivity studies in the presentation of the application results. The result of this assessment concluded that 3 gaps required sensitivity studies, all of which showed no change in the conclusions of the RISI evaluation (see Section 2.2 above).

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6.0 References

[1] *Revised Risk-Informed Inservice Inspection Evaluation Procedure,* EPRI TR-112657, Revision B-A, December 1999.

[2] Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,* Revision 2, March 2009.

[3] Framatome Technologies, Inc., *PSA Peer Review Certification Process: PSA Self-Assessment Process,* 47-5005658-00, September 1999.

[4] American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,* (ASME RA-S-2002), Addenda RA-Sb-2005, and Addenda RA-Sc-2007, August 2007.

[5] TMI MSPI Basis Document, TMI-2006-004 Rev. 2, September 2009.

[6] ASME Committee on Nuclear Risk Management in collaboration with ANS Risk Informed Standards Committee, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,* ASME/ANS RA-Sa-2009, March 2009.

[7] U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, Revision 1, November 2002.

[8] NUREG/CR-6595, An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, Revision 1, October 2004.

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Table 2-1
Open Peer Review Findings

Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
IE-A4-01	The list of systems examined seems to be generated from a high level PRA system significance standpoint and does not seem to provide a complete list of all plant systems.	IE-A5	Open	Table 5 in the Initiating Event Notebook (TMI-PRA-002, Rev.1) shows the results of a systematic review of all the systems in the PRA. The impact of failure of systems not modeled in the PRA that cause an IE are subsumed in other events (e.g., reactor trip, Loss of offsite power, LOCA, etc.). However, this is not explicitly documented in the IE Notebook. Therefore, this is considered a documentation issue not affecting the technical adequacy of the PRA model.
IE-A4a-01	For the systematic evaluation required in IE-A4, the examination of potential initiating events resulting from common cause failures is not documented.	IE-A6	Open	The potential for common cause failures (CCF), including CCFs from routine system alignments that could result from preventive and corrective maintenance, was included in the systematic evaluation for potential initiating events. This is a documentation issue not affecting the technical adequacy of the PRA model.
IE-A6-01	No documentation was found of interviews with plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	IE-A8	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
IE-C10-01	No comparison of the initiating event fault tree results with generic data has been identified. Compare plant initiating event fault tree results to generic frequency sources (i.e., NUREG/CR-5750, NUREG/CR-6928, WOG PSA Database, etc.) and explain differences.	IE-C12	Open	The initiating event fault tree results were compared to generic industry frequencies and with the PWROG database. However, the results of the review are not documented; Therefore, this is a documentation issue not affecting the technical adequacy of the PRA model.

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Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
IE-D1-01	The initiating event analysis has not been documented in a manner that facilitates PRA applications, upgrades, and peer review. The IE analysis is very difficult to trace and relies heavily on the ABS 2003 documentation, without proper reference in the IE notebook.	IE-D1	Open	The IE notebook was updated, although several documentation-related IE F&Os remain open. Therefore, this F&O remains open, but it is a documentation issue not affecting the technical adequacy of the PRA model.
AS-C1-01	Much of the AS-related documentation is located in the ABS 2003 report, with updates identified in the Event Tree notebook. In many cases, bases could not be verified without support of the TMI PRA personnel to aid in tracking down the documentation. To facilitate reviews, upgrades, etc., it is necessary to either include all the documentation in the event tree notebook or to reference the material in other documents.	AS-C1	Open	The Event Tree Notebook was updated, although several documentation-related AS F&Os remain open. Therefore, this F&O remains open, but it is a documentation issue not affecting the technical adequacy of the PRA model.
AS-C2-01	The process used to develop the accident sequences is not provided. Incorporation of plant specific information is therefore not demonstrated. Provide the process description and include the discussion of use of procedures, etc.	AS-C2, AS-A4, AS-A5	Open	The process for developing accident sequences used plant-specific information such as procedures. This is a documentation issue not affecting the technical adequacy of the PRA model.
SC-B2-01	Tables 3-1 through 3-8 of TMI PRA-003 include several instances of use of "Judgment" as the basis for success criteria. These applications of judgment do not use section 4.3 of the ASME std. to attain CCII and are not discussed in the report as required by SC-C2 to attain CCI. Do not use judgment as basis for success criteria or apply para. 4.3 of the ASME standards when implementing expert judgment.	SC-B2	Open	Expert judgment was NOT used in determining the success criteria. Therefore, this is a documentation issue not affecting the technical adequacy of the PRA model.

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SC-B4-01	Many of the T/H success criteria were developed using the MAAP computer code, including large break LOCAs greater than 10" diameter (see Table 3-3 of Success Criteria notebook). However, FAI has identified a limitation/precaution using MAAP for the large break LOCA analyses. "...the results of the code should not be used for a definitive determination of the primary system pressure response, mass and energy releases, and peak cladding temperatures during this time frame." Do not use MAAP to develop large LOCA success criteria due to limitations associated with the code.	SC-B4	Open	The documentation is misleading. MAAP was not used to develop the success criteria for Large LOCAs. Therefore, this is a documentation issue not affecting the technical adequacy of the PRA model.
SC-B5-01	No documentation of a check for the reasonableness and acceptability of the results (i.e., comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features). Compare TMI results with results of the same analyses performed for similar plants, accounting for differences in unique plant features	SC-B5	Open	Reasonableness and acceptability of the results were checked; this is a documentation issue not affecting the technical adequacy of the PRA model.
SC-C1-01	Documentation does not facilitate PRA application, upgrades, or peer review. Though it appears the information exists, one must piece together information in multiple notebooks and calculations with no correlation reference to understand how success criteria were evaluated or developed in the model. In application and model upgrade one could easily make an error due to the disconnected nature of the documentation.	SC-C1	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.

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SC-C2-01	There is an implied process in the latest Success Criteria Notebook, but not a clear process for evaluating and documenting success criteria, this can be easily related to the other SC-C criteria not being met. Documentation of core damage could be clarified. Though calculations and other references are used to develop success criteria they are not easily found in the documentation. Computer codes are identified in some cases, however there is no description of limitations or potential conservatisms. The use of expert judgment is used without rational or basis. There is in many cases no basis for the time given for human actions such as operator interviews or simulator runs or MAAP analysis. There is no summery of success criteria for mitigating systems and HEP's used,	SC-C2	Open	The HRA and its notebook were revised to include documented bases for times available to perform operator actions and times needed to perform the actions. Expert judgment was NOT used in the development of success criteria. Although not incorporated in the Success Criteria NB, the following applies to computer codes used for success criteria: *For success criteria that were developed for the PRA, instead of using design basis success criteria, MAAP 4 is used. The overall conclusion from the EPRI MAAP Thermal-Hydraulic Qualification Studies was that MAAP had a wide range of applicability; however, a few limitations were identified. The current position on MAAP code limitations can be found on the MAAP4 web site. The significant limitation of MAAP for PWRs is Large LOCA behavior prior to reflood. The TMI PRA uses design basis criteria for Large LOCAs, so this limitation of MAAP 4 has been addressed.* The Success Criteria NB still requires updating, so this F&O remains open. This is a documentation issue not affecting the technical adequacy of the PRA model.
SC-C3-01	Documentation of sources of uncertainty has not been accomplished. This is a recognized/acknowledged gap for the TMI PRA.	SC-C3	Open	To be determined once the new USNRC/EPRI guidance is implemented. However, the EPRI RISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.

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Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
SY-A20-01	In general, the system notebooks do not discuss room cooling. The EFW system considered the impact of a steam line break and the diesel generators are assumed to require the room fan for success. However, other notebooks (e.g., HPI, DHRW/CCW, LPI/DHR) do not mention room cooling. HVAC systems are discussed in Appendix D to the 2003 TMI update, which presents the TMI responses to the 2000 peer review. In that document, the response to F&O DE-2 presents a review of various HVAC systems. Some PRA component areas are excluded with a good basis (e.g., NSCCW pump areas reference results from loss of ventilation tests). However, it is difficult to evaluate each area's HVAC requirements by reading the responses to the F&Os.	SY-A22, SY-B6, SY-B7, AS-B7	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
SY-C1-01	This SR is not met due to SY-C2 and SY-C3 not being met. In general the system notebooks did not supply sufficient information to evaluate SY effectively. Continue developing system documentation as a stand-alone document representing the current model. It is recommended that system notebooks like the Electrical Systems be broken up to discuss specific systems in more detail, for example the Diesel Generators, 4.160Kv, 480VAC, DC etc.	SY-C1, SY-C2, SY-C3, SY-A2	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
SY-C2-01	Documentation of the systems analysis was not sufficient reasonably assess the associated supporting requirements.	SY-C2	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
QU-B5-01	Logic loops have been broken, as none appear in the TMI1042 model. However, no record can be found of how the logic loops were broken. Document how logic loops were identified and broken.	QU-B5	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.

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QU-C1-01	Multiple HFE identification only considers HFE in the quantified model fault tree. Recovery event applied post quantification by the recovery tree were not addressed. Include recovery tree event for dependency identification.	QU-C1, HR-H3	Open	Only three recoveries are used in the TMI PRA. Dependency with other HFEs is considered for two of them, but not the third (REO). A sensitivity analysis was performed as described in Section 2.2. There was no impact on the RISI rankings based on this sensitivity. Therefore, there is no impact on the RISI assessment from this gap.
QU-D3-01	Comparison of the results of the model with other similar plants was not documented. Perform the comparison.	QU-D4	Open	A comparison of the model results with other plants was performed at a high level for other B&W plants and at a more detailed level for ANO-2. This review is not documented. Therefore, this is a documentation issue not affecting the technical adequacy of the PRA model nor this application.
QU-D5-01	Contribution to CDF of SSCs/operator actions are not provided in a manner to distinguish between initiating events vs. event mitigation. Expand the results discussion to include additional discussion of contributors at lower level of resolution and provide the contributions for IEs and for mitigation.	QU-D6	Open	Some SSCs that are significant contributors to initiating events, but not to mitigation, are not explicitly identified in the documentation of significant contributors. Significant contributors to initiating events were identified through a review of support system initiating event cutsets, but the individual contributors and cutsets were omitted from the quantification notebook. It should be noted that initiating event fault trees are re-quantified for any application affecting the components or configurations represented by these fault trees. However, this is a documentation issue not affecting the technical adequacy of the PRA model nor this application.

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QU-E4-01	There is no evidence that an evaluation was performed of the sensitivity of the results to key model uncertainties and key assumptions.	QU-E4	Open	To be determined once the new USNRC/EPRI guidance is implemented. However, the EPRI RISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.
QU-F1-01	The documentation of the quantification included only minimal information. Many of the requirements of the stated in the SRs are not included. (Examples: Reviews are not documented. Contributors are very minimally documented.) The documentation does not meet the minimum requirements of the ASME standard and thus does not facilitate applications/upgrades/reviews. The documentation does not describe the approach for identification and breaking of logic loops in the model. Revise the quantification documentation to include the requirements of the SRs.	QU-F1	Open	The Quantification Notebook has been updated and significantly improved. However, several documentation-related QU F&Os remain open. Therefore, this F&O remains open, but it is a documentation issue not affecting the technical adequacy of the PRA model.
QU-F2-01	The documentation of the quantification is missing significant sections such as reviews, sequence discussions, lower level results, uncertainty analyses. Update the results documentation to include all needed information. Use the SR to provide guidance regarding needed and suggested content.	QU-F2	Open	Many of the sections listed in the F&O have been added to the Quantification Notebook, but not all items in QU-F2 have been documented and this F&O remains open. However, it is a documentation issue not affecting the technical adequacy of the PRA model.
QU-F5-01	In the quantification notebook, other than the LERF truncation limitation, no evaluations of limitations were presented. Explicitly consider limitations of the model as they may apply to applications.	QU-F5, LE-G5	Open	LERF truncation is the only identified limitation to the TMI PRA model for applications. This is a documentation issue not affecting the technical adequacy of the PRA model.

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Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
LE-B1-01	The LERF contributors from Table 4.5.9-3 of the ASME Standard are considered in the TMI Containment Event Tree. Of the items applicable for Large, Dry Containments such as TMI, containment isolation is addressed in CET heading B, ISLOCA, SGTR, and induced SGTR in heading A, and HPME/core debris impingement in heading E. The item "In-vessel recovery" is considered in preventing late containment failures (per TMI-PRA-015.2, page 5-109), but no credit is given (failure event set to 1.0). It is conservative to take no credit for in-vessel recovery; the conservative modeling in the late analysis does not impact LERF, but failure to consider in the early analysis could potentially overstate impact of early containment failure after vessel breach.	LE-B1	Open	LE-B1 does not meet Capability Category II, but is considered adequate for this application. As this addresses a model issue resulting in conservative LERF results, the result of this gap is to bias CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.
LE-C2-01	This F&O applies to several LE SRs that involve reviewing significant LERF sequences for potential credit for equipment repair, additional recovery actions, engineering evaluations, etc. There is no formal record of a review of the LERF results for such items. Document reviews of the significant accident progression sequences that result in a large, early release to determine if repair, additional recoveries, additional engineering evaluations, etc. can be credited. If any credit is given, provide justification for the credit.	LE-C3, LE-C10, LE-C11	Open	LE-C3, LE-C10 and LE-C11 do not meet Capability Category II, but they are treated conservatively. As these address model issues resulting in conservative LERF results, the result of this gap is to bias CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.

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Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
LE-C7-01	System level operator actions are described in the Level 1 System Analysis notebooks. Offsite power recovery data is consistent with the Level 1 analysis. Other human actions in the TMI CET were estimated using qualitative judgment in Table 5-1 of the CET notebook. This qualitative evaluation is acceptable for some uncertain phenomenological issues, but more detailed HRA analyses are needed for actions that can be quantified, as per the requirements of the ASME Standard paragraph 4.5.5. Identify operator actions in the Level 2 for which a more detailed HRA is possible. One example would be comparing the time at which PORVs can be opened to reduce RCS pressure to the time at which an induced SGTR might occur. Consider sensitivity analyses on uncertain parameters.	LE-C7	Open	Conservative screening values are used for the CET HEPs. Therefore, the impact of these HEPs is to bias CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.
LE-C8a-01	Equipment survivability is considered for the containment fans in Section 5 of the CET notebook. For before, soon after, and long after vessel failure containment conditions, the fans are assumed to have a 0% chance of failure due to the accident environment. As a basis, the analysis states that the Oconee fans are expected to remain functional throughout an accident. The Oconee reference is from 1990, and may have been updated since that time. As the fans are important in controlling containment pressure and temperature, which impacts the EARLY evaluation, more detailed justification should be examined to credit their survivability.	LE-C9	Open	The Reactor Building Emergency Cooling System has no impact on CCDP or CLERP for RISI (CCDP and CLERP = 0.0 for RI-D23 in Table 3-3). Therefore, correcting this gap would have no impact on the PRA case runs. Therefore, this gap is considered acceptable for the RISI evaluation.

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LE-D1b-01	The TMI containment comparison to the Oconee containment evaluation (Appendix B of TMI-PRA-015.2, Rev. 0) provided a good basis for utilizing the Oconee analyses of the personnel airlocks and purge penetrations. However, the report identified that additional analyses were necessary for evaluation of the equipment hatch, mechanical penetrations and electrical penetrations. A discussion of a qualitative evaluation of these is provided in the latter portion of Appendix C of TMI-PRA-015.2, Rev. 0. A detailed evaluation of the TMI equipment hatch, personnel airlock and containment purge valves are performed, providing good plant-specific basis for their evaluation. However, the evaluation of the electrical and mechanical penetrations is very subjective, stating simply that it is assumed that their failure pressures will be higher than the containment structure. While these assumptions are likely true, some additional basis should be provided.	LE-D2	Open	A sensitivity was performed to determine the impact of assuming a substantially lower failure pressure for the electrical and mechanical penetrations. See Section 2.2 for details of the assessment. This sensitivity resulted in no changes to the rankings for the RISI. Therefore, this gap is considered acceptable for the RISI evaluation.
LE-D4-01	The secondary side isolation is evaluated in the Level 1 analysis. However, the SG relief valve was evaluated only for the pre-core damage failures to isolate. Should core damage occur, the relief valve would experience many additional challenges (either passing steam or water depending on whether or not there is FW flow to the SG). The Level 2 analysis does not account for this elevated potential for a stuck open relief valve.	LE-D5	Open	All accident progression sequences involving SGTR (either as an initiator or induced following core damage) are assumed to be LERF. No credit is taken for SG isolation for any SGTR accident progression sequence. Therefore, SGTR is treated conservatively. The result of this gap is to bias CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.

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Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
LE-D5-01	Induced SGTR is considered in CET top event node Bypass, but it does not appear that a specific ISGTR methodology was utilized. The most significant issue with the ISGTR model is the assumption that operators would start the RCPs with dry SGs. The CET notebook states that operators are directed to do so with no caution about SG status. Clearing the loop seal results in significant convective heat transfer to the SG tubes, yielding the assumed 0.9 conditional probability of ISGTR. However, the current TMI SAMG guidance (ER-TM-TSC-0010, Rev. 1) directs operators to turn on the RCPs as a SAMG action but has a caution on the step 3.3 that turning on the RCPs when the SGs are dry can result in an induced SGTR. The caution states that if the SG cannot be adequately protected, then don't turn on the RCPs.	LE-D6	Open	The operator action to clear seals was determined to be considerably less likely than previously assumed, based on a review of the latest TMI SAMG guidance. Changing the operator action to reflect the SAMG guidance reduced ISGTR contribution to LERF. This F&O is still open, although the excessive conservatism relating to ISGTR has been removed. The representation of ISGTR is still considered to be conservative (but to a lesser extent), which bias the CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.
LE-E2-01	The TMI CET parameter estimates are conservative in general. The probabilities of early containment failure from DCH, rapid steam generation, and combustible gas burns are conservative and are all based on references from 1992 and earlier. Studies since that time (e.g., NUREG/CR-6075, NUREG/CR-6109 and NUREG/CR-6338) have recommended greatly reduced probabilities or even eliminated early containment HPME failures from large, dry containments.	LE-E2	Open	The TMI CET parameter estimates are conservative, resulting in conservative LERF results. The result of this gap is to bias CLERP results higher, which is conservative. Therefore, this gap is considered acceptable for the RISI evaluation.

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LE-F2-01	The CET document notes some MAAP sensitivity analyses that were performed to aid in determining the split fractions in the CET. The sensitivity analyses were not specifically referenced, but were performed to address some of the MAAP uncertainties. No sensitivities on the other phenomenological Level 2 uncertainties (e.g., induced SGTR assumptions and probabilities) have been performed. No uncertainty calculation was documented in the Level 2 notebooks. Perform LERF uncertainty and sensitivity calculations. Characterize LERF uncertainties consistent with the applicable requirements of ASME Standard tables 4.5.8-2(d) and 4.5.8-2(e).	LE-F3	Open	To be determined once the new USNRC/EPRI guidance is implemented. However, the EPRI RISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.

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Table 2-2
Gaps to Capability Category II
For the DA Technical Element

Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
DA-B1-01	Although components are grouped according to type and characteristic, there is no difference in failure rates for most groups of the same type, unless the group was updated with plant specific data.	DA-B1	Open	This SR is Met Capability Category I, since the components are grouped by type. The most recent generic data (NUREG/CR-6928) does not support more specific groups, since most component failure rates are based on type (e.g., there is only one category of MOVs). However, there is gradation between standby and normally operating components. For the important systems (MU, DH, EF, DC, DH, DR, NR, NC, DG), plant specific data is used to determine the failure rates by system and component type. The component type codes that were updated with plant specific data were not changed much from the generic data. Therefore, it is not expected that more complete grouping of components by characteristic in order to meet Capability Category II would have much impact on PRA and the results of the RISI evaluation. Therefore, this gap is considered acceptable for the RISI evaluation.
DA-B2-01	Although the component failure rates are grouped by system and component type, that does not guarantee that outliers are not included in a group.	DA-B2	Open	There is no indication of outliers due to testing or operational characteristics (except for potentially manual valves, which are not risk significant), nor due to poor performance of certain components or systems. Therefore, this gap is considered acceptable for the RISI evaluation.

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Table 2-2
Gaps to Capability Category II
For the DA Technical Element

Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
DA-C2-01	Plant specific data is collected for component failures and success for all components in the scope of MSPI. Although this is a smaller set of component types and failure modes than all the significant basic events (e.g, F-V >.005 or RAW >2), it is considered an acceptable scope of data for a model update. Unavailability data is collected for all MR equipment for which unavailability data is maintained.	DA-C2	Open	The impact of updating with plant specific data on failure rates was minor (e.g., changes by several percent). There is no indication of particularly poor performers in any risk significant system. Therefore, there would be no significant impact expected on the PRA or RISI if plant specific data was used for all risk significant component types. Therefore, this gap is considered acceptable for the RISI evaluation.
DA-C4-01	The MSPI rules are used for data collection of failures, as described in the Data Notebook. The failure definitions are generally consistent with the PRA failure definitions. However, that is currently an assumption, since there is no documented basis.	DA-C4	Open	This is considered a documentation issue not affecting the technical adequacy of the PRA model.
DA-C7-01	The number of Surveillance Tests are estimated based on plant requirements. Data is obtained from the MSPI Derivation Reports as described in Section 2.4 of the Data Notebook.	DA-C7	Open	This SR meets Capability Category I. Although the number of Surveillance Tests are estimated, the estimation is expected to be very close to the actual value. If actual numbers of tests were used, the final failure rates should not be significantly different from the failure rates calculated with estimated demands. Therefore, this gap is considered acceptable for the RISI evaluation.

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Table 2-2
Gaps to Capability Category II
For the DA Technical Element

Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
DA-C8-01	Time that the component is in standby is estimated based on plant requirements (e.g., nominal time between surveillance tests). Documentation of standby mission times is lacking in the current data notebook.	DA-C8	Open	This SR meets Capability Category I. Although the standby time of components is estimated, the estimation is expected to be very close to the actual value. If actual standby times were used, the final failure probabilities should not be significantly different from the failure probabilities calculated with estimated times. Therefore, this gap is considered acceptable for the RISI evaluation.
DA-C10-01	Surveillance tests were reviewed to determine demands and operational time. Successes were estimated based on surveillance schedules.	DA-C10	Open	This SR meets Capability Category I. Although the demands and operational time for components is estimated, the estimation is conservative and expected to be reasonably close to the actual value. If actual operation times and successes were used, the final failure rates should be lower but not significantly different from the failure rates calculated with estimated times. Therefore, this gap is considered acceptable for the RISI evaluation.
DA-C12-01	A review of support system unavailability to ensure that double counting did not occur was performed. However, the documentation is lacking in the Data Notebook.	DA-C12	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
DA-C14-01	A review of coincident unavailability was performed. However, the documentation is lacking in the notebook.	DA-C14	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.

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Table 2-2
Gaps to Capability Category II
For the DA Technical Element

Title	Description of Gap	Applicable SRs [6]	Current Status	Comment
DA-D1-01	The decision was made to only update MSPI components with plant-specific data. However, this does not meet the requirements to update all Significant BEs, since there are significant BEs that are not within the scope of MSPI. For the BEs within the scope of MSPI, CC II is Met. However, the full scope of significant BEs does not use both generic and plant-specific data in a Bayes process.	DA-D1	Open	This SR meets Capability Category I. The impact of updating with plant specific data on failure rates was minor (e.g., changes by several percent). There is no indication of particularly poor performers in any risk significant system. Therefore, there would be no significant impact expected on the PRA or RISI if plant specific data was used for all risk significant component types. Therefore, this gap is considered acceptable for the RISI evaluation.
DA-D4-01	Although a Bayesian Approach is used, there is no evidence that a check of the posterior distribution was made as required by this SR. On review, it can be seen that the type codes which were updated with plant specific information have reasonable values, but there is no documentation of the check.	DA-D4	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
DA-E3-01	This SR is not met. Parametric uncertainty values are provided, but sources of model uncertainty and related assumptions are not.	DA-E3	Open	To be determined once the new USNRC/EPRI guidance is implemented. However, the EPRI RISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.

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Table 2-3
Identified Gaps to Capability Category II
For the IF Technical Element

IFPP-A2-01	Documentation is lacking in details for several parts of the flood analysis, such as flood area determination and screening criteria and results.	IFPP-A2, IFSO-B2, IFSN-B2 IFQU-B2	Open	This is a documentation issue not affecting the technical adequacy of the PRA model.
IFPP-B3-01	Documentation and evaluation of sources of uncertainty has not been accomplished. This is a recognized/acknowledged gap for the TMI PRA.	IFPP-B3, IFSO-B3 IFSN-B3 IFEV-B3	Open	To be determined once the new USNRC/EPRI guidance is implemented. However, the EPRI RISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.
IFEV-A5-01	Several requirements in establishing flood initiating event frequencies are not met. 1) Recent pipe data is not used 2) Effect of plant specific features and experience are not factored into the initiating event frequencies 3) Human-induced flooding does not appear to be evaluated.	IFEV-A5, IFEV-A6, IFEV-A7	Open	The importance of flood initiating events are affected by this gap. The only impact on the RISI calculations associated with flood initiating events are the CCDP and CLERP calculations for demand impacts. A sensitivity was performed for these calculations based on assuming all flood initiators were an order of magnitude higher. See Section 2.2 for details of the assessment. This sensitivity resulted in no changes to the rankings for the RISI. Therefore, this gap is considered acceptable for the RISI evaluation.

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Request for Relief for Inservice Inspection Impracticality of Pressure Testing the Reactor Pressure Vessel Head Flange Connection Lines In Accordance with 10 CFR 50.55a(g)(5)(iii)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	2
Reference:	Table IWC-2500-1, IWC-5200
Examination Category:	C-H
Item Number:	C7.10
Description:	Pressure Testing the Reactor Pressure Vessel Head Flange Connection Lines
Component Number:	Reactor Pressure Vessel Head Flange Connection Lines
Drawing Number:	Figure I4R-03.1

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The ISI program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda.

3.0 APPLICABLE CODE REQUIREMENT:

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires ISI Class 2 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWC-5220. This pressure test is to be conducted once each inspection period.

4.0 IMPRACTICALITY OF COMPLIANCE:

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that pressure testing the Reactor Pressure Vessel Head Flange Connection Line is deemed impractical.

The two Reactor Pressure Vessel Head Flange Connection Lines are separated from the reactor pressure boundary by one passive membrane, an O-ring located on the reactor pressure vessel closure head flange. A second O-ring is located on the opposite side of the tap in the vessel flange (see Figure I4R-03.1). This line runs from the flange to a normally closed 1" isolation valve and is not pressurized during normal operation.

The configuration of this system precludes manual testing while the vessel head is removed. The configuration of the vessel tap, combined with the small size of the tap and

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the high test pressure requirement (approximately 2155 psig), prevents the tap from being temporarily plugged. Also, when the reactor pressure vessel closure head is installed, an adequate pressure test cannot be performed due to the fact that the inner O-ring is designed to withstand pressure in one direction only. Due to the groove that the O-ring sits in and the clip assembly (See Figure I4R-03.1), pressurization in the opposite direction into the recessed cavity and retainer clips would likely damage the O-ring.

5.0 BURDEN CAUSED BY COMPLIANCE:

Pressure testing of this line during the System Leakage Test is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. Purposely failing the inner O-ring to perform the ASME Section XI required test would require purchasing a new set of O-rings, additional time and radiation exposure to detension the reactor pressure vessel head, installation of the new O-rings, and then reset and retension the reactor pressure vessel head. This is considered to impose an undue burden.

Based on the above, TMI requests relief from the ASME Section XI requirements for system leakage testing of the Reactor Pressure Vessel Head Flange Connection Lines.

6.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

A VT-2 visual examination on the ISI Class 2 portion of the Reactor Pressure Vessel Head Flange Connection Lines will be performed once each inspection period when the reactor pressure vessel head is off and the fuel transfer canal is filled above the vessel flange. The static head developed with the connection lines filled with water will allow for the detection of any gross leakage in the lines. This examination will be performed on the accessible, exposed portion of the lines out to the closed isolation Class 2 boundary valves once each inspection period as per the frequency specified by Table IWC-2500-1.

7.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the fourth ten-year ISI interval for TMI, Unit 1.

8.0 PRECEDENTS:

Similar relief requests have been approved for:

Peach Bottom Atomic Power Station, Units 2 and 3, fourth ISI interval Relief Request I4R-25 was granted per U.S. Nuclear Regulatory Commission (USNRC) Safety Evaluation Report (SER) dated February 26, 2009.

Limerick Generating Station, Units 1 and 2, third ISI interval Relief Request I3R-08 was granted per USNRC SER dated March 11, 2008.

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LaSalle County Station, Units 1 and 2, third interval Relief Request I3R-08 was granted per USNRC SER dated January 30, 2008.

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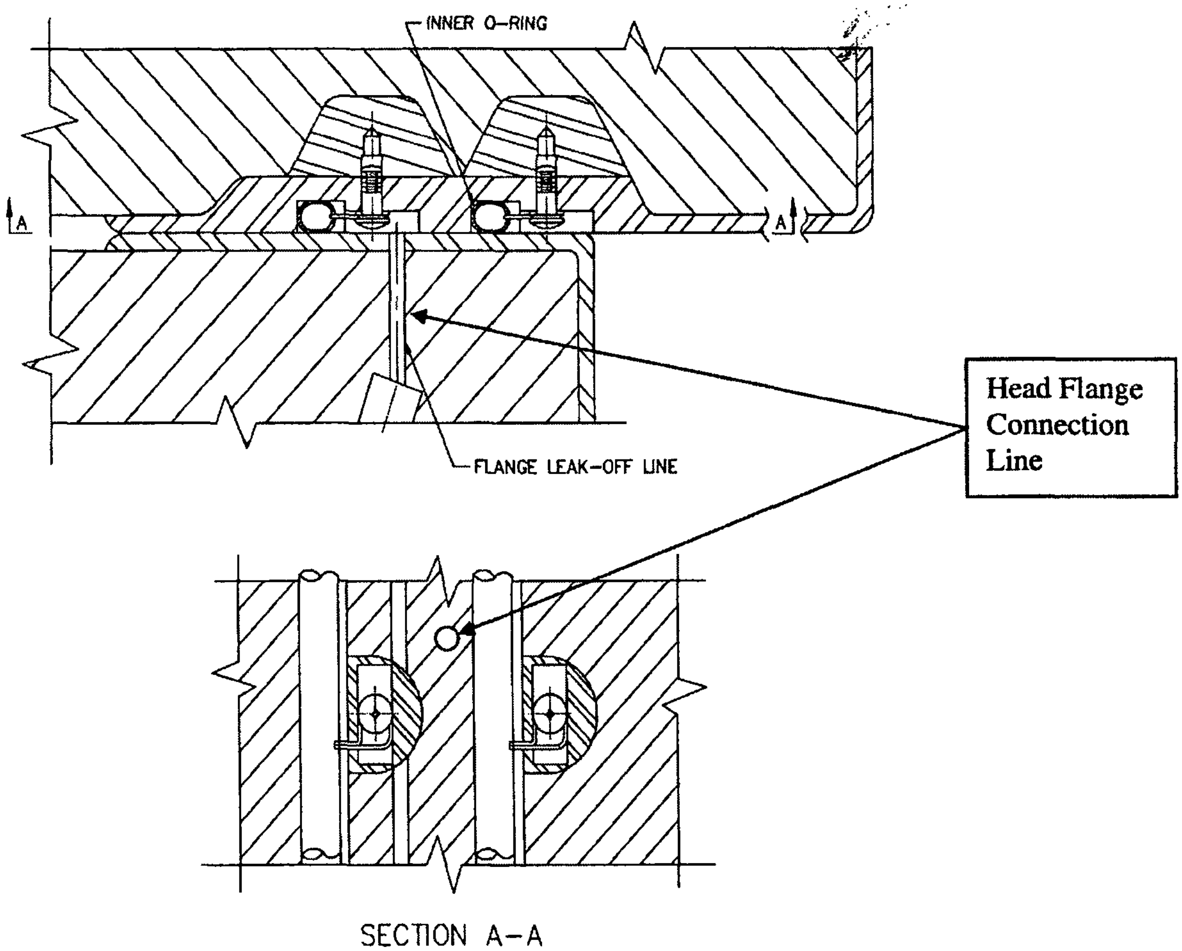

FIGURE I4R-03.1
O-RING CONFIGURATION

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Request for Relief for ISI Snubbers Included in the Technical Specifications Snubber Visual Examination and Functional Testing Program In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1, 2, and 3
Reference:	IWF-5200(a) and IWF-5300(a) IWF-5200(b) and IWF-5300(b)
Examination Category:	NA
Item Number:	NA
Description:	ISI Snubbers Included in the Technical Specifications Snubber Visual Examination and Functional Testing Program
Component Number:	Various Safety Related Snubbers

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The ISI program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda. Table IWA-1600-1 of the ASME Section XI specifically references the 1987 Edition with OMa-1988 Addenda of the ASME/ANSI OM Code (Part 4).

3.0 APPLICABLE CODE REQUIREMENT:

Paragraphs IWF-5200(a) and IWF-5300(a) require Preservice and Inservice examinations to be performed in accordance with ASME/ANSI OM, Part 4, using the VT-3 visual examination method described in Paragraph IWA-2213.

Paragraphs IWF-5200(b) and IWF-5300(b) require Preservice and Inservice tests to be performed in accordance with ASME/ANSI OM, Part 4.

4.0 REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative, utilizing TMI Technical Specifications, Section 4.17, will provide an acceptable level of quality and safety.

ASME/ANSI OM (Part 4) specifies three functional test plans. This Code was completely revised in the 1988 Addenda to incorporate three snubber functional testing

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sampling plans, identified as the 10% testing sample plan, the 37 testing sample plan, and the 55 testing sample plan.

The TMI Technical Specifications provide similar requirements for visual examination, scheduling, re-examinations, and functional testing requirements.

"To provide assurance of snubber functional reliability, one of the two sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.17-1"

The TMI, Unit 1 Technical Specifications 10% testing sample plan differs from the ASME OM 10% plan in that the Technical Specifications require an additional 10% of that type of snubber be tested for each functional test failure. The ASME OM 10% testing plan requires additional snubber testing as follows: "For any snubber(s) determined to be unacceptable as a result of testing, an additional sample of at least one-half the size of the initial sample lot shall be tested until the total number tested is equal to the initial sample size multiplied by the factor 1 + C/2, where C is the total number of snubbers found to be unacceptable." The TMI, Unit 1 Technical Specifications testing plan results in an increase in the overall level of plant quality and safety based on a larger testing population should unacceptable testing results be encountered.

The TMI, Unit 1 Technical Specifications contain requirements for a snubber seal service life monitoring program. The TMI, Unit 1 Technical Specifications (4.17.1.i) require the following:

"A snubber seal service life program shall be developed whereby the seal service life of hydraulic snubbers is monitored to ensure that the service life is not exceeded between surveillance inspections. The designated service life for the various seals shall be established based on engineering information. The seals shall be replaced so that the indicated service life will not be exceeded during a period when the snubber is required to be OPERABLE."

The TMI safety-related snubber population is comprised exclusively of hydraulic snubbers. TMI has procedures in place to implement the snubber program as described in Technical Specifications, Section 4.17.

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The TMI, Unit 1 Technical Specifications snubber visual examination frequency is based on an operating cycle of 24 months. The TMI, Unit 1 Technical Specifications require visual examination of all safety-related snubbers to be performed in accordance with the following schedule:

Number of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period**#
0	24 months ± 25%
1	16 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

** The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

The provisions of Technical Specifications Table 1.2 are not applicable.

The TMI, Unit 1 Technical Specifications visual examination frequency is similar to the visual examination frequency defined in OM Part 4. OM Part 4, 2.3.2.2 requires essentially identical visual examination frequencies when two or more unacceptable snubbers are identified. The TMI, Unit 1 Technical Specifications visual examination frequency for 0 unacceptable snubbers is 24 months, and for 1 unacceptable snubber is 16 months. The OM Part 4, 2.3.2.2 visual examination frequency for 0 or 1 unacceptable snubber is 18 months and 12 months, respectively. The OM Part 4 requirement did not initially account for 24 month fuel cycles which were not predominant at the time of issuance. The OM Part 4 visual examination frequency could require plant shutdown just for snubber visual examinations when 24-month operating cycles are implemented.

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The TMI, Unit 1 Technical Specifications do not specifically address preservice visual examination of snubbers. TMI will perform preservice visual examination of snubbers following maintenance activities (e.g., replacement, repair, modification, etc.).

In conclusion, the visual examination and functional testing of snubbers at TMI, Unit 1 will be performed in accordance with Technical Specifications, Section 4.17. These visual examinations and functional tests will be performed in lieu of the inspection and testing requirements of Paragraphs IWF-5200(a) and (b) and Paragraphs IWF-5300(a) and (b). The general requirements of Subsection IWA, such as examination methods, personnel qualifications, etc. remain applicable. Based on this approach, TMI, Unit 1 has determined that implementation of the Technical Specifications, Section 4.17, snubber program will assure an acceptable level of quality and safety.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

TMI, Unit 1 requests approval to use Technical Specifications, Section 4.17, "Shock Suppressors (Snubbers)", and associated Bases, as found within the TMI, Unit 1 Technical Specifications for visual examination, scheduling, re-examinations, and functional testing requirements.

Snubber preservice and inservice visual examinations will be conducted using the VT-3 visual examination method (i.e., VT-3 qualified procedures and personnel) described in Paragraph IWA-2213 of ASME Section XI.

Repair/replacement activities performed on snubbers shall be in accordance with Article IWA-4000 of ASME Section XI.

Snubbers installed, corrected, or modified by repair/replacement activities shall be preservice examined and preservice tested in accordance with the applicable Technical Specifications requirements prior to return to service.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the fourth ten-year ISI interval for TMI, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

Limerick Generating Station, Units 1 and 2, third ISI interval Relief Request I3R-05 was authorized per U.S. Nuclear Regulatory Commission (USNRC) Safety Evaluation Report (SER) dated March 11, 2008.

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Hope Creek Generating Station, third ISI interval Relief Request HC-I3R-02 was authorized per USNRC SER dated October 16, 2008.

Susquehanna Steam Electric Station, Units 1 and 2, third ISI interval Relief Request 3RR-03 was authorized per USNRC SER dated September 24, 2004.

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Request for Relief From Qualification Requirements of ASME Section XI, Appendix VIII, Supplement 11, for Examination of Structural Weld Overlays (SWOLs) In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1
Reference:	ASME Section XI, Mandatory Appendix VIII, Supplement 11
Examination Category:	B-F, B-J
Item Number:	B5.40, B9.11
Description:	Qualification Requirements of ASME Section XI, Appendix VIII, Supplement 11, for Examination of Structural Weld Overlays (SWOLs)
Drawing Number:	1D-ISI-RC-002, 1D-ISI-RC-012, 1D-ISI-DH-001

2.0 APPLICABLE CODE EDITION AND ADDENDA:

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda.

2. ASME Section XI, 2001 Edition, No Addenda, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," mandated through 10 CFR 50.55a(b)(2)(xxiv).

3. ASME Section XI, 2001 Edition, No Addenda, Appendix VIII, Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds," mandated through 10 CFR 50.55a(b)(2)(xxiv).

3.0 APPLICABLE CODE REQUIREMENT:

Full structural weld overlays of austenitic piping welds shall be examined using procedures and examiners qualified in accordance with ASME Section XI, Mandatory Appendix VIII, Supplement 11.

10 CFR 50.55a(b)(2)(xxiv) prohibits use of Appendix VIII and associated supplements beyond the 2001 Edition, No Addenda of ASME Section XI.

TMI, Unit 1 has applied full structural weld overlays (SWOL) on three dissimilar metal weld locations (Reference Table 1 for specific locations). In Relief Request I4R-02, it was

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indicated that ultrasonic examination of completed weld overlay repaired welds will be performed in accordance with ASME Section XI, Non-Mandatory Appendix Q. Article Q-4000 states: "Ultrasonic examination personnel shall be certified in accordance with the Owner's written practice. Procedures and personnel shall be qualified in accordance with Appendix VIII." Appendix VIII, Supplement 11 is applicable to ultrasonic examination of weld overlay repaired locations.

4.0 REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(a)(3)(i), TMI, Unit 1 requests relief from the qualification requirements of Supplement 11 and proposes instead to use the ultrasonic (UT) qualification program for weld overlay inspections developed and administered through the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) qualification program.

U.S. nuclear utilities created the PDI program to implement performance demonstration requirements contained in Section XI, Appendix VIII. PDI has developed a program for qualifying equipment, procedures, and personnel for examinations of weld overlays in accordance with the UT criteria of Supplement 11. Prior to the Supplement 11 program, EPRI maintained a performance demonstration program for weld overlay examination qualification under the Tri-party Agreement. Instead of having two programs with similar objectives, the USNRC staff recognized the PDI program for weld overlay examination qualifications as an acceptable alternative to the Tri-party Agreement (Reference 3).

Although the PDI program does not fully conform with the existing requirements of Supplement 11, it is routinely assessed by the USNRC staff for consistency with the current ASME Code and proposed changes. The major differences between the PDI program compared to Supplement 11 are associated with flaw locations contained in test specimens and fabricated flaw tolerances. The changes in flaw locations within the test specimens allowed using the test specimens from the Tri-party Agreement, and changes in fabricated flaw tolerances provide UT acoustic responses similar to those associated with Intergranular Stress Corrosion Cracking (IGSCC).

Table 2 of this relief request provides the requirements of Supplement 11 along with the associated EPRI PDI requirement. Discussion of the differences between the two programs is as follows:

Paragraph 1.1.(b) of Supplement 11 states limitations to the maximum thickness for which a procedure may be qualified. The ASME Code states that "The specimen set must include at least one specimen with overlay thickness within -0.10-inch to +0.25-inch of the maximum nominal overlay thickness for which the procedure is applicable." While the ASME Code requirement addresses the specimen thickness tolerance for a single specimen

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set, it is confusing when multiple specimen sets are used. The PDI proposed alternative states that "the specimen set shall include specimens with overlays not thicker than 0.10-inch more than the minimum thickness, nor thinner than 0.25-inch of the maximum nominal overlay thickness for which the examination procedure is applicable." The proposed alternative provides clarification on the application of the tolerance. The tolerance is unchanged for a single specimen set; however, the proposed alternative clarifies the tolerance for multiple specimen sets by providing tolerances for both the minimum and the maximum thicknesses.

Paragraph 1.1(d)(1) requires that all base metal flaws be cracks. PDI determined that certain Supplement 11 requirements pertaining to location and size of cracks would be extremely difficult to achieve. For example, flaw implantation requires excavating a volume of base material to allow a pre-cracked coupon to be welded into this area. This process would add weld material to an area of the specimen that typically consists of only base material, and could potentially make ultrasonic examination more difficult and not representative of actual field conditions. In an effort to satisfy the requirements, PDI developed a process for fabricating flaws that exhibit crack-like reflective characteristics. Instead of all flaws being cracks, as required by Paragraph 1.1(d)(1), the PDI weld overlay performance demonstrations contain at least 70% cracks with the remainder being fabricated flaws exhibiting crack-like reflective characteristics. The fabricated flaws are semi-elliptical with tip widths of less than 0.002-inches. PDI limits flaws in cases where implantation of cracks produces spurious reflectors that are uncharacteristic of actual flaws.

Paragraph 1.1(e)(1) requires that at least 20% but not less than 40% of the flaws shall be oriented within ±20 degrees of the axial direction (of the piping test specimen). Flaws contained in the original base metal heat-affected zone satisfy this requirement; however, PDI excludes axial fabrication flaws in the weld overlay material. PDI has concluded that axial flaws in the overlay material are improbable because the overlay filler material is applied in the circumferential direction (parallel to the girth weld); therefore, fabrication anomalies would also be expected to have major dimensions in the circumferential direction.

Paragraph 1.1(e)(1) also requires that the rules of Subarticle IWA-3300 shall be used to determine whether closely spaced flaws should be treated as single or multiple flaws. PDI treats each flaw as an individual flaw and not as part of a system of closely spaced flaws. PDI controls the flaws going into a test specimen set such that the flaws are free of interfering reflections from adjacent flaws. In some cases this permits flaws to be spaced closer than what is allowed for classification as a multiple set of flaws by Subarticle IWA-3300, thus potentially making the performance demonstration more challenging than the existing requirement.

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Paragraph 1.1(e)(2) requires that specimens be divided into base metal and overlay grading units. The PDI program adds clarification with the addition of the word "fabrication" and ensures that flaw identification will not be masked by other flaws with the addition of "Flaws shall not interfere with ultrasonic detection or characterization of other flaws." The PDI alternative provides clarification and assurance that the flaws are identified.

Paragraph 1.1(e)(2)(a)(1) requires that a base grading unit shall include at least three inches of the length of the overlaid weld, and the base grading unit includes the outer 25% of the overlaid weld and base metal on both sides. The PDI program reduced the criteria to one inch of the length of the overlaid weld and eliminated from the grading unit the need to include both sides of the weld. The proposed change permits the PDI program to continue using test specimens from the existing weld overlay program which have flaws on both sides of the welds. These test specimens have been used successfully for testing the proficiency of personnel for over 16 years. The weld overlay qualification is designed to be a near-side (relative to the weld) examination, and it is improbable that a candidate would detect a flaw on the opposite side of the weld due to the sound attenuation and re-direction caused by the weld microstructure. However, the presence of flaws on both sides of the original weld (outside the PDI grading unit) may actually provide a more challenging examination, as candidates must determine the relevancy of these flaws, if detected.

Paragraph 1.1(e)(2)(a)(2) requires, when base metal cracking penetrates into the overlay material, that a portion of the base grading unit shall not be used as part of the overlay grading unit. The PDI program adjusts for the changes in Paragraph 1.1(e)(2)(a)(2) and conservatively states that when base metal flaws penetrate into the overlay material, no portion of it shall be used as part of the overlay fabrication grading unit.

Paragraph 1.1(e)(2)(a)(3) requires that for unflawed base grading units, at least one inch of unflawed overlaid weld and base metal shall exist on either side of the base grading unit. This is to minimize the number of false identifications of extraneous reflectors. The PDI program stipulates that unflawed overlaid weld and base metal exists on all sides of the grading unit and flawed grading units must be free of interfering reflections from adjacent flaws which addresses the same concerns as the ASME Code.

Paragraph 1.1(e)(2)(b)(1) requires that an overlay grading unit shall include the overlay material and the base metal-to-overlay interface of at least six square inches. The overlay grading unit shall be rectangular, with minimum dimensions of two inches. The PDI program reduces the base metal-to-overlay interface to at least one inch (in lieu of a minimum of two inches) and eliminates the minimum rectangular dimension. This criterion is necessary to allow use of existing examination specimens that were fabricated in order to meet USNRC Generic Letter 88-01 (Tri-party Agreement, July 1984). This criterion may be more challenging to meet than that of the ASME Code because of the variability associated with the shape of the grading unit.

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Paragraph 1.1(e)(2)(b)(2) requires that unflawed overlay grading units shall be surrounded by unflawed overlay material and unflawed base metal-to-overlay interface for at least one inch around its entire perimeter. The PDI program redefines the area by noting unflawed overlay fabrication grading units shall be separated by at least one inch of unflawed material at both ends and sufficient area on both sides to preclude interfering reflections from adjacent flaws. This change may provide a more challenging demonstration than required by ASME Code because of the possibility for having a parallel flaw on the opposite side of the weld.

Paragraph 1.1(e)(2)(b)(3) requirements are retained in the PDI program. In addition, the PDI program requires that initial procedure qualification contain three times the number of flaws required for a personal qualification. To qualify new values of essential variables, the equivalent of at least one personnel qualification is required.

Paragraph 1.1(f)(1) requirements are retained in the PDI program, with the clarification change of the term "flaws" for "cracks." In addition, the PDI program includes the requirements that sizing sets shall contain a distribution of flaw dimensions to verify sizing capabilities. The PDI program also requires that initial procedure qualification contain three times the number of flaws required for a personnel qualification. To qualify new values of essential variables, the equivalent of at least one personnel qualification is required.

Paragraphs 1.1(f)(3) and 1.1(f)(4) requirements are clarified by the PDI program by replacing the term "cracking" with "flaws" because of the use of alternative flaws.

Paragraphs 2.1 and 2.2(d) requirements are clarified by the PDI program by the addition of the terms "metal" and "fabrication". These terms were added to clarify the description of the grading units present in a specimen. "Metal" was added to "base" to read "base metal" and "fabrication" was added to "overlay" to read "overlay fabrication".

Paragraph 2.3 requires that, for depth sizing tests, 80% of the flaws shall be sized at a specific location on the surface of the specimen identified to the candidate. This requires detection and sizing tests to be performed separately. The PDI revised the weld overlay program to allow sizing to be conducted either in conjunction with, or separately from, the flaw detection test. If performed in conjunction with detection and the detected flaws do not meet the Supplement 11 range criteria, additional specimens will be presented to the candidate with the regions containing flaws identified. Each candidate will be required to determine the maximum depth of the flaw in each region. For separate sizing tests, the regions of interest will also be identified and the maximum depth and length of each flaw in the region will similarly be determined. In addition, PDI stated that grading units are not applicable to sizing tests, and that each sizing region will be large enough to contain the target flaw, but small enough such that candidates will not attempt to size a different flaw.

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Paragraph 3.1 requires that examination procedures, equipment and personnel (as a complete ultrasonic system) are qualified for detection or sizing of flaws, as applicable, when certain criteria are met. The PDI program allows procedure qualification to be performed separately from personnel and equipment qualification. Historical data indicate that, if ultrasonic detection or sizing procedures are thoroughly tested, personnel and equipment using those procedures have a higher probability of successfully passing a qualification test. In an effort to increase this passing rate, PDI has elected to perform procedure qualifications separately in order to assess and modify essential variables that may affect overall system capabilities. For a procedure to be qualified, the PDI program requires three times as many flaws to be detected (or sized) as shown in Supplement 11 for the entire ultrasonic system. The personnel and equipment are still required to meet the Supplement 11 requirement.

Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.10-inch are reported as being intrusions into the overlay material. The PDI program omits this criterion because of the difficulty in actually fabricating a flaw with a 0.10-inch minimum extension into the overlay, while still knowing the true state of the flaw dimensions. However, the PDI program requires that cracks be depth-sized to the tolerance specified in the ASME Code which is 0.125-inches. Since the ASME Code tolerance is close to the 0.10-inch value of Paragraph 3.2(b), any crack extending beyond 0.10-inch into the overlay material would be identified as such from the characterized dimensions.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Pursuant to 10 CFR 50.55a(a)(3)(i), UT examination technique and personnel qualifications for the existing and future SWOLs will be performed using EPRI PDI demonstrated procedures in conjunction with PDI qualified examiners in lieu of the ASME Section XI, Appendix VIII, Supplement 11 requirements. The EPRI PDI qualification program provides a recognized acceptable alternative to the requirements of Supplement 11 and provides an acceptable level of quality and safety.

6.0 PERIOD FOR WHICH RELIEF IS REQUESTED:

Relief is requested for the fourth ten-year ISI interval for TMI, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

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TMI, Unit 1, third inspection interval Relief Request was authorized per U.S. Nuclear Regulatory Commission (USNRC) Safety Evaluation Report (SER) dated October 17, 2007 (Reference 4).

TMI, Unit 1, third inspection interval Relief Request from flaw removal, heat treatment, and nondestructive examination requirements was authorized per USNRC SER dated July 21, 2004 (Reference 5).

Braidwood Station, Units 1 and 2, second inspection interval Relief Request I2R-48 was authorized per USNRC SER dated September 17, 2007 (Reference 6).

8.0 REFERENCES:

1. ASME Code, Section XI, 2001 Edition, No Addenda, Appendix VIII, Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds".

2. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, October 2007.

3. Letter from W. Bateman (USNRC) to M. Bratton (Entergy Nuclear Southwest), "Weld Overlay Performance Demonstration Administered by PDI as an Alternative for Generic Letter 88-01 Recommendations," dated January 15, 2002 (ML020160532).

4. Letter from H. Chernoff (USNRC) to C. Crane (AmerGen Energy Company, LLC), "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Relief Request 2007-TMI-01, Regarding Structural Weld Overlays on Pressurizer Surge, Pressurizer Spray, and Hot Leg Decay Heat Drop Line Nozzles, (TAC NO. MD5427)," dated October 17, 2007 (ML072770051).

5. Letter from R. Laufer (USNRC) to C. Crane (AmerGen Energy Company, LLC), "Three Mile Island Nuclear Station, Unit 1 (TMI-1) Request for Relief from Flaw Removal, Heat Treatment, and Nondestructive Examination Requirements for the Third 10-Year Inservice Inspection (ISI) Interval (TAC NO. MC1201)," dated July 21, 2004 (ML041670510).

6. Russell Gibbs (USNRC) to C. Crane (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 Evaluation of Inservice Inspection Program Relief Request I2R-48 Pertaining to Structural Weld Overlays on Pressurizer Spray, Relief, Safety, and Surge Nozzle Safe Ends (TAC NOS. MD4590, and MD4591)," dated September 17, 2007 (ML072430034).

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Table 1

WELD IDENTIFICATION	ITEM #	SIZE	ADJACENT WELD IDENTIFICATION	DESCRIPTION
SR0010BM	B9.11	10"	NA	Pressurizer surge nozzle to pipe dissimilar metal weld at hot leg "A".
PR0021BM	B5.40	10"	NA	Pressurizer surge nozzle to safe end dissimilar metal weld at the pressurizer.
DH0001BM	B9.11	12"	DH0498	Decay heat nozzle to safe end weld at hot leg "B". This location also includes the adjacent safe end to pipe weld.

Note: The SWOL application SER approvals for these welds were provided in References 5 and 6.

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Appendix VIII, Supplement 11	PDI Modification
1.0 SPECIMEN REQUIREMENTS	
Qualification test specimens shall meet requirements listed herein, unless a set of specimens is designed to accommodate specific limitations stated in the scope of the examination procedure (e.g., pipe size, weld joint configuration, access limitations). The same specimen may be used to demonstrate both detection and sizing qualification.	No Change
1.1 General. The specimen set shall conform to the following requirements.	No Change
1.1(a) Specimens shall have sufficient volume to minimize spurious reflections that may interfere with the interpretation process.	No Change
1.1(b) The specimen set shall consist of at least three specimens having different nominal pipe diameters and overlay thicknesses. They shall include the minimum and maximum nominal pipe diameters for which the examination procedure is applicable. Pipe diameters within a range of 0.9 to 1.5 times a nominal diameter shall be considered equivalent. If the procedure is applicable to pipe diameters of 24 inches or larger, the specimen set must include at least one specimen 24 inches or larger but need not include the maximum diameter. The specimen set must include at least one specimen with overlay thickness within -0.1 inches to +0.25 inches of the maximum nominal overlay thickness for which the procedure is applicable.	The specimen set shall consist of at least three specimens having different nominal pipe diameters and overlay thicknesses. They shall include the minimum and maximum nominal pipe diameters for which the examination procedure is applicable. Pipe diameters within a range of 0.9 to 1.5 times a nominal diameter shall be considered equivalent. If the procedure is applicable to pipe diameters of 24 inches or larger, the specimen set must include at least one specimen 24 inches or larger but need not include the maximum diameter. The specimen set shall include specimens with overlays not thicker than 0.1 inches more than the minimum thickness, nor thinner than 0.25 inches of the maximum nominal overlay thickness for which the procedure is applicable.
1.1(c) The surface condition of at least two specimens shall approximate the roughest surface condition for which the examination procedure is applicable.	No Change
(d) Flaw Conditions	
1.1(d)(1) Base metal flaws. All flaws must be	Base metal flaws. All flaws must be in or near the

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cracks in or near the butt weld heat-affected zone, open to the inside surface, and extending at least 75% through the base metal wall. Flaws may extend 100% through the base metal and into the overlay material; in this case, intentional overlay fabrication flaws shall not interfere with ultrasonic detection or characterization of the cracking. Specimens containing IGSCC [intergranular stress corrosion cracking] shall be used when available.	butt weld heat-affected zone, open to the inside surface, and extending at least 75% through the base metal wall. Intentional overlay fabrication flaws shall not interfere with ultrasonic detection or characterization of the base metal flaws. Specimens containing IGSCC shall be used when available. At least 70% of the flaws in the detection and sizing tests shall be cracks and the remainder shall be alternative flaws. Alternative flaw mechanisms, if used, shall provide crack-like reflective characteristics and shall be limited by the following: (a) The use of alternative flaws shall be limited to when the implantation of cracks produces spurious reflectors that are uncharacteristic of actual flaws. (b) Flaws shall be semielliptical with a tip width of less than or equal to 0.002 inches.
1.1(d)(2) Overlay fabrication flaws. At least 40% of the flaws shall be non-crack fabrication flaws (e.g., sidewall lack of fusion or laminar lack of bond) in the overlay or the pipe-to-overlay interface. At least 20% of the flaws shall be cracks. The balance of the flaws shall be either type.	No Change
(e) Detection Specimens	
1.1(e)(1) At least 20% but less than 40% of the flaws shall be oriented within +/- 20 degrees of the pipe axial direction. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access. The rules of Subarticle IWA-3300 shall be used to determine whether closely spaced flaws should be treated as single or multiple flaws.	At least 20% but less than 40% of the base metal flaws shall be oriented within +/- 20 degrees of the pipe axial direction. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access.
1.1(e)(2) Specimens shall be divided into base and overlay grading units. Each specimen shall contain one or both types of grading units.	Specimens shall be divided into base metal and overlay fabrication grading units. Each specimen shall contain one or both types of grading units. Flaws shall not interfere with ultrasonic detection or

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	characterization of other flaws.
1.1(e)(2)(a)(1) A base grading unit shall include at least 3 inches of the length of the overlaid weld. The base grading unit includes the outer 25% of the overlaid weld and base metal on both sides. The base grading unit shall not include the inner 75% of the overlaid weld and base metal overlay material, or base metal-to-overlay interface.	A base metal grading unit includes the overlay material and outer 25% of the original overlaid weld. The base metal grading unit shall extend circumferentially for at least 1 inch and shall start at the centerline and be wide enough in the axial direction to encompass one half of the original weld crown and a minimum of 0.50 inch of the adjacent base material.
1.1(e)(2)(a)(2) When base metal cracking penetrates into the overlay material, the base grading unit shall include the overlay metal within 1 inch of the crack location. This portion of the overlay material shall not be used as part of any overlay grading unit.	When base metal flaws penetrate into the overlay material, the base metal grading unit shall not be used as part of any overlay fabrication grading unit.
1.1(e)(2)(a)(3) When a base grading unit is designed to be unflawed, at least 1 inch of unflawed overlaid weld and base metal shall exist on either side of the base grading unit. The segment of weld length used in one base grading unit shall not be used in another base grading unit. Base grading units need not be uniformly spaced around the specimen.	Sufficient unflawed overlaid weld and base metal shall exist on all sides of the grading unit to preclude interfering reflections from adjacent flaws.
1.1(e)(2)(b)(1) An overlay grading unit shall include the overlay material and the base metal-to-overlay interface of at least 6 square inches. The overlay grading unit shall be rectangular, with minimum dimensions of 2 inches.	An overlay fabrication grading unit shall include the overlay material and the base metal-to-overlay interface for a length of at least 1 inch.
1.1(e)(2)(b)(2) An overlay grading unit designed to be unflawed shall be surrounded by unflawed overlay material and unflawed base metal-to-overlay interface for at least 1 inch around its entire perimeter. The specific area used in one overlay grading unit shall not be used in another overlay grading unit. Overlay grading units need not be spaced uniformly about the specimen.	Overlay fabrication grading units designed to be unflawed shall be separated by unflawed overlay material and unflawed base metal-to-overlay interface for at least 1 inch at both ends. Sufficient unflawed overlaid weld and base metal shall exist on both sides of the overlay fabrication grading unit to preclude interfering reflections from adjacent flaws. The specific area used in one overlay fabrication grading unit shall not be used in another overlay fabrication grading unit. Overlay fabrication grading units need not be spaced

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	uniformly about the specimen.
1.1(e)(2)(b)(3) Detection sets shall be selected from Table VIII-S2-1. The minimum detection sample set is five flawed base grading units, ten unflawed base grading units, five flawed overlay grading units, and ten unflawed overlay grading units. For each type of grading unit, the set shall contain at least twice as many unflawed as flawed grading units.	Detection sets shall be selected from Table VIII-S2-1. The minimum detection sample set is five flawed base metal grading units, ten unflawed base metal grading units, five flawed overlay fabrication grading units, and ten unflawed overlay fabrication grading units. For each type of grading unit, the set shall contain at least twice as many unflawed as flawed grading units. For initial procedure qualification, detection sets shall include the equivalent of three personnel qualification sets. To qualify new values of essential variables, at least one personnel qualification set is required.
(f) Sizing Specimen	
1.1(f)(1) The minimum number of flaws shall be ten. At least 30% of the flaws shall be overlay fabrication flaws. At least 40% of the flaws shall be cracks open to the inside surface.	The minimum number of flaws shall be ten. At least 30% of the flaws shall be overlay fabrication flaws. At least 40% of the flaws shall be open to the inside surface. Sizing sets shall contain a distribution of flaw dimensions to assess sizing capabilities. For initial procedure qualification, sizing sets shall include the equivalent of three personnel qualification sets. To qualify new values of essential variables, at least one personnel qualification set is required.
1.1(f)(2) At least 20% but less than 40% of the flaws shall be oriented axially. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface which the candidate has physical or visual access.	No Change
1.1(f)(3) Base metal cracking used for length sizing demonstrations shall be oriented circumferentially.	Base metal flaws used for length sizing demonstrations shall be oriented circumferentially.
1.1(f)(4) Depth sizing specimens sets shall include at least two distinct locations where cracking in the base metal extends into the overlay material by at least 0.1 inch in the through-wall direction.	Depth sizing specimen sets shall include at least two distinct locations where a base metal flaw extends into the overlay material by at least 0.1 inch in the through-wall direction.

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Appendix VIII, Supplement 11	PDI Modification
2.0 CONDUCT OF PERFORMANCE DEMONSTRATION	
The specimen inside surface and identification shall be concealed from the candidate. All examinations shall be completed prior to grading the results and presenting the results to the candidate. Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited.	The specimen inside surface and identification shall be concealed from the candidate. All examinations shall be completed prior to grading the results and presenting the results to the candidate. Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited. The overlay fabrication flaw test and the base metal flaw test may be performed separately.
2.1 Detection Test	
Flawed and unflawed grading units shall be randomly mixed. Although the boundaries of specific grading units shall not be revealed to the candidate, the candidate shall be made aware of the type or types of grading units (base or overlay) that are present for each specimen.	Flawed and unflawed grading units shall be randomly mixed. Although the boundaries of specific grading units shall not be revealed to the candidate, the candidate shall be made aware of the type or types of grading units (base metal or overlay fabrication) that are present for each specimen.
2.2 Length Sizing Test	
2.2(a) The length sizing test may be conducted separately or in conjunction with the detection test.	No Change
2.2(b) When the length sizing test is conducted in conjunction with the detection test and the detected flaws do not satisfy the requirements of 1.1(f), additional specimens shall be provided to the candidate. The regions containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the length of the flaw in each region.	No Change
2.2(c) For separate length sizing test, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the length of the flaw in each region.	No Change
2.2(d) For flaws in base grading units, the candidate shall estimate the length of that part of the flaw that is in the outer 25% of the base wall thickness.	For flaws in base metal grading units, the candidate shall estimate the length of that part of the flaw that is in the outer 25% of the base metal wall thickness.

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Appendix VIII, Supplement 11	PDI Modification
2.3 Depth Sizing Test	
For the depth sizing test, 80% of the flaws shall be sized at a specific location on the surface of the specimen identified to the candidate. For the remaining flaws, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region.	(a) The depth sizing test may be conducted separately or in conjunction with the detection test. (b) When the depth sizing test is conducted in conjunction with the detection test and the detected flaws do not satisfy the requirements of 1.1(f), additional specimens shall be provided to the candidate. The regions containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region. (c) For a separate depth sizing test, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region.
3.0 ACCEPTANCE CRITERIA	
3.1 Detection Acceptance Criteria	
Examination procedures, equipment, and personnel are qualified for detection when the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for both detection and false calls. The criteria shall be satisfied separately by the demonstration results for base grading units and for overlay grading units.	a) Examination procedures are qualified for detection when; 1) All flaws within the scope of the procedure are detected and the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for false calls. (a) At least one successful personnel demonstration has been performed meeting the acceptance criteria defined in (b). (b) Examination equipment and personnel are qualified for detection when the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for both detection and false calls. (c) The criteria in (a), (b) shall be satisfied

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Appendix VIII, Supplement 11	PDI Modification
	separately by the demonstration results for base metal grading units and for overlay fabrication grading units.
3.2 Sizing Acceptance Criteria	
Examination procedures, equipment, and personnel are qualified for sizing when the results of the performance demonstration satisfy the following criteria.	No Change
3.2(a) The RMS error of the flaw length measurements, as compared to the true flaw lengths, is less than or equal to 0.75 inch. The length of base metal cracking is measured at the 75% through-base-metal position.	The RMS error of the flaw length measurements, as compared to the true flaw lengths, is less than or equal 0.75 inch. The length of base metal flaws is measured at the 75% through-base-metal position.
3.2(b) All extensions of base metal cracking into the overlay material by at least 0.1 inch are reported as being intrusions into the overlay material.	This requirement is omitted.
3.2(c) The RMS error of the flaw depth measurements, as compared to the true flaw depths, is less than or equal to 0.125 inch.	(b) The RMS error of the flaw depth measurements, as compared to the true flaw depths, is less than or equal to 0.125 inch.

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Request for Relief for Alternative Requirements due to Applicability of ASME Code Case N-649, Alternative Requirements for IWE-5240 Visual Examination In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	MC
Reference:	ASME Code Case N-649 and IWE-5240
Examination Category:	NA
Item Number:	NA
Description:	Applicability of ASME Code Case N-649, Alternative Requirements for IWE-5240 Visual Examination
Component Number:	Various

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The ISI program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition, No Addenda.

3.0 APPLICABLE CODE REQUIREMENT:

IWE-5240, "Visual Examination", requires that a detailed visual examination be performed during any IWE-5220 leakage test on areas affected by repair/replacement activities.

ASME Code Case N-649, Alternative Requirements for IWE-5240 Visual Examination, allows for a VT-3, VT-1, or detailed visual examination depending on the timing of the leakage test.

4.0 REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

The "Applicability Index for Section XI Cases" states that ASME Code Case N-649 is applicable up to and including the 1998 Edition with the 2000 Addenda of ASME Section XI. The Edition/Addenda references in the Code Case text itself also stop at the 1998 Edition with the 2000 Addenda. However, the requirements of Paragraph IWE-5240 are identical in both the 1998 Edition with the 2000 Addenda and the 2004 Edition, No

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Addenda. Paragraph IWE-5240 requires that a detailed visual examination of repaired areas be completed during a post repair pressure test. TMI has a concrete containment with a metal liner that is inaccessible <u>during</u> a post repair pressure test. ASME Code Case N-649 was issued to allow this visual examination to be performed during or after the pressure test in recognition of the impracticality of performing the visual examinations of concrete containment liners during the post repair pressure test. ASME did not address this impracticality in the Code until the 2004 Edition with the 2006 Addenda of ASME Section XI was issued, so ASME Code Case N-649 is actually needed for the 2004 Edition, No Addenda of ASME Section XI.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

TMI, Unit 1 requests the applicability of ASME Code Case N-649 be extended to the 2004 Edition, No Addenda for use in the plant's fourth ISI interval. USNRC Regulatory Guide 1.147, Revision 15, lists ASME Code Case N-649 as acceptable for use with no conditions or limitations. The only issue being addressed by this relief request is the applicability listed in the "Applicability Index for Section XI Cases."

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the fourth ten-year ISI interval for TMI, Unit 1.

7.0 PRECEDENTS:

None.

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***** NOTE *****

Third ISI Interval Relief Request RR-09-01, Revision 0, was previously authorized under the Third Interval ISI Program Plan in a SER from the USNRC to TMI, Exelon Nuclear, dated September 21, 2010. Relief was authorized for the deferral of the volumetric examination for the Reactor Vessel welds (Examination Categories B-A and B-D). The deferral of these examinations applies until the Fall 2015 refueling outage T1R21. At that time, TMI will have to resubmit the requested alternative for future intervals after the results of the Fall 2015 refueling outage T1R21 examinations have been evaluated. All ASME Section XI references were made in accordance with the 1995 Edition through the 1996 Addenda of ASME Section XI. No changes to the actual approved relief request have been made and no further or revised submittal is required.

RR-09-01
Proposed Alternative In Accordance with 10 CFR 50.55a(a)(3)(i)
Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

These examination categories and item numbers are from Table IWB-2500-1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI:

Examination Category	Item No.	Component	Description
B-A	B1.11	RCT0001RV0012	Circumferential Shell Welds
		RCT0001RV0013	
		RCT0001RV0014	
		RCT0001RV0015	
B- A	B1.12	RCT0001RV0018L	Longitudinal Shell Welds
		RCT0001RV0013L	
		RCT0001RV0020L	
		RCT0001RV0021L	
B-A	B1.21	RCT0001RV0016	Circumferential Head Welds
B-A	B1.30	RCT0001RV0011	Shell-to-Flange Weld
B-A	B1.51	RCT0001A12052	Beltline Region Repair Weld
B-D	B3.90	RCT0001RV0001N	Nozzle-to-Vessel Welds
		RCT0001RV0002N*	
		RCT0001RV0003N	
		RCT0001RV0004N	
		RCT0001RV0005N*	
		RCT0001RV0006N	

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		RCT0001RV0007N RCT0001RV0008N	
B-D	B3.100	RCT0001RV0001N RCT0001RV0002N* RCT0001RV0003N RCT0001RV0004N RCT0001RV0005N* RCT0001RV0006N RCT0001RV0007N RCT0001RV0008N	Nozzle Inner Radius Areas

(Throughout this request the above examination categories are referred to as "the subject examinations" and ASME Section XI, is referred to as "the Code.")

* Note that the Third ISI Interval examinations for Identification RCT0001RV0002N and RCT0001RV0005N were completed in the 2001 refueling outage. The remaining examinations during the 2001 refueling outage were credited as Second ISI Interval examinations. This is allowed per ASME Section XI, Paragraph IWA-2430(d)(2).

2. Applicable Code Edition and Addenda

The applicable Code Edition and Addenda for TMI is the ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 Edition with Addenda through 1996 (Reference 1) for the Third ISI Interval. The applicable ASME Section XI Edition and Addenda for subsequent intervals will be as required by 10 CFR 50.55a.

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor vessel pressure retaining welds identified in Table IWB-2500-1 once each ten year interval. The TMI Third ISI Interval is scheduled to end April 19, 2011; however, TMI will be applying ASME Section XI, Paragraph IWA-2430(d)(1) to extend the current interval through the Fall 2011 refueling outage (T1R19).

4. Reason for Request

An alternative is requested from the requirement of Paragraph IWB-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor vessel pressure retaining, Examination Category B-A and B-D welds, be performed once each ten-year interval. Extension of the inspection interval for Examination Category B-A and B-D welds

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from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Exelon Generation Company, LLC (EGC) proposes to defer the ASME Section XI required volumetric examination of the TMI reactor vessel full penetration pressure retaining Examination Category B-A and B-D welds for the Third ISI Interval (Fourth ISI Interval for two nozzles identified in Section 1 above), currently scheduled for 2011. EGC currently proposes to perform the required volumetric examinations during the Fall 2015 refueling outage concurrent with other scheduled in-vessel examinations that will occur at that time. The 2015 outage is three outages earlier than that provided to the Staff in the PWROG letter 06-06-356 (Reference 2) and is a deviation to the PWROG letter. This deviation is needed so that license renewal commitments (Reference 3) to perform inspections of the reactor vessel internals can be satisfied while minimizing the number of times that removal of the reactor vessel internals is needed.

In accordance with 10 CFR 50.55a(a)(3)(i), an alternate inspection interval is requested on the basis that the current inspection interval can be extended based on a negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.1 74 (Reference 4).

The methodology used to demonstrate the acceptability of extending the inspection intervals for Examination Category B-A and B-D welds based on a negligible change in risk is contained in WCAP-16168-NP-A, Revision 2 (Reference 5). This methodology was used to develop a pilot plant analysis for Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs and is an extension of the work that was performed as part of the USNRC's Pressurized Thermal Shock (PTS) Risk Re-Evaluation (Reference 6). The critical parameters for demonstrating that this pilot plant analysis is applicable on a plant specific basis, as identified in WCAP-16168-NP-A, Revision 2, are identified in Table 1. By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology to the TMI reactor vessel is acceptable as shown in Table 1 below.

Table 1 Critical Parameters for Application of the Bounding Analysis

Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required?**
Dominant Pressurized Thermal Shock (PTS) Transients in the USNRC	USNRC PTS Risk Study (Reference 6)	PTS Generalization Study (Reference 7)	No

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Table 1 Critical Parameters for Application of the Bounding Analysis

Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required?**
PTS Risk Study are applicable			
Through Wall Cracking Frequency (TWCF)	4.42E-07 Events per year (Reference 5)	1.81E-12 events per year (calculated per Reference 5)	No
Frequency and Severity of Design Basis Transients	12 heatup/cooldowns per year (Reference 5)	Bounded by 12 heatup/cooldowns per year	Yes (As required by Reference 5 for B&W plants. See Appendix A).
Cladding Layers (Single/Multiple)	Single Layer (Reference 5)	Single Layer	No

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Additional information relative to the TMI reactor vessel inspections is provided in Table 2. This information confirms that satisfactory examinations have been performed on the TMI reactor vessel.

Table 2	**Additional Information Pertaining to the Reactor Vessel Inspections**
Inspection methodology:	Past inspections have been performed per the requirements of Regulatory Guide 1.150 (Reference 8) and ASME Section XI. The reactor vessel beltline region welds have also been inspected in accordance with Appendix VIII of ASME Section XI. Future inspections will be in accordance with Appendix VIII of ASME Section XI and/or other equivalent requirements approved by the USNRC.
Number of past inspections:	Two (2) inspections have been performed to date with the exception that three (3) examinations have been completed on two nozzles identified in Section 1 above.
Number of indications found:	26 recordable indications were detected in the reactor vessel beltline region during the most recent inservice inspection. All indications were acceptable per Table IWB-3510-1 of ASME Section XI. Only one indication is in the inner 1" of the reactor vessel thickness. This indication has a through-wall extent of 0.14". Based on the area of plate inspected in the beltline region, 42 flaws of this size would be acceptable per the proposed PTS rule (Reference 9).
Proposed inspection schedule for balance of plant life:	The next ten year inservice inspection is currently required to be completed no later than the 2011 refueling outage. This inspection will be performed in the 2015 refueling outage.

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Table 3 provides additional information relative to the calculation of the TWCF for TMI.

Table 3 Details of TWCF Calculation for 60 EFPY of Operation

Inputs

Reactor Coolant System Temperature, T_{RCS} [°F]:	N/A	T_{wall} [inches]:	8.565

#	Region/Component Description	Material/Flux Type	Cu [wt%]	Ni [wt%]	CF [°F]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Nozzle Belt Forging ARY-59	A508 Cl. 2	0.08	0.72	51	3	2.118
2	Upper Shell Plate C2789-1	SA-302 Gr BM	0.09	0.57	58	1	2.273
3	Upper Shell Plate C2789-2	SA-302 Gr BM	0.09	0.57	58	1	2.273
4	Lower Shell Plate C3307-1	SA-302 Gr BM	0.12	0.55	82	1	2.274
5	Lower Shell Plate C3251-1	SA-302 Gr BM	0.11	0.50	73	1	2.274
6	Nozzle Belt to Upper Shell Circ. Weld WF-70	ASA/Linde 80	0.32	0.58	199.3	-31.1	2.118
7	Upper Shell Axial Weld WF-8	ASA/Linde 80	0.19	0.57	167	-47.6	1.513
8	Upper Shell Axial Weld WF-8	ASA/Linde 80	0.19	0.57	167	-47.6	1.513
9	Upper to Lower Shell Circ. Weld WF-25	ASA/Linde 80	0.34	0.68	220.6	-74.3	2.208
10	Lower Shell Axial Weld SA-1526	ASA/Linde 80	0.34	0.68	220.6	-74.3	1.358
11	Lower Shell Axial Weld SA-1526	ASA/Linde 80	0.34	0.68	220.6	-74.3	1.358

Outputs

Methodology Used to Calculate ΔT_{30}:	Regulatory Guide 1.99, Revision 2

	Controlling Material Region # (From Above)	RT_{MAX-XX} [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	Fluence Factor	ΔT_{30} [°F]	$TWCF_{95-XX}$
Axial Weld - AW	10, 11	624.75	1.358	1.085	239.36	3.92E-13
Circumferential Weld - CW	6	668.56	2.118	1.204	239.97	2.90E-13
Plate - PL	4	560.92	2.274	1.222	100.23	6.92E-14
Forging - FO	1	524.10	2.118	1.204	61.41	1.87E-15
$TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW} + \alpha_{FO}TWCF_{95-FO}$):						1.81E-12

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6. Duration of Proposed Alternative

EGC proposes to perform the required volumetric examinations during the Fall 2015 refueling outage concurrent with other scheduled in-vessel examinations that will occur at that time. This request is for the remainder of the current Third ISI Interval, and subsequent ten year interval.

7. Precedents

1. "Donald C. Cook Nuclear Plant, Unit 2 (CNP-2) - Evaluation of Relief Request (ISIR-29) to Extend the Third 10-Year Inservice Inspection (ISI) Interval for Reactor Vessel Weld Examination (TAC MD9934)," dated June 8, 2009 (ML091260163).

2. "Safety Evaluation For Relief Requests ISI-020 & 021 Reactor Vessel Weld Examination Extension - Calvert Cliffs Nuclear Power Plant, Unit No. 2 (TAC Nos. MD9773 AND MD9774), dated April 8, 2009 (ML090920077).

3. "Palisades Plant - Evaluation of Relief Request to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examination (TAC NO. MD9265)," dated February 11, 2009 (ML090120896).

4. "R.E. Ginna Nuclear Power Plant: Safety Evaluation for Relief Request No. 18, Reactor Vessel Weld Examination Extension (TAC NO. MD9962)," dated July 31, 2009 (ML092080229).

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 Edition with Addenda through 1996.

2. OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." MUHP 5097-99, Task 2059," October 31, 2006.

3. "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1," Appendix A: Long Term Commitments for License Renewal of TMI-1, Commitment No. 36, dated June 2009.

4. USNRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.

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5. Pressurized Water Reactor Owners Group Topical Report, WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," dated June 2008.

6. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 1, 2007.

7. USNRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004.

8. USNRC Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," February 1983. Used for prior examinations.

9. SECY-07-0104, "Proposed Rulemaking - Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," June 25, 2007 (ML070570141).

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Appendix A: Assessment of Design Basis Transients

WCAP-16168-NP-A, Revision 2, states the following:

"Licensees with B&W plants must (a) verify that the fatigue crack growth of 12 heat-up/cool-down transients per year that was used in the PWROG fatigue analysis bound the fatigue crack growth for all its design basis transients and (b) identify the design basis transients that contribute to significant fatigue crack growth."

In accordance with this requirement, an evaluation was performed of the TMI design basis transients. The listing of design basis transients for TMI is shown in Table A-1.

Table A-1 TMI Reactor Vessel Design Basis Transients for 40 Years

Transient Description	# of Evaluated Cycles
Normal Operating Transients	
Heatup from 70°F to 8% Full Power and Cooldown from 8% Full Power	240
Power Changes between 0% and 15% Power	1440
Power Loading and Unloading between 8% and 100% Power	18,000
10% Step Load Increase and Decrease	8,000
Feed and Bleed Operations	4,000
Test Transients - High Pressure Injection System and Core Flooding Check Valve	288
Steam Generator Filling, Draining, Flushing, and Cleaning	540
Upset Transients	
Step Load Reduction	310
Reactor Trip	400
Rapid Depressurization	40
Change of Flow	20
Rod Withdrawal Accident	40
Control Rod Drop	40
Loss of Station Power	40

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Table A-1 TMI Reactor Vessel Design Basis Transients for 40 Years	
Transient Description	**# of Evaluated Cycles**
Loss of Feedwater to One Steam Generator	20
Loss of Feedwater Heater	40
Emergency Transients	
Stuck Open Turbine Bypass Valve	10
Faulted Transients	
Steam Line Failure	1
Loss of Coolant	1
Tests	
Hydrotest	20
Hot Functional Testing	1

Previous flaw tolerance evaluations of the reactor vessel have shown that only the following design basis transients generate large enough changes in stress and/or occur with enough frequency to contribute to fatigue crack growth:

- Cooldown from 8% full power
- Rapid depressurization
- Power changes between 15% to 0% power
- Power loading and unloading between 8% and 100% power
- Loss of feedwater to one steam generator
- Rod withdrawal accident
- Control rod drop

To determine whether the TMI reactor vessel was bounded by the pilot plant analysis, the surface breaking flaw density from the B&W pilot plant "10-Year ISI Only" case (Page K-3 of WCAP-16168-NP-A, Revision 2) was compared against the surface breaking flaw density calculated based on the TMI design basis transients identified above as contributors to fatigue crack growth. The following process, which is consistent with the methods described on pages 3-14 to 3-16 of WCAP-16168-NP-A, Revision 2, was used to make this comparison:

- The pressure, temperature, and film coefficient versus time histories for each transient were input into the FAVLOAD Module of the FAVOR Code along with TMI specific

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dimensions of the reactor vessel. The FAVLOAD module generated stress intensity factors as a function of flaw depth for input to the PROBSBFD Code.

- The PROBSBFD Code was used to determine the average surface flaw density resulting from the design basis transients analyzed with the FAVLOAD module. One run was performed for each of the design basis transients. The PROBSBFD Code was used to generate the surface breaking flaw density distribution files used in the WCAP-16168-NP-A, Revision 2. The following were used as inputs to the PROBSBFD Code:

  - o The initial flaw depth was set to the TMI cladding thickness rounded up to the nearest 1% of the total vessel wail thickness. To determine this depth, a cladding thickness of 0.188" was used in combination with a base metal thickness of 8.44". This combination was used because it results in the deepest initial flaw depth as a percentage of the wall thickness. This equated to 2.18%, which was rounded up to 3% and was consistent with the initial flaw depth used for the B&W pilot plant. PROBSBFD inputs for the B&W pilot plant can be found in Appendix K of WCAP-16168-NP-A, Revision 2.

  - o The inservice inspection input was chosen such that one inservice inspection was performed at 10 years and no subsequent inservice inspections were performed. This corresponded to the "10-Year ISI Only" case evaluated in the WCAP pilot plant analyses.

  - o The number of transients per year was chosen based on the design number per year over a 40 year plant life as identified in Table A-1. However, consistent with the pilot plant analyses, the fatigue crack growth analysis was performed for 80 calendar years, or for twice the number of transients in Table A-1.

  - o All other inputs were consistent with those used in WCAP-16168-NP-A, Revision 2.

- The average of the 1000 surface breaking flaw densities, which is output by PROBSBFD, for each design basis transient and, for their respective number of cycles per year, were added together to obtain a total average surface breaking flaw density distribution. This was done by combining the density at each specific amount of flaw growth, for each aspect ratio, for each transient.

The total average surface breaking flaw density distribution was then compared to the average distribution used in the B&W pilot plant evaluation (Page K-3 of WCAP-16168-NP-A, Revision 2) at each specific amount of flaw growth for each aspect ratio. As shown in Table A-2, the TMI surface breaking flaw densities are bounded by the values in the B&W pilot plant distribution. Therefore, it can be concluded that the fatigue crack growth of 12 heat-up/cool-down transients

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per year used in the PWROG fatigue analysis bounds the fatigue crack growth for the TMI reactor vessel design basis transients.

Table A-2 Comparison of TMI and B& W Pilot Plant Average Surface Breaking Flaw Density Distributions

("Y" Indicates TMI Flaw Density Bounded by Pilot Plant Flaw Density)

Flaw Growth#	**Aspect Ratio (l/a)**			
(a/t)	**2**	**6**	**10**	**99**
+0.25%	Y	Y	Y	Y
+0.50%	Y	Y	Y	Y
+0.75%	Y	Y	Y	Y
+1.00%	Y	Y	Y	Y
+1.25%	Y	Y	Y	Y
+1.50%	Y	Y	Y	Y
+1.75%	Y	Y	Y	Y
+2.00%	Y	Y	Y	Y
+2.25%	Y	Y	Y	Y
+2.50%	Y	Y	Y	Y
+2.75%	Y	Y	Y	Y
+3.00%	Y	Y	Y	Y

It should be noted that the TMI transients have been evaluated for license renewal and it has been determined that the actual number of transient occurrences is not expected to exceed the corresponding number in the design basis duty cycle before the end of the period of extended operation. Furthermore, TMI has set reduced administrative limits on the design basis transients to meet requirements with regards to fatigue on plant components other than the reactor vessel. The TMI fatigue management program will track these transient occurrences to ensure that these administrative limits are not violated. Given these administrative limits, and the fact that the TMI fatigue crack growth analysis was done for 80 calendar years, the analysis discussed above is conservative.

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***** NOTE *****

Third ISI Interval Relief Request RR-09-02, Revision 0, was previously authorized under the Third Interval ISI Program Plan in a SER from the USNRC to TMI, Exelon Nuclear, dated September 21, 2010. Relief was authorized for the deferral of the visual examination for the Reactor Vessel internals (Examination Categories B-N-2 and B-N-3). The deferral of these examinations applies until the Fall 2015 refueling outage T1R21. At that time, TMI will have to resubmit the requested alternative for future intervals after the results of the Fall 2015 refueling outage T1R21 examinations have been evaluated. All ASME Section XI references were made in accordance with the 1995 Edition through the 1996 Addenda of ASME Section XI. No changes to the actual approved relief request have been made and no further or revised submittal is required.

RR-09-02
Proposed Alternative In Accordance with 10 CFR 50.55a(a)(3)(ii)
Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

These examination categories and item numbers are from Table IWB-2500-1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI:

Examination Category	Item No.	Description
B-N-2	B13.50	Interior Attachments Within Beltline Region
B-N-3	B13.70	Core Support Structure

(Throughout this request the above examination categories are referred to as "the subject examinations" and ASME Section XI is referred to as "the Code".)

2. Applicable Code Edition and Addenda

ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 Edition with Addenda through 1996 (Reference 1).

3. Applicable Code Requirement

In accordance with Paragraph IWA-2430(d)(1), each inspection interval may be reduced or extended by as much as one year. Adjustments shall not cause successive intervals to be altered more than one year from the original pattern of intervals. Additionally, Table IWB-2500-1, Examination Categories B-N-2 and B-N-3, Item Numbers B13.50 and B13.70

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require visual examination of the accessible interior attachment welds within and beyond the beltline region and a visual examination of the accessible core support structure surfaces of the Reactor Pressure Vessel (RPV) once each ten year interval. The TMI Third ISI Interval is scheduled to end April 19, 2011; however, TMI will be applying ASME Section XI, Paragraph IWA-2430(d)(1) to extend the current interval through the Fall 2011 refueling outage.

4. Reason for Request

In Pressurized Water Reactor Owners Group (PWROG) Topical Report WCAP-1616-NP-A, Revision 2 (Reference 2), the PWROG provided the technical and regulatory basis for decreasing the frequency of inspections by extending the ASME Section XI ISI interval from the current 10 years to 20 years for Examination Categories B-A and B-D RPV welds. The USNRC approved the topical report by letter dated May 8, 2008 (Reference 3). To implement the change presented in Reference 2, Exelon Generation Company, LLC (EGC) is submitting RR-09-01, in accordance with the Safety Evaluation (Reference 3) to request an alternative from the Code requirements pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the alternative inspection interval (20 years) provides an acceptable level of quality and safety. In RR-09-01, EGC identified 2015 as the year in which future inspections of the Examination Categories B-A and B-D RPV welds are scheduled to be performed. The intent of this relief request (RR-09-02) is to allow deferral of the subject examinations to the same time (2015 refueling outage) as the Examination Categories B-A and B-D RPV welds described in RR-09-01.

During the ten year inservice inspection of the RPV shell, lower head, and nozzle welds in 2001, TMI also performed visual examinations of the RPV interior attachments and the core support structure. Since the core support structure (called a core support assembly on B&W designed plants) requires removal to facilitate examination of the RPV shell, lower head, and nozzle welds, the visual examinations of Examination Categories B-N-2 and B-N-3 have historically been performed during the same outage at the end of the ISI interval.

TMI has also committed to the development and implementation of a plant specific Reactor Vessel Internals (RVI) inspection program and subsequent submittal to the USNRC two years prior to the period of extended operation (Reference 5). TMI may elect to perform the enhanced examinations for the RVI inspection program coincident with the core barrel removal in 2015. To complete the full scope of the RVI examination it is expected to require a complete core offload and removal of all internals to facilitate implementation of the examinations. Portions of the RVI inspection may be performed prior to this time as may be prescribed in that program.

Performing examinations related to core barrel removal during the same refueling outage will result in significant savings in dose and outage duration since the same equipment and

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personnel used for visual and volumetric examination of the RPV shell welds and nozzle welds from the RPV interior can be used to implement the required RVI examinations. Additionally, removing the RPV internals only once to accommodate all the examinations discussed in this relief request will result in significant savings in radiation exposure.

5. Proposed Alternative and Basis for Use

The Third ISI Interval for TMI began on April 20, 2001 and is scheduled to conclude on April 19, 2011. With the allowed one-year extension of Paragraph IWA-2430(d)(1), the interval may be extended to April 19, 2012.

TMI proposes to perform the subject Examination Category B-N-2 and B-N-3 examinations for the Third ISI Interval prior to exiting the Fall 2015 refueling outage. Completing examinations no later than the 2015 refueling outage also assures compliance with License Renewal Commitments to perform RVI examinations. MRP-227 ("Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," Revision 0, December 2008 (Reference 4)) requires the RVI examinations to be completed within two refueling outages from the beginning of the period of extended operation (by Fall 2017). The proposed alternative inspection schedule would enable the subject examinations to be performed during the 2015 refueling outage with the risk-informed extension of the RV inservice inspection and other required inspections. In accordance with 10 CFR 50.55a(a)(3)(ii), this alternative is requested on the basis that performing the examination of the RPV interior attachments and core support structure separate in time from the RPV shell, lower head, and nozzle welds would result in hardship or unusual difficulty without a compensating increase in quality or safety.

The examinations required by Examination Categories B-N-2 and B-N-3 require the removal of all the fuel and the core barrel from the RPV. An unnecessary risk is created by removal of the core barrel to perform a visual examination without a compensating increase in quality or safety. Further, the radiation exposure to establish the conditions for and perform the Examination Categories B-N-2 and B-N-3 examinations would essentially double if the subject examinations were performed separate in time from the RPV shell, lower head, and nozzle weld examinations.

The ASME Section XI visual examinations of the RPV interior attachments and the core support assembly have been performed several times at TMI with no relevant indications noted during the examinations. The examinations were last performed during the 2001 refueling outage with acceptable results.

As stated in Reference 2, "*...it must be recognized that all reactor coolant pressure boundary failures occurring to date have been identified as a result of leakage, and were discovered by visual examination. The proposed RV ISI interval extension does not alter the*

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visual examination interval. The reactor vessel would undergo, as a minimum, the ASME Section XI Examination Category B-P pressure tests and visual examinations conducted at the end of each refueling before plant start-up, as wel as leak tests with visual examinations that precede each start-up following maintenance or repair activities." The visual examinations discussed in Reference 2 are not the subject examinations (i.e., B-N-2 and B-N-3) of this relief request. During the 2011 and 2013 refueling outages, TMI will perform the required Examination Category B-N-1 visual examinations. These examinations will include the space that is made accessible for examination by the removal of components during normal refueling outages. These examinations are required once each period and will provide reasonable assurance of structural integrity. As discussed further in Reference 2, defenses against human errors are preserved with the increase in inspection interval. Specifically, the increase in the inspection interval reduces the frequency for which the RV lower internals need to be removed thereby reducing the possibility for human error and damage to the core support assembly.

Therefore, in accordance with 10 CFR 50.55a(a)(3)(ii), this change in inspection timing from 2011 to 2015 for the subject examinations is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

6. Duration of Proposed Alternative

EGC currently proposes to perform the required B-N-2 and B-N-3 examinations during the Fall 2015 refueling outage concurrent with other scheduled in-vessel examinations that will occur at that time. RVI examinations in outages subsequent to the 2015 refueling outage will be in accordance with ASME Section XI and the TMI RVI inspection program. This request is for the remainder of the current Third ISI Interval, and the next ten year interval.

7. Precedents

1. "Safety Evaluation For Relief Requests ISI-020 & 021 Reactor Vessel Weld Examination Extension - Calvert Cliffs Nuclear Power Plant, Unit No. 2 (TAC Nos. MD9773 AND MD9774)," dated April 8, 2009 (ML090920077).

2. "Donald C. Cook Nuclear Plant, Unit 2 (CNP-2) - Evaluation of Relief Request (ISIR-30) To Extend the Third 10-Year Inservice Inspection (ISI) Interval for Visual Examination of the Reactor Pressure Vessel Interior Attachments Beyond the Beltline Region and Core Support Structure (TAC NO. ME0770)," dated June 8, 2009 (ML091320549).

8. References

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1. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 Edition with Addenda through 1996.

2. Pressurized Water Reactor Owners Group Topical Report, WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," dated June 2008.

3. Letter from H. K. Nieh (USNRC) to G. Bischoff (PWROG), dated May 8, 2008, "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16168-NP, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (TAC No. MC9768)," dated May 8, 2008.

4. MRP-227 "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," Revision 0, December 2008.

5. "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1," Appendix A: Long Term Commitments for License Renewal of TMI-1, Commitment No. 36, dated June 2009.

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Request for Relief for Alternative Requirements of Full Structural Weld Overlays (SWOLs) of the Pressurizer Spray Dissimilar Metal Welds In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED

Code Class:	1
Reference:	IWA-4000, *"Repair/Replacement Activities"*
Examination Category:	R-A
Item Number:	See Table 1A for listing
Description:	Full SWOLs of the Pressurizer Spray Dissimilar Metal Welds
Component Number(s):	See Table 1A for listing
Drawing Number(s):	1D-ISI-RC-005 1D-ISI-RC-012

2.0 APPLICABLE CODE EDITION AND ADDENDA

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2004 Edition, no Addenda.
2. Pressurizer Code of Construction, ASME Code Section III, 1965 Edition, through Summer 1967 Addenda.
3. USAS B31.7, Nuclear Power Piping, February 1968 Draft Including June 1968 Errata.
4. USAS B31.1, Power Piping, 1967 Edition.

3.0 APPLICABLE CODE REQUIREMENT

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2004 Edition no Addenda, IWA-4000, *"Repair/Replacement Activities"*.
2. Nuclear Regulatory Commission (NRC) conditionally approved Code Case N-504-3, *"Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1,"* with condition as specified in Regulatory Guide (RG) 1.147, Revision 15.
3. NRC conditionally approved Code Case N-638-1, *"Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique,*

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Section XI, Division 1," with condition as specified in Regulatory Guide (RG) 1.147, Revision 15.

4. ASME Code, Section XI, 2001 Edition, no Addenda, Appendix VIII, Supplement 11, *"Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds".*

4.0 REASON FOR REQUEST

Dissimilar metal welds on U.S. Pressurized Water Reactor (PWR) Reactor Coolant Systems (RCS) often consist of Alloy 82/182 weld material to connect stainless steel pipe and safe-ends to vessel and piping nozzles, which are generally constructed using carbon or low alloy ferritic steel. These welds have shown a propensity for Primary Water Stress Corrosion Cracking (PWSCC) degradation, especially in components subjected to higher operating temperatures, such as the pressurizer or hot leg.

Three Mile Island Nuclear Station (TMI), Unit 1 plans to install a full structural weld overlay (SWOL) on the pressurizer (PZR) spray nozzle to safe-end and safe-end to elbow dissimilar metal welds during the T1R19 refueling outage in Fall 2011.

Exelon Generation Company, LLC (Exelon) is taking a proactive approach in addressing Alloy 600 PWSCC degradation by applying a preemptive SWOL to the pressurizer spray nozzle to safe-end and safe-end to elbow dissimilar metal welds. SWOLs have been used for several years on both Boiling Water Reactors (BWR) and PWRs to arrest existing (or postulated) flaws from propagating while establishing a new structural pressure boundary. In some cases, SWOLs have been used to reestablish structural integrity of a dissimilar metal weld containing through wall leaking flaws. In addition to proactively mitigating PWSCC in the dissimilar metal weld, the SWOL can provide an acceptable geometry that allows ultrasonic examination in accordance with Performance Demonstration Initiative (PDI), Materials Reliability Program (MRP) (Reference 6), and ASME Section XI requirements.

The welding will be utilizing a mechanized Gas Tungsten Arc Welding (GTAW) process and the ambient temperature temper bead method with ERNiCrFe-7A (referred to as Alloy 52M in subsequent discussion in this document) weld metal. When temper bead welding is not required, manual GTAW with Alloy 52M may be used if local repairs of weld defects are necessary or if additional weld metal is required locally to form the final SWOL contour. Shielded metal arc welding (SMAW), using Alloy 152 (ENiCrFe-7), would only be used to repair indications in the existing dissimilar metal welds prior to overlay initiation.

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As discussed herein, there is no comprehensive criterion for a licensee to apply a SWOL repair to a dissimilar metal weld that is constructed of Alloy 82/182 weld material. The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2004 Edition, no Addenda, IWA-4000, is used for the TMI, Unit 1 Section XI Repair/Replacement Program, but it does not contain the needed requirements for this type of weld overlay repair.

Repair/replacement activities associated with weld overlays of this type are required to address materials, welding parameters, personnel radiation exposure concerns, operational constraints, examination techniques, and procedural requirements.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE

Pursuant to 10 CFR 50.55a(a)(3)(i), Exelon proposes applying a SWOL designed in accordance with Code Case N-504-3 (Reference 1) including NRC conditions identified in Regulatory Guide 1.147, Revision 15 (References 2 and 4) with the modifications proposed in Table 2 of this relief request. In addition, Code Case N-638-1 (Reference 3) (including NRC conditions identified in Regulatory Guide 1.147 (Reference 2)) for temper bead welding will be used with modifications proposed in Table 3 of this relief request. Final ultrasonic (UT) examination of the finished SWOL will be performed using Electric Power Research Institute (EPRI) PDI demonstrated ultrasonic examination procedures and personnel in lieu of the ASME Section XI, Appendix VIII, Supplement 11 (Reference 5) as proposed in an associated relief request (Reference 13).

Code Case N-504-3 is currently approved for use in RG 1.147, Revision 15 with additional conditions that ASME Code Section XI, 2004 Edition with 2005 Addenda, Appendix Q, be applied to use of the Code Case. The PZR spray nozzle SWOL, described herein, will extend around the full circumference of the dissimilar metal welds as required by Code Case N-504-3. The specific thickness and length will be determined according to the guidance provided in Code Case N-504-3 and Regulatory Guide 1.147, Revision 15. The SWOL will completely cover the dissimilar metal welds, Inconel 600 safe-end, ferritic steel nozzle, and adjacent stainless steel material with Alloy 52M material to the extent that PWSCC susceptible material is mitigated and examination capability is maintained for adjacent welds. The purpose for this SWOL approach is to produce a single weld overlay covering both welds thereby also providing the weld geometry required to perform the final volumetric examinations and obtain the required examination volume coverage.

Prior to installation of the SWOL, TMI, Unit 1 will complete a bare metal visual examination of the nozzle to safe-end and safe-end to elbow dissimilar metal welds immediately after the insulation is removed in the area around the nozzle and dissimilar metal weld areas to ensure that no through-wall cracks exist prior to applying the overlay.

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The visual examination will be completed in accordance with 10 CFR 50.55a(g)(6)(ii)(E). Exelon intends to complete a PDI qualified UT examination of the dissimilar metal welds prior to application of this overlay.

Per N-504-3, Paragraph (d), a liquid penetrant (PT) examination will be performed of the overlay area with an acceptance criteria that no indication greater than 1/16" is permitted. If any indication is found greater than 1/16", the indication will be removed or reduced below the acceptance criteria, and the PT performed again. If any indication(s) do require repair, the repair will be completed and the area will again have a PT completed for final acceptance. Refer to Question 1 from the Exelon Response to a Request for Additional Information (RAI) concerning additional information on PT examinations (Reference 15).

Per N-504-3, flaw evaluations and shrinkage stress effects analyses will be addressed through the approved overlay designs that are currently in development.

These documents will be completed and approved for use prior to application at TMI, Unit 1. Full structural weld overlays are considered physical plant changes and will be approved in accordance with Exelon procedures.

Monitoring of preheat and interpass temperature is necessary to assure the field welding heat input remains within qualified parameters. Weld preheat and heat input are monitored through the use of calibrated contact pyrometers or thermocouples. The interpass temperature is measured multiple times within each layer.

Subsequent inservice examinations will be scheduled and performed in accordance with the requirements of Nonmandatory Appendix Q or alternate schedules accepted by the NRC (for example adoption of ASME Code Case N-770 as proposed by 75FR24324). If a PDI qualified UT examination of welds SP-021BM and PR-009BM is completed prior to application of the weld overlay and no PWSCC or other indications unacceptable to ASME Section XI are identified, the welds will be considered "uncracked" per Code Case N-770. If a pre-overlay PDI qualified UT examination is not completed, or PWSCC is detected, or other ASME Section XI unacceptable indications are identified, then the overlay repaired welds will be considered "cracked" per Code Case N-770. Regardless of pre-overlay UT examination status or results, the applied weld overlay will be a full structural weld overlay (assumed circumferential flaw that is 100% through wall extending around the entire circumference of the weld). The design life flaw growth analysis will assume that a 75% through-wall flaw exists at the time of weld overlay deposit, based on flaw detection capability in the outer 25% through-wall of the original weld. If a flaw is detected in the outer 25% through-wall of the repaired welds the design life flaw growth analysis will be revised to address the actual flaw condition detected as required by ASME Section XI, Appendix Q. In all cases the overlay design will comply

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with Code Case N-504-3, including ASME Section XI, Appendix Q requirements, as specified in this relief.

Exelon intends to install one or more weld barrier layer(s) to prevent hot cracking in stainless steel materials similar to that installed during the 2007 weld overlays (Reference 15).

6.0 DURATION OF THE PROPOSED ALTERNATIVE

The duration of the proposed alternative associated with the SWOL is the remaining service life of the components including future plant life extension. Relief from the fourth ten-year ISI interval ASME Section XI, Appendix VIII, Supplement 11 inspection requirements is addressed in Reference 13 and is applicable to this overlay application. The fourth ten-year ISI interval is scheduled to begin on April 20, 2011.

7.0 PRECEDENTS

A similar relief request was approved for use in the Reference 14 NRC Safety Evaluation Report. However, this overlay was not performed in the 2007 refueling outage and is being resubmitted for approval.

Additionally, similar relief requests for SWOLs of dissimilar metal weld (both PWR and BWR) have been approved for a number of units throughout the industry (see References 7 through 10). A number of units have submitted relief requests citing similar proposed relief request methodology. These relief requests were associated with welding over detected or postulated flaws outside the acceptance criteria of Section XI utilizing proposed modifications to existing Code Cases N-504 and N-638 and NRC conditions for use.

8.0 REFERENCES

1. ASME Code Case N-504-3, *"Alternative Rules for Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping,"* dated August 4, 2004.
2. Regulatory Guide 1.147, *"Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,"* Revision 15, October 2007.
3. ASME Code Case N-638-1, *"Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique,"* dated February 13, 2003.
4. American Society of Mechanical Engineers (ASME) Section XI Nonmandatory Appendix Q, *"Nonmandatory Appendix Q Weld Overlay Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Weldments,"* 2004 Edition with 2005 Addenda.

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5. ASME Code, Section XI, 2001 Edition, no Addenda, Appendix VIII, Supplement 11, *"Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds"*.

6. *"Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139, Revision 1),"* EPRI Report Number 1015009, Final Report, December 2008.

7. Letter from Richard Laufer (NRC) to Christopher M. Crane (AmerGen Energy Company, LLC), *"Three Mile Island Nuclear Station, Unit 1 (TMI-1) Request for Relief From Flaw Removal, Heat Treatment, and Nondestructive Examination Requirements for the Third 10-year Inservice Inspection (ISI) Interval (TAC No. MC1201),"* dated July 21, 2004, ADAMS Accession Number ML041670510.

8. Letter from Richard J. Laufer (NRC) to Bryce L. Shriver (PPL Susquehanna), *"Susquehanna Steam Electric Station, Unit 1 – Relief from American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code), Section XI Appendix VIII, Supplement 11, Requirements and Cases N-504-2 and N-638 Requirements (TAC Nos. MC2450, MC2451, and MC2594),"* dated June 22, 2005, ADAMS Accession Number ML051220568.

9. Richard J. Laufer (NRC) to Michael R. Kansler (Entergy Nuclear Operations, Inc.), *"Pilgrim Nuclear Power Station Relief Request No. PRR-9 (TAC No. MC8292),"* dated March 22, 2006, ADAMS Accession Number ML060240055.

10. Michael L. Marshall Jr. (NRC) to Christopher M. Crane (Exelon Generation Company, LLC), *"Byron Station, Unit No. 1 – Evaluation of Relief Request I3R-08 Pertaining to Structural Weld Overlays (TAC No. MD1761),"* dated January 29, 2007, ADAMS Accession Number ML062510169.

11. William H. Bateman (NRC) to Michael Bratton (PDI Chairman) (Entergy Nuclear Southwest), *"Weld Overlay Performance Demonstration Administered By PDI as an Alternative for Generic Letter 88-01 Recommendations,"* dated January 15, 2002, ADAMS Accession Number ML020160532.

12. *Repair and Replacement Applications Center: "Temperbead Welding Applications 48-Hour Hold Requirements for Ambient Temperature Temperbead Welding,"* EPRI Report Number 1013558, December 2006.

13. Pamela B. Cowan (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, *"Submittal of Relief Requests Associated with the Fourth Inservice Inspection (ISI) Interval," Relief Request I4R-05 - Request for Relief From Qualification Requirements of ASME Section XI, Appendix VIII, Supplement 11, for Examination of Structural Weld Overlays (SWOLs) In Accordance with 10 CFR 50.55a(a)(3)(i)*, dated August 10, 2010, ADAMS Accession Number ML102290162.

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14. Harold K. Chernoff (NRC) to Christopher M. Crane (AmerGen Energy Company, LLC), *"Three Mile Island Nuclear Station, Unit 1 (TMI-1), Relief Request 2007-TMI-01, Regarding Structural Weld Overlays on Pressurizer Surge, Pressurizer Spray, and Hot Leg Decay Heat Drop Line Nozzles, (TAC No. MD5427),"* dated October 17, 2007, ADAMS Accession Number ML072770051.

15. Russell G. West (AmerGen Energy Company, LLC) to U.S. Nuclear Regulatory Commission, *"Response to Request for Additional Information – Relief Request No. 2007-TMI-01 – Structural Weld Overlays (SWOLs) of the Pressurizer Surge, Pressurizer Spray, and Hot Leg Decay Heat Drop Line Nozzle Dissimilar Metal Welds including the SWOL of Adjacent Welds,"* dated August 13, 2007, ADAMS Accession Number ML072320404.

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TABLE 1A
COMPONENT IDENTIFICATION

For TMI, Unit 1 PZR Nozzle to Safe-End and Safe-End to Elbow Weld SWOL Scheduled for Overlay Repair During TMI Outage T1R19

NOZZLE	NOZZLE TO SAFE-END WELD #	ITEM #	SIZE	SAFE-END TO ELBOW ADJACENT WELD #	ITEM #
Spray	PR-009BM	R1.15 B15.150	4"	SP-021BM	R1.11 R1.15 B15.150

Note: Item numbers reflect Risk-Informed classification per RISI Program and ASME Code Case N-722.

R1.11: Elements Subject to Thermal Fatigue.

R1.15: Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC).

B15.150: Spray nozzle-to-pipe connection (B15.150 is the Code Case N-722 assigned number).

The NRC has not approved Leak Before Break (LBB) application to these welds therefore LBB does not apply to this overlay.

TABLE 1B
COMPONENT MATERIAL

Weld Number	Component	Component	Component
PR-009BM	Nozzle – P1	Weld – F43	Safe-End – P43
SP-021BM	Safe-End – P43	Weld – F43	Elbow – P8

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TABLE 2
DESIGN/MATERIAL/NONDESTRUCTIVE EXAMINATION
Modifications to Code Case N-504-3 and ASME Section XI, Appendix Q

CODE CASE N-504-3 AND ASME SECTION XI APPENDIX Q	PROPOSED MODIFICATIONS
N-504-3: "*Reply:* It is the opinion of the Committee that, in lieu of the requirements of IWA-4120 in Editions and Addenda up to and including the 1989 Edition with the 1990 Addenda, in IWA-4170(b) in the 1989 Edition with the 1991 Addenda up to and including the 1995 Edition, and in IWA-4410 in the 1995 Edition with the 1995 Addenda up to and including the1996 Addenda, and in IWA-4420 in the 1995 Edition with the 1997 Addenda and later Editions and Addenda, in IWA-4810(a) in the 1992 Edition with the 1994 Addenda through the 1995 Edition, and in IWA-4520(a) in the 1995 Edition with the 1995 Addenda and later Editions and Addenda, a defect in austenitic stainless steel piping may be reduced to a flaw of acceptable size in accordance with IWB-3640 from the 1983 Edition with the Winter 1985 Addenda, or later Editions and Addenda, by deposition of weld reinforcement (weld overlay) on the outside surface of the pipe, provided the following requirements are met:"	Modification: Code Case N-504-3 and Appendix Q will be used for the weld overlay of the P1, P8, F43 and/or P43 materials as defined in Table 1B. Basis: Code Case N-504-3 is accepted for use in the current NRC Regulatory Guide 1.147 Rev. 15, and previous revisions have been used extensively in BWR primary system piping. More recently, N-504-3 (with modifications approved under relief request submittals) has been applied to PWR applications as a PWSCC mitigation technique. Industry operating experience in the area has shown that PWSCC in Alloy 82/182 will blunt at the interface with stainless steel base metal, ferritic base metal, or Alloy 52M weld metal. The 360 deg. full structural weld overlay will control growth in a PWSCC crack and maintain RCS integrity. The applied SWOL will also induce compressive stress in the existing dissimilar metal welds, thus potentially impeding growth of cracks. Furthermore, the SWOL will be sized to meet structural requirements without crediting integrity of the existing welds.

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TABLE 2
DESIGN/MATERIAL/NONDESTRUCTIVE EXAMINATION
Modifications to Code Case N-504-3 and ASME Section XI, Appendix Q

CODE CASE N-504-3 AND ASME SECTION XI APPENDIX Q	PROPOSED MODIFICATIONS
N-504-3: "(b) Reinforcement weld metal shall be low carbon (0.035% max.) austenitic stainless steel applied 360 deg. around the circumference of the pipe, and shall be deposited in accordance with a qualified welding procedure specification identified in the Repair Program." **Note: This requirement is similar to Appendix Q, Q-2000(a).**	Modification: Weld overlay filler metal shall be an austenitic nickel alloy (28% Cr min.) applied 360 deg. around the circumference of the item, and shall be deposited using a Welding Procedure Specification for groove welding, qualified in accordance with the Repair/Replacement Code and Owner's requirements and identified in the Repair/Replacement Plan. Basis: Industry operational experience has shown that PWSCC in Alloy 82/182 will blunt at the interface with stainless steel base metal, ferritic base metal, or Alloy 52M weld metal.
N-504-3: "(e) The weld reinforcement shall consist of a minimum of two weld layers having as-deposited delta ferrite content of [at] least 7.5 FN. The first layer of weld metal with delta ferrite content of at least 7.5 FN shall constitute the first layer of the weld reinforcement design thickness. Alternatively, first layers of at least 5 FN may be acceptable based on evaluation." **Note: This requirement is similar to Q-2000(d) except that Q-2000(d) alternatively allows the deposited first layers of weld metal have a carbon content of <0.02% and an FN value of at least 5 FN.**	Modification: Delta ferrite measurements will not be performed for weld overlay repairs using Alloy 52M weld metal. Basis: The deposited Alloy 52M is 100% austenitic and contains no delta ferrite due to the high nickel composition (approximately 60% nickel). The austenitic nickel alloy weld overlay shall consist of at least two weld layers deposited from a filler metal with a Cr content of at least 28%. When welding over an austenitic base material or austenitic filler material weld and the associated dilution zone from an adjacent ferritic base material, a diluted first layer of at least 24% Cr is acceptable, provided the Cr content of the deposited weld metal is determined by chemical analysis of a representative coupon. Alternatively, the first weld layer may be considered "sacrificial", and will not be credited towards the reinforcement design thickness.

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TABLE 2
DESIGN/MATERIAL/NONDESTRUCTIVE EXAMINATION
Modifications to Code Case N-504-3 and ASME Section XI, Appendix Q

CODE CASE N-504-3 AND ASME SECTION XI APPENDIX Q	PROPOSED MODIFICATIONS
N-504-3: "(h) The completed repair shall be pressure tested in accordance with IWA-5000. If the flaw penetrated the original pressure boundary prior to welding, or if any evidence of the flaw penetrating the pressure boundary is observed during the welding operation, a system hydrostatic test shall be performed in accordance with IWA-5000. If the system pressure boundary has not been penetrated, a system leakage, inservice, or functional test shall be performed in accordance with IWA-5000." **Nonmandatory Appendix Q:** (mandated through Regulatory Guide 1.147, Revision 15 as a condition of using Code Case N-504-3) states: "Ultrasonic examination personnel shall be certified in accordance with the Owner's written practice. Procedures and personnel shall be qualified in accordance with Appendix VIII."	Modification: A system leakage test at system nominal operating pressure in accordance with IWA-5000 shall be performed in accordance with the TMI, Unit 1 ISI Program. Prior to the system leakage test, ultrasonic examination of the finished SWOL using EPRI PDI demonstrated weld overlay (Reference 11) examination procedures and qualified examiners shall be performed. Basis: The TMI, Unit 1 fourth interval ISI program utilizes the 2004 Edition, no Addenda, of ASME Code Section XI (including 10 CFR 50.55a(b)(2)(xx)(B)) for NDE and pressure testing of welded repairs and replacements. ASME Section XI (including 10 CFR 50.55a(b)(2)(xx)(B)) permits a system leakage test in lieu of a hydrostatic test provided NDE is performed in accordance with IWA-4540(a)(2) of the 2002 Addenda of ASME Section XI. IWA-4540(a)(2) requires NDE to be performed in accordance with the 1992 Edition of ASME Code Section III, Subsection NB, however, Subsection NB does not address the structural weld overlay configuration, so the NDE requirements of Nonmandatory Appendix Q performed using EPRI PDI demonstrated procedures with qualified examiners will be used. The use of the EPRI PDI demonstration and qualification program in lieu of Appendix VIII is addressed in Reference 13, as part of an associated relief request.

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TABLE 3 AMBIENT TEMPERATURE TEMPER BEAD WELDING Modifications to Code Case N-638-1	
CODE CASE N-638-1	**PROPOSED MODIFICATIONS**
1.0 General Requirements "(a) The maximum area of an individual weld based on the finished surface shall be 100 sq. in., and the depth of the weld shall not be greater than one-half of the ferritic base metal thickness."	Modification: The maximum area of an individual weld based on the finished surface over the ferritic material shall not exceed 300 sq. in. If any of the TMI, Unit 1 SWOL repairs exceed 300 sq. in. over the ferritic material, additional relief will be requested. The one half base metal thickness limitation applies only to excavation and repair and is not applicable to this repair. Basis: Although the final design for the TMI, Unit 1 SWOL was not completed at the time of development of this relief request, it is possible that the SWOL will require welding on more than 100 sq. in. of surface on the carbon steel base material. The SWOL will extend to the transition taper of the carbon steel nozzle to provide a weld geometry that allows qualified UT examination of the required examination volume. There have been a number of temper bead SWOL repairs successfully applied to safe-end to nozzle welds in the nuclear industry, and a SWOL repair having a 300 sq. in. area was previously approved for TMI, Unit 1 (Reference 14).
4.0 Examination "(b) The final weld surface and the band around the area defined in para. 1.0(d) shall be examined using a surface and ultrasonic methods when the completed weld has been at ambient temperature for at least 48 hours. The ultrasonic examination shall be in accordance with Appendix I.[3]" "[3]Refer to the 1989 Edition with the 1989 Addenda and later Editions and Addenda."	Modification: For the SWOLs, full UT of the 1.5T band will not be performed. UT will be performed on the actual weld overlay, meeting the requirements of ASME Code Section XI, Nonmandatory Appendix Q-4100. When austenitic filler materials are used, the final SWOL will be examined using the surface and ultrasonic methods after three tempering weld layers (i.e., layers 1, 2, and 3) are completed and

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TABLE 3
AMBIENT TEMPERATURE TEMPER BEAD WELDING
Modifications to Code Case N-638-1

CODE CASE N-638-1	PROPOSED MODIFICATIONS
	have been in place for at least 48 hours. Basis: Later editions of the code as well as later revisions to the Code Case (N-638-2 and later) removed the requirement for the 1.5T examination band. This is in line with the less restrictive requirements for UT of the ferritic nozzle because hydrogen cracking away from the temper bead weld is not considered a concern in later editions of the code and Code Case N-638. The code case applies to any type of welding where a temper bead technique is to be employed (which includes weld repairs of excavated flaws) and is not specifically written for a SWOL repair. However, it is believed that for this type of repair, any major base material cracking would take place in the ferritic heat-affected zone directly below or adjacent to the weld overlay and not in the required 1.5T examination band of ferritic material beyond the overlay. If this type of cracking were to occur it should be detected by the NDE of the SWOL and adjacent ferritic steel surfaces. As supported by EPRI's white paper (Reference 12) providing the technical basis for Code Case N-638-4, the 48 hour hold time prior to final NDE examinations will start upon completion of the third temper bead weld overlay layer. Reference 12 addresses previous concerns regarding the 48-hour hold time prior to final NDE examinations. Areas of concern imposing the 48-hour hold time addressed through this report include: material microstructure; sources for hydrogen introduction; tensile stress and temperature; and diffusivity and solubility of hydrogen in steels. The report concludes there is

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TABLE 3
AMBIENT TEMPERATURE TEMPER BEAD WELDING
Modifications to Code Case N-638-1

CODE CASE N-638-1	PROPOSED MODIFICATIONS
	no technical basis for waiting 48 hours after the weld overlay cools to ambient temperature before performing final NDE of the completed weld overlay. Based on past and recent NDE experience on temper bead weld overlays, hydrogen cracking of these welds was not identified during the initial NDE after a 48-hour hold time or subsequent inservice inspection examinations. Appendix I does not specifically address weld overlay ultrasonic examinations. Ultrasonic examinations shall be performed using EPRI PDI weld overlay demonstrated examination procedures with PDI qualified inspectors.

9.0 REFERENCES

The references used to develop this Inservice Inspection Program Plan include:

9.5 USNRC References

9.5.1 Code of Federal Regulations, Title 10, Energy.
- Part 50, Paragraph 50.55a, "Codes and Standards."
- Part 50, Paragraph 2, "Definitions," the definition of "Reactor Coolant Pressure Boundary."
- Part 50, Appendix J, Primary Reactor Containment Testing for Water Cooled Power Reactors.
- SECY-96-080, Issuance of Final Amendment to 10 CFR 50.55a to Incorporate by Reference the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL.

9.5.2 Branch Technical Position MEB 3-1, dated November 24, 1975, "High Energy Fluid Systems, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

9.5.3 Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity."

9.5.4 Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste- Containing Components of Nuclear Power Plants."

9.5.5 Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

9.5.6 Regulatory Guide 1.193, "ASME Code Cases Not Approved For Use."

9.5.7 USNRC NUREG 0612, "Control of Heavy Loads,"

9.5.8 USNRC SER related to EPRI Topical Report TR-112657, Rev. B, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, July 1999," dated October 28, 1999.

9.6 Industry References

9.6.1 ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Inservice Inspection of Nuclear Power Plant Components," the 2004 Edition, No Addenda. (4th ISI Interval and 2nd CISI Interval).

9.6.2 ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Rules For Construction of Nuclear Power Plant Components," the 2004 Edition, No Addenda.

9.6.3 ANSI N18.2a "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

9.6.4 MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Revision 1, Report 1015009, December 2008.

9.6.5 MRP-146, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines," Report 1011955, June 2005.

9.6.6 MRP-192, "Assessment of RHR Mixing Tee Thermal Fatigue in PWR Plants," Revision 1, Report 1018395, November 2008.

9.6.7 EPRI Topical Report TR-112657, Rev. B-A, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999.

9.7 Licensee References

9.7.1 TMI, Updated Final Safety Analysis Report (UFSAR).

9.7.2 TMI, Technical Specifications (TS).

9.7.3 TMI, ISI Classification Basis Document (TMI04.G04), Fourth Ten-Year Inspection Interval.

9.7.4 TMI, ISI Selection Document (TMI04.G05), Fourth Ten-Year Inspection Interval.

9.7.5 TMI, Risk-Informed Inservice Inspection Evaluation (Final Report), Revision 3.

9.7.6 Exelon Procedure ER-AA-330, "Conduct of Inservice Inspection Activities".

9.7.7 Letter from Pamela B. Cowan, Exelon Nuclear, to the USNRC, dated October 29, 2009, Subject: Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld and Internal Examinations - Relief Requests RR-09-01 and RR-09-02.

9.7.8 Letter from Pamela B. Cowan, Exelon Nuclear, to the USNRC, dated August 10, 2010, Subject: Submittal of Relief Requests Associated with the Fourth Inservice Inspection (ISI) Interval.

9.7.9 Safety Evaluation Report from Harold K. Chernoff, USNRC, to Michael J. Pacilio, Exelon Nuclear, dated September 21, 2010, Subject: Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld and Internal Examinations - Relief Requests RR-09-01 and RR-09-02 (TAC NOS. ME2483 and ME2484).

9.7.10 Letter from Pamela B. Cowan, Exelon Nuclear, to the USNRC, dated September 39, 2010, Subject: Submittal of Relief Request RR-10-02 Concerning the Weld Overlay of the Pressurizer Spray Nozzle to Safe-End and Safe-End to Elbow Dissimilar Metal Welds.

9.8 Commitments

9.8.1 CM-1 Action Tracking Item AR603573.03, License Renewal Reactor Head Closure Studs Program (Step 1.1).

9.8.2 CM-2 Action Tracking Item AR603573.01, License Renewal ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Step 1.1).

9.8.3 CM-3 Action Tracking Item AR603573.07, License Renewal Bolting Integrity Program (Step 1.1).

9.8.4 CM-4 Action Tracking Item AR603573.26, License Renewal ASME Section XI, Subsection IWF Program (Step 1.1).

9.8.5 CM-5 Action Tracking Item AR603573.24, License Renewal ASME Section XI, Subsection IWE (Step 1.1).

9.8.6 CM-6 Action Tracking Item AR603573.25, License Renewal ASME Section XI, Subsection IWL (Step 1.1).

9.8.7 CM-7 Action Tracking Item AR603573.38, License Renewal Concrete Containment Tendon Prestress (Step 1.1).

9.8.8 CM-8 Action Tracking Item AR603573.05, License Renewal Nickel Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program (Step 1.1).