Constellation Energy®
Nine Mile Point Nuclear Station

P.O. Box 63
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March 16, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 1; Docket No. 50-220

American Society of Mechanical Engineers (ASME) Code, Section XI, Inservice Inspection Program for the Fourth Ten-Year Inservice Inspection Interval and Associated 10 CFR 50.55a Requests

REFERENCES:

(a) Letter from G. J. Laughlin (NMPNS) to Document Control Desk (NRC), dated September 16, 2008, Extension of Permanent Relief from Inservice Inspection Requirements of 10 CFR 50.55a(g) for the Volumetric Examination of Reactor Pressure Vessel Shell Circumferential Welds for the License Renewal Period of Extended Operation

(b) Letter from G. J. Laughlin (NMPNS) to Document Control Desk (NRC), dated August 29, 2008, Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation

This letter submits the Nine Mile Point Unit 1 (NMP1) Fourth Ten-Year Inservice Inspection (ISI) Plan and Schedule (see Enclosure) and requests NRC approval of an associated 10 CFR 50.55a request pursuant to 10 CFR 50.55a(a)(3).

In accordance with 10 CFR 50.55a(g)(4)(ii), the NMP1 ISI program has been updated to comply with the latest edition and addenda of the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) twelve (12) months before the start of the interval. The NMP1 fourth ten-year ISI interval will commence on August 23, 2009 and end on August 22, 2019; therefore, the 2001 Edition with 2003 Addenda of ASME Section XI is the applicable Code.

There are three (3) 10 CFR 50.55a requests associated with the fourth ten-year ISI interval program. The details of these requests are contained in Appendix H of the enclosed ISI Plan and Schedule and are summarized below.

- Request Number 1ISI-001A regarding volumetric examination of reactor pressure vessel shell circumferential welds has previously been submitted for NRC review (see Reference a).

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NRR

- Request Number 1ISI-02 regarding the use of ASME Code Case N-730 for the repair of control rod drive bottom head penetrations has previously been submitted for NRC review (see Reference b). During preparation of the present ISI program submittal, it was noted that there was a minor error in Figure 1 of Request Number 1ISI-02 (page ISI 02-6 of ISI 02-6) that was previously submitted in Reference (b). The figure showed a dimension of "0.10 TO .015" for the gap between the reactor vessel bore inside diameter and the control rod drive (CRD) housing outside diameter. The correct dimension for this gap is ".010 TO .015." A corrected copy of Figure 1 is provided in the copy of Request Number 1ISI-02 that is included in Appendix H of the enclosed ISI Plan and Schedule.

- Request Number 1ISI-003 requests authorization for implementation of a risk-informed/safety based inservice inspection program for Class 1 and Class 2 piping based on ASME Code Case N-716.

This letter does not contain any regulatory commitments.

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

Peter Mazzaferro

Peter A. Mazzaferro
Acting Manager Engineering Services

PAM/DEV

Enclosure: Nine Mile Point Nuclear Station, Unit 1 - Fourth Ten-Year Inservice Inspection Plan and Schedule (CNG-NMP1-ISI-004, Revision 00)

cc:
S. J. Collins, NRC
R. V. Guzman, NRC
Resident Inspector, NRC

ENCLOSURE

NINE MILE POINT NUCLEAR STATION, UNIT 1

FOURTH TEN-YEAR INSERVICE INSPECTION PLAN AND SCHEDULE

(CNG-NMP1-ISI-004, Revision 00)

Nine Mile Point Nuclear Station, LLC
March 16, 2009

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FOURTH TEN-YEAR INSERVICE INSPECTION INTERVAL
August 23, 2009 To August 22, 2019

FOURTH TEN-YEAR INSERVICE INSPECTION PLAN AND SCHEDULE

Commercial Service Date:			December 26, 1969
NRC Docket Number:			50-220
Document Number:			CNG-NMP1-ISI-004
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RECORD OF REVISION

REV. No.	SECTION	DATE OF REVISION	SECTION DESCRIPTION
00	All	January 15, 2009	Initial issue of updated Fourth Ten-Year In-service Inspection Interval Plan and Schedule

ABBREVIATIONS

Listed below are the abbreviations utilized in this document:

AMP	Aging Management Program
ANII	Authorized Nuclear In-service Inspector
ANSI	American Nuclear Standard Institute
ASME	American Society of Mechanical Engineers
B&PV	Boiler & Pressure Vessel Code
BC	Branch Connection
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CC	Crevice Corrosion
CEG	Constellation Energy Group
CFR	Code of Federal Regulations
CLB	Current License Basis
CRC	Corrosion Resistant Cladding
CRD	Control Rod Drive
CRA	Code Required Area (Surface)
CRS	Core Spray System
CT	Condensate Transfer
CTN-SP	Containment Spray System
CRV	Code Required Volume
CU	Reactor Water Clean-Up System
DPI	Drywell Inerting CAD and Purge System
EC	Erosion-Cavitation
ECS	Emergency Cooling System
ECSCC	External Chloride Stress Corrosion Cracking

ABBREVIATIONS

FAC	Flow Accelerated Corrosion
FMEA	Failure Modes and Effects Analysis
FS	Flow Sensitive
FSAR	Final Safety Analysis Report
FWS	Feedwater System
FPS	Spent Fuel Pool Filtering and Cooling Sytem
GALL	Generic Aging Lessons Learned Report
GE	General Electric
GL	Generic Letter
HPCI	High Pressure Coolant Injection System
IEB	Inspection and Enforcement Bulletin (USNRC)
IEN	Inspection and Enforcement Notice (USNRC)
IHSI	Induction Heat Stress Improvement
ISI	In-service Inspection
IGSCC	Intergranular Stress Corrosion Cracking
IVVI	In-Vessel Visual Inspections
LC	Localized Corrosion
LPS	Liquid Poison System
LRA	License Renewal Application
MIC	Microbiologically-Induced Corrosion
MSS	Main Steam System
MT	Magnetic Particle Examination
N/A	Not Applicable
NBVI	Nuclear Boiler Vessel Instrumentation

ABBREVIATIONS

NDE	Nondestructive Examination
NMP1	Nine Mile Point Nuclear Station Unit 1
NPS	Nominal Pipe Size
NMPNS	Nine Mile Point Nuclear Station, LLC
NSSS	Nuclear Steam Supply System
NWT	Nominal Wall Thickness
OD	Outside Diameter
P&ID	Piping and Instrumentation Diagram
PDI	Performance Demonstration Initiative
PIT	Pitting
PT	Liquid Penetrant Examination
RBCLC	Reactor Building Closed Loop Cooling System
RHR	Residual Heat Removal System
RHSI	Resistant Heat Stress Improvement
RICSIL	Rapid Information Communication Services Information Letter
RI-ISI (RIS_B)	Risk-Informed In-service Inspection
RR	Reactor Recirculation System
RG	Regulatory Guide (USNRC)
RPV	Reactor Pressure Vessel
RWC	Reactor Water Cleanup System
RXVI	Reactor Vessel Instrumentation
SCC	Stress Corrosion Cracking
SD	Structural Discontinuity
SDC	Shutdown Cooling Water System
SIL	Services Information Letter

ABBREVIATIONS

SRP	Standard Review Plan (USNRC)
SWS	Service Water System
SURF	Surface Examination
SI	Stress Improvement
TASCS	Thermal Stratification, Cycling and Striping
TE	Terminal End
TF	Thermal Fatigue
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specifications
TT	Thermal Transients
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
USNRC	United States Nuclear Regulatory Commission
VOL	Volumetric Examination
VT	Visual Examination (suffix number denotes type of exam, (VT-1, VT-2, VT-3))
WinISI 2004	Computerized Window-Based In-service Inspection Data Base Management Software

GLOSSARY OF TERMS

ACCIDENT SEQUENCE - a combination of events leading from an initiating event that challenges safety systems to an undesired consequence, such as core damage or breach of containment integrity.

ASSESS - to determine by evaluation of data compared with previously obtained data such as operating data or design specifications.

AUTHORIZED INSPECTION AGENCY - an organization that is empowered by an enforcement authority to provide inspection personnel and services as required by Section XI.

AUTHORIZED NUCLEAR INSERVICE INSPECTOR - a person who is employed and has been qualified by an Authorized Inspection Agency to verify that examinations, tests and repair/replacement activities (that do not include welding or brazing) are performed in accordance with the requirements of Section XI.

AUTHORIZED NUCLEAR INSPECTOR - an employee of an Authorized Inspection Agency who has been qualified in accordance with NCA-5000 of Section III.

BELTLINE REGION – the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

COMPONENT – a vessel, concrete containment, pump, valve, storage tank, piping system, or core support structure.
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COMPONENT SUPPORT - a metal support designed to transmit loads from a component to the load-carrying building or foundation structure. Component supports include piping supports and encompass those structural elements relied upon to either support the weight or provide structural stability to components.

CONSTANT LOAD TYPE SUPPORT - spring type support that produces a relatively constant supporting force throughout a specified deflection

CONSTRUCTION - an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection and certification required in the manufacturer and installation of items.

CONSTRUCTION CODE – nationally recognized Codes, Standards, and Specifications (e.g., ASME, ASTM, USAS, ANSI, API, AWWA, AISC, MSS, AWS) including designated Cases, providing construction requirements for an item.

CORE DAMAGE - prolonged oxidation and severe fuel damage due to uncovering and heat up of the reactor core.

CORE DAMAGE FREQUENCY - an estimate of the likelihood of a severe accident associated with core damage.

CORE SUPPORT STRUCTURES - those structures or parts of structures that are designed to provide direct support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel

CORRECTIVE ACTION - action taken to resolve flaws and relevant conditions, including supplemental examinations, analytical evaluation, repair/replacement activities, and corrective measures.

GLOSSARY OF TERMS

CORRECTIVE MEASURES – actions (such as maintenance) taken to resolve relevant conditions, but not including supplemental examinations, analytical evaluations, and repair/replacement activities.

DEFECT - a flaw (imperfection or unintentional discontinuity) of such size, shape, orientation, location, or properties as to be rejectable.

DISCONTINUITY - a lack of continuity or cohesion; an interruption in the normal physical structure of material or a product

DISSIMILAR METAL WELD – a weld between (a) carbon or low alloy steels to high alloy steels, (b) carbon or low alloy steels to high nickel alloys, or (c) high alloy steels to high nickel alloys.

ENFORCEMENT AUTHORITY - a regional or local governing body, such as a State or Municipality of the United States or a Province of Canada, empowered to enact and enforce Boiler and Pressure Vessel Code legislation.

ENGINEERING EVALUATION - an evaluation of indications that exceed allowable acceptance standards to determine if the margins required by the Design Specification and the Construction Code are maintained.

EVENT TREE - a quantifiable logical network that begins with an initiating event or condition and progresses through a series of branches (usually binary) that represents expected system or operator performance that either succeeds or fails and arrives at either a success or failed condition (e.g., core damage) at the end of the tree.

EVALUATION - the process of determining the significance of examination or of test results, including the comparison of examination or test results with applicable acceptance criteria or previous results.

EXAMINATION CATEGORY - a grouping of items to be examined or tested.

FAILURE - events involving leakage, rupture, or conditions that would disable a components ability to perform its intended safety function.

FAILURE MODE - a condition or degradation mechanism that can cause a failure.

FAILURE MODES AND EFFECTS ANALYSIS - an analysis intended to identify the conceivable failure modes of a component and the impact of the failure on operations, the system, and surrounding components along with the likelihood of the failure and consequences.

FLAW - an imperfection or unintentional discontinuity that is detectable by nondestructive examination

GALL – The Generic Aging Lessons Learned Report that contains the Nuclear Regulatory Commission staffs generic evaluation of the existing plant programs and documents the technical basis for determining where existing programs are adequate without modification and where existing programs should be augmented for the extended period of operation. NUREG-1801, Volume 1 and 2, July 2001.

HANGER - an item that carries the weight of components or piping from above with the supporting members being mainly in tension

IMPERFECTION - a condition of being imperfect; a departure of a quality characteristic from its intended condition

GLOSSARY OF TERMS

INDICATION - the response or evidence from the application of a nondestructive examination

INSERVICE EXAMINATION - the process of visual, surface, or volumetric examination performed in accordance with the rules and requirements of Section XI.

INSERVICE INSPECTION - methods and actions for assuring the structural and pressure-retaining integrity of safety-related nuclear power plant components in accordance with the rules of Section XI.

INSPECTION - verification of the performance of examinations and tests by an Inspector.

INSPECTION PROGRAM- the plan and schedule for performing examinations or tests.

INSPECTOR - an Authorized Nuclear In-service Inspector, except for those instances where so designated as Authorized Nuclear Inspector.

INSPECTION INTERVAL - duration of time, typically 10-years.

INSPECTION PERIOD - duration of time within an ten-year inspection interval, as determined by applicable edition/addenda of Section XI, Program B.

ITEM - a material, part, appurtenance, piping sub-assembly, component or component support.

MAINTENANCE - routine servicing or work undertaken to correct, adjust or prevent an abnormal or unsatisfactory condition.

NONDESTRUCTIVE EXAMINATION - an examination by the visual, surface or volumetric method.

OPEN ENDED - a condition of piping or lines that permits free discharge to atmospheric or containment atmosphere

OWNER - the organization legally responsible for the construction and/or operation of a nuclear facility including but not limited to one who has applied for, or who has been granted, a construction permit or operating license by the regulatory authority having lawful jurisdiction.

PIPING SEGMENT - a portion of piping for which a failure at any point in the segment results in the same consequence (e.g., loss of the system, loss of a pump train) as a failure at any other point in the segment and includes piping structural elements between major discontinuities, such as pumps and valves.

PIPING STRUCTURAL ELEMENT - an item within a specified piping segment, such as a straight length of pipe, a pipe elbow, a coupling, a fitting, a flanged joint, or a weld.

PIPING SYSTEM - an assembly of piping segments, piping supports, and other components that may consist of one or more Code classes with a defined function as described within the In-service Inspection Program.

PROBABILISTIC RISK ASSESSMENT (PRA) - a quantitative assessment of the risk associated with plant operation and maintenance and measured in terms of frequency of occurrence of different events, including severe core damage or a breach of containment integrity.

RELEVANT CONDITION - a condition observed during a visual examination that requires supplemental examination, corrective measure, correction by repair/replacement activities, or analytical evaluation

GLOSSARY OF TERMS

REGULATORY AUTHORITY - a federal government agency, such as the United States Nuclear Regulatory Commission, that is empowered to issue and enforce regulations affecting the design, construction, and operation of nuclear power plants.

SUBSEQUENT PERIOD - is the next following period, even if it is in the following ten-year inspection interval.

SUPPORT - (1) an item used to position components, resist gravity, resist dynamic loading, or maintain equilibrium of components; (2) an item that carries the weight of a component or piping from below with the supporting members being mainly in compression.

SUPPORT PART - a part of subassembly of a component support or piping support.

TERMINAL ENDS - the extremities of piping runs that connect structures, components, or pipe anchors, each of which acts as a rigid restraint or provides at least 2 degrees of restraint to piping thermal expansion.

VARIABLE SPRING TYPE SUPPORT - a spring type support providing a variable supporting force throughout a specified deflection

VIBRATION CONTROL AND SWAY BRACE - a spring type support providing a variable force along its axis.

VERIFY - to determine that a particular action has been performed in accordance with the rules and requirements of Section XI either by witnessing the action or by reviewing records.

ABSTRACT

This document provides the Updated Fourth Ten-Year In-service Inspection Plan and Schedule for Nine Mile Point Nuclear Station, Unit 1 (NMP1).

This document provides the basis for those pressure retaining components and/or systems (including their supports), which are classified as Quality Group A, B or C, (ASME Code Class 1, Class 2, and Class 3), and subject to examination, as set forth in the applicable Edition of the ASME Boiler and Pressure Vessel Code (B&PVC), Section XI, to the extent practical within the limitations of design, geometry and materials of construction of the components pursuant to Title 10, Part 50, Section .55a (b)(2) of the Code of Federal Regulations.

The ASME Boiler and Pressure Vessel Code, Edition applicable to the Nine Mile Point Nuclear Station, Unit 1, Fourth In-service Inspection Interval Plan and Schedule is the 2001 Edition through the 2003 Addenda of Section XI, hereafter referred to as the Code.

This document also provides the augmented in-service inspections required by Generic Letter 88-01 as provided in Section 6 of this document.

Aging Management

For the purpose of license renewal, the NRC staff has evaluated the appropriate ASME Section XI programs based on the ten program elements described in the GALL report. Except where noted, the NRC staff has determined that the ASME Section XI programs provide processes for identifying degradation that is attributable to applicable aging effects and are therefore acceptable for managing the effects of aging during the period of extended operation. The aging management programs (AMPs) as referenced in the GALL report that are being credited and are managed in whole or in part by the ASME Section XI monitoring programs for license renewal during the period of extended operation are listed below.

AMPs **Title of Aging Management Programs**

All or portions of these AMP's have been credited to manage the effects of aging for systems, structures and components within the scope of license renewal.

XI.M1	ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD
XI.M3	Reactor Head Closure Studs
XI.M5	BWR Feedwater Nozzle
XI.M6	BWR Control Rod Drive Return Line Nozzle
XI.M7	BWR Stress Corrosion Cracking
XI.S1	ASME Section XI, Subsection IWE
XI.S2	ASME Section XI, Subsection IWL
XI.S3	ASME Section XI, Subsection IWF

Actions required for Aging Management are incorporated within the updated Ten-Year In-service Inspection Plan and Schedule.

SECTION 1 – INTRODUCTION

1.0 INTRODUCTION

This document details the basis and plans for the Fourth Ten-Year In-service Inspection Interval for components, welds, supports, bolting, pump casings, valve bodies, and reactor pressure vessel internals for the Nine Mile Point Nuclear Station, Unit 1. The following historical information has been considered in the development of the overall In-service Inspection Program:

• Issuance of Construction Permit	April 12, 1965
• Issuance of Full Power Operating License (DPR-63)	December 26, 1974
• Commercial Operation Date	December 26, 1969
• First Ten-Year Inspection Interval	December 26, 1974 to June 26, 1986
• Second Ten-Year Inspection Interval	June 26, 1986 to December 25, 1999
• Third Ten-Year Inspection Interval	December 26, 1999 to August 22, 2009

1.1 Inspection Interval

a. The First In-service Inspection Interval began on December 26, 1974 and ended on June 26, 1986, and was developed to meet the Winter 1972 Edition and the 1974 Edition thru Summer 1975 Addenda of Section XI.

b. The Second In-service Inspection Interval began on June 25, 1986 and ended on December 26, 1999, and was developed to meet the 1983 Edition through Summer 1983 Addenda of Section XI.

c. The Third In-service Inspection Interval began on December 26, 1999 and was scheduled to end on December 25, 2009. The plant operating license expired on August 22, 2009, the end of the Third Inspection Interval and was developed to meet the 1989 Edition, no Addenda of Section XI.

d. The Fourth In-service Inspection Interval becomes effective on August 23, 2009 and is scheduled to end on August 22, 2019 and was developed to meet the 2001 Edition through the 2003 Addenda of Section XI. See paragraph 1.4 for applicable Code Edition and Addenda for the Fourth In-service Inspection Interval.

1.2 Inspection Periods

The Fourth In-service Inspection Interval is divided into three successive inspection periods as determined by calendar years of plant service within the inspection interval. Identified below are the period dates for the fourth ten-year inspection interval as defined by Inspection Program "B".

In accordance with IWA-2430(d)(3), the inspection period specified below may be decreased or extended by as much as 1 year to enable inspections to coincide with NMP1's plant outages. This adjustment shall not alter the requirements for scheduling inspection intervals.

TABLE 1-2
NMP1 INSERVICE INSPECTION PERIODS

INSPECTION PERIODS	PERIOD START DATES	PERIOD END DATES	REFUEL OUTAGE	REFUEL OUTAGE YEAR
1	August 23, 2009	August 22, 2012	RFO-21	2011
2	August 23, 2012	August 22, 2016	RFO-22 RFO-23	2013 2015
3	August 23, 2016	August 22, 2019	RFO-24 RFO-25	2017 2019

1.3 Applicable Documents

The Fourth In-service Inspection Program for ASME Code Class 1, 2 and 3, systems and components (including their supports) was developed after giving due consideration to the following documents and subject to the limitations and modifications listed in 10 CFR 50.55a(b), and to the extent practical within the limitations of design, geometry and materials of construction. Specific areas within this plan where these documents are used in the preparation of the inspection program are addressed within each area that is affected.

- **Code of Federal Regulations**

  10 CFR 50.55(a) Code of Federal Regulations; (Reference: February 14, 2007)

  Federal Register, Volume 73, Number 176, dated September 10, 2008. 10 CFR 50 Industry Codes and Standards, Amended Requirements, Final Rule.

  10 CFR 54 Requirements for Renewal of Operating License for Nuclear Power Plants.

- **ASME Code Editions and Addenda**

  ASME Boiler and Pressure Vessel Code, Sections V, 2001 Edition, "Nondestructive Examination"

  ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition, through 2003 Addenda, "Rules for In-service Inspection of Nuclear Power Plant Components"

- **USNRC Regulatory Guides**

  The following list of Regulatory Guides are applicable to the Nine Mile Point Nuclear Station Fourth In-service Inspection Program:

  1.26 Quality Group Classifications and Standards for Water-Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 2, June 1975.

  1.65 Materials and Inspections for Reactor Vessel Closure Studs, dated October 1973, Regulatory Position C.4.b.

1.84 Design and Fabrication Code Case Acceptability ASME Section III, Division 1, Latest Revision.

1.85 Material Code Case Acceptability ASME Section III, Division 1, Latest Revision.

1.147 In-service Inspection Code Case Acceptability ASME Section XI, Division 1, Revision 15, October 2007.

- **NMP1 Plant Specific Documents**

Nine Mile Point Unit 1 Updated Safety Analysis Report, Sections 1, 5, 7, 9, 12 and 16.

Nine Mile Point Unit 1 Plant Technical Specifications

USNRC Docket Number 50-220, NMP1 Facility Operating License DPR-63

EPRI TR-112657, Electric Power Research Institute Report for Alternative Requirements of Risk-Informed In-service Inspection Methodology. Revision B-A, dated December 1999.

Constellation Energy NMPNS Amended License renewal Application.

- **Performance Demonstrative Initiative Documents**

NRC Assessment of the PDI Program, March 6, 1996, Jack Strosnider, Chief Materials and Chemical Engineering Branch, to Bruce Sheffel, Chairmen PDI, March 6, 1996.

IGSCC Performance Demonstration Administered by PDI as an Alternative for Generic Letter 88-01 Recommendations, Edmund J. Sullivan Jr. Acting Branch Chief to Frank Leonard chairman PDI, September 2, 1998.

PDI Program Description, Rev 4

EPRI Guideline for the Implementation of Appendix VIII and 10 CFR 50.55a, Revision D, dated 04/18/2000 Draft.

- **USNRC NUREGS/SRP's**

USNRC NUREG 0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Revision 2.

USNRC NUREG 1801, Volume 1 and 2, Generic Aging Lessons Learned (GALL) Report.

USNRC NUREG 1900, Volume 1 and 2, Safety Evaluation Report Related to the License renewal of NMPNS, Units 1 and 2.

USNRC NUREG 0619, BWR Feedwater Nozzle and CRD Return Lines.

- **USNRC Bulletins**

82-03 Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants, Revision 1, October 28, 1982

- **USNRC Generic Letters**

81-11	Feedwater and Control Rod Drive Nozzle Cracking,
88-01	USNRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, January 25, 1988.
88-01	NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, Supplement 1, February 4, 1992.
90-05	Guidance for Performing Temporary Non-Code Repairs to ASME (ISI) Code Class 1, 2 and 3 Piping and Components, June 15, 1990.
98-05	Boiling Water Reactor Licensees Use of the BWRVIP 05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds, December 10, 1998.

- **Boiling Water Reactor Vessel Inspection Program (BWRVIP) References or Commitments**

Requirements of the BWRVIP Reactor Vessel and Internals (IVVI) Inspection Program are contained in applicable Nuclear Engineering Reports (NER's), and are outside the scope of this document.

BWRVIP 75 - Technical Basis for Revision to Generic Letter 88-01 Inspection Schedules, EPRI TR-113932, dated October 1999.

- **USNRC Informational Notices**

98-44	Ten-Year In-service Inspection (ISI) Program Update for Licensees That Intend to Implement Risk-Informed ISI of Piping, dated August 11, 1999.

- **ASME Code Cases**

Code Cases approved for use through Regulatory Guide 1.147 may be proposed for revision to the inspection plan. Specific Code Cases used in the preparation of this inspection plan or included for proposed use are identified in Appendix I of this document.

1.4 Applicable Code Editions and Addenda

1.4.1 Fourth Inspection Interval

Pursuant to Title 10, Part 50, Section 55a(g)(4), of the Code of Federal Regulations, the In-service Inspection requirements applicable to nondestructive examination for the Fourth In-service Inspection Interval are based on the rules set forth in the 2001 Edition through the 2003 Addenda of Section XI, that was endorsed twelve months prior to the start of the Fourth In-service Inspection Interval.

1.4.2 Subsequent Code Editions and Addenda

As permitted by 10 CFR 50.55a(g)(4)(iv), NMPNS may elect to meet the requirements set forth in subsequent Editions and Addenda of Section XI that are incorporated by reference into 10 CFR 50.55a(b)(2), and subject to the applicable limitations and modification and subject to USNRC approval.

Portions of Editions and Addenda may also be used provided that all related requirements to the respective Editions and Addenda are met. NMPNS intends to continually evaluate and apply, as appropriate, changes in adopted Code Editions and Addenda which provide the continuing assurance of the quality and safety of pressure retaining components and systems.

Note: On July 28, 2004 the USNRC issued NRC Regulatory Issue Summary 2004-12, Clarification on use of later editions and addenda to ASME OM Code and Section XI. If NMPNS plans to use later editions and addenda to ASME OM Code and Section XI that have been incorporated by reference into 10 CFR 50.55a must obtain prior approval pursuant to 10 CFR 50.55a(f)(4)(iv) or (g)(4)(iv). NMPNS may request this approval by submitting a letter to the NRC Document Control Desk.

1.5 ASME Code Classifications

System safety classifications, design and fabrication requirements meet the intent of 10 CFR 50.2v and Regulatory Guide 1.26, to the extent practical within the limitations of design, geometry and materials of construction of the components, as identified within the Nine Mile Point Nuclear Station, Unit 2, Updated Safety Analysis Report (USAR).

Water, steam and radioactive containing components (other than turbines and condensers) are designated Quality Group A, B, or C, (ASME Code Class 1, 2 or 3), and that are safety-related

1.5.1 ASME Code Class 1

Quality Group A system boundaries were developed based on 10 CFR 50.2(v), and the NMP1 FSAR, and apply to the reactor coolant pressure boundary components.

The Reactor Coolant system includes a single cycle, forced circulation, General Electric Boiling Water Reactor.

1.5.2 ASME Code Class 2

Quality Group B system boundaries were developed based on Regulatory Guide 1.26 and the NMP1 FSAR, and apply to those components of the Reactor Coolant System not classified as Quality Group A, (ASME Code Class 1), and that are safety-related.

1.5.3 ASME Code Class 3

Quality Group C system boundaries were developed based on Regulatory Guide 1.26 and the NMP1 FSAR, and apply to those components that are not classified as Quality Group A or B, (ASME Code Class 1 or 2), and that are safety-related

1.5.4 Non-Nuclear Safety-Related

Quality Group D applies to those components not related to nuclear safety, and as such is not included within this document. Exception, Reactor Water Cleanup System welds located outside the containment isolation valves. See Section 6 Augmented Examinations for details.

1.5.5 Application

Application of the rules of Section XI are governed by the group classification criteria as defined above and applied as follows:

1. The rules of IWB were applied to those systems whose components are classified ASME Class 1.
2. The rules of IWC were applied to those systems whose components are classified ASME Class 2.

3. The rules of IWD were applied to those systems whose components are classified ASME Class 3.

4. The requirements of IWE were applied to components classified ASME Class MC and to metallic shell and penetration liners classified ASME Class CC.

5. The requirements of IWF were applied to supports classified ASME Class 1, 2, 3, or MC.

1.5.6 Optional Construction of a Component

Optional construction of a component within a system boundary to a classification higher than the minimum class established in the component Design Specification (either upgrading from Class 2 to Class 1 or from Class 3 to Class 2) shall not affect the overall system classification by which the applicable rules of Section XI are determined.

1.5.7 Piping Penetrating Containment

The portions of piping that penetrates a containment vessel, which is required by Section III to be constructed to be Class 1 or 2 rules for piping and which may differ from the classification of the balance of the piping system, need not affect the overall system classification that determines the applicable rules of Section XI.

1.5.8 Classification Diagrams

The system Quality Group A, B and C, (ASME Code Class 1, 2 and 3) classification interfaces between components of different quality groups/classes applicable to Nine Mile Point Nuclear Station, are designated on various ASME Section XI Boundary Diagrams (P&I Ds). These designations identify the system class breaks by color coding.

The rules of IWB, IWC and IWD were applied to these diagrams in order to determine those components/systems subject to examination/test. Components subject to surface, volumetric and visual examination are listed in the ASME Section XI Summary Tables in Appendix A, B, C and D.

Appendix G provides a list of the applicable ASME Section XI Boundary diagrams (P&I Ds). Copies of these diagrams are available through the NMPNS drawing control system.

1.6 Inspection Program B

The Nine Mile Point Nuclear Station inspection intervals comply with IWA-2432, Inspection Program B. With the exceptions of the examinations identified in 1.6.1, the required examinations in each examination category shall be completed in accordance with Table 1-6.

1.6.1 Class 1, 2 and 3 Components

(a) The required percentages of examinations in each Examination Category shall be completed in accordance with Table IWB-2412-1, IWC-2412-1, and IWD-2412-1, with the following exceptions:

  (1) Examination Categories B-N-1, B-P, B-Q, C-H, and the system pressure test requirements of D-A, D-B and D-C;

  (2) Examinations partially deferred to the end of the inspection interval, as allowed by Examination Categories B-A, B-D and B-F.

  (3) Examinations deferred to the end of an inspection interval, as allowed by Examination

Categories B-A, B-L-1, B-M-1, B-N-2, B-N-3, and B-O;

(4) Examination deferred until disassembly of a component for maintenance, repair/replacement activity, or volumetric examination as allowed by Examination Categories B-G-1, B-G-2, B-L-2 and B-M-2.

(5) Welded attachments examined as a result of component support deformation under Examination Category B-K;

If there are less than three items or welds to be examined in an Examination Category, the items or welds may be examined in any two periods or in any one period if there is only one item or weld, in lieu of the percentage requirements of Table 1-6 below.

(b) If items or welds are added to the Inspection Program, during the service lifetime of a plant, examination shall be scheduled as follows:

(1) When items or welds are added during the first period of an interval, at least 25% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during each of the second and third periods of the interval. Alternatively, if deferral of the examinations is permitted for the Examination Category and Item Number, the second period examinations may be deferred to the third period and at least 50% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during the third period.

(2) When items or welds are added during the second period of the interval, at least 25% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during the third period of the interval.

(3) When items or welds are added during the third period of an interval, examinations shall be scheduled in accordance with IWB-2412(a) for successive intervals.

1.6.2 Component Supports

The required examinations shall be completed in accordance with the inspection schedule established for the components under IWB, IWC, and IWD.

TABLE 1-6
INSPECTION PROGRAM B

Inspection Interval	Inspection Period Calendar Years of Plant Service Within the Interval	Minimum Examination Completed, %	Maximum Examination Credited, %
NMP2 3^{RD}. In-service Inspection Interval	3	16%	50%
	7	50[1]%	75%
	10	100%	100%

Note: (1) If the first period completion percentage for any examination Category exceeds 34%, at least 16% of the required examinations shall be performed in the second period.

1.7 Development of Inspection Program

Sections 2 through 6 provides a narrative description of the Nine Mile Point Unit 1 Fourth In-service Inspection Program basis for ASME Code Class 1, 2 and 3, (including their Supports and Augmented Examinations), of components and/or systems subject to examination/test.

1.8 Substitute Examinations

NMPNS may substitute items scheduled in the Inspection Plan for others not previously scheduled . This substitution may be done due to such conditions as limited physical access, high radiation levels, etc. Such changes will be documented in accordance with ASME Section XI requirements.

1.9 Exclusions/Exceptions

This paragraph defines the exclusions/exceptions; NMPNS has taken due to the unit being docketed prior to June 1978.

1.9.1 Examination Category B-J

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:

(1) Primary plus secondary stress intensity range of 2.4Sm for ferritic and stainless steel

(2) Cumulative usage factor U of 0.4

1.9.2 Containment Inspection Program

The Containment In-service Inspection Program for Class MC, Subsection IWE is addressed in document CNG-NMP1-CISI-002, Latest revision.

1.9.3 Alternative Risk-Informed In-service Inspection Program

The Alternate Risk-Informed In-service Inspection Program for ASME Code Class 1 and 2, Examination Categories B-F, B-J, C-F-1, C-F-2 and IGSCC Category A piping welds are addressed in Section 7 of this document.

1.9.4 Additional Programs

In addition to the above items, the following Programs are outside the scope of this document. They are addressed in separate documents.

- In-service Pump and Valve Test Program
- IWE Class MC Containment Examination Program
- System Pressure Test Program
- Containment Pressure Test Program
- Snubber Inspection and Testing Program

1.9.5 USNRC Limitations and Modifications

The NRC amended 10 CFR 50.55a in the Federal Register, on September 10, 2008, Volume 73, Number 176, Final Rule. Based on the amendment, the NRC staff has imposed the following limitations and modification.

a. 10 CFR 50.55a(b)(2)(xv) Appendix VIII Specimen Set and Qualification Requirements

The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition through the 2001 Edition. Licensees choosing to apply these provisions shall apply all of the following provisions under this paragraph except for those in 10 CFR 50.55a(b)(2)(iv)(F) which are optional.

Licensees who use later editions and addenda than the 2001 Edition of Section XI of the ASME Code shall use the 2001 Edition of Appendix VIII.

b. 10 CFR 50.55a(b)(2)(xx) System Leakage Test

The NRC imposed a condition in 10 CFR 50.55a(b)(2)(xx) requiring Section III NDE be performed following repair and replacement activities if a system leakage test was to be used in lieu of a hydrostatic test under the 2003 Addenda through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(b)(2). See Section 9 for additional details.

SECTION 2 - ASME CODE CLASS 1 SYSTEMS/COMPONENTS

2.0 ASME CODE CLASS 1 SYSTEMS/COMPONENTS

The ASME Code Class 1 system boundaries subject to examination and testing were developed based upon the requirements of 10 CFR 50.2(v) and Nine Mile Point Unit 1 (NMP1), Final Safety Analysis Report (FSAR). The ASME Code Class 1 components and systems (including their supports) subject to examination and testing is described in detail below:

Note: Systems subject to the examination requirements of this section are identified in the Table below:

ASME Code Class 1 Systems	System Identification
Reactor Pressure Vessel (RPV)	00.0, 09.0
Main Steam System (MSS)	01.0
Feedwater System (FWS)	31.0
Reactor Recirculation System (RCS)	32.0
Reactor Water Cleanup System (RWCUS)	33.0, 33.2, 37.0, 37.1
Reactor Vessel Instrumentation System (RXVI)	36.0
Shutdown Cooling System (SDCS)	38.0
Emergency Cooling System (ECS)	39.0
Core Spray System (CSS)	40.0, 40.1
Liquid Poison System (LPS)	42.1
Control Rod Drive System (CRDS)	44.1, 44.2

2.1 ASME Code Exemptions

IWB-1220 - The following components or parts of components are exempted from the volumetric and surface examination requirements of IWB-2500:

(a) Components that are connected to the Reactor Coolant System and part of the reactor coolant pressure boundary, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the Reactor Coolant System under normal plant operating conditions is within the capacity of makeup systems that are operable from on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.

(b) (1) Piping of NPS 1" (DN25) and smaller, except steam generator tubing;

(2) Components and their connections in piping of NPS 1"(DN25) and smaller;

(c) Reactor vessel head connections and associated piping, NPS 2" (DN50) and smaller, made inaccessible by control rod drive penetrations.

(d) Welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration, or encapsulated by guard pipe.

2.2 Component/Piping Examination Development

A narrative discussion of Class 1 components subject to examination and testing are described in detail below:

2.2.1 Category B-A, Pressure Retaining Welds in Reactor Vessel

All volumetric examinations are performed from the inside and/or outside surface of the vessel using manual/automated inspection equipment, (as applicable).

Items B1.11, B1.12 - Shell Welds

Scope of Examination - includes volumetric examination of essentially 100% of all (12) longitudinal and approximately 2 to 3% of the (4) circumferential shell weld lengths that intersect with longitudinal welds. (Excludes the shell to flange weld). Deferral of examinations to the end of the interval is permissible.

- (4) B1.11 Circumferential shell welds, (2 to 3% required)
- (12) B1.12 Longitudinal shell welds, (12) required

Subject to Request for Alternate: 1ISI-001A, Submitted under letter, dated September 16, 2008.

Note: ASME Code Case N-526 Alternate Requirements for Successive Inspections of Class 1 and 2 Vessels, weld RV-WD-140, Lower Intermediate Shell Long Weld is being implemented. The three conditions of the Code Case have been met. See paragraph 2.3 for additional details and Appendix I for Code Case.

Items B1.21, B1.22 - Bottom Head Welds

Scope of Examination - includes volumetric examination of essentially 100% of accessible length of (2) circumferential and (14) meridional bottom head welds. Deferral of examinations to the end of the interval is permissible

- (2) B1.21 Circumferential head welds, (2) required
- (14) B1.22 Meridional head welds, (14) required

Items B1.21, B1.22 - Top Head Welds

Scope of Examination - includes volumetric examination of essentially 100% of accessible length of (1) circumferential and (8) meridional top head welds. Deferral of examinations to the end of the interval is permissible.

- (1) B1.21 Circumferential head welds, (1) required
- (8) B1.22 Meridional head welds, (8) required

Item B1.30 - Shell-to-Flange Weld

Scope of Examination - Volumetric examination of 100% of the shell to flange weld. 50% of weld is performed from seal surface and 50% of the weld performed from the shell side. Deferral of examinations to the end of the interval is permissible provided (a) no welded repairs/replacement activities have been performed either on the shell-to-flange weld or head-to-flange weld and (b) neither the shell-to-flange weld nor the head-to-flange weld contains identified flaws or relevant conditions that require successive inspections in accordance with IWB-2420(b).

- (1) B1.30 Circumferential shell to flange weld, (1) required

Note: ASME Code Case N-526 Alternate Requirements for Successive Inspections of Class 1 and 2 Vessels, weld RV-WD-099, Shell to Flange Weld is being implemented. The three conditions of the Code Case have been met. See paragraph 2.3 for additional details and Appendix I for Code Case.

The examination will be performed during the first and third inspection periods in conjunction with the nozzle examinations of Examination Category B-D (Program B). At least 50% of the weld shall be examined by the end of the First Inspection Period from the flange surface (seal surface), and the remainder by the end of the Third Inspection Period from the shell side.

Item B1.40 - Head to Flange Weld

Scope of Examination - includes volumetric and surface examination of essentially 100% of the reactor vessel head to flange weld length. Deferral of examinations to the end of the interval is permissible provided (a) no welded repairs/replacement activities have been performed either on the shell-to-flange weld or head-to-flange weld and (b) neither the shell-to-flange weld nor the head-to-flange weld contains identified flaws or relevant conditions that require successive inspections in accordance with IWB-2420(b).

- (1) B1.40 Circumferential head to flange weld, (1) required, Examine 1/3 Each Period

Note: Code Case N-623, Deferral of Inspections of shell-to-flange and head-to-flange welds of reactor vessel is applicable to RV-WD-001 Closure Head to Flange Weld provided the three conditions of the Code Case are met. See paragraph 2.3 for additional details and Appendix I for Code Case.

Item B1.51 - Repair Welds (Beltline Region)

- Not applicable to Nine Mile Point Nuclear Station

2.2.2 Category B-B, Pressure Retaining Welds in Vessels Other Than Reactor Vessels.

- This Examination Category is not applicable to Nine Mile Point Nuclear Station.

2.2.3 Category B-D, Full Penetration Welded Nozzles in Vessels (Inspection Program B)

Reactor Vessels: Items B3.90, B3.100 - Nozzle to Vessel Welds and Nozzle Inside Radius Section

Scope of Examination - Volumetric examination of 100% of all nozzles with full penetration welds to vessel shell (or head) and integrally cast nozzles.

- (40) B3.90 RPV Nozzle-to-Vessel Welds, (40) required
- (40) B3.100 RPV Nozzle Inner Radius Sections, (23) required

Note: **Note (2) of IWB-2500-1** - At least 25% but not more than 50% of the nozzles shall be examined by the end of the First Inspection Period, and the remainder by the end of the inspection interval.

Note: Code Case N-613-1, Ultrasonic examination of full penetration nozzles (B3.90) in vessel welds may be examined using the reduced examination volume (A-B-C-D-E-F-G-H). See Appendix I for Code Case requirements)

Note: Code Case N-648-1, Perform Visual examination in lieu of Ultrasonic examination on Nozzle Inner Radius Sections. Code Case is limited to those nozzle inner radius that can be 100% examined. In order to track examinations performed in accordance with Code Case, an Examination Item Number B3.200 has been assigned. See Appendix I for Code Case requirements. Feedwater and Control Rod Drive Nozzles are excluded from Code Case.

- (17) B3.200 RPV Nozzle Inner Radius Sections, (17) required

Pressurizer: Items B3.110, B3.120

- Not applicable to Nine Mile Point Nuclear Station.

Steam Generators: Items B3.130, B3.140

- Not applicable to Nine Mile Point Nuclear Station.

Heat Exchanger: Items B3.150, B3.160

- Not applicable to Nine Mile Point Nuclear Station.

2.2.4 Category B-F, Pressure Retaining Dissimilar Metal Welds In Vessel Nozzles

Reactor Vessel: Items B5.10, B5.20, B5.30

Volumetric and surface or surface examinations are required of all dissimilar metal safe end welds in each loop and connecting branch of the Reactor Coolant System. For the reactor vessel nozzle dissimilar metal safe end welds, the examination may be performed coincident with the vessel nozzle examinations required by Examination Category B-D.

Note: Examination Category B-F welds are also scheduled and examined as part of the IGSCC Augmented Inspection Program. The extent and frequency of the augmented inspections are defined in Section 6 of this document. Completed examinations shall be used to satisfy the percentage requirements of Inspection Program B, as applicable.

Scope of Examination - The selection and scheduling of examination items shall be in accordance with the Alternate Risk-Informed In-service Inspection Program, Section 7 and the Augmented IGSCC Program, Section 6 of this document.

- (33) B5.10 NPS 4" or Larger Nozzle-to-Safe end butt welds, (None) required
- (11) B5.20 Less than NPS 4" Nozzle-to-Safe end butt welds, (None) required
- (0) B5.30 Nozzle-to-Safe end socket welds, Not applicable to NMP1

Pressurizer Items B5.40, B5.50, B5.60

- Not applicable to Nine Mile Point Nuclear Station.

Steam Generators: Items B5.70, B5.80, B5.90

- Not applicable to Nine Mile Point Nuclear Station.

Heat Exchanger: Items B5.100, B5.110, B5.120

- Not applicable to Nine Mile Point Nuclear Station.

Piping: Items B5.130, B5.140. B5.150

- These item numbers have been removed from Examination Category B-F and been relocated under Examination Category B-J.

2.2.5 Category B-G-1 - Pressure Retaining Bolting, Greater Than 2 in. In Diameter

Examination includes all bolts, studs, nuts, bushings, and threads in flange stud holes.

(1) Bolting may be examined in place under tension, when the connection is disassembled, or when the bolting is removed.

(2) Bushings and threads in base material of flanges are required to be examined only when the connections are disassembled, Bushings may be examined in place.

(3) Volumetric examination of bolting for heat exchangers, pumps, or valves may be conducted on one heat exchanger, one pump, or one valve among group of heat exchangers, pumps, or valves that are similar in design, type, and function. In addition, when the component to be examined contains a group of bolted connections of similar design and size, such as flanged connections, the examination may be conducted on one bolted connection among the group.

(4) Visual examination of bolting for heat exchangers, pumps, or valves is required only when the component is examined under Examination Category B-B, B-L-2 or B-M-2. Examination of a bolted connection is required only once during the interval.

(5) The examination of flange bolting in piping system may be limited to one bolted connection among a group of bolted connections that are similar in design, size, function, and service.

(6) Examination includes 1 in. annular surface of flange surrounding each stud.

(7) When bolts or studs are removed for examination, surface examination meeting the acceptance standards of IWB-3515 may be substituted for volumetric examination.

Reactor Vessel: Items B6.10, B6.20, B6.40, B6.50

Scope of Examination - Examinations consist of visual (VT-1) examination of reactor vessel closure head nuts, volumetric examination of closure head studs, volumetric examination of the threads in the base material of the reactor vessel flange only when the connections is disassembled, and visual (VT-1) examination of the closure washers and bushings.

- (64) B6.10 Closure Nuts, (64) required, 1/3 each period

- (64) B6.20 Closure Studs, (64) required, 1/3 each period
- (64) B6.40 Threads in Flange, (64) required, 1/3 each period
- (128) B6.50 Closure Washers, (128) required, 1/3 each period
- (64) B6.50 Bushings, (64) required, 1/3 each period

Pressurizer: Items B6.60, B6.70, B6.80

- Not applicable to Nine Mile Point Nuclear Station.

Steam Generators: Items B6.90, B6.100, B6.110

- Not applicable to Nine Mile Point Nuclear Station.

Heat Exchanger: Items B6.120, B6.130, B6.140

- Not applicable to Nine Mile Point Nuclear Station.

Piping: Items B6.150, B6.160, B6.170

- Not applicable to Nine Mile Point Nuclear Station.

Pumps: Items B6.180, B6.190, B6.200

All bolts, studs, nuts, bushings, and flange surfaces. Examinations are applicable to five (5) Reactor Recirculation Pumps 32-187, 32-188, 32-189, 32-190 and 32-191.

Scope of Examination - Examinations consist of volumetric examination of bolts and studs, visual (VT-1) examination of flange surface when connection is disassembled, and visual (VT-1) examination of nuts, bushings and washers.

Note: Pump bolting is limited to the one pump selected under Examination Category B-L-2. Bolting may be examined in place under tension, when the connection is disassembled, or when the bolting is removed.

Bushings and threads in base material of flanges are required to be examined only when the connections are disassembled. Bushings may be examined in place. Flange surface requires 1 inch annular surface of flange surrounding each stud hole.

- (80) B6.180 Studs, 16 studs per pump, (16) one pump required
- (5) B6.190 Flange surfaces, (1) per pump, one pump required
- (240) B6.200 Nuts, Bushings, and Washers, (48) per pump, one pump required

Valves: Items B6.210, B6.220, B6.230

All bolts, studs, nuts, bushings, and flange surfaces. Examinations applicable to Feedwater, Core Spray and Shutdown Cooling systems and are limited to valve selected under Category B-M-2.

Scope of Examination - Examinations consist of volumetric examination of bolts and studs, visual (VT-1) examination of flange surface when connection is disassembled, and visual (VT-1) examination of nuts, bushings and washers.

- (5) B6.210 Valve Bolts, (3) required
- (5) B6.220 Valve Flanges, (3) required
- (9) B6.230 Valve Nuts, Bushings, washers, (5) required

2.2.6 Category B-G-2, Pressure Retaining Bolting, 2 in. And Less in Diameter

Items: B7.10, B7.20, B7.30, B7.40, B7.50, B7.60, B7.70

Visual VT-1 examination each interval of all bolts, studs, and nuts

Scope of Examination -. Examinations are limited to bolting only when a connection is disassembled or bolting removed.

(a) For vessels, pumps, or valves, examination of bolting is required only when the component is examined under Examination Category B-B, B-L-2, or B-M-2. Examination of bolted connection is required only once during the interval.

(b) The examination of flange bolting in piping systems may be limited to one bolted connection among group of bolted connections that are similar in design, size, function, and service. Examination required only when a flange is disassembled. Examination of bolted connection is required only once during the interval.

- (N/A) B7.10 Reactor Pressure Vessel, Not applicable to Nine Mile Point Nuclear Station
- (N/A) B7.20 Pressurizer, Not applicable to Nine Mile Point Nuclear Station
- (N/A) B7.30 Steam Generator, Not applicable to Nine Mile Point Nuclear Station
- (N/A) B7.40 Heat Exchanger, Not applicable to Nine Mile Point Nuclear Station
- (30) B7.50 Piping Flange Bolting, (7) required
- (80) B7.60 Pump, (5) pumps, 16 cap screws per pump, (1) pump required
- (80) B7.70 Valves, (15) required

2.2.7 Category B-J, Pressure Retaining Welds in Piping

Items: B9.11, B9.21, B9.22, B9.31, B9.32, B9.40

All dissimilar metal pipe welds, terminal ends, plus an additional number of piping welds so that 25% of all non-exempt circumferential and branch connection pipe welds are examined.

Note: All augmented Main Steam and Feedwater System welds, to the extent practical shall be used to satisfy the percentage requirements of Inspection Program B and the augmented requirements of NUREG 0313, Generic Letter 88-01, Supplement 1, shall also be used for satisfying the percentage requirements of Inspection Program B, to the extent practical. See Section 6.0 Augmented In-service Inspections/Examinations of this document for details.

Scope of Examination - The selection and scheduling of examination items shall be in accordance with the Alternate Risk-Informed In-service Inspection Program, Section 7 of this document.

- (344) B9.11 Circumferential welds, (None) required
- (114) B9.21 Circumferential welds, (None) required
- (N/A) B9.22 Circumferential welds of PWR High Pressure Safety Injection Systems, Not applicable to Nine Mile Point Nuclear Station
- (9) B9.31 Branch Conn. NPS 4" or Larger, (None) required

- (26) B9.32 Branch Conn. Less than NPS 4", (None) required
- (91) B9.40 Socket welds, (None) required

2.2.8 Category B-K, Welded Attachments For Vessels, Piping, Pumps and Valves

Items: B10.10, B10.20, B10.30, B10.40 Pressure Vessels, Piping, Pumps and Valve Welded Attachments

Scope of Examination - Surface examination to include essentially 100% of the length of the attachment weld at each integrally welded attachment subject to examination. For multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessel shall be selected for examination, Limited to Reactor Pressure Vessel skirt weld and stabilizers. For piping, pumps and valves, a sample of 10% of the welded attachments associated with the component supports selected for examination under IWF-2510 shall be examined.

- (6) B10.10 Vessel Integral Attachments, (3) required
- (76) B10.20 Piping Integral Attachments, (8) required
- (N/A) B10.30 Pump Integral Attachments, Not applicable to Nine Mile Point Nuclear Station
- (8) B10.40 Valve Integral Attachments, (1) required

2.2.9 Category B-L-1, Pressure Retaining Welds in Pump Casings, B-L-2, Pump Casings

Item: B12.10 Pump Casing Welds

Scope of Examination - 100% visual (VT-1) examination of all welds in one Pump in each group of pumps performing similar functions in the system.

- Not applicable to Nine Mile Point Nuclear Station Unit 1. The five (5) Reactor Recirculation Pumps do not have pump casing welds.

Item: B12.20 Pump Casing

Scope of Examination – Visual (VT-3) examination of the interior surfaces of one of the five (5) Reactor Recirculation Pumps, (32-187, 32-188, 32-189, 32-190 and 32-191), when disassembled for maintenance, repair, or volumetric examination. Examination of the internal pressure boundary shall include the internal pressure retaining surfaces made accessible for examination by disassembly. Pump to be identified when pump is disassembled.

- (5) B12.20 Recirculation. Pumps (1) Pump Required

2.2.10 Category B-M-1, Pressure Retaining Welds in Valve Bodies, B-M-2, Valve Bodies

Items: B12.30 Valve Body Welds

- Not applicable to Nine Mile Point Nuclear Station Unit 1. Valves less than NPS 4 do not have any valve body welds.

Items: B12.40 Valve Body Welds NPS 4" or Larger

Scope of Examination - Volumetric examination to include essentially 100% of weld length. Examinations are limited to at least one valve within each group of valves that are the same size, constructional design, and manufacturing method, and perform similar functions in the system. Applicable to Main Steam Relief Valves, valve group 03.

- (6) Valve Body Welds, (1) required

Item: B12.50 Valve Body Interior

Scope of Examination - Visual VT-3 examination of at least one valve in a group of valves that are the same size, constructional design (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system (such as containment isolation and system over pressure protection). Examinations are performed once per interval when disassembled for maintenance or repair. Valves are identified when the valve is disassembled.

- (68) B12.50 valves, (20) required

Table 2-1 below list all valves grouped by system, size, and type and subject to examination under B-M-2.

Table 2-1
VALVE GROUPINGS

Grp NO.	Sys ID	Size	Type Valve	B-G-1	B-G-2	B-M-2	B-M-1	Remarks	Selected
V01	01.0	24.0"	Globe	N/A	01-01-VB 01-02-VB 01-03-VB 01-04-VB	01-01-VBY 01-02-VBY 01-03-VBY 01-04-VBY	N/A	1 Valve among a group of valves	X
V02	01.0	6.0"	Gate	N/A	01-07-VB 01-08-VB 01-09-VB 01-10-VB 01-11-VB 01-12-VB	01-07-VBY 01-08-VBY 01-09-VBY 01-10-VBY 01-11-VBY 01-12-VBY	N/A	1 Valve among a group of valves	X
VO3	01.0	6.0"	Relief	N/A	01-102-A-SVB 01-102-B-SVB 01-102-C-SVB 01-102-D-SVB 01-102-E-SVB 01-102-F-SVB 01-102-*1-SVB 01-102-*2-SVB 01-102-F3-SVB	01-102-A-SVBY 01-102-B-SVBY 01-102-C-SVBY 01-102-D-SVBY 01-102-E-SVBY 01-102-F-SVBY 01-102-*1-SVBY 01-102-*2-SVBY 01-102-*F3-SVBY	01-102-A-WD-001 01-102-B-WD-001 01-102-C-WD-001 01-102-D-WD-001 01-102-E-WD-001 01-102-F-WD-001 01-102-*1-WD-001 01-102-*2-WD-001 01-102-F3-WD-001	1 Valve among a group of valves	X
V04	31.0	18.0"	Check	31-01R-VB 31-02R-VB	N/A CKV-31-01R1-VB2 CKV-31-02R1-VB2	31-01R-VBY 31-02R-VBY	N/A	1 Valve among a group of valves	X
V05	31.0	18.0"	Gate	N/A	31-07-VB 31-08-VB	31-07-VBY 31-08-VBY	N/A	1 Valve among a group of	X

Table 2-1
VALVE GROUPINGS

Grp NO.	Sys ID	Size	Type Valve	B-G-1	B-G-2	B-M-2	B-M-1	Remarks	Selected
								valves	
V06	32.0	28.0"	Gate	N/A	32-380-VB 32-381-VB 32-382-VB 32-383-VB 32-384-VB 32-375-VB 32-376-VB 32-377-VB 32-378-VB 32-379-VB	32-380-VBY 32-381-VBY 32-382-VBY 32-383-VBY 32-384-VBY 32-375-VBY 32-376-VBY 32-377-VBY 32-378-VBY 32-379-VBY	N/A	1 Valve among a group of valves	X
V07	33.0	6.0"	Gate	N/A	33-01R-VB 33-02R-VB	33-01R-VBY 33-02R-VBY	N/A	1 Valve among a group of valves	X
V08	38.0	14.0"	Gate	N/A	38-01-VB 38-02-VB 38-13-VB IV-38-13-VB	38-01-VBY 38-02-VBY 38-13-VBY	N/A	1 Valve among a group of valves	X
V09	39.0	10.0"	Globe	N/A	39-05-VB 39-06-VB	39-05-VBY 39-06-VBY	N/A	1 Valve among a group of valves	X
V10	38.0	14.0"	Check	38-12-VB	N/A	38-12-VBY	N/A	1 Valve among a group of valves	X
V11	40.0	12.0"	Check	40-03-VB 40-13-VB	N/A	40-03-VBY 40-13-VBY	N/A	1 Valve among a group of valves	X
V12	39.0	10.0"	Check	N/A	39-03-VB 39-04-VB	39-03-VBY 39-04-VBY	N/A	1 Valve among a group of valves	X
V13	33.0	6.0"	Check	N/A	33-03-VB	33-03-VBY	N/A	1 Valve among a group of valves	X
V14	40.0	12.0"	Gate	N/A	40-01-VB 40-02-VB 40-09-VB 40-10-VB 40-11-VB 40-12-VB	40-01-VBY 40-02-VBY 40-09-VBY 40-10-VBY 40-11-VBY 40-12-VBY	N/A	1 Valve among a group of valves	X

Table 2-1
VALVE GROUPINGS

Grp NO.	Sys ID	Size	Type Valve	B-G-1	B-G-2	B-M-2	B-M-1	Remarks	Selected
V15	39.0	10.0"	Gate	N/A	39-01R-VB 39-02R-VB	39-01R-VBY 39-02R-VBY	N/A	1 Valve among a group of valves	X
V16	40.0	6.0"	Gate	N/A	40-05-VB *40-06-VB	40-05-VBY 40-06-VBY	N/A	1 Valve among a group of valves	X
V17	00.0	6.0"	Safety Relief	N/A	CH-576-12A-B V-BK-01-119A CH-576-12B-B V-BK-01-119B CH-576-12C-B V-BK-01-119C CH-576-12D-B V-BK-01-119D CH-576-12F-B V-BK-01-119F CH-576-12G-B V-BK-01-119G CH-576-12H-B V-BK-01-119H CH-576-12J-B V-BK-01-119J CH-576-12M-B V-BK-01-119M	PSV-01-119A PSV-01-119B PSV-01-119C PSV-01-119D PSV-01-119F PSV-01-119G PSV-01-119H PSV-01-119J PSV-01-119M	N/A	1 Valve among a group of valves	X
V18	39.0	10.0"	Gate	N/A	N/A	39-07R-VBY 39-08R-VBY 39-09R-VBY 39-10R-VBY	N/A	1 Valve among a group of valves	X
V19	33.0	6.0"	Gate	N/A	33-04-VB 33-04-VB2	33-04-VBY	N/A	1 Valve among a group of valves	X

X = 1 Valve among a group of valves is required
* Required examinations that have been completed
** Valve was replaced

2.2.11 Category B-N-1, Interior of Reactor Vessel, B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, B-N-3, Removable Core Support Structures.

Item: B13.10 Vessel Interior

Scope of Examination - Visual VT-3 examination of accessible areas (areas above and below the reactor core made accessible for examination by removal of components during normal refueling). Accessible areas include but not limited to:

(**Above the Reactor Core**); Top Guide (Core Grid), Shroud Head and Steam Separator Assembly, Steam Dryer Assembly, Feedwater Sparger Assembly, Core Spray Lines and Sparger Assembly, Upper Core Grid;

(**Below the Reactor Core)**; Core Plate, Core Differential Pressure and Standby Liquid Control Line, Fuel Support Pieces, Control Rod Guide Tubes and Housings, and In-Core Flux Monitor Assemblies;

In addition, also includes the Closure Head Interior surface, RPV Nozzle Surface Entrance. Examinations are performed once each inspection period.

- (55) B13.10 Accessible Areas, once each period

Items: B13.20 Interior Attachments Within Beltline Region

Reactor Vessel (BWR)

Scope of Examination - Visual VT-1 examination of accessible welds of interior attachments within the Beltline region, to include the Lower Surveillance Specimen Holders (once per interval).

- (3) B13.20 Interior Attachments, (3) required

Item: B13.30 Interior Attachments - Beyond Beltline Region

Scope of Examination - Visual VT-3 examination of accessible welds of interior attachments beyond the Beltline region, to include the Conical Support to RPV weld, Dryer Support Brackets, Guide Rod Support Brackets, Feedwater Sparger Brackets, Shroud Support Stubs, Steam Dryer Hold Down Brackets, and Upper Surveillance Specimen Holders. (once per interval).

- (18) B13.30 Interior Attachments, (18) required

Item: B13.40 Core Support Structure

Scope of Examination - Visual VT-3 examination of accessible surfaces of core support structures, to include the Tie Rod Stabilizer, Shroud Ring to Conical Support weld, Upper Core Grid, Integrally Welded Core Support Structure, and Lower Core Grid. (once per interval).

- (7) B13.40 Accessible surfaces, (7) required

Item: B13.50, B13.60, B13.70 RPV (PWR's)

Reactor Vessel (PWR)

- Not applicable to Nine Mile Point Nuclear Station.

2.2.12 Category B-O, Pressure Retaining Welds in Control Rod Housings

There are thirty-two (32) peripheral CRD housings, and each housing has two (2) welds.

Item: B14.10 Welds in CRD Housings

Scope of Examination - Volumetric or surface examination of 10% of the peripheral CRD housings. Perform partial examination of 10% of the welds.

- (129) (32) Peripheral CRD Housings, (4) Peripheral CRD Housings (10%) required

2.2.13 Category B-P, All Pressure Retaining Components

Pressure Retaining Components: Items B15.10

Scope of Examination – Perform visual (VT-2) examination in conjunction with a system leakage test, each refueling outage prior to plant startup following reactor refueling outage. System pressure tests are conducted on All Class 1 systems and components in accordance with the Nine Mile Point Unit 1 System Pressure Testing Program.

2.2.14 Category B-Q, Steam Generator Tubing

Item: B16.10, B16.20 - Steam Generator Tubing in Straight Tube and U-Tube Design

- Not applicable to Nine Mile Point Nuclear Station.

2.3 Successive Inspections

The sequence of component examinations which was established during the first inspection interval shall be repeated during the fourth inspection interval, to the extent practical.

In accordance with IWB-2420(b), several welds examined during the third inspection interval were evaluated in accordance with IWB-3142.4, and were determined by analysis to qualify as acceptable for continued service. The areas containing these flaw indications shall require reexamination during the fourth inspection interval. Applicable welds are uniquely identified within the Ten-Year Implementation Schedule Tables.

1. **32-WD-050** - identified a surface planar indication during the 1997 refueling outage (RFO-14), on a 28.0" recirculation suction circumferential weld, reporting to be 1.25" long, and 0.25" deep, and located in the root area where a weld repair had been performed during the original weld fabrication. In a letter NMP1L 1201 dated 04/07/97, NMPNS submitted a flaw evaluation on 04/07/97, and the USNRC concluded in a letter and Safety Evaluation 04/30/97, that the flaw was acceptable for the next two (2) year fuel cycle.

   By letter 06/01/99 NMPNS reported the re-inspection results conducted during the 1999 refueling outage (RFO-15), that the flaw had not essentially changed, and will be re-examined during the next three inspection periods per IWB-2420(b). The USNRC letter and Safety Evaluation of 07/23/99 TAC No. MA5696 finds this weld acceptable for another two (2) year fuel cycle.

   Weld 32-WD-050 was re-inspected during the 2001 and 2005 refueling outage (RFO-16, RFO-18), and determined that the flaw had not essentially changed.

	Second Inspection Interval		Third Inspection Interval		
Weld ID	2ND. Period	3RD. Period	1ST. Period	2ND. Period	3RD. Period
32-WD-050	RFO-12 – 1997	RFO-15 - 1999	RFO-16 - 2001	RFO-18 - 2005	None

Re-examination Status: In accordance with IWB-2420(c) the re-examinations of the flaw indications remained essentially unchanged for three successive inspection periods; therefore weld 32-WD-050 will revert to the original schedule of successive inspections.

2. **RV-WD-140 and RV-WD-099** - identified two (2) subsurface flaws in weld RV-WD-140 and (7) subsurface flaws in RV-WD-099 during 1999 refueling outage (RFO-15).NMPNS submitted the flaw evaluation under letter NMP1L 1467, 09/14/99.

	Second Inspection Interval		Third Inspection Interval		
Weld ID	2ND. Period	3RD. Period	1ST. Period	2ND. Period	3RD. Period
RV-WD-140	None	RFO-15 – 1999	None	None	RFO-19 – 2007
RV-WD-099	None	RFO-15 – 1999	None	None	Scheduled

Weld RV-WD-140 was re-inspected during the 2007 refueling outage (RFO-19), and determined that the flaw indications had not essentially changed.

NMPNS by letter dated August 22, 2007, as supplemented by letter dated April 2, 2008 submitted for NRC staff review, a re-evaluation of flaws detected in the RPV vessel welds. On July 29, 2008, the NRC staff issued a Safety Evaluation (TAC No. MD6620) and concluded that NMP1 can be operated without repair of subject RPV welds for 46 EFPY.

<u>**Re-examination Status:**</u> Re-examinations of the flaw indications in accordance with IWB-2420(b) of vessel volumes found by volumetric examination to contain subsurface flaws are not required. (Ref. ASME Code Case N-526).

3. **Welds 32-WD-046 (surface planar), 32-WD-086 (surface planar), 32-WD-126 (surface planar), and 32-WD-168 (surface planar) - identified flaw indications.**

NMPNS submitted the flaw evaluation under letter NMP1L 1467, 09/14/99. USNRC Letter and Safety Evaluation of January 14, 2000 TAC No. MA6511 finds these welds acceptable till the end of operating cycle.

Welds 32-WD-050, 046, 086, 126 and 168 were re-examined during refueling outage sixteen (RFO-16), 2001. The results of the volumetric examinations determined that the flaws essentially did not change.

NMPNS submitted letter NMP1L 1673, dated September 13, 2002, notifying the NRC that changes to the regulatory commitments regarding re-inspection schedule was being changed from every refueling outage (Generic Letter 88-01, Category F) to IWB-2420(b) once each inspection period for three successive inspection periods.

USNRC Safety Evaluation dated 03-27-03; TAC MB6435 authorized use of IWB-2420(b) schedule for re-examinations.

Weld 32-WD-046, 086, 126 and 168 was re-inspected during the 2005 refueling outage (RFO-18), and determined that the flaw had not essentially changed.

	Second Inspection Interval		Third Inspection Interval		
Weld ID	2ND. Period	3RD. Period	1ST. Period	2ND. Period	3RD. Period
32-WD-050	RFO-12 – 1997	RFO-15 - 1999	RFO-16 - 2001	RFO-18 - 2005	None
32-WD-046	N/A	RFO-15 - 1999	RFO-16 -2001	RFO-18 - 2005	Scheduled
32-WD-086	N/A	RFO-15 – 1999	RFO-16 – 2001	RFO018- 2005	Scheduled
32-WD-126	N/A	RFO-15 - 1999	RFO-16 - 2001	RFO-18– 2005	Scheduled
32-WD-168	RFO-12 – 1997	RFO-15 - 1999	RFO-16 - 2001	RFO-18– 2005	Scheduled

Re-examination Status: In accordance with IWB-2420(c) the re-examinations of the flaw indications remained essentially unchanged for three successive inspection periods; therefore weld 32-WD-050, 046, 086, 126, and 168 will revert to the original schedule of successive inspections.

4. **RV-WD-005(C),** Closure Head Meridional weld, identified during RFO-17 a flaw indication located near the ID surface of the weld. The dimensions are 0.3" in the trough-wall direction and 7.0" in length. This flaw indication was determined to be a result of an original fabrication defect. This determination was confirmed by review of the original Combustion Engineering construction radiographs (RT). The film was digitized /enhanced and revealed two (2) locations indicative of a weld repair region. The flaw indications were indicative of lack of fusion. By letter NMP1L 1776, dated September 19, 2003, NMPNS requested USNRC approval of the structural flaw evaluation of the subsurface flaw indication. By E-mail on April 8, 2004, the staff forwarded a list of questions for discussion, directed toward treatment of cladding stress in the methodology supporting the submitted evaluation. A telephone conversation was held May 6, 2004 for which NMPNS provided justification for not considering cladding stresses in the evaluation. By letter NMP1L 1846, dated July 2, 2004, NMPNS provided responses to the NRC staff preliminary questions regarding structural flaw evaluation Methodology. USNRC Safety Evaluation dated December 21, 2004; TAC MC0930 authorized the evaluation of a detected subsurface flaw in reactor pressure vessel closure head meridional weld.

	Second Inspection Interval		Third Inspection Interval		
Weld ID	2ND. Period	3RD. Period	1ST. Period	2ND. Period	3RD. Period
RV-WD-005C	N/A	N/A	RFO-17 - 2003	N/A	RFO-19 – 2007

Re-examination Status: In accordance with IWB-2420(b) the re-examinations of the flaw indications will continue to be performed in the Fourth Inspection Interval.

5. **Weld 32-WD-164** – identified a surface planar flaw indication during the 2007 refueling outage (RFO-19), on a 28.0" recirculation nozzle N2D to Safe-End circumferential weld, reporting to be 1.59" long, and 0.27" deep, and located in an area where previous weld repairs had been performed during the original piping and safe end replacement activity in 1982/1983. In a letter NMP1L 2136, NMPNS submitted a flaw evaluation on 05/10/07, and the USNRC concluded in a letter and Safety Evaluation 05/05/08 and 05/20/08, (MD5700 / MD 5709) that the flaw was acceptable for the next two (2) year fuel cycle.

	Third Inspection Interval		Fourth Inspection Interval		
Weld ID	2ND. Period	3RD. Period	1ST. Period	2ND. Period	3RD. Period
32-WD-164	N/A	RFO-19 - 2007	Scheduled	Scheduled	Scheduled

Re-examination Status: Re-examinations of the flaw indications will continue to be performed during the Fourth Inspection Interval.

2.4 Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the updated in-service inspection program to identify that this Section of the existing inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.M1, ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD, and has been credited for aging management.

SECTION 3 - ASME CODE CLASS 2 SYSTEMS/COMPONENTS

3.0 ASME CODE CLASS 2 SYSTEMS/COMPONENTS

The ASME Code Class 2 System Boundaries were developed based upon the requirements of Regulatory Guide 1.26 and the NMP1 USAR.

The ASME Code Class 2 components and systems (including supports) subject to examination and testing are described in detail below:

__Note :__ Systems subject to examination requirements of this section are identified in the Table below:

Quality Group B, ASME Code Class 2 System Identifications	System Identification Numbers
Emergency Cooling System (ECS)	39.0
Core Spray System (CSS)	81.0, 81.1
Control Rod Drive System (CRDS)	44.2
Reactor Building Closed Loop Cooling System (RBCLCS)	70.0
Containment Spray System (CTN-SP)	80.0, 93.0, 93.1

Refer to Appendix B for Section XI Summary Table details.

3.1 ASME Code Exemptions

IWC-1220 - The following components or parts of components are exempted from the volumetric and surface examination requirements of IWC-2500;

3.1.1 IWC-1221 - Components within RHR, ECC and CHR Systems or portions of systems.

(a) For systems, except high pressure safety injection systems in pressurized water reactor plants:

- (1) piping NPS 4 (DN 100) and smaller
- (2) vessels, pumps, and valves and their connections in piping NPS 4 (DN 100) and smaller

(b) For high pressure safety injection systems in pressurized water reactor plants:

- (1) piping NPS 1-1 /2 (DN 40) and smaller
- (2) vessels, pumps, and valves and their connections in piping NPS 1- 1 / 2 (DN 40) and smaller

(c) Vessels, piping, pumps, valves, other components and component connections of any size in statically pressurized, passive (i.e., no pumps) safety injection systems of pressurized water reactor plants.

(d) Piping and other components of any size beyond the last shutoff valve in open ended portions of systems that do not contain water during normal plant operating conditions.

3.1.2 IWC-1222 - Components within systems or portions of systems other than RHR, ECC and CHR Systems

(a) For systems, except auxiliary feedwater systems in pressurized water reactor plants:

  (1) piping NPS 4 (DN 100) and smaller

  (2) vessels, pumps, and valves and their connections in piping NPS 4 (DN 100) and smaller

(b) For auxiliary feedwater systems in pressurized water reactor plants:

  (1) piping NPS 1-1 / 2 (DN 40) and smaller

  (2) vessels, pumps, and valves and their connections in piping NPS 1-1 / 2 DN 40) and smaller

(c) Vessels, piping, pumps, valves, other components, and component connections of any size in systems or portions of systems that operate (when the system function is required) at a pressure equal to or less than 275 psig (1900 kPa) and at a temperature equal to or less than 200 degrees F (93 degrees C).

(d) Piping and other components of any size beyond the last shutoff valve in open ended portions of systems that do not contain water during normal plant operating conditions.

3.1.3 IWC-1223 – Inaccessible Welds

Welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration, or encapsulated by guard pipe.

3.2 Component/Piping Examination Development

A narrative discussion of Class 2 components subject to examination and testing are described in detail below:

3.2.1 Category C-A, Pressure Retaining Welds in Pressure Vessels

Item C1.10 - Shell Circumferential Welds

Scope of Examination: 100% of all welds at gross structural discontinuities only. The examinations are limited to one vessel among a group of vessels.

- Not applicable to Nine Mile Point Nuclear Station

Item C1.20 - Head Circumferential Welds

Scope of examination: 100% of head-to-shell welds, (limited to one vessel of multiple vessels). .

- Not applicable to Nine Mile Point Nuclear Station

Item C1.30 - Tubesheet to Shell Welds

Scope of examination: 100% of Tubesheet to shell welds (limited to one vessel of multiple vessels). Components applicable to this examination category are the four (4) Emergency Condenser Heat Exchanger's (111, 112, 121, and 122), and four (4) Reactor Containment Spray Heat Exchanger's (111, 112, 121 and 122).

- (16) C1.30 Shell Circ. Welds, (4) welds required

3.2.2 Category C-B, Pressure Retaining Nozzle Welds in Vessels

Item C2.10 and C2.11 Nozzles in Vessels < ½ in. Nominal Thickness

Scope of examination: All nozzles at terminal ends of piping runs. Components applicable to four (4) Reactor Containment Spray Heat Exchanger's (80-HE111, 112, 121, and 122).

- (8) C2.11 Nozzle to Shell Welds, (2) required

Item C2.20 - Nozzles Without Reinforcing Plate in Vessels > ½ in. Nominal Thickness

Components applicable to four (4) Emergency Condenser Heat Exchanger's (39-HE111, 112, 121, and 122).

Item C2.21 - Nozzle to Shell or Head Welds

Scope of Examination - All nozzles at terminal ends of piping runs (limited to one vessel of multiple vessels). Includes only those piping runs selected for examination under Examination Category C-F.

- (8) C2.21 Nozzle to Shell or Head Welds, (2) required

Item C2.22 - Nozzle Inside Radius Section

Scope of Examination - All nozzles at terminal ends of piping runs (limited to one vessel of multiple vessels).

- Not applicable to Nine Mile Point Nuclear Station

Note: The Emergency Condenser Heat Exchanger Nozzles do not have inner radius sections.

Item C2.30 – Nozzles with Reinforcing Plate

Nozzles in vessels > 1/2" Nominal Thickness, not applicable to Nine Mile Point Nuclear Station

Item C2.31 – Nozzles with Reinforcing Plate in vessel > ½ in., Reinforcing Plate Welds to Nozzle and Vessel

Scope of Examination - All nozzles at terminal ends of piping runs (limited to one vessel of multiple vessels).

- Not applicable to Nine Mile Point Nuclear Station.

Item C2.32 Nozzle to Shell (Nozzle to Head or Nozzle to Nozzle) Welds When Inside of Vessel is Accessible

- Not applicable to Nine Mile Point Nuclear Station.

Item C2.33 - Nozzle to Shell (Nozzle to Head or Nozzle to Nozzle) Welds When Inside of Vessel is Inaccessible

Scope of Examination - Visual VT-2 of tell tale hole in reinforcing plates (limited to one vessel of multiple vessels). Examination performed in accordance with system pressure test program.

- Not applicable to Nine Mile Point Nuclear Station

3.2.3 Category C-C, Welded Attachments for Vessels, Piping, Pumps & Valves

Examination includes essentially 100% of the length of the attachment weld at each attachment subject to examination.

Item C3.10 - Pressure Vessels, Welded Attachments

Scope of Examination - 100% of the length of the attachment weld of only one integrally welded attachment of only one of the multiple vessels selected. Applicable to four (4) Reactor Containment Spray Heat Exchangers, (80-HE111, 112, 121, and 122).

- (8) C3.10 Integral Welded Attachments, (2) required

Item C3.20 - Piping, Welded Attachments

Scope of Examination - 100% of the length of the attachment weld of 10% of the welded attachments associated with the component supports selected for examination under IWF-2510 is required.

- (201) Integrally welded attachments, (21) required

Note: NMP1 is unique in the way that the integral attachments have been identified. Each integral attachment may provide more than one identification number. Example: One lug may identify the weld as 001, 002, 003 and 004 etc. for the same weld on the same lug. .

Item C3.30 - Pumps, Welded Attachments

Scope of Examination - 100% of the length of the attachment weld of 10% of the welded attachments associated with the component supports selected for examination under IWF-2510 is required.

- Not applicable to Nine Mile Point Nuclear Station.

Item C3.40 - Valves, Welded Attachments

Scope of Examination - 100% of the length of the attachment weld of 10% of the welded attachments associated with the component supports selected for examination under IWF-2510 is required.

- Not applicable to Nine Mile Point Nuclear Station

3.2.4 Category C-D, Pressure Retaining Bolting > 2" in Diameter

Items C4.10 Pressure Vessel

Scope of Examination – Volumetric examination of 100% bolts and studs at each bolted connection of components required to be inspected. The examination of bolting for vessels may be performed on one vessel in a group of vessels.

- Not applicable to Nine Mile Point Nuclear Station.

Items C4.20 Piping

- Not applicable to Nine Mile Point Nuclear Station

Items C4.30 Pumps

- Not applicable to Nine Mile Point Nuclear Station

Items C4.40 Valves

- Not applicable to Nine Mile Point Nuclear Station

3.2.5 Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping

Items C5.10, C5.11, C5.20, C5.21, C5.22, C5.30, C5.40, C5.41

Welds were initially selected for examination, included 7.5%, but not less than 28 welds, of all austenitic stainless steel of high alloy welds not exempted by IWC-1220.

- (75) C5.11 Circ. Welds, (28) welds were initially required
- (121) C5.12 Long Welds
- (0) C5.41 Circ. Welds, Not applicable to NMPNS

Scope of Examination - The selection and scheduling of examination items shall be in accordance with the Alternate Risk-Informed In-service Inspection Program, Section 7 of this document.

3.2.6 Category C-F-2, Pressure Retaining Welds in Carbon or Low Alloy Steel Piping

Items C5.50, C5.51, C5.60, C5.61, C5.70, C5.80, C5.81

Welds initially selected for examination, included 7.5%, but not less than 28 welds, of all carbon and low alloy steel welds not exempted by IWC-1220.

- (642) C5.51, Circ. Welds
- (16) C5.52, Long Welds
- (0) C5.61, Circ. Welds
- (0) C5.62, Long welds

- (0) C5.70, Socket Welds
- (22) C5.81, Circ. Welds
- (0) C5.82, Long Welds

Scope of Examination - The selection and scheduling of examination items shall be in accordance with the Alternate Risk-Informed In-service Inspection Program, Section 7 of this document.

3.2.7 Category C-G, Pressure Retaining Welds in Pumps and Valves

Items C6.10 & C6.20

Scope of Examination - 100% of welds in all components in each piping run examined under Examination Category C-F. This Category is applicable to four (4) Reactor Containment Spray Pumps (80-P111, 112, 121, and 122), and four (4) Reactor Core Spray Pumps (40-P111, 112, 121 and 122). In the case of multiple pumps and valves of similar design, size, function, and service, the examination of only one pump and one valve among each group of multiple pumps and valves is required. The examination may be performed from either the inside or outside surface of the component.

- (80) C6.10 Pump Casing Welds, (20) required
- (0) C6.20 Valve Casing Welds, Not applicable to NMPNS

System	Pump No.	Affected Welds	Reason Affected	Requirements
80.0 Reactor Containment Spray	122	80-23-WD-009 80-23-WD-012 80-23-WD-014 80-23-WD-010 80-23-WD-011	Embedded Embedded Embedded When disassembled When disassembled	Visual and surface examinations when disassembled
81.0 Reactor Core Spray	121	81-03-WD-009 81-03-WD-012 81-03-WD-014 81-03-WD-010 81-03-WD-011	Embedded Embedded Embedded When disassembled When disassembled	Visual and surface examinations when disassembled

Note: ASME Code Case N-617, Alternative Examination Distribution Requirements for Table IWC-2500-1, Examination Category C-G, Pressure Retaining Welds in Pumps and Valves, Section XI, Division 1 may be used. See Appendix I for Code Case requirements.

3.2.8 Category C-H, All Pressure Retaining Components

Items C7.10, C7.20, C7.30, C7.40, C7.50, C7.60, C7.70 & C7.80

Scope of Examination – System leakage tests are conducted on all Class 2 pressure retaining boundaries each inspection period in accordance with the Nine Mile Point Unit 1 System Pressure Testing Program.

3.3 Successive Inspections

The sequence of component examinations established during the first inspection interval will be repeated during the forth inspection interval, to the extent practical.

Note: Code Case N-624, modification of the sequence of examinations established during the first interval. See Appendix I for Code Case requirements

3.4 Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the updated in-service inspection program to identify that this Section of the existing inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.M1, ASME Section XI In-service Inspection Subsections IWB, IWC, and IWD, and has been credited for aging management as part of the NMPNS License renewal Application (LRA).

SECTION 4 - ASME CODE CLASS 3 SYSTEMS/COMPONENTS

4.0 CLASS 3 SYSTEMS/COMPONENTS

The ASME Code Class 3 system boundaries subject to examination and testing were developed based upon the requirements of Regulatory Guide 1.26, and the NMP1 USAR. The ASME Code Class 3 components and systems subject to examination and testing are described in detail below:

Note: Systems subject to examination requirements of this section are identified in the Table below.

ASME Code Class 3 Systems	System Identification
Emergency Cooling System (ECS)	39.0
Spent Fuel Pool Cooling System (SFPC)	54.0
Condensate Transfer System (CTS)	57.0
Reactor Building Closed Loop Cooling System (RBCLCS)	70.0
Service Water System (SWS)	72.0
Containment Spray System (CSS)	93.0

Refer to Appendix C for Section XI Summary Table Details.

4.1 ASME Code Exemptions Employed

4.1.1 IWD-1220

The following components or parts of components are exempted from the VT-1 visual examination requirements of IWD-2500:

(a) Piping NPS 4 (DN100) and smaller;

(b) Vessels, pumps, and valves and their connections in piping NPS 4 (DN100) and smaller;

(c) Components that operate at a pressure of 275 psig (1900 kPa) or less and at a temperature of 200° F (93°C) or less in systems (or portions of systems) whose function is not required in support of reactor residual heat removal, containment heat removal, and emergency core cooling;

(d) Welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration, or encapsulated by guard pipe.

[1] In piping is defined as having a cumulative inlet and a cumulative outlet pipe cross-sectional area neither of which exceeds the nominal OD cross-sectional area of the designed size

4.2 Component/System Examination Development

A narrative discussion of Class 3 components subject to examination and testing are described in detail below:

4.2.1 Category D-A Welded Attachments for Vessels, Piping, Pumps and Valves

Selected samples of welded attachments shall be examined each inspection interval. All welded attachments selected for examination shall be those most subject to corrosion, as determined by NMPNS.

For multiple vessels of similar design, function and service, the welded attachments of only one of the multiple vessels shall be selected for examination.

For welded attachments of piping, pumps, and valves, a 10% sample shall be selected for examination. This percentage sample shall be proportional to the total number of nonexempt welded attachments connected to piping, pumps, and valves in each system subject to these examinations.

Item D1.10 - Pressure Vessel Welded Attachments

Scope of Examination - Perform Visual (VT-1) examination of 100% of the weld length of each welded attachments required each interval. Applicable to Emergency Condenser Heat Exchanger's (39-HE111, 112, 121 and 122). Reactor Building Closed Loop Cooling Heat Exchanger's (70-HE13, 14, and 15). Shutdown Cooling Water Heat Exchanger's (38-HE11, 12, and 13). Spent Fuel Pool Cooling Heat Exchanger's (54-HE11 and 12).

- (28) D1.10 Pressure Vessel Integral Welded Attachments, (3) Required

Item D1.20 - Piping Welded Attachments

Scope of Examination - Perform Visual (VT-1) examination of 100% of the weld length of each welded attachment required each interval.

- (157) D1.20 Piping Integral Welded Attachments, (16) required

Item D1.30 - Pumps Welded Attachments

Scope of Examination - Perform Visual (VT-1) examination of 100% of the weld length of each welded attachment required each interval.

- Not applicable to Nine Mile Point Nuclear Station

Item D1.40 - Valve Welded Attachments

Scope of Examination - Perform Visual (VT-1) examination of 100% of the weld length of each welded attachment required each interval.

- Not applicable to Nine Mile Point Nuclear Station

4.2.2 Category D-B, All Pressure Retaining Components

The pressure retaining components within the boundary of each system are visually examined (VT-2) in conjunction with a system leakage test each inspection period in accordance with the Nine Mile Point Nuclear Station Unit 1 System Pressure Test Program.

Item D2.10 – Pressure Retaining Components

Scope of Examination/Testing – Perform visual (VT-2) examination in conjunction with a system leakage test each inspection period.

4.3 Successive Inspections

The sequence of component examinations which was established during the first inspection interval will be repeated during the Forth Inspection Interval, to the extent practical.

Note: Code Case N-624, modification of the sequence of examinations established during the first interval. (See Appendix I for requirements)

4.4 Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the existing in-service inspection program to identify that Section 4 of the inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.M1, ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD, and has been credited for aging management as part of the NMP License Renewal Application (LRA).

SECTION 5 - ASME CODE CLASS 1, 2 AND 3 COMPONENT SUPPORTS

5.0 CLASS 1, 2 AND 3 COMPONENT SUPPORTS - IWF

Component supports selected for examination shall be the supports of those components that are required to be examined under IWB, IWC and IWD. Class 1, 2, 3, and MC supports receive a Visual (VT-3) examination to determine their general mechanical and structural condition, and when required, conditions relating to their operability. (Refer to Appendix D for Section XI Summary Table details).

5.1 Supports Exempt from Examination

Supports exempt from the examination requirements of IWF-2000 are those connected to piping and other items exempted from volumetric, surface, or VT-1 or VT-3 visual examination by IWB-1220, IWC-1220, IWD-1220, and IWE-1220.See Sections 2, 3 and 4 of this document for exemptions.

a. In addition, portions of supports that are inaccessible by being encased in concrete, buried underground, or encapsulated by guard pipe are also exempt from the examination requirements of IWF-2000.

b. NMPNS has determined that a support that does not fully meet the definition of a component support, as defined within ASME Section XI, Article IWA-9000, Glossary definition for Component Support, is exempt for examination. Pipe whip restraints, insulation lugs, or unused pipe supports, which do not provide structural stability or support the weight of the pipe, are exempt.

5.2 Support Examination Development

A narrative discussion of Class 1, 2 and 3 component supports subject to examination are described in detail below:

In order to assure that a representative sample of supports within each ASME Code Class is examined, (Code Examination Category F-A, Examination Item Numbers F1.10 Class 1, F1.20 Class 2, F1.30 Class 3, and F1.40 other than piping), selection was based on Class, System and Type, to the extent practical [1].

Table 5-1 provides the Examination Category selection process.

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[1] All component supports subject to examination have been classified (a, b, c, d, etc.), to the extent practical. As these supports could be classified by one or more of the suffixes for the same support, only one suffix was selected. These classifications are identified in the 10-year inspection Tables.

Table 5-1
CATEGORY F-A SELECTION PROCESS

Exam Item No.	ASME Code Class	Applicable System	Type of Supports
F1.10	Code Class 1	44.2 Control Rod Drive 39.0 Emergency Condenser	A. Supports such as one directional rod hangers
F1.20	Code Class 2	31.0 Feedwater	B. Supports such as Multidirectional restraints
F1.30	Code Class 3	42.1 Liquid Poison	
F1.40	Other than piping	01.0 Main Stream 33.0 Reactor Clean Up 40.0 Reactor Core Spray 32.0 Reactor Recirculation 38.0 Reactor Shutdown Cooling 80.0 Reactor Containment Spray 54.0 Spent Fuel Pool Cooling 72.0 Service Water 70.0 Reactor Building Water Closed Loop Cooling	C. Supports that allow thermal movement,.springs

5.2.1 Examination Category F-A Supports

Item F1.10 - Class 1 Piping Supports

Scope of Examination – Visual (VT-3) examination of 25% of all non-exempt Class 1 piping supports.

- (37) F1.10A Supports, (9) required
- (22) F1.10B Supports; (6) Required
- (99) F1.10C Supports, (25) Required

(158) Class 1 Supports Times 25% = 40 Required

Note: The total percentage sample (25% or 40 of Class 1 piping supports) shall be comprised of supports from each system, where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system

(a) To the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval.

(b) Additional selections as may be determined to assure that a representative sample of each type and function within each system is examined.

Item F1.20 - Class 2 Piping Supports

Scope of Examination – Visual (VT-3) examination of 15% of all non-exempt Class 2 piping supports.

- (114) F1.20A Supports, (17) Required
- (144) F1.20B Supports, 22) Required
- (94) F1.20C Supports, (14) Required

(352) Class 2 Supports Times 15% = 53 Required

Note: The total percentage sample (15% or 53 of Class 2 piping supports) shall be comprised of supports from each system, where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system and distributed and selected as follows:

(a) To the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval.

(b) Additional selections as may be determined to assure that a representative sample of each type and function within each system is examined.

Item F1.30 - Class 3 Piping Supports

Scope of Examination - Visual (VT-3) examination of 10% of all non-exempt Class 3 piping supports

- (305) F1.30A Supports, (31) Required
- (187) F1.30B Supports, (19) Required
- (10) F1.30C Supports, (1) Required

(502) Class 3 Supports Times 10% = (50) Required

Note: The total percentage sample (10% or 50 of Class 3 supports) shall be comprised of supports from each system, where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system and distributed and selected as follows:

(a) To the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval.

(b) Additional selections as may be determined to assure that a representative sample of each type and function within each system is examined.

Item F1.40 - Supports Other than Piping Supports (Class 1, 2, 3 and MC)

Scope of Examination - Visual (VT-3) examination of 100% of all non-exempt Supports, other than piping supports. This item is applicable to the Emergency Condenser Heat Exchanger's (39-HE111, 112, 121, and 122); Emergency Service Water Pump (72-P11 and 12); Reactor Building Closed Loop Cooling Heat Exchanger (70-HE13, 14, and 15), Pump (70-P01, 02, and 03) and MU Tank (T126); Reactor Containment Spray Heat Exchanger's (80-HE111, 112, 121 and 122); Reactor Containment Spray(80-P111, 112, 121, and 122) and Raw Water Pumps (111, 112, 121 and 122); Reactor Core Spray Pumps (40-P111,121 and 122) and Reactor Core Spray Topping Pumps (111, 112, 121 and 122); Reactor Recirculation Pumps (32-P11, 12, 13, 14 and 15); Reactor Vessel Supports; Shutdown Cooling Water Heat Exchanger, (38-HE11, 12, and 13); Spent Fuel Pool Cooling Filter (11 and 12) ; Spent Fuel Pool Cooling Heat Exchanger and Pumps (54-HE11 and 12) and the Spent Fuel Pool Cooling Surge Tank. For multiple components, only one of the multiple components are required to be examined.

Note: For multiple components, other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined.

The recording of Hot or Cold positions will be performed in conjunction with the visual VT-3 examination.

- (58) F1.40B Supports, (20) Required
- (5) F1.40C Supports, (1) Required

Note: For multiple components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined.

The recording of Hot or Cold positions will be performed in conjunction with the VT-3 examination.

5.3 Snubber Examination and Performance Testing Program

The Nine Mile Point Unit 1 Snubber Examination and Performance Testing of Dynamic Restraints (Snubbers) are outside the scope of this inspection plan and schedule.

5.4 Multiple Component Support Equivalency Tables

For multiple components within a system of similar design, function and service, the supports of only one of the multiple components are required to be examined.

5.4.1 Class 1 Multiple Supports

The five Reactor Recirculation Pumps are the only Class 1 pumps at NMP1. The Reactor Pressure Vessel is the only Class 1 vessel.

Table 5-4-1
CLASS 1 MULTIPLE SUPPORTS

Group	System ID	Component ID	Support ID	Reference Drawing	Selected √
RPV-01	00.1 Reactor Pressure Vessel	N/A	RV-SB-1A	F-45183-C S28	
			RV-SB-1B		
			RV-SB-2A		
			RV-SB-2B		
			RV-SB-3A		

Table 5-4-1
CLASS 1 MULTIPLE SUPPORTS

Group	System ID	Component ID	Support ID	Reference Drawing	Selected √
			RV-SB-3B		
			RV-SB-4A		
			RV-SB-4B		
			RPV-NR02-SKT/BLT	E-231-569-1	√
PMP-01	32.0 Reactor Recirculation System Pumps	PMP-32-187	32-187-11	C-26846-C S1	√
			32-11-H1	ISI-PUMP-001	
		PMP-32-188	32-188-12		
			32-12-H1		
		PMP-32-189	32-189-13		
			32-13-H1		
		PMP-32-190	32-190-14		
			32-14-H1		
		PMP-32-191	32-191-15		
			32-15-H1		

5.4.2 Class 2 Multiple Supports

Table 5-4-2
CLASS 2 MULTIPLE SUPPORTS

Group	System ID	Component ID	Support ID	Reference Drawing	Selected √
HTX-01	80.0 Containment Spray Heat Exchanger	HTX-80-122	80-13	F45183C-15B	√
		HTX-80-112	80-14		
		HTX-80-121	80-33		
		HTX-80-111	80-34		
PMP-02	81.0 Reactor Core Spray Pumps	PMP-81-111	81-23	F-45183C-13	√
		PMP-81-112	81-24		
		PMP-81-121	81-03		
		PMP-81-122	81-04		
PMP-03	81.0 Reactor Core Spray Topping Pumps	PMP-81-111	PMP-81-50	F-45183C-13	√
		PMP-81-112	PMP-81-49		
		PMP-81-121	PMP-81-51		
		PMP-81-122	PMP-81-52		

5.4.3 Class 3 Multiple Supports

Table 5-4-3
CLASS 3 MULTIPLE SUPPORTS

Group	System ID	Component ID	Support ID	Reference Drawing	Selected √
HTX-02	38.0 Shutdown Cooling Heat Exchanger	HTX-38-135	38-SDC-HX-11	C-26855C-7	√
		HTX-38-132	38-SDC-HX-12		
		HTX-38-129	38-SDC-HX-13		
HTX -03	60.0 Emergency Cooling	HTX-60-111	HX-60-46	F-45183C-11A	

Table 5-4-3
CLASS 3 MULTIPLE SUPPORTS

Group	System ID	Component ID	Support ID	Reference Drawing	Selected √
	Heat Exchanger	HTX-60-112	HX-60-45		
		HTX-60-121	HX-60-44		
		HTX-60-122	HX-60-42		√
HTX-04	54.0 S pent Fuel Pool Heat Exchanger	HTX-54-11	FP-54-05	C-27171-C	
		HTX-54-12	FP-54-04		√
HTX-05	72.0 Reactor Building Closed Loop Cooling Heat Exchanger	HTX-72-11	70-13R	F-63027-C	√
		HTX-72-12	70-14R		
		HTX-72-13	70-15R		
PMP-04	54.0 Spent Fuel Pool Pumps	PMP-54-11	PMP-54-01	F-63041-C	√
		PMP-54-12	PMP-54-02		
PMP-05	57.0 Condensate Transfer Pump	PMP-57-12	57-11	F-63048	√
		PMP-57-11	57-12		
PMP-06	70.0 Reactor Building Closed Loop Cooling Pump	PMP-70-11	70-01	C-26855-C	√
		PMP-70-12	70-02		
		PMP-70-13	70-03		
PMP-07	72.0 Emergency Service Water Pump	72-11	72-04	F-45235-C	√
		72-12	72-03		
PMP-08	80 Containment Spray Pump	PMP-80-122	80-23	F-45183C-15C	
		PMP-80-121	80-03		√
		PMP-80-111	80-04		
		PMP-80-112	80-24		
PMP-09	93.0 Containment Spray Pump	PMP-93-02	RWP-111	F-45183C-16	√
		PMP-93-01	RWP-112		
		PMP-93-04	RWP-121		
		PMP-93-03	RWP-122		
Filters	54.0 Spent Fuel Pool	FP-54-03	Filter No. 11	C-27171-C	√
		FP-54-06	Filter No. 12		
Tank-01	54.0 Spent Fuel Pool Tank	TK-54-68	Surge Tank	C-27171-C	√
		TK-54-69			
Tank-02	57.0 Condensate Storage Tank	57-01	57-11 North Surge	F-63003-C	√
		57-02	57-12 South Surge		
Tank-03	60.0 Emergency Cooling Tank	TK-60-10	MU 60-11	F-63017.C	√
		TK-60-09	MU 60-12		
Tank-04	71.0 Reactor Building Closed Loop Cooling Tank	71-126	CLC M.U. Tank	C-26855-C	√

5.5 Successive Inspections

The sequence of component support examinations established during the first inspection interval shall be repeated during each successive inspection interval, to the extent practical.

5.6 Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the in-service inspection program to identify that Section 5 of the inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.S3, ASME Section XI , Subsection IWF, and has been credited for aging management as part of the NMP License Renewal Application (LRA).

SECTION 6 - AUGMENTED INSPECTIONS/EXAMINATIONS

6.0 AUGMENTED INSPECTIONS/EXAMINATIONS

This section of the Fourth In-service Inspection Plan and schedule provides a summary description of augmented in-service examination requirements. Augmented in-service inspections are those additional inspections/examinations required by the USNRC for systems and components for which the USNRC deems that added assurance of structural reliability is necessary. Augmented in-service inspections are performed in addition to the ASME Code, Section XI inspection/examination requirements. NMPNS plans on utilizing, where applicable, the results of the augmented in-service inspections to satisfy the requirements of the ASME Section XI required examinations.

Refer to Appendix E for Summary Table details.

6.1 Generic Letter 88-01, Augmented IGSCC Examinations

The requirements for an augmented IGSCC inspection program were initially provided by Generic Letter GL 88-01, GL 88-01 Supplement 1, "Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping" and NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

Generic Letter 88-01, USNRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, presents the USNRC staff positions on materials, processes, water chemistry, weld overlay reinforcement, partial replacement, stress improvement of cracked weldments, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion, leak detection, and reporting requirements. The technical bases for these positions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and process Guidelines for BWR Coolant Pressure Boundary Piping."

Generic Letter 88-01, Supplement #1 - In its first supplement to GL 88-01, issued February 4, 1992, the USNRC provided several acceptable alternative staff positions to those originally in the Generic Letter. NMPNS elected to use one of those alternative staff positions.

One of those positions allowed sample expansion for Category D welds to be limited to the piping system where the crack was found. NMPNS elected to examine 50% of Category D welds, by system loop, each cycle and used this relaxation of sample expansion criteria should cracking be found.

BWRVIP-75 BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules

The Electric Power Research Institute (EPRI) report TR-113932, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," dated October 1999, was submitted to the U. S. Nuclear Regulatory Commission (USNRC) for staff review by letter dated October 27, 1999. The BWRVIP-75 report proposed revisions to the extent and frequency for piping inspection contained in Generic Letter (GL) 88-01. The BWRVIP-75 report also provides justification for the proposed inspection criteria for Category A through E welds for the respective conditions of normal water chemistry (NWC) and hydrogen water chemistry (HWC).

The USNRC staff reviewed the BWRVIP-75 report, and proposed revisions and determined that the revised guidance for revisions of Generic Letter 88-01 inspection schedules, is acceptable for inspection of safety-related BWR piping welds, (USNRC Safety Evaluation, TAC No. MA5012, dated May 14, 2002), in lieu of the inspection guidance in GL 88-01 and NUREG-0313, Rev. 2, or as the technical basis for plant-specific request for a license amendment to change technical specifications requiring GL 88-01 or NUREG-0313, Rev. 2

inspections. Further the staff determined that the revised BWRVIP-75 guidance is acceptable for NMPNS referencing as the technical basis for relief from, or as an alternative to, the ASME Code and 10 CFR 50.55a, in order to use the sample schedules and frequencies specified in the revised BWRVIP-75 report that are less than those required by the ASME Code.

NMPNS has determined that the proposed reduction of inspection frequency is supported by the improved quality of reactor water chemistry. Based upon the staff recommendations, in order to qualify for the reduced inspection frequency as defined below, the average conductivity in reactor water coolant should not exceed the recommendations in the BWRVIP-29 report, or later revisions. The average conductivity can be calculated from the measurements made during the entire inspection interval based on the total operating time at a temperature at or above 200 degrees F.

GL 88-01 divided piping welds into 7 categories lettered A through G, of which three are applicable to NMPNS. They are categories A, D, and E. A summary of these categories as well as their applicability and scope of examinations based on normal water chemistry (NWC) is defined below:

6.1.1 IGSCC Category A Weldments - Identifies welds which are fabricated from resistant materials.

IGSCC Category A weldments are those welds with no known cracks, which have a low probability of incurring IGSCC problems, because they are made entirely of IGSCC resistant material or have been solution heat treated after welding. Augmented examinations required by GL 88-01 are identified as defined below.

Scope of Examination - IGSCC Category A welds have been incorporated within the Alternate Risk-Informed Inservice Inspection Program, Section 7 of this document. There are one hundred thirty two (132) Category A welds at NMP1, one hundred sixteen (116) are ASME Code Class 1 and twelve (12) welds are ASME Code Class 2.

As a result of Guidance to the industry in document BWRVIP-75-A, NMPNS issued Condition Report CR-NM-2007-3533 that required NMPNS to re-evaluate the categorization of subject welds (A to D) and the examination frequency during the development of the fourth ten-year in-service inspection program update. As a result of the re-evaluation, eleven welds were re-categorized from Category A to Category D.

Note: See Appendix E for ASME Section XI Summary Tables.

6.1.2 IGSCC Category B Weldments - Identifies welds which are fabricated from non-resistant material

Category B weldments are those welds made of resistant materials, but have had an SI performed either before service or within two years of operation.

Scope of Examination - There are no welds in this category at the Nine Mile Point Nuclear Station.

6.1.3 IGSCC Category C Weldments - Identifies welds which are fabricated from non-resistant materials.

Category C weldments are those welds not made of resistant materials, and have been given an SI process after more than two years of operation.

Scope of Examination - There are no welds in this category at the Nine Mile Point Nuclear Station.

6.1.4 IGSCC Category D Weldments - non-resistant materials: no stress improvement

Category D weldments are those welds not made with resistant materials, and have not been given an SI treatment, but have been examined and found to be free of cracks. Included in this category are all bimetallic nozzle weldments made with non-resistant material and 182 Inconel weld butter.

Previous Inspection Requirements- All welds were initially examined at least once every two refueling outages. Approximately 50% of all Category D welds were examined each refueling outage. Welds classified as category D were examined in accordance with GL 88-01, Table 1, as modified by alternative staff position number 4 for sample expansion, as contained in Supplement 1 to GL 88-01. All 142 Category D welds were scheduled to be ultrasonically examined every other refueling outage, with sample expansion limited to the piping system loop where cracking was found.

Scope of Examination - Category D welds will be examined at a frequency of 100% of the population once every six years. NMPNS is on a two year fuel cycle, so the frequency is 100% every three outages.

Note: In order to track these examinations and assure compliance, NMPNS has further defined Category D welds as follows:

a. G-L-D1 are ASME Code Class 1 welds that require ultrasonic examinations.

b. G-L-D2 are ASME Code Class 2 welds that require ultrasonic examinations.

TABLE 6 - 1
IGSCC CATEGORY D WELDS

Category	System No.	Total No. Welds	Total No. Welds Req'd	Total Welds Sched	Total Welds Sched 1ST Period	Total Welds Sched 2ND Period	Total Welds Sched 3RD Period	Totals Welds Sched. Interval
G-L-D1	32.0	10		17	2	8	7	17
G-L-D1	33.0	3		3	0	3	0	3
	37.0	1		1	0	1	0	1
	38.0	12		22	7	5	10	22
	39.0	24		48	19	5	24	48
	40.0	72		138	36	33	69	138
G-L-D2	39.0	31		62	2	29	31	62
Totals		153	153	291	66	84	141	291

6.1.5 IGSCC Category E Weldments - All welds included in this category are weld overlays.

Category E weldments are those welds with known cracks that have been reinforced by an acceptable weld overlay or have been mitigated by an SI treatment welding.

Weld 33-FW-22 in the Reactor Water Cleanup system is a non-safety related, non-ASME Section XI weld that is located outboard of the primary containment isolation valves. Weld 33-FW-22 was repaired in 1997 due to a through-wall pin hole leak by performing a weld overlay (reference NMPNS letters dated May 15, 1997 and May 19, 1997), and is classified as IGSCC Category E. It was last inspected during RFO-15 in 1999.

Note: Subject to NMPNS letter NMP1L 1673, dated September 13, 2002, notifying NRC of change to commitments, to use EPRI PDI program for weld overlays in lieu of Generic Letter 88-01.

Weld 33-FW-23A in the Reactor Water Cleanup system is a non-safety related, non-ASME Section XI weld that is located outboard of the primary containment isolation valves. Weld 33-FW-23A was repaired in 2004 due to a through-wall leak by performing a weld overlay, and is classified as IGSCC Category E. USNRC Safety Evaluation, dated 03-27-03, TAC MB6435 authorized use of PDI Program.

Previous Inspection Requirements - should be inspected once every two refueling outages after repair. Approximately 50% shall be inspected during the first refueling outage and subsequent outages shall be maintained for the rest of the interval.

Scope of Examination - should be inspected using a 25% sample every ten years. Fifty percent (50%) of these examinations are to be completed within the first 6 years of the interval.

- (2) Category E welds, (1) required within the first 6 years

Note: In order to track these examinations and assure compliance, NMPNS has further defined Category E welds as follows:

a. G-L-E1 are Non Safety-Related, Non ASME Code Classed welds that require ultrasonic examinations.

b. Defined below is the breakdown of Category E welds (1 times 2 equals 2 required)

TABLE 6 - 2
IGSCC CATEGORY E WELDS

Category	System No.	Total No. Welds	Total No. Welds Req'd	Total Welds Sched	Total Welds Sched 1ST Period	Total Welds Sched 2NDPeriod	Total Welds Sched 3RD Period	Totals Welds Sched. Interval
G-L-E1	33.0	2	1	1	0	1	1	2
Totals		2	1	1	0	1	0	1

6.1.6 IGSCC Category F Weldments - Cracked, inadequate or no repair

Category F weldments are those welds with known cracks that have been approved by analysis for limited additional service without repair.

Scope of Examination - inspected each refueling outages. See section 2, page 2-20 for additional information.

- (5) Category F welds, all (5) required once each inspection period for three consecutive periods in accordance with IWB-2420(b).

a. USNRC Safety Evaluation, TAC No. MA6511, dated January 14,2000, accepted

b. USNRC Safety Evaluation, TAC No. MA5696, dated July 23, 1999, accepted

c. NMPNS Letter NMP1L 1673, dated September 13, 2002, notification of change to regulatory commitments

6.1.7 IGSCC Category G Weldments - Non-resistant and not inspected by UT

Category G weldments are those welds not made of resistant materials, have not been given an SI treatment.

Scope of Examination - Welds classified as Category G are examined in accordance with GL 88-01. An additional seventeen (17) welds, seven (7) of which are inside penetrations and nine (9) can not be examined due to configuration, receive a visual (VT-2) examination for evidence of leakage at each refueling outage.

- (19) Category G welds, (95) required

In order to track these examinations and assure compliance, NMPNS has further defined Category G welds as follows:

a. G-L-G1 are ASME Code Class 1 welds that require visual (VT-2) examinations each outage for evidence of leakage.

b. Defined below is the breakdown of Category G welds

TABLE 6 - 3
EXAMINATION CATEGORY G WELDS

Category	System No.	Total No. Welds	Total No. Welds Req'd	Total Welds Sched	Total Welds Sched 1ST Period	Total Welds Sched 2ND Period	Total Welds Sched 3RD Period	Totals Welds Sched. Interval
G-L-G1	33.0	2		8	2	4	4	10
	37.0	1		5	1	2	2	5
	38.0	4		20	4	8	8	20
	39.0	6		30	6	12	12	30
	40.0	6		30	6	12	12	30
Totals		19		95	19	38	38	95

6.1.8 IGSCC Category S Weldments - Outboard of CI's

Category S weldments are those safety and/or non-safety-related welds that are located on the Reactor Water Clean Up system outboard of the Containment Isolation Valve.

USNRC Report 50-220/97-09, dated November 4, 1997 closed USNRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" requirements for NMP1. Therefore, the inspection requirements associated with the Reactor Water Cleanup System outside of the containment isolation valves are no longer applicable to NMP1.

Inspection Schedule - The extent and frequency of inspection for various weldment categories are detailed in Table below.

Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the existing augmented in-service inspection program to identify that Section 6, paragraph 6.1.8 of the existing inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program AMP XI.M7 BWR Stress Corrosion Cracking and (AMP) XI.M25, BWR Reactor Water Cleanup System has been credited for aging management as part of the NMP License Renewal Application (LRA).

TABLE 6 - 5
IGSCC EXAMINATION REQUIREMENTS

IGSCC CAT	Weld Description	EXISTING INSPECTION FREQUENCY GL 88-01	REVISED INSPECTION FREQUENCY (NWC) BWRVIP-75		Scope Expansion NWC	NMP1 Applicability
			NWC	HWC		
A	Resistant Materials	25%, Every 10 Year Interval, At least 12% in 6 years	B-F = 25% every 10 years B-J = 10% every 10 years	10% every 10 years	Note 1	Incorporated under Alternate Risk-Informed Program Currently 143 welds
B	Non-Resistant Materials; Stress Improvement (SI) within 2 years of operation	50%, Every 10 Year Interval, At least 25% in 6 years	25% every 10 years	10% every 10 years	Note 1	Not applicable to NMP1
C	Non-resistant Materials; Stress Improvement (SI) after 2 years of operation	All Within Two Refueling Cycles after the Post-SI Inspection, and All Every 10 Years thereafter, At least 50% in 6 years	25% every 10 years	10% every 10 years	Note 1	Not applicable to NMP1
D	Non-resistant Materials; No Stress Improvement	All Every Two Refueling Outage, 50% each refueling outage	100% every 6 years	100% every 10 years, at least 50% in 1ST. 6 years	Note 1	Currently 142 welds
E	Cracked Reinforced by weld overlay	All Every Two Refueling Outages, 50% each refueling outage	25% every 10 years	10% every 10 years	Note 1	Currently 2 weld
	Cracked: mitigated by SI		100% every 6 years	100% every 10 years, at least 50% in 1ST. 6 years	Note 1	No welds
F	Cracked; Inadequate or no repair	All Every Refueling Outage	Every refueling outage	Every refueling outage	N/A	Currently 5 welds Re-inspect per ASME Section

TABLE 6 - 5
IGSCC EXAMINATION REQUIREMENTS

IGSCC CAT	Weld Description	EXISTING INSPECTION FREQUENCY GL 88-01	REVISED INSPECTION FREQUENCY (NWC) BWRVIP-75		Scope Expansion NWC	NMP1 Applicability
			NWC	HWC		
						XI
G	Non-resistant; Not inspected (by UT)	All Next refueling Outage	Next outage	Next outage	N/A	Currently 19 welds All due to inaccessibility and high radiation
S	N/A	3 welds each outage	Each refueling Outage	N/A	N/A	Currently 30 welds RWCU outboard of CIV's, No longer required

Note 1 See Section 8 for expansion criteria

6.2 Generic Letter 98-05, Boiling Water Reactor Licensees Use of the BWRVIP - 05 Report to Request Relief From Augmented In-service Examination Requirements on Reactor Vessel Shell Welds.

In NMPNS letter NMP1L 1391, dated December 10, 1998, relief was requested pursuant to GL 98-05. Nine Mile Point Nuclear Station requested relief from the in-service inspection requirements of 10CFR50.55 (g) for volumetric examination of circumferential reactor pressure vessel (RPV) welds (ASME Code Section XI, Table IWB-2500-1, Category B-A, Item B1.11, Circumferential welds). This relief request also includes an alternative to the required inspections for RPV shell welds specified in 10CFR50.55a(g)(6)(ii)(A)(2).

Additionally, NMPNS requested approval of an alternative to the examination requirements specified in 10CFR50.55a(g) for volumetric examination of longitudinal RPV shell welds and the shell-to-flange weld (ASME Code Section XI, Table IWB-2500-1, Category B-A, Item B1.12, and Item B1.30). NMPNS proposes to perform an automated inspection of certain RPV welds using personnel and procedures qualified to the Performance Demonstration Initiative, (PDI). The use of these inspection procedures is a alternative to 10CFR50.55a(b)(2).

NMPNS has incorporated the information in BWRVIP-05 into these alternative requirements and addressed the USNRC positions in the USNRC's July 28, 1998 safety evaluation report. See NMPNS letter NMP1L 1391 for specific details.

Scope of Examination - NMP1 has received a final Safety Evaluation Report from the USNRC. See Examination Category B-A, Section 2 of this document for additional information.

6.3 NUREG 0619 BWR Feedwater and Control Rod Drive Return Line (CRDRL) Nozzle Cracking, Generic Letter 81-11 and GE NE-523-A71-0594-A

Third In-service Inspection Interval - On June 5, 1998, the USNRC staff issued a safety evaluation for GE-NE-523-A71-0594, Revision 0. The BWROG revised the original submittal to address recommendations in the USNRC staff's safety evaluation. The BWROG letter of September 24, 1999, responded to the USNRC recommendations by adopting the forthcoming schedule for implementation of the ASME Code, Section XI, in 10 CFR 50.55a. The USNRC staff completed its review and determined that the proposed inspection program and schedule in GE-NE-523-A71-0594, Revision 1, is justified and therefore, GE-NE-523-A71-0594-A, Revision 1, is an acceptable alternative to the inspection guidelines in NUREG 0619 (TAC No. MA6787, dated March 10, 2000).

Scope of Examination – NMPNS has completed examinations in compliance with Appendix VIII, therefore, NMPNS examinations will be in accordance with the provisions of ASME Section XI, Appendix VIII as mandated by 10 CFR 50.55a. The examination frequency from this point forward will be the ASME Section XI examination frequency.

Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the existing in-service inspection program to identify that Section 6, paragraph 6.3 of the existing inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.M5, BWR Feedwater Nozzle and XI.M6, BWR Control Rod Drive Return Line Nozzle, and has been credited for aging management as part of the NMP License Renewal Application (LRA).

6.4 NUREG-0803/ Generic Letter 81-34 and 86-01 - BWR Scram System Pipe Break

Generic Letter 81-34 transmitted NUREG-0803 to all BWR licensees.

NUREG-0803, Generic SER Regarding Integrity of BWR Scram System Piping, addresses the need for improvement in procedures, periodic in-service inspection and surveillance for the scram discharge volume (SDV) system. These guidelines were developed to address the consequences of a postulated leakage crack in the SDV piping and the resulting large leakage (up to 550 gpm) downstream of the system isolation valves.

Generic Letter 86-01, Safety Concerns associated with Pipe Breaks in the BWR Scram System, addressed the staff's position based on information provided in BWROG and General Electric Company supplied generic information (NEDO-22209, BWROG-8420) and staff generic analysis of the SDV piping system integrity. The staff has concluded that SDV piping satisfies BTP MEB 3-1, position B.2.C (1). A through wall leak need not be postulated. Also BWROG emergency procedure guidelines and visual verification of the SDV integrity provide sufficient measures to verify the detecting and mitigating the consequences of leakage.

Scope of Examinations - Based on the above information NMPNS will perform the examinations and tests required by the 2001 Edition through 2003 Addenda of Section XI for Class 2 systems. No additional augmented examinations are required.

Augmented Inspections

Augmented inspections are those additional inspections determined by NMPNS for systems and components where added assurance of structural reliability is recommended. Augmented Inspections are outside the scope of ASME Section XI and this inspection plan.

6.5 In-Vessel Visual Examinations (BWRVIP)

NMPNS has voluntary committed to additional inspections of the NMP1 reactor vessel and internals. These additional inspections are outside the scope of the ASME Section XI Code requirements. Specific inspection requirements are addressed in Nuclear Engineering Reports (NERs).

6.6 ASME Code Class 1 CRD Bottom Head Penetrations

The lower head of the Nine Mile Point Unit 1 reactor vessel contains one-hundred twenty nine (129) penetrations for the control rod drive (CRD) mechanism. During the last eighteen years of operation, through-wall cracking has occurred in various stub tube penetrations. This cracking has resulted in the leakage of reactor coolant from the vessel through the gap between the housing outside diameter and the vessel penetration bore. Repairs have been made by roll expanding the CRD housing in order to eliminate this gap and stop or limit the reactor coolant leakage. In the event the roll expansion does not seal a CRD penetration within allowable leakage criteria, an alternate repair method will be required.

During the License Renewal Application process NMPNS was mandated to commit to implement ASME Code Case N-730.

Scope of Examinations: - A UT examination of roll-expanded CRD housings shall be performed in accordance with Figures 1 or 2 of the Code Case on at least 10% of previously rolled housings, during each inspection interval.

- (33) Expanded CRD Housings, (4) required

Note: Future ISI CRD UT examinations will be coordinated with CRDM maintenance/replacements.

6.7 NSSS Supplier Recommendations

In addition to ASME Code requirements, General Electric Nuclear Energy (GENE) implements the Service Information Letter (SIL) and the Rapid Information Communication Services Information Letters (RICSIL) program. NMPNS may voluntary commit to additional inspections. These additional inspections are outside the scope of ASME Section XI requirements. Specific inspection requirements are addressed in Nuclear Engineering Reports (NER's).

6.8 Aging Management Program

The License Renewal Rule (10 CFR 54) required NMPNS to be able to demonstrate that the effects of aging will be adequately managed so that the intended function(s) of each structure and component (SC) will be maintained consistent with the Current License Basis (CLB) for the period of extended operation.

This paragraph has been added to the in-service inspection program to identify that all or portions of Section 6 of the existing inspection plan and schedule was compared against the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Aging Management Program (AMP) XI.M3 Reactor Head Closure Studs, XI.M5 BWR Feedwater Nozzle, XI.M6 BWR CRD return Line Nozzle, XI.M7 BWR Stress Corrosion Cracking, and XI.M25 BWR RWCU, and were determined to be applicable have been credited for aging management as part of the NMP License Renewal Application (LRA).

SECTION 7 - ALTERNATE RISK-INFORMED (RIS_B) INSPECTION PROGRAM

7.0 ALTERNATIVE RISK-INFORMED INSPECTION

As an alternative to the 2001 Edition through the 2003 Addenda of ASME Section XI Code, ASME Code Class 1 and 2 piping weld examinations, NMPNS developed an alternative risk-informed (RIS_B) in-service inspection program. The alternative risk-informed in-service inspection process (RI-ISI) summarized in this section was developed based on the Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A titled "Revised Risk-Informed In-service Inspection Evaluation Procedure", here after referred to as the EPRI-ISI-TR. In addition this section converts the initial RI-ISI Code Case N-578-1 to Code Case N-716 Risk-Informed / Safety-Based.

The implementation of the Alternative Risk-Informed (RIS_B) In-service Inspection Program, as defined in this section of the inspection plan, will be implemented in accordance with the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code Case N-716, titled "Alternative Piping Classification and Examination Requirements, Section XI, Division 1, approved by ASME on April 19, 2006, as defined below. (See Appendix I for Code Case requirements)

7.1 Systems Subject to Risk-Informed Inspection

Systems subject to examination under the RIS_B program were based on the ASME Code Class 1 and 2 piping system classifications.

The ASME Code Class 1 and 2 components and systems subject to RIS_B examination are identified below.

A comparison between the proposed Alternative RIS_B Program and the current ASME Section XI In-service Inspection Program requirements for in-scope piping is given in Table 7-1 below.

Table 7-1 Comparison between RIS_B and ASME Code

System-Zone No.[1]	RI-ISI Inspection Locations				ASME Section XI 2001 Edition thru 2003 Addenda Examination Requirements			
	Class 1		Class 2		B-F	B-J	C-F-1	C-F-2
	HSS	LSS	HSS	LSS				
MS – 01.0	66	-	-	-	-	66	-	-
FW – 31.0	50	-	-	-	-	50	-	-
RR – 32.0	187	-	-	-	10	177	-	-
RWCU – 33.0	33	-	-	-	-	33	-	-
RXVI – 36.0	39	-	-	-	27	12	-	-
RD/RHV – 37.0	52				1	51		
SDC – 38.0	16	-	-	-	-	16	-	-
EC – 39.0	50	-	-	75	2	48	75	-
CS – 40.0, 81.0	(-1)94	-	-	242	2	93	-	242
LP – 41.0, 42.1	21	-	-	-	1	20	-	-
CRD/SDV – 44.1	21	-	-	(+1) 60	1	20	-	59
CTN-SP – 80.0,93.0	-	-	-	363	-	-	-	363
TOTAL	629			740	44	586	75	664

1. Systems are described in Sections 2 and 3
2. RPV (17) moved to system 36.0

7.2 Piping Subject to Examination

The alternative examination requirements of this Section shall be used for ASME Code Class 1 and 2 piping evaluated by the EPRI-TR risk-informed process. For ASME Code Class 1 and 2 piping, these requirements are an alternative to the requirements of Examination Categories B-F, B-J, Table IWB-2500-1, Examination Categories C-F-1, or C-F-2, Table IWC-2500-1 and IGSCC Category "A", Generic Letter 88-01 piping welds.

7.3 Element Selection

Low Safety Significant (LSS) piping welds are exempt from volumetric and surface examination. Ten percent of the High Safety Significant (HSS) piping welds shall be selected for examination. The existing plant flow accelerated corrosion program and localized corrosion program, excluding crevice corrosion, may not be credited toward the 10% requirement. The selection of HSS piping welds for examination is based on the following:

(1) HSS piping welds are subject to a degradation mechanism (DM) evaluation per the criteria in ASME Section XI Code Case N-716.

(2) Examinations must be prorated equally among systems to the extent practical and each system must meet the following criteria.

   a. A minimum of 25 percent of the piping weld population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected for examination.

   b. If the examinations selected above exceed 10% of the total number of HSS piping welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS piping weld population is inspected.

   c. If the examinations selected above are not at least 10% of the HSS piping weld population, additional piping welds are selected so that the total number selected for examination is at least 10%.

(3) At least 10% of the RCPB piping weld population must be selected for examination.

(4) At least 2/3 of the RCPB piping welds selected for examination must be located between the first isolation valve and the reactor pressure vessel.

(5) A minimum of 10% of the RCPB piping welds that lie outside containment must be selected for examination.

(6) A minimum of 10% of the BER piping welds must be selected for examination [NMP1 does not have a plant-specific BER Program].

Table 7-3 provides the number of items attributed to specific degradation mechanisms.

A review of ASME Code Class 1 RI-ISI selections was made to ensure that the percentage was not significantly reduced below 10 percent of the ASME Code Class 1 piping population. The results of this review indicate that the Alternative RI-ISI program will be inspecting greater than 10 percent of the ASME

Code Class 1 piping systems.

Table 7-3 Degradation Mechanisms by ASME Code Class

Abbreviation	Degradation Mechanism	ASME Code Class 1 Items	ASME Code Class 2 Items	Total Number of Items
TASCS	Thermal Transients	89	0	89
TT	Thermal Transients	0	0	0
N/A	Thermal Stripping, Cycling and Stratification	0	0	0
IGSCC	Intergranular Stress Corrosion Cracking	142	31	173
TGSCC	Transgranular Stress Corrosion Cracking	0	0	0
ECSCC	External Chloride Stress Corrosion Cracking	0	0	0
PWSCC	Primary Water Stress Corrosion Cracking	0	0	0
MIC	Microbiologically Influenced Corrosion	0	0	0
Pitting	Pitting	0	0	0
CC	Crevice Corrosion Cracking	11	0	11
EC	Cavitation	0	0	0
FAC	Flow Accelerated Corrosion	7	0	7
None	No Degradation Mechanism	421	709	1130

7.4 Alternative Risk-Informed In-service Inspection Plan and Schedule

The initial Alternative Risk-Informed In-service Inspection Plan and Schedule was implemented starting with refueling outage (RFO-17), the first refueling outage of the second in-service inspection period of the third inspection interval. The Alternative RIS_B program requires sixty-eight (68) element examinations to be completed over the fourth ten-year in-service inspection interval. Ninety-one (91) element examinations have been scheduled for completion in the fourth ten-year in-service inspection interval.

The applicable ASME Code Case to be used for the Alternative Risk-Informed RIS_B Inspection Program for ASME Code Class 1 and Code Class 2 piping examinations is ASME Code Case N-716, Alternative Piping Clarification and Examination Requirements, Section XI, Division 1, approved by ASME on April 19, 2006. The requirements of the Code Case were applied to the EPRI-TR process.

7.5 Category R-A, Risk-Informed Piping Examinations

All examinations are performed from the outside surface using manual/automated inspection equipment, (as applicable), and utilizing volumetric examination techniques.

All high safety significant (HSS) piping structural elements have been classified in accordance with Examination Category R-A. In order to incorporate these examinations into the current in-service inspection database and still maintain alignment with the current ASME Code Class requirements and for accounting/percentage purposes, NMPNS has classified the examination elements in accordance with

Table 7-5 below. Refer to Appendix F for RIS_B Summary Table details.

Table 7-5 Classification Criteria

Abbreviation	Degradation Mechanism	Examination Category	Examination Item Number	
			HSS-Class 1	LSS-Class 2
TT / TASCS	Thermal Transient	R-A	R1.11	R2.11
N/A	Not Used	R-A	R1.12	R2.12
EC	Erosion-Cavitation	R-A	R1.13	R2.13
CC	Crevice Corrosion Cracking	R-A	R1.14	R2.14
PWSCC	Primary Water Stress Corrosion Cracking	R-A	R1.15	R2.15
IGSCC / TGSCC	Intergranular / Transgranular Stress Corrosion Cracking	R-A	R1.16	R2.16
MIC	Microbiologically Influenced Corrosion	R-A	R1.17	R2.17
FAC	Flow Accelerated Corrosion	R-A	R1.18	R2.18
ECSCC	External Chloride Stress Corrosion Cracking	R-A	R1.19	R2.19
None	No Degradation Mechanism	R-A	R1.20	R2.20

The elements subject to examination under Examination Category R-A shall be completed during each inspection interval in accordance with Table IWB-2411-1 or Table IWB-2412-1.

A narrative discussion of ASME Code Class 1 and Class 2 components subject to RI-ISI examination are described in detail below:

Items R1.10 High-Safety-Significant Piping Structural Elements

7.5.1 Items R1.11 and R2.11 - Elements Subject to Thermal Fatigue

Scope of Examination - Volumetric examination, (8) Socket welds of any size and branch pipe connection welds NPS 2 (DN 50) and smaller selected for examination require a volumetric examination of the piping base metal within 1 / 2 in, (1.3 mm) of the toe of the weld, and the fitting itself shall receive a VT-2 visual examination. Examination requirements and figure numbers IWB-2500-8(c), IWB-2500-9, 10, 11 as applicable.

- (89) R1.11 Elements subject to Thermal Fatigue (18) selected
- (0) R2.11 Elements subject to Thermal Fatigue (0) selected

7.5.2 Items: R1.12 and R2.12 Elements Subject to High Cycle Mechanical Fatigue

Scope of Examination – This item is not used in Code Case N-716

7.5.3 Items: R1.13 and R2.13 Elements Subject to Erosion Cavitation

Scope of Examination - Volumetric examination; (6) Includes base metal, welds, and weld HAZ in the affected regions of carbon and low alloy steel, and the welds and weld HAZ of austenitic steel examinations shall verify the minimum wall thickness required. Acceptance criteria for localized thinning is in the course of preparation. The examination method and examination region shall be sufficient to characterize the extent of the element degradation; (7) In accordance with the NMPNS existing programs, such as PWSCC, IGSCC, MIC, or FAC programs, as applicable.

- (0) R1.13 Elements subject to Erosion Cavitation, (0) selected
- (N/A) R2.13 Elements subject to Erosion Cavitation

7.5.4 Items: R1.14 and R2.14 Elements Subject to Crevice Corrosion Cracking

Scope of Examination - Volumetric examination, (9) Socket welds of any size and branch pipe connection welds NPS 2 (DN 50) and smaller require only a VT-2 visual examination. (5) The examination volume shall include the volume surrounding the weld, weld HAZ, and base metal, where applicable, in the crevice region. Examination should focus on detection of cracks initiating and propagating from the inner surface;

- (11) R1.14 Elements subject to Crevice Corrosion Cracking, (3) selected
- (N/A) R2.14 Elements subject to Crevice Corrosion Cracking

7.5.5 Items: R1.15 and R2.15 Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC)

Scope of Examination – Not applicable to Nine Mile Point Nuclear Station.

- (N/A) R1.15 Elements subject to PWSCC
- (N/A) R2.15 Elements subject to PWSCC

7.5.6 Items: R1.16 and R2.16 Elements Subject to Intergranular or Transgranular Stress Corrosion Cracking (IGSCC, TGSCC)

Scope of Examination - Volumetric examination, (7) In accordance with NMPNS existing IGSCC program; (9) Socket welds of any size and branch pipe connection welds NPS 2 (DN 50) and smaller require only a VT-2 visual examination; (10) Paragraph and Figure numbers IWB-2500-8(c), 9, 10 and 11.

- (142) R1.16 Elements subject to IGSCC (33) selected, Examinations performed in accordance with the current IGSCC program.
- (31) R2.16 Elements subject to IGSCC (0) selected.

7.5.7 Items: R1.17 and R2.17 Elements Subject to Localized Microbiologically Corrosion (Microbiologically-Induced Corrosion (MIC), or Pitting)

Scope of Examination – Not applicable to Nine Mile Point Nuclear Station

- (N/A) R1.17 Elements subject to MIC
- (N/A) R2.17 Elements subject to MIC

7.5.8 Items: R1.18 and R2.18 Elements Subject to Flow Accelerated Corrosion (FAC)

Scope of Examination - Examination, (7) In accordance with the NMPNS existing FAC program;

- (7) R1.18 Elements subject to FAC, (2) selected. Examinations performed in accordance with current FAC program.
- (0) R2.18 Elements subject to FAC.

7.5.9 Items: R1.19 and R2.19 Elements Subject to External Chloride Stress Corrosion Cracking (ECSCC)

Scope of Examination – Not applicable to Nine Mile Point Nuclear Station.

- (N/A) R1.19 Elements subject to ECSCC
- (N/A) R2.19 Elements subject to ECSCC

7.5.10 Items: R1.20 and R2.20 Elements not Subject to a Damage Mechanism

Scope of Examination - Volumetric examination to, (9) 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; (2) Includes all elements identified in accordance with the risk-informed selection process; (4) The examinations shall include any longitudinal weld at the location selected for examination in (2). The longitudinal weld examination requirements shall be met for both transverse and parallel flaws examination volume defined in (2); (5) Initially selected elements are to be examined in the same sequence during successive inspection intervals, to the extent practical; (10) Paragraph and Figure numbers IWB-2500-8(c), 7(a), 9, 10, 11, .

- (421) R1.20 Elements subject to No Degradation, (35) selected
- (709) R2.20 Elements subject to No Degradation, (0) selected

7.6 Category R-B, Risk-Informed Piping Examinations

All high safety significant (HSS) piping structural elements have been classified in accordance with Examination Category R-A, Table 7-5 as defined above. For those examinations for which no examination item number is prescribed, Category R-B has been designated in order to incorporate these examinations into the current in-service inspection database and still maintain alignment with the current ASME Code Class requirements and for accounting/percentage purposes. A narrative discussion of the components subject to this classification is described in detail below:

7.6.1 Items: R1.21 and R2.21 Socket Weld Elements Located in High Safety Significant Systems

Scope of Examination - Visual (VT-2) examination, For those socket welds located in (HSS) risk categories 1, 2, and 3, perform visual examination each refueling outage as prescribed in footnote (12) of Table 1 of Code Case N-578-1. VT-2 examinations are performed in accordance with the system pressure test program.

- (43) R1.21 Socket welds subject to a VT-2 examination, None required.
- (0) R2.21 Socket welds subject to a VT-2 examination, None required

Note: Not applicable to NMPNS, not located on any HSS piping structural elements.

7.6.2 Items: R1.22 and R2.22 Intersecting Longitudinal Weld Seams

Scope of Examination - All longitudinal pipe welds intersecting any of the selected circumferential welds will also be examined.

- (0) Class 1 and 2 longitudinal welds subject to examination

7.7 Augmented Inspections

Augmented inspection requirements of NUREG-0313, Rev. 2, USNRC Generic Letter 88-01, Supplement. 1, are discussed in detail within Section 6 of the In-service Inspection Plan and Schedule.

Augmented inspections, to the extent practical shall be used to satisfy the percentage requirements of Inspection Program "B".

Consistence with the EPRI-TR, Category A, Generic Letter 88-01 (NUREG-0313, Rev 2) welds are integrated into the proposed alternative RIS_B program. As such, NMPNS response to Generic Letter 88-01 and its supplement remains unchanged for IGSCC Categories B through G at this time. Another augmented inspection program, Generic Letter 89-08 – Flow Accelerated Corrosion Program (FAC), is credited in the proposed RIS_B program but is not affected or changed by the proposed RIS_B program. Any other existing augmented inspection programs are unaffected by this section.

7.8 Examination Category B-P and C-H, Pressure Retaining Components

System pressure tests are conducted on all ASME Code Class 1, Class 2 and Class 3 systems and components in accordance with the Nine Mile Point Unit 2 System Pressure Testing Program. Visual examinations required by the RIS_B program shall be performed in accordance with the current system pressure testing program.

7.9 Successive Inspections

The sequence of risk-informed piping weld examinations established during the third inspection interval will be repeated during the subsequent inspection interval, to the extent practical.

Note: Code Case N-624, modification of the sequence of examinations established during the first interval. (See Appendix I for requirements)

In accordance with IWB-2420(b), piping welds examined during the third inspection interval are evaluated in accordance with IWB-3142.4, and were determined by analysis to qualify as acceptable for continued service, the areas containing these indications shall require reexamination during the fourth inspection interval. Applicable welds are uniquely identified within the RIS_B inspection plan Tables.

7.10 Unaffected Portions of ASME Section XI

Non-related portions of the ASME Section XI Code, (inspection intervals, acceptance criteria for evaluation of flaws, expansion criteria for flaws discovered, and qualification on examination techniques and personnel are essentially unaffected by the Alternative Risk-Informed RIS_B Inspection Program.

SECTION 8 - ACCEPTANCE STANDARDS

8.0 ACCEPTANCE STANDARDS

Flaws and/or relevant conditions detected during an in-service examination shall be compared against the acceptance standards of Section XI, 2001 Edition, through 2003 Addenda as defined in Tables 8-1 through 8-5 below.

8.1 Acceptance by Volumetric or Surface Examination

A component whose volumetric or surface examination either reconfirms the absence of flaws/conditions or detects flaws indications that do not exceed the acceptance criteria identified in Tables 8-1 through 8-4, shall be acceptable for continued service. Verified changes of flaws/conditions from prior examinations shall be recorded.

Acceptance of components for continued service with indications/conditions exceeding the acceptance criteria above shall be corrected in accordance with paragraphs 8.2 through 8.6.

8.2 Acceptance by Repair/Replacement Activity

A component whose volumetric or surface examination reveals flaws/conditions that exceed the acceptance criteria of Tables 8-1 through 8-4 shall be unacceptable for continued service until the additional examination requirements are satisfied and the component is corrected by a repair/replacement activity to the extent necessary to meet the acceptance criteria in 8.1.

Note: The additional examination requirements of IWB-2430, IWC-2430, or IWF-2430, (as applicable) shall be performed for service induced defects/conditions, and/or those construction or manufacturing defects determined by Nuclear Engineering to be detrimental to the quality or safety of the component/system.

8.3 Acceptance by Analytical Evaluation

A component whose volumetric or surface examination reveals flaws that exceed the acceptance criteria of Tables 8-1 through 8-4 are acceptable for continued service without a repair/replacement activity if an analytical evaluation meets the acceptance criteria of IWB-3600 or IWC-3600 as applicable.

Where the acceptance criteria of IWB-3600 or IWC-3600 are satisfied, the area containing the flaw shall be subsequently reexamined in accordance with 8.4.1, or 8.4.2 .

Note: Reexamination shall be accomplished only on service induced defects/conditions.

8.4 Successive Inspections

8.4.1 Class 1 Components

Pursuant to the Section XI Code, sub-article IWB-2420 (b), in the case, where examinations reveal the presence of service-induced defects that exceed the acceptance standards and the component is analyzed as acceptable for continued service, the areas containing such flaws shall be reexamined during the next three (3) inspection periods of Inspection Program B (IWB-2412-1). Provided the flaw remains essentially unchanged for three successive inspection periods, the component examination schedule will revert to the original schedule of successive inspections.

8.4.2 Class 2 Components

Pursuant to the Section XI Code, sub-article IWC-2420 (b), in the case, where examinations reveal the presence of service-induced defects that exceed the acceptance standards and the component is analyzed as acceptable for continued service, the areas containing such flaws shall be reexamined during the next inspection period of Inspection Program B (IWC-2412-1). Provided the flaw remains essentially unchanged for the next inspection period, the component examination schedule will revert to the original schedule of successive inspections.

8.4.3 Class 3 Components

Pursuant to the Section XI Code, sub-article IWD-2420 (b), in the case, where examinations reveal the presence of service-induced defects that exceed the acceptance standards and the component is analyzed as acceptable for continued service, the areas containing such flaws shall be reexamined during the next inspection period of Inspection Program B (IWD-2412-1). Provided the flaw remains essentially unchanged for the next inspection period, the component examination schedule will revert to the original schedule of successive inspections.

8.4.4 Class MC Components

For ASME Code Class MC, see Containment Inspection Plan and Schedule CNG-NMP1-CISI-002.

8.4.5 Class 1, 2 and 3 Component Supports

Pursuant to ASME Code, sub-article IWF-2420(b), when a component support is accepted for continued service, the component support shall be reexamined during the next inspection period of Inspection Program B. When the re-examination do not require additional corrective measures, the component support will revert to the original schedule of successive inspections.

8.5 Acceptance by Engineering Evaluation

Examinations that reveal indications/conditions exceeding the acceptance criteria identified in Tables 8-1 through 8-4 will be submitted to Nuclear Engineering for evaluation and disposition:

A. Indications/conditions found to be acceptable by the materials and welding criteria specified in the Construction Code and/or Section III Edition applicable to the construction of the component shall be acceptable for continued service.

B. Indications/conditions determined to be acceptable by the NMPNS Design and/or Manufacturer's Specifications shall be acceptable for continued service.

C. Indications/conditions believed to be surface anomalies (e.g., fabrication marks, scratches, surface abrasion, material roughness or other conditions are acceptable for continued service provided the indication/condition is removed by light flapping and/or grinding (surface preparation), and the material removed does not violate the design minimum wall thickness.

D. If the evaluations conducted on a component support demonstrates that the support was functional for its intended safety function, additional exams are not required.

E. When supplemental examinations of 8.7 are required, if either the thickness of the base metal is reduced by no more than 10% of the nominal plate thickness or the reduced thickness can be shown by analysis to satisfy the requirements of the Design Specifications, the component shall be acceptable by evaluation.

F. Where the flaw or area of degradation are accepted by engineering evaluation, the area containing the flaw or degradation shall be reexamined in accordance with 8.4.

Nuclear Engineering evaluation and/or disposition may include the need for corrective measures, repairs, maintenance, analytical evaluation, or replacement, as appropriate.

8.6 Acceptance by Corrective Measures or Repair/Replacement Activity

A Component supports whose examinations reveal conditions described in -3410(a) is unacceptable for continued service until such conditions are corrected by one or more of the following:

(a) Adjustment and reexamination in accordance with IWF-2200 for conditions such as:

  (1) Detached or loosened mechanical connections;

  (2) Improper hot or cold settings of spring supports and constant load supports;

  (3) Misalignment of supports; or

  (4) Improper displacement settings of guides and stops.

(b) Repair/replacement activities in accordance with IWA-4000 and re-examination in accordance with IWF-2200;

8.7 Acceptance by Evaluation or Test

A component support or portion of a component support which is unacceptable per Table 8-4, for continued service may be analyzed and/or tested to the extent necessary to substantiate its integrity for its intended service.

8.8 Acceptance by Supplemental Examination

Components containing indications and/or relevant conditions shall be acceptable for continued service if the results of supplemental examinations meet the acceptance requirements of 8.1.

Examinations that detect flaws or evidence of degradation that requires evaluation in accordance with the requirements of 8.5 may be supplemented by other examination methods and techniques (IWA-2240) to determine the character of the flaw (i.e., size, shape, and orientation) or degradation. Visual examinations that detect surface flaws or areas that are suspect shall be supplemented by either surface or volumetric examination.

8.9 Acceptance Criteria in Course of Preparation

If acceptance criteria for a particular component, examination category, or examination method are not specified, defects that exceed the acceptance criteria for materials and welds specified in the Construction Code and/or Section III Edition applicable to the construction of the component shall be evaluated to determine disposition.

8.10 Additional Examinations

8.10.1 Class 1 Sample Expansion

The additional examination requirements identified in IWB-2430 shall be performed for service induced flaws or relevant conditions, and/or those construction or manufacturing defects determined by Nuclear Engineering to be detrimental to the quality or safety of the component/system only. When this situation exists, additional examinations shall include the following:

a. The additional examinations shall include an additional number of welds, areas, or parts included in the inspection item equal to the number of welds, areas, or parts included in the inspection item that were scheduled to be performed during the present inspection period. The additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaw or relevant conditions.

b. If the additional examinations reveal flaws or relevant conditions exceeding the acceptance standards, the examinations shall be further extended to include additional examinations during the current outage. These additional examinations shall include the remaining number of welds, areas, or parts of similar material and service subject to the same type of flaw or relevant conditions.

Additional examinations will be performed before the end of the outage.

Note: Code Case N-586-1, Alternative additional examination requirements for Class 1, 2 and 3 piping, components and supports. (See Appendix I for requirements)

8.10.2 Class 2 Sample Expansion

The additional examination requirements identified in IWC-2430 shall be performed for service induced flaws or relevant conditions, and/or those construction or manufacturing defects determined by Nuclear Engineering to be detrimental to the quality or safety of the component/system only. When this situation exists, additional examinations shall include the following:

a. The additional examinations shall include an additional number of welds, areas, or parts included in the inspection item equal to 20% of the number of welds, areas, or parts included in the inspection item that were scheduled to be performed during the interval. The additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaw or relevant conditions.

b. If the additional examinations reveal flaws or relevant conditions exceeding the acceptance standards, the examinations shall be further extended to include additional examinations during the current outage. These additional examinations shall include the remaining number of welds, areas, or parts of similar material and service subject to the same type of flaw or relevant conditions.

Additional examinations will be performed before the end of the outage.

Note: Code Case N-586-1, Alternative additional examination requirements for Class 1, 2 and 3 piping, components and supports. (See Appendix I for requirements)

8.10.3 Class 3 Sample Expansion

The additional examination requirements identified in IWD-2430 shall be performed for service induced flaws or relevant conditions, and/or those construction or manufacturing defects determined by Nuclear Engineering to be detrimental to the quality or safety of the component/system only. When this situation exists, additional examinations shall include the following:

a. The additional examinations shall include an additional number of welds, areas, or parts included in the inspection item equal to 20% of the number of welds, areas, or parts included in the inspection item that were scheduled to be performed during the interval. The additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaw or relevant conditions.

b. If the additional examinations reveal flaws or relevant conditions exceeding the acceptance standards, the examinations shall be further extended to include additional examinations during the current outage. The extent of the additional shall be determined by NMPNS based upon engineering evaluation of the root cause of the flaws or relevant conditions.

Additional examinations will be performed before the end of the outage.

Note: Code Case N-586-1, Alternative additional examination requirements for Class 1, 2 and 3 piping, components and supports. (See Appendix I for Code Case requirements)

8.10.4 Component Supports Sample Expansion

The additional examination requirements identified in IWF-2430 shall be performed for service induced defects/condition, and/or those construction or manufacturing defects determined by Nuclear Engineering to be detrimental to the quality or safety of the component/system only. When this situation exists, additional examinations shall include the following:

a. The additional examinations shall include the component supports immediately adjacent to those component supports for which corrective action is required. The additional examinations shall be extended to include additional supports within the system, equal in number and of the same type and function as those scheduled for examination during the inspection period.

b. If the additional examinations reveal flaws or relevant conditions exceeding the acceptance standards, the examinations shall be further extended to include additional examinations during the current outage. These additional examinations shall include the remaining component supports within the system of the same type and function.

c. When the additional examinations reveal flaws or relevant conditions, the examinations shall be extended during the current outage, to include all nonexempt supports potentially subject to the same failure modes that required corrective action. Also these additional examinations shall include nonexempt component supports in other systems when the support failures requiring corrective action indicate non-system related support failure modes.

Additional examinations will be performed before the end of the outage.

Note: Code Case N-586-1, Alternative additional examination requirements for Class 1, 2 and 3 piping, components and supports. (See Appendix I for Code Case requirements)

8.10.5 Class MC Sample Expansion

For ASME Code Class MC, see Containment Inspection plan and schedule CNG-NMP1-CISI-002.

8.10.6 IGSCC Sample Expansion

A. IGSCC Categories A, B or C

If one or more cracked welds in IGSCC Categories A, B. or C, are found by a sample inspection during the 10 year interval, an additional sample of the welds in that category shall be inspected, approximately equal in number to the original sample. This additional sample should be similar in distribution (according to pipe size, system, and location) to the original sample, unless it is determined that there is a technical reason to select a different distribution. If any cracked welds are found in this sample, all of the welds in that IGSCC Category should be inspected.

B. IGSCC Category E

If significant crack growth, or additional cracks are found during the inspection of one or more IGSCC Category E welds, all other Category E welds should be examined.

a. Significant crack growth for overlayed welds is defined as crack extension to deeper than 75% of the original wall thickness, or for cracks originally deeper than 75% of the pipe wall, evidence of crack growth into the effective weld overlay.

b. Significant crack growth for SI mitigated Category E welds is defined as growth to a length or depth exceeding the criteria for SI mitigation. (10% of circumference or 30% in depth).

Note: Based on discussions held with the BWRVIP during the public meeting, the NRC staff requested that the criteria for sample expansion for Category E welds (resistant material) be modified such that, for the first expansion, an equal number to the original inspection population be examined; for the second expansion, fifty percent (50%) of the total population is examined; and for the third expansion, the full population (100%) is examined.

C. IGSCC Category D

Category D weld expansions are limited to piping systems where cracking was identified.

If cracking is detected, the sample size will be expanded to a sample equal in number to the size of the initial sample. If cracking is detected in the additional sample, all remaining Category welds will be examined. Sample expansion can be limited, with technical justification, to the system or type component (i.e., safe-end to nozzle) in which flaws were detected. However, the sample size should include a number equal to the original sample or otherwise include all the welds within the system or component type that expansion is being limited.

8.10.7 Risk-Informed (RIS_B) Sample Expansion

A. Examinations performed in accordance with Table 1 of CC N-716 that reveal flaws or relevant conditions exceeding the acceptance criteria of 8-1 shall be extended to include a first sample of additional examinations during the current outage.

1. The piping structural elements (welds) to be examined in the first sample of additional examinations shall include HSS elements with the same postulated degradation mechanism in systems whose materials and service conditions are similar to the element that exceeded the acceptance standards.

2. The number of examinations required is the number of HSS elements with the same postulated degradation mechanism scheduled for the current inspection period. If there are not enough HSS elements to equal this number, NMPNS shall include remaining HSS elements and LSS elements up to and including this number that are subject to the same degradation mechanism.

B. If additional examinations required by 2 above reveal flaws or relevant conditions exceeding the acceptance standards of 8-1, the examinations shall be extended to include a second sample of additional examinations during the current outage.

1. The second sample of additional piping structural elements to be examined shall include all remaining HSS piping structural elements in Table 1 subject to the same degradation mechanism.

2. NMPNS shall also examine LSS piping structural elements subject to the same degradation mechanism or document the basis for their exclusion.

C. For the inspection period following the period in which the examination of A and B were completed, the examinations shall be performed as originally scheduled in accordance with IWB-2400.

8.11 Defects Found Outside Section XI Examination

Defects/conditions that are found outside the course of an ASME Section XI examination, shall be compared against the acceptance standards of Tables 8-1 through 8-4, as applicable.

8.12 Roll Expansion Acceptance Standards

In accordance with ASME Code Case N-730, the following acceptance criteria shall be used:

a. If planar flaws are discovered in the roll region (Region 2), this Code Case may not be used.
b. The examination results shall be evaluated in accordance with IWB-3523 (Table 8-1). If flaws exceed the acceptance standards, they shall be evaluated to show that the requirements of IWB-3640 are satisfied.
c. If the requirements of IWB-3640 are not met, the defect shall be corrected by a repair/replacement activity.

8.12.1 Roll Expansion Additional Examinations

If flaws are detected that fail to meet the acceptance standards of IWB-3523 (Table 8-1), the additional examinations requirements of IWB-2430 (8.10) shall be met.

TABLE 8-1 - CLASS 1 ACCEPTANCE STANDARDS

EXAMINATION CATEGORY	COMPONENT OR PART EXAMINED	ACCEPTANCE STANDARD
B-A	Vessel Welds	IWB-3510
B-B	Vessel Welds	IWB-3510
B-D	Full Penetration Welded Nozzles in Vessels	IWB-3512
B-F, B-J	Dissimilar and Similar Metal Welds in Piping and Vessel Nozzles	IWB-3514
B-G-1	Bolting > 2" dia.	IWB-3515/3517
B-G-2	Bolting ≤ 2" dia.	IWB-3517
B-K	Welded Attachments for Vessels, Piping, Pumps & Valves	IWB-3516
B-L-1, B-M-1	Welds in Pumps & Valves	IWB-3518
B-L-2, B-M-2	Pump Casings & Valve Bodies	IWB-3519
B-N-1 B-N-2 B-N-3	Interior Surfaces & Internal Components of Reactor Vessels	IWB-3520
B-O	Control Rod Drive and Instrument Nozzle Housing Welds	IWB-3523
B-P	Pressure Retaining Boundary	IWB-3522
B-Q	Steam Generator Tubing	IWB-3521

TABLE 8-2 – CLASS 2 ACCEPTANCE STANDARDS

EXAMINATION CATEGORY	COMPONENT OR PART EXAMINED	ACCEPTANCE STANDARD
C-A	Welds in Pressure Vessels	IWC-3510
C-B	Vessel Nozzle Welds	IWC-3511
C-C	Welded Attachments for Vessels, Piping, Pumps and Valves	IWC-3512
C-D	Bolting	IWC-3513
C-F-1,C-F-2	Welds in Piping	IWC-3514
C-G	Welds in Pumps and Valves	IWC-3515
C-H	Pressure Retaining Components	IWC-3516

TABLE 8-3 - CLASS 3 ACCEPTANCE STANDARDS

EXAMINATION CATEGORY	COMPONENT OR PART EXAMINED	ACCEPTANCE STANDARD
D-A	Integral Attachments (VT-1)	Mfg. Code & Applicable Standards

* ASME Section XI Acceptance Standard in course of preparation. The requirements of IWC-3200 may be used.

TABLE 8-4 - COMPONENT SUPPORT ACCEPTANCE STANDARDS

EXAMINATION CATEGORY	COMPONENT OR PART EXAMINED	ACCEPTANCE STANDARD
F-A	Supports	IWF-3410

SECTION 9 - REPAIRS / REPLACEMENTS ACTIVITIES

9.0 Repairs / Replacement Activities

This section supplements the requirements established in Constellation Nuclear Generation Fleet Administrative Procedure CNG-AM-1.01-1020, ASME Section XI Repair/Replacement Program and site Interface Procedure NIP-IIT-02, ASME Section XI Repair and Replacement Activities, for the Nine Mile Point Nuclear Station. The managerial and administrative controls over the implementation and completion of repairs, replacement (including modifications) are identified in NIP-IIT-02.

The applicable ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, hereafter referred to as the Code, is the 2001 Edition through the 2003 Addenda.

9.1 Repairs

Repairs for which these provisions apply are restricted to those performed on systems and components classified Quality Group A, B or C, (ASME Code Class 1, 2, 3) pressure retaining components and their integral attachments.

9.1.1 Pressure Testing

After repairs by welding on the pressure retaining boundary, a pressure test shall be performed in accordance with the requirements of the In-service Pressure Testing Program, Document NMP1-PT-003.

Note: 10 CFR 50.55a(b)(2)(xx) System Leakage Test - The NRC imposed a condition in 10 CFR 50.55a(b)(2)(xx) requiring Section III NDE be performed following repair and replacement activities if a system leakage test was to be used in lieu of a hydrostatic test under the 2003 Addenda through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(b)(2).

9.1.2 Baseline Examinations

When required by ASME Section XI, the repaired area shall be reexamined to establish a new baseline record. The examination shall include the method that detected the flaw.

9.2 Replacements

Replacements (including modifications) for which these provisions apply are restricted to those performed on systems and components classified Quality Group A, B or C, (ASME Code Class 1, 2, 3) pressure retaining components and their integral attachments.

9.2.1 Intergranular Stress Corrosion Cracking - NUREG-0313, Revision 2

As a result of NMPNS commitments made to the NRC to conform to the material requirements of NUREG-0313, Revision 2, the following material/processes should be used for future replacements of stainless steel piping in contact with reactor water. These materials are considered resistant to sensitization and IGSCC in BWR piping systems. Use of the following materials and processes are subject to engineering review for reconciliation of material properties (i.e., stress allowables, modulus of elasticity, thermal expansion coefficients) when they differ from the currently specified materials for each system.

A. Carbon wrought austenitic stainless steel, which includes nuclear grades 304NG and 316NG with a maximum carbon content of 0.02% or types 304L, 316L and similar low carbon grades with a maximum carbon content of 0.035%. Type 347, as modified for nuclear use, will be resistant with somewhat higher carbon content, the usual maximum of 0.04% is adequate. These materials are generally tested for resistance to sensitization in accordance with ASME A262-A, E1, or equivalent standard.

B. Austenitic stainless steel piping that does not meet the requirements above is considered to be resistant if it is given a solution heat treatment after welding.

C. Piping weldments are considered resistant to IGSCC if the weld heat affected zone on the inside of the pipe is protected by a cladding of resistant weld metal. This is often referred to as corrosion resistant cladding (CRC).

D. The use of other austenitic material, including nickel base alloys such as inconel 600, shall be evaluated on an individual case basis. Inconel 82 is the only commonly used nickel base weld considered to be resistant.

9.2.2 Reactor Vessel Closure Studs – Regulatory Guide 1.65

As a result of the License Renewal process, NMPNS is incorporating the commitment made to implement the preventive measures of NRC Regulatory Guide 1.65, dated October 1973, Materials and Inspections for Reactor Vessel Closure Studs, to mitigate cracking.

Note: Closure stud bolting is defined to include all studs (stud bolts), nuts, and washers used to fasten the pressure vessel head to the pressure vessel.

Provided below are the Regulatory Positions as defined in RG 1.65 that NMPNS will consider during a Reactor Vessel Closure Stud replacement:

1. Bolting Materials

a. Reactor vessel closure stud bolting should be fabricated from materials which have adequate toughness throughout the life cycle of the reactor. Stud bolting should meet the requirements set forth in Subsection NB, Requirements for Class 1 Components, Section III of the ASME Code. Bolting materials should meet the requirements of one of the following ASME specifications:

   (1) SA-540 Grade B-23 and B-24 bar (AISI 4340).

   (2) SA-193 Grade B-7 bar (AISI 4140, 4142, 4145).

   (3) SA-194 Grade 7 (nuts for bolting) (AISI 4140, 4142, 4145).

   (4) SA-320 Grade L-43 bar (AISI 4340).

b. The requirements of the specification in a above should be supplemented by the following:

   (1) The maximum measured ultimate tensile strength of the stud bolting material should not exceed 170 ksi.

(2) Charpy V impact testing should be performed according to ASME SA-370, Methods and Definitions for Mechanical Testing of Steel Products, and to be acceptable, the results must satisfy the requirements of Paragraph IV.A.4. of Appendix G to 10 CFR Part 50.

In case a test fails, one retest may be conducted according to sub-article NB-2350 of Section III, ASME Code.

(3) Stud bolting should not be metal-plated unless it has been demonstrated that the plating will not degrade the quality of the material in any significant way (e.g., corrosion, H2 embrittlement) or reduce the quality of results attainable by the various required inspection procedures. The stud bolting may have a manganese phosphate (or other acceptable) surface treatment. Lubricants for the stud bolting are permissible provided they are stable at operating temperatures and are compatible with the bolting and vessel material and the surrounding environment.

2. Inspection

The nondestructive examination of stud bolts and nuts should be performed according to sub-article NB-2580 of Section III of the ASME Code as supplemented by the following:

a. The stud bolts and nuts should be ultrasonically examined after final heat treatment and prior to threading.

b. The ultrasonic examination (paragraph NB-2584) should be conducted according to ASME Specification SA-388, Ultrasonic Examination of Heavy Steel Forgings.

c. The calibration standard used to establish the first back reflection for the ultrasonic testing should be based on good sound representative material. To assure that the material is representative, the selection of the standard should be based on a preliminary ultrasonic examination of a number of specimens (a minimum of three per standard).

d. The magnetic particle or liquid penetrant examination (paragraph NB-2583) should be performed on the studs and nuts after final heat treatment and threading.

e. The requirements of paragraph NB-2585 should be applied to all closure stud bolts and nuts.

3. Protection Against Corrosion

During the venting and filling of the pressure vessel and while the head is removed, the stud bolts and stud bolt holes in the vessel flange should be adequately protected from corrosion and contamination.

4. In-service Inspection

The in-service examination of pressure vessel stud bolting should be performed in accordance with the requirements of Section XI of the ASME Code as defined in Section 2 of this inspection plan.

9.2.3 Pressure Testing

Pressure testing shall be performed on replacements in accordance with the In-service Pressure Testing Program, Document NMP1-PT-003.

Note: 10 CFR 50.55a(b)(2)(xx) System Leakage Test - The NRC imposed a condition in 10 CFR 50.55a(b)(2)(xx) requiring Section III NDE be performed following repair and replacement activities if a system leakage test was to be used in lieu of a hydrostatic test under the 2003 Addenda through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(b)(2).

9.2.4 Pre-service Examinations

Prior to the systems return to service, a pre-service examination shall be made in accordance with IWB-2200, IWC-2200, IWD-2200, or IWF-2200.

9.3 Repair/Replacement Activities for IWE Class MC Components

Repair/Replacement activities for ASME Code Class MC shall be in accordance with the ASME Code Section XI 2001 Edition through the 2003 Addenda. Repair and Replacement supplemental guidelines are located in document CNG-NMP1-CISI-002.

Note: Section IWL is not applicable to Nine Mile Point Nuclear Station, as the NMP1 Plant uses a steel primary containment.

9.4 Roll Expansion Repair of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs

In accordance with a License Renewal Commitment, NMPNS was committed to implement ASME Code Case N-730, Roll Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs to eliminate leakage. Provided below are the examination and test requirements:

9.4.1 Prior to Roll Expansion

a. Prior to roll expansion, ultrasonic (UT) examination of the regions specified in Fig. 1 or Fig. 2 of Code Case N-730, shall be performed.
b. For stub-tube configuration (Fig. 1), the rolled region (Region 2) and the stub-tube-to-housing J-groove-weld region (Region 1) shall be examined.
c. If leakage is due to through-wall cracking of the housing, this Code Case shall not be used.
d. For roll region (Region 2) shall not have any planar flaws.
e. For housing indications in the area of the stub-tube-to-housing J-grove weld region or the housing-to-vessel-weld region (Region1) the housing including weld region shall be evaluated as a housing weld for the purpose of determining flaw acceptance.
f. The examination results shall be evaluated in accordance with IWB-3253.
g. If flaws exceed the acceptance standards of IWB-3523, they shall be evaluated to show that the requirements of IWB-3640 are satisfied. The Code Case may not be used if the requirements of IWB-3640 are not met.

9.4.2 Following Completion of Roll Expansion

(a) After completion of the roll expansion, UT examination of the stub-tube-to-housing J-groove weld region or housing-to-vessel-weld region (Region 1) shall be performed.
(b) The examination results shall be evaluated in accordance with IWB-3523.
(c) If flaws exceed the acceptance standards of IWB-3523, they shall be evaluated to show that the requirements of IWB-3640 are satisfied.
(d) This Code Case may not be used if the requirements of IWB-3640 are not met.

(e) After completion of the roll expansion, UT examination of the rolled region (Region 2) shall be performed to establish that no planar flaws exist in the rolled region.

9.4.3 Ultrasonic Examination Procedure

The UT procedure used in the examinations shall be demonstrated on a plant-specific mockup, with flaws located in the area of interest, in accordance with Appendix I of this Code Case.

9.4.4 Leakage

If the location of the leakage has not been determined, an in-vessel VT-1 visual examination of the leaking CRD penetration shall be made before the end of the next scheduled refueling outage, to attempt to locate the leakage source and to determine the general condition of the housing cracks, wear, or localized accumulation of corrosion products shall require corrective action. Roll expansion satisfies the corrective action requirement.

After completion of the post-roll-expansion UT examination, the CRD housing penetration shall be subjected to VT-2 visual examination in conjunction with a system leakage test in accordance with IWB-5000 and the system pressure test program. For CRD housings subjected to roll expansion, the acceptance criterion is no leakage.

9.4.5 CRD Penetrations with Continued Leakage Following Roll Expansion Repair

During the period of extended operation, should a CRD stub tube rolled in accordance with the provisions of the N-730 roll repair code case resume leaking, Nine Mile Point will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected:

1. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in RG 1.147.
2. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1.
3. A future developed mechanical/welded repair method subject to the approval of the NRC.

9.5 Authorized Nuclear In-service Inspector

The services of an Authorized Nuclear In-service Inspector (ANII) shall be used when making all repairs/replacements. The repair plan shall be made available for review by the ANII for all welded repairs/replacements. The ANII shall determine what hold points, if any, are required to monitor the repair/replacement activity. NMPNS shall notify the ANII prior to starting the repair / replacement and keep the inspector informed of the progress of the work so that necessary inspections may be performed.

9.6 Implementation

Managerial and administrative controls for ASME Section XI Class 1, 2 and 3 Repairs and Replacements are provided in fleet and site specific procedures.

9.7 ASME Code Cases

Appendix I of this document provides a listing of approved ASME Code Cases that are authorized under Regulatory Guide 1. 147, latest revision, and that may be used during a Repair or Replacement activity. All ASME Code Cases used during a Repair or Replacement activity shall be identified on the ASME Repair/Replacement Plan Certification Record and/or within the Repair/Replacement Plan. The ASME

Repair/Replacement Plan Certificate is provided in Section 10 of this document.

Note: For snubbers and pressure relief valves rotated from stock and installed on components (including piping systems), see Appendix I, ASME Code Case N-508-1 for alternative.

SECTION 10 - RECORDS AND REPORTS

10.0 RECORDS AND REPORTS

This section provides the requirements for the preparation and submittal of In-service Inspection records and reports as required by the applicable Edition and Addenda of the ASME Boiler and Pressure Vessel Code. This section also addresses additional reporting requirements for IGSCC components.

10.1 General

Examinations, tests, replacements, and repair records are prepared in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

As a alternate to the requirements of IWA-6000, NMPNS is implementing ASME Code Case N-532-4, "Repair/Replacement Activity Documentation Requirements and In-service Summary Report Preparation and Submission, Section XI, Division 1".

10.2 Owner's Activity Report

An Owner's Activity Report Form OAR-1 (Figure 10-1), shall be prepared and certified upon completion of each refueling outage.

Each Form OAR-1 shall contain the following:

a. A listing of item(s) with flaws or relevant conditions that exceeded the acceptance criteria of Section XI and that required evaluation to determine acceptability for continued service shall be provided with the information and format of Table 1. The information is required whether or not the flaw or relevant condition was discovered during a scheduled examination or test.

b. An abstract for repair/replacement activities that were required due to an item containing a flaw or relevant condition that exceeded Section XI acceptance criteria shall be provided with the information and format of Table 2. This information is required even if the discovery of the flaw or relevant condition that necessitated the repair/replacement activity did not result from an examination or test required by Section XI. If no acceptance criteria for a particular item is not specified in Section XI, the provisions of IWA-3100(b) shall be used to determine which repair/replacement activities are required to be included in the abstract.

c. If no items met the criteria of 2(a) or (b), the term "None" should be recorded in the applicable table.

d. If there are multiple inspection plans with different intervals, periods, Editions or Addenda, they shall be identified on Form OAR-1.

e. Form OAR-1 shall be certified by the Owner and presented to the Inspector for the required signature.

10.3 Cover Sheet

Each Owner's Activity Report will have a cover sheet that provides the following information:

a. Date of document completion

b. Name and address of Owner

c. Name and address of generating plant

d. Name and number designation of the plant

e. Commercial service date for the unit

10.4 Owner's Activity Report Submittal

Each Form OAR-1 shall be submitted to the USNRC within 90 days of the completion of the refueling outage.

10.5 Reporting Requirements for IGSCC

If any cracks are identified that do not meet the criteria for continued operation without evaluation given in Section XI of the Code, USNRC approval of flaw evaluation and/or repairs in accordance with IWB-3000 and IWA-4000 is required.

10.6 Reporting Requirements for Class MC and CC

Reporting requirements for ASME Code Class MC and CC are incorporated in CNG-NMP1-CISI-002 Containment Inspection Plan and Schedule.

10.7 Reporting Requirements for NUREG 0619

NUREG-0619 reporting requirements are no longer required for Nine Mile Point Nuclear Station Unit 1.

10.8 Reporting Requirements for Risk-Informed Examinations

As Risk-Informed weld examinations are considered an alternative to the ASME Section XI requirements, the reporting requirements shall be as required for ASME Section XI welds.

10.9 Inspection Program B Compliance

Following completion of each refueling outage or as an minimum following each inspection period the required percentages of examinations in each Examination Category shall be completed in accordance with Table IWB-2412-1, IWC-2412-1, IWD-2412-1and IWE-2412-1. A listing of examination categories, examinations required and completed as required by Section XI shall be prepared and determine to be acceptable for continued service shall be provided with the information and format of Figure 10-3.

FIGURE 10-1

FORM OAR-1
OWNER'S ACTIVITY REPORT

As required by the provisions of the ASME Code Case N-532-4 Page of

Report Number ____________________

Plant Nine Mile Point Nuclear Station, P. O. Box 63, Lycoming, New York 13093
(Name and Address of Plant)

Plant Unit 1 Commercial Service Date December 26, 1969 Refueling Outage Number RFO-21

Current Inspection Interval Fourth Inservice Inspection Interval August 22, 2009 to August 23, 2019
(1st, 2nd, 3rd, 4th, Other)

Current Inspection Period First Inservice Inspection Period August 23, 2009 to August 22, 2012
(1st, 2nd, 3rd)

Edition and Addenda of Section XI applicable to the Inspection Plans 2001 Edition, through 2003 Addenda

Date and Revision of Inspection Plans CNG-NMP1-ISI-004, Revision 00, January 15, 2009

Edition and Addenda of ASME Section XI applicable to Repairs and Replacements, if different than the Inspection Plan ____
____
(if applicable)

CERTIFICATE OF CONFORMANCE

I certify that the statements made in this report are correct: (b) the examinations and tests meet the Inspection Plan as required by the ASME Code, Section XI; and (c) the repair/replacement activities and evaluations supporting the completion of ____________ conform to the requirements of Section XI.
(refueling outage number)

Signed ____________________ Date ____________
(General Supervisor – Corporate Engineering Programs)

CERTIFICATE OF INSERVICE INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and the State or Province of ____________ and employed by ____________ of ____________ have inspected the items described in this Owner's Activity Report and state that to the best of my knowledge and belief, the Owner has performed all activities represented by this report in accordance with the requirements of Section XI.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the repair/replacement activities and evaluation described in this report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

____________________ Commissions ____________________
Inspector's Signature National Board, State, Province, and Endorsements

Date ____________ 19____

FIGURE 10-1 (Continued)

TABLE 1

ITEMS WITH FLAWS OR RELEVANT CONDITIONS THAT REQUIRE EVALUATION FOR CONTINUED SERVICE

As required by the provisions of the ASME Code Case N-532-4

Page of

EXAMINATION CATEGORY	ITEM NUMBER	ITEM DESCRIPTION	EVALUATION DESCRIPTION

FIGURE 10-1 (Continued)

TABLE 2
ABSTRACT OF REPAIR/REPLACEMENT ACTIVITIES REQUIRED FOR CONTINUED SERVICE

As required by the provisions of the ASME Code Case N-532-4

Page of

CODE CLASS	ITEM DESCRIPTION	DESCRIPTION OF WORK	DATE COMPLETED	REPAIR/REPLACEMENT PLAN NUMBER

FIGURE 10-2

FORM NIS-2A
REPAIR/REPLACEMENT CERTIFICATION RECORD

As required by the provisions of the ASME Code Case N-532-4

Page ____ of ____

OWNER'S CERTIFICATE OF COMPLIANCE

I certify that the ______________________________ activities represented by Repair/Replacement Plan Number______________________________ conforms to the requirements of Section XI.

Edition and Addenda of Section XI used: ______________________________

Code Case used: .______________________________
(if applicable)

Signed______________________________Date____________________,19_____
Owner or Owner's designee, Title

CERTIFICATE OF INSERVICE INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and the State or Province of______________________________ and employed by ______________________________ of ______________________________ have inspected the items described in Repair/Replacement Plan No.____________________ , and state that to the best of my knowledge and belief, the Owner has performed all the activities described in the Repair/Replacement Plan in accordance with the requirements of Section XI.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the activities described in the Repair/Replacement Plan. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or loss of any kind arising from or connected with this inspection.

______________________________Commissions______________________________
Inspector's Signature
National Board, State, Province, and Endorsements

Date____________________19_____

Figure 10-3 – ASME Section XI Summary Table

Date: 09/28/07
Revision: 00
Code Edition: A03

Nine Mile Point – Unit 1
Document ID / Section XI Summary

Page #

B-A Pressure Retaining Welds in Reactor Vessel

Item Number	Item Description	Zone	Tbl	No of Comp	No Req	Number of Components Scheduled / Completed 1st. Period	2nd. Period	3rd. Period
a.	b	c	d	e	f	g, / h, l	j, / k,l	m, / n,o

Legend

Date: Date of Summary Table print out
Revision: Revision number of inspection plan and schedule
Code Edition: A03 (2003 Addenda

a	ASME Examination Item Number
b.	Examination Item Description
c.	Zone or system identification number
d.	Table number identifier
e.	Number of components subject to examination
f.	Number of components required for the interval
g / h, l	Number items scheduled in first period / Number items completed in first period, Percent completed for first period
j / k, l	Number items scheduled in second period / Number items completed in second period, Percent completed for second period
m / n, o	Number items scheduled in third period / Number items completed in third period, Percent completed for third period

APPENDIX A - CLASS 1 SUMMARY TABLES

ASME CODE CLASS 1
SUMMARY TABLES

Page 1 through 15

B-A PRESSURE RETAINING WELDS IN REACTOR VESSEL

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
B1.11	CIRCUMFERENTIAL SHELL WELDS All Welds, 100% weld length, Deferral permissible										
		00.1	A	4	4	0 / 0	0.0 %	0 / 0	0.0 %	4 / 0	100.0 %
B1.12	LONGITUDINAL SHELL WELDS All Welds, 100% of weld length Deferral permissible										
		00.1	A	12	12	0 / 0	0.0 %	0 / 0	0.0 %	12 / 0	100.0 %
B1.21	CIRCUMFERENTIAL HEAD WELDS Access. length of All welds, 100% weld length, Deferral permissible.										
		00.0	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		00.1	A	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		Item Total:		3	3	0 / 0	0.0 %	1 / 0	33.3 %	2 / 0	100.0 %
B1.22	MERIDIONAL HEAD WELDS Accessible length of All welds, 100% weld length, Deferral permissible.										
		00.0	A	8	8	6 / 0	75.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		00.1	A	14	14	0 / 0	0.0 %	0 / 0	0.0 %	14 / 0	100.0 %
		Item Total:		22	22	6 / 0	27.3 %	2 / 0	36.4 %	14 / 0	100.0 %
B1.30	SHELL-TO-FLANGE WELD 100% of weld length, Partial Deferral Permissable See footnote(3) and (5).										
		00.1	A	2	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
B1.40	HEAD-TO-FLANGE WELD 100% of weld length, Partial Deferral Permissable see footnote(4) and (5).										
		00.0	A	3	3	1 / 0	33.3 %	1 / 0	66.7 %	1 / 0	100.0 %
	Category Total:			46	46	8 / 0	17.4 %	4 / 0	26.1 %	34 / 0	100.0 %

B-D FULL PENETRATION WELDS OF NOZZLES IN VESSELS (INSPECTION PROGRAM B)

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B3.100	REACTOR VESSEL-NOZZLE INSIDE RADIUS SECTION All Nozzles, See notes (2) and (5), 25% but not more than 50% First Period										
		031-N4	A	4	4	0 / 0	0.0 %	0 / 0	0.0 %	4 / 0	100.0 %
		032-N2	A	5	5	0 / 0	0.0 %	0 / 0	0.0 %	5 / 0	100.0 %
		036-N7	A	10	10	10 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		037-N8	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-N5	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		040-N6	A	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		44.1-N9	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
	Item Total:			23	23	11 / 0	47.8 %	2 / 0	56.5 %	10 / 0	100.0 %
B3.200	REACTOR VESSEL-NOZZLE INSIDE RADIUS SECTION All nozzles, Code Case N-648-1 VT in lieu of UT										
		001-N3	A	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		032-N1	A	5	5	0 / 0	0.0 %	2 / 0	40.0 %	3 / 0	100.0 %
		036-N7	A	8	8	0 / 0	0.0 %	4 / 0	50.0 %	4 / 0	100.0 %
		039-N5	A	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		040-N6	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
	Item Total:			17	17	1 / 0	5.9 %	7 / 0	47.1 %	9 / 0	100.0 %
B3.90	REACTOR VESSEL-NOZZLE-TO-VESSEL WELDS All Nozzles, 25% to 50% 1st period, Remainder by end of Interval, See note (2), (3), and (5)										
		001-N3	A	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		031-N4	A	4	4	0 / 0	0.0 %	0 / 0	0.0 %	4 / 0	100.0 %
		032-N1	A	5	5	0 / 0	0.0 %	2 / 0	40.0 %	3 / 0	100.0 %
		032-N2	A	5	5	0 / 0	0.0 %	0 / 0	0.0 %	5 / 0	100.0 %
		036-N7	A	18	18	10 / 0	55.6 %	4 / 0	77.8 %	4 / 0	100.0 %
		037-N8	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-N5	A	2	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		040-N6	A	2	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		44.1-N9	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
	Item Total:			40	40	12 / 0	30.0 %	9 / 0	52.5 %	19 / 0	100.0 %
	Category Total:			80	80	24 / 0	30.0 %	18 / 0	52.5 %	38 / 0	100.0 %

B-F PRESSURE RETAINING DISSIMILAR METAL WELDS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B5.10	REACTOR VESSEL-NOZZLE-TO-SAFE END BUTT WELDS NPS 4 or LARGER All welds, May coincide with Cat.B-D examinations, incorporated under RI-ISI program, See Notes 1, 2										
		32-N1	A	5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32-N2	A	5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36-N7	A	18	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37-N8	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39-N5	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40-N6	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Item Total:			33	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
B5.20	REACTOR VESSEL-NOZZLE-TO-SAFE END BUTT WELDS, LESS THAN NPS 4 All welds May coincide with Category B-D examinations, See note 1										
		36-N13	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36-N14	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36-N15	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36-N16	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36-N17	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1-N12	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.1-N9	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Item Total:			11	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Category Total:			44	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

B-G-1 PRESSURE RETAINING BOLTING GREATER THAN 2 INCHES IN DIAMETER

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B6.10	REACTOR VESSEL-CLOSURE HEAD NUTS All nuts, Deferral permissible.										
		00.0	A	64	64	22 / 0	34.4 %	21 / 0	67.2 %	21 / 0	100.0 %
B6.180	PUMPS-BOLTS AND STUDS All bolts & studs Limit to one pump among group of valves and under Cat B-L-2										
		032-187	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		032-188	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-189	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-190	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-191	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
B6.190	PUMPS-FLANGE SURFACE, WHEN CONNECTION DISASSEMBLED All flange surf, Limit to one flange among group of flanges and under Cat B-L-2										
		032-187	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		032-188	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-189	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-190	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-191	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
B6.20	REACTOR VESSEL-CLOSURE STUDS All studs, Deferral permissible										
		00.0	A	64	64	22 / 0	34.4 %	21 / 0	67.2 %	21 / 0	100.0 %
B6.200	PUMPS-NUTS, BUSHINGS, AND WASHERS All nuts, bush. & wash, Limit to one pump among group pumps and Cat B-L-2										
		032-187	A	3	3	0 / 0	0.0 %	0 / 0	0.0 %	3 / 0	100.0 %
		032-188	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-189	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-190	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-191	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		15	3	0 / 0	0.0 %	0 / 0	0.0 %	3 / 0	100.0 %
B6.210	VALVES-BOLTS AND STUDS All bolts and studs, Limited to one valve among group of valves and under Category BM-2,										
		031-V04	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		038-V10	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		040-V11	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		5	3	2 / 0	66.7 %	0 / 0	66.7 %	1 / 0	100.0 %
B6.220	VALVES-FLANGE SURFACE, WHEN CONNECTION DISASSEMBLED All flange surfaces, Limited to valve selected under Category B-M-2										
		031-V04	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		038-V10	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		040-V11	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		5	3	2 / 0	66.7 %	0 / 0	66.7 %	1 / 0	100.0 %
B6.230	VALVES-NUTS, BUSHINGS, AND WASHERS All nuts, bushings, washers, Limited to valve selected under B-M-2										
		031-V04	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		038-V10	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		040-V11	A	6	3	3 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		9	5	4 / 0	80.0 %	0 / 0	80.0 %	1 / 0	100.0 %

B-G-1 **PRESSURE RETAINING BOLTING GREATER THAN 2 INCHES IN DIAMETER**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B6.40	REACTOR VESSEL-THREADS IN FLANGE All threads in flange, Deferral permissible										
		00.1	A	64	64	22 / 0	34.4 %	21 / 0	67.2 %	21 / 0	100.0 %
B6.50	REACTOR VESSEL-CLOSURE WASHERS, BUSHINGS All washers & Bushings, Deferral permissible										
		00.0	A	128	128	44 / 0	34.4 %	42 / 0	67.2 %	42 / 0	100.0 %
		00.1	A	64	64	22 / 0	34.4 %	21 / 0	67.2 %	21 / 0	100.0 %
	Item Total:			192	192	66 / 0	34.4 %	63 / 0	67.2 %	63 / 0	100.0 %
	Category Total:			428	400	140 / 0	35.0 %	126 / 0	66.5 %	134 / 0	100.0 %

B-G-2 PRESSURE RETAINING BOLTING, 2 INCHES AND LESS IN DIAMETER

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B7.50	PIPING-BOLTS, STUDS, AND NUTS One flange among group of flanges and only when disassembled or removed										
		000-F01	A	6	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		000-F02	A	12	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		032-F03	A	7	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		032-F04	A	2	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		032-F05	A	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		037-F06	A	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		30	7	1 / 0	14.3 %	4 / 0	71.4 %	2 / 0	100.0 %
B7.60	PUMPS-BOLTS, STUDS, AND NUTS Limited under B-L-2. Only when disassembled or removed										
		032-187	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-188	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-189	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-190	A	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		032-191	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		5	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
B7.70	VALVES-BOLTS, STUDS, AND NUTS Limited under B-M-2 and only when disassembled or removed										
		000-V17	A	9	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		001-V01	A	4	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		001-V02	A	6	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		001-V03	A	6	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		01.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		031-V04	A	4	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		031-V05	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		032-V06	A	10	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		033-V07	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		033-V13	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		033-V19	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		038-V08	A	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		039-V09	A	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-V12	A	4	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		039-V15	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		040-V14	A	6	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		040-V16	A	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		32.0	A	5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.0	A	4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.3	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		80	15	4 / 0	26.7 %	7 / 0	73.3 %	4 / 0	100.0 %
		Category Total:		115	23	6 / 0	26.1 %	11 / 0	73.9 %	6 / 0	100.0 %

B-J PRESSURE RETAINING WELDS IN PIPING

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B9.11	CIRCUMFERENTIAL PIPE WELDS, NPS 4 or LARGER At least 25% of the welds										
		01.0	A	34	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		31.0	A	50	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	96	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.0	A	6	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.2	A	23	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		38.0	A	16	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	A	48	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	A	70	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		346	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
B9.21	CIRCUMFERENTIAL PIPE WELDS, LESS THAN NPS 4 At least 25% of the welds										
		01.0	A	14	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	32	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36.0	A	6	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.0	A	14	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	A	9	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	A	19	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.1	A	15	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.3	A	5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		114	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
B9.31	BRANCH CONNECTION WELDS, NPS 4 or LARGER At least 25% of the welds										
		01.0	A	6	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		9	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
B9.32	BRANCH CONNECTION WELDS, LESS THAN NPS 4 At least 25%										
		01.0	A	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	18	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	A	4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		26	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
B9.40	SOCKET WELDS At least 25% of the welds										
		01.0	A	9	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	30	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36.0	A	6	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.0	A	18	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	A	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.1	A	16	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		91	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		586	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

B-K INTEGRAL ATTACHMENTS FOR VESSELS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed		
						1st Period	2nd Period	3rd Period
B10.10	PRESSURE VESSEL-INTEGRALLY WELDED ATTACHMENTS 100% of the length of the welded attachments and only one attachment amoung group of vessels							
		00.1-RPV	A	6	3	2 / 0 66.7 %	0 / 0 66.7 %	1 / 0 100.0 %
B10.20	PIPING-INTEGRALLY WELDED ATTACHMENTS 10% of all Welded attachments associated with component supports selected under IWF2510							
		01.0	A	8	1	0 / 0 0.0 %	1 / 0 100.0 %	0 / 0 100.0 %
		02.0	A	2	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		31.0	A	10	1	0 / 0 0.0 %	0 / 0 0.0 %	1 / 0 100.0 %
		32.0	A	10	2	0 / 0 0.0 %	1 / 0 50.0 %	1 / 0 100.0 %
		33.0	A	2	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		33.2	A	10	1	0 / 0 0.0 %	0 / 0 0.0 %	1 / 0 100.0 %
		33.3	A	3	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		37.1	A	2	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		38.0	A	4	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		39.0	A	8	1	0 / 0 0.0 %	0 / 0 0.0 %	1 / 0 100.0 %
		40.0	A	16	2	0 / 0 0.0 %	1 / 0 50.0 %	1 / 0 100.0 %
		44.1	A	1	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		Item Total:		76	8	0 / 0 0.0 %	3 / 0 37.5 %	5 / 0 100.0 %
B10.40	VALVES-INTEGRALLY WELDED ATTACHMENTS 10% of all Welded attachments associated with component supports selected under IWF2510							
		038-V08	A	1	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		038-V10	A	1	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		33.0	A	2	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		40.0	A	2	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		42.1	A	1	0	0 / 0 0.0 %	0 / 0 0.0 %	0 / 0 0.0 %
		44.3	A	1	1	1 / 0 100.0 %	0 / 0 100.0 %	0 / 0 100.0 %
		Item Total:		8	1	1 / 0 100.0 %	0 / 0 100.0 %	0 / 0 100.0 %
		Category Total:		90	12	3 / 0 25.0 %	3 / 0 50.0 %	6 / 0 100.0 %

B-L-2 PUMP CASINGS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
B12.20	PUMPS-PUMP CASINGS Internal surfaces of one pump, among group of pumps and only when disassembled for maint, repair or										
		032-187	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		032-188	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-189	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-190	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-191	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		Category Total:		5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %

B-M-1 PRESSURE RETAINING WELDS IN VALVE BODIES

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
B12.40	VALVES-VALVE BODY WELDS, NPS 4 or LARGER Limited to one valve among a group of valves and100% of weld length										
		001-V03	A	6	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		Category Total:		6	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %

B-M-2 VALVE BODIES

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
B12.50	VALVES-VALVE BODIES EXCEEDING 4 INCHES NOMINAL PIPE SIZE										
	Internal surfaces of one valve among each group of valves if diss for maint, repair or Vol exam, on										
		000-V17	A	9	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		001-V01	A	4	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		001-V02	A	6	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		001-V03	A	6	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		031-V04	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		031-V05	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		032-V06	A	10	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		033-V07	A	3	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		033-V13	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		033-V19	A	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		038-V08	A	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		038-V10	A	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		039-V09	A	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-V12	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		039-V15	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		039-V18	A	4	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		040-V11	A	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		040-V14	A	6	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		040-V16	A	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		68	20	6 / 0	30.0 %	8 / 0	70.0 %	6 / 0	100.0 %
		Category Total:		68	20	6 / 0	30.0 %	8 / 0	70.0 %	6 / 0	100.0 %

B-N-1 INTERIOR OF REACTOR VESSEL

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B13.10	REACTOR VESSEL-VESSEL INTERIOR First refuel outage then once each inspection period areas above & below reactor core	00.2	A	55	165	55 / 0	33.3 %	55 / 0	66.7 %	55 / 0	100.0 %
	Category Total:			55	165	55 / 0	33.3 %	55 / 0	66.7 %	55 / 0	100.0 %

B-N-2 INTEGRALLY WELDED CORE SUPPORT STRUCTURES AND INTERIOR ATTACHMENTS TO REACTOR VESSEL

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B13.20	REACTOR VESSEL (BWR)-INTERIOR ATTACHMENTS WITHIN BELTLINE REGION All welds Deferral Permissable	00.2	A	3	3	0 / 0	0.0 %	0 / 0	0.0 %	3 / 0	100.0 %
B13.30	REACTOR VESSEL (BWR)-INTERIOR ATTACHMENTS BEYOND BELTLINE REGION Accessible welds	00.2	A	18	18	4 / 0	22.2 %	10 / 0	77.8 %	4 / 0	100.0 %
B13.40	REACTOR VESSEL (BWR)-CORE SUPPORT STRUCTURE Accessible surfaces	00.2	A	7	7	2 / 0	28.6 %	0 / 0	28.6 %	5 / 0	100.0 %
	Category Total:			28	28	6 / 0	21.4 %	10 / 0	57.1 %	12 / 0	100.0 %

B-O **PRESSURE RETAINING WELDS IN CONTROL ROD HOUSINGS**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
B14.10	REACTOR VESSEL-WELDS IN CONTROL ROD DRIVE HOUSINGS 10% peripheral CRD Housings, Deferral Permissable										
		44.0	A	8	4	0 / 0	0.0 %	4 / 0	100.0 %	0 / 0	100.0 %
	Category Total:			8	4	0 / 0	0.0 %	4 / 0	100.0 %	0 / 0	100.0 %

B-P ALL PRESSURE RETAINING COMPONENTS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
B15.10	Pressure Retaining Components System Leakage Test each Refueling Outage										
		00.1	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		01.0	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		31.0	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	A	15	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36.0	A	11	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		38.0	A	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.0	A	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		40	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		40	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

APPENDIX B - CLASS 2 SUMMARY TABLES

ASME CODE CLASS 2
SUMMARY TABLES
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C-A PRESSURE RETAINING WELDS IN PRESSURE VESSELS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
C1.30	TUBESHEET-TO-SHELL WELDS Limited to one vessel in a group of vessels										
		039-HE111	B	2	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-HE112	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE121	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE122	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE111	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE112	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE121	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE122	B	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		Item Total:		16	4	1 / 0	25.0 %	1 / 0	50.0 %	2 / 0	100.0 %
		Category Total:		16	4	1 / 0	25.0 %	1 / 0	50.0 %	2 / 0	100.0 %

C-B PRESSURE RETAINING NOZZLE WELDS IN VESSELS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
C2.11	NOZZLE-TO-SHELL (NOZ TO HEAD or NOZ TO NOZ) WELD <= 1/2 IN. NOMINAL THICKNESS All nozzles at Terminal Ends of piping runs selected under GF, Limited to one vessel among vessels										
		080-HE111	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE112	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE121	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE122	B	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		Item Total:		8	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
C2.21	NOZZLE-TO-SHELL (NOZZLE TO HEAD or NOZ TO NOZ) WELD > 1/2 IN. NOMINAL THICKNESS WITHOUT REINFORCING PLATE All nozzles at terminal ends of piping runs selected under Category GF, Limited to one vessel										
		039-HE111	B	2	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		039-HE112	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE121	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE122	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		8	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Category Total:		16	4	1 / 0	25.0 %	1 / 0	50.0 %	2 / 0	100.0 %

C-C INTEGRAL ATTACHMENTS FOR VESSELS, PIPING, PUMPS, AND VALVES

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
C3.10	PRESSURE VESSELS-INTEGRALLY WELDED ATTACHMENTS Only one welded attachment shall be selected among group of vessels or single vessel										
		080-HE111	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE112	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE121	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE122	B	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		Item Total:		8	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
C3.20	PIPING-INTEGRALLY WELDED ATTACHMENTS 10% of welded attachments associated with component supports selected under IWF2510										
		39.0	B	34	4	2 / 0	50.0 %	0 / 0	50.0 %	2 / 0	100.0 %
		44.2	B	4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		80.0	B	63	9	3 / 0	33.3 %	4 / 0	77.8 %	2 / 0	100.0 %
		81.0	B	80	9	3 / 0	33.3 %	2 / 0	55.6 %	4 / 0	100.0 %
		93.0	B	20	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		Item Total:		201	24	8 / 0	33.3 %	7 / 0	62.5 %	9 / 0	100.0 %
		Category Total:		209	26	8 / 0	30.8 %	7 / 0	57.7 %	11 / 0	100.0 %

C-F-1 PRESSURE RETAINING WELDS IN AUSTENITIC STEEL OR HIGH ALLOY PIPING

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
C5.11	CIRCUMFERENTIAL PIPE WELDS> 3/8 IN. NOMINAL WALL THICKNESS FOR PIPING>NPS 4 7.5% but not less than 28 welds										
		39.0	B	75	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Category Total:			75	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

C-F-2 PRESSURE RETAINING WELDS IN CARBON OR LOW ALLOY STEEL PIPING

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
C5.51	CIRCUMFERENTIAL WELD 7.5%, But not less than 28 welds										
		44.2	B	59	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		80.0	B	292	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		81.0	B	174	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		81.1	B	60	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		93.0	B	8	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		93.1	B	49	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		642	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
C5.81	CIRCUMFERENTIAL PIPE BRANCH CONNECTIONS OF BRANCH PIPING> NPS 2 7.5%, But not less than 28 welds										
		80.0	B	14	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		81.0	B	8	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		22	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		664	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

C-G PRESSURE RETAINING WELDS IN PUMPS AND VALVES

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
C6.10	PUMP CASING WELDS Only one pump among each group of pump										
		080-P111	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-P112	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-P121	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-P122	B	10	10	5 / 0	50.0 %	5 / 0	100.0 %	0 / 0	100.0 %
		081-P111	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		081-P112	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		081-P121	B	10	10	0 / 0	0.0 %	0 / 0	0.0 %	10 / 0	100.0 %
		081-P122	B	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		80	20	5 / 0	25.0 %	5 / 0	50.0 %	10 / 0	100.0 %
		Category Total:		80	20	5 / 0	25.0 %	5 / 0	50.0 %	10 / 0	100.0 %

C-H ALL PRESSURE RETAINING COMPONENTS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
C7.10	Pressure Retaining Components System Leakage Test each inspection period										
		121.0	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		41.0	B	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.0	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.2	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		63.0	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		70.0	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		80.0	B	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		12	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		12	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

APPENDIX C - CLASS 3 SUMMARY TABLES

ASME CODE CLASS 3
SUMMARY TABLES
Page 23 through 24

D-A WELDED ATTACHMENTS FOR VESSELS, PIPING, PUMPS, AND VALVES

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
D1.10	PRESSURE VESSEL WELDED ATTACHMENTS 100% of weld length, Only one welded attachment among group of vessels or single vessel										
		038-HE11	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		038-HE12	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		038-HE13	C	2	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		039-HE111	C	3	3	0 / 0	0.0 %	0 / 0	0.0 %	3 / 0	100.0 %
		039-HE112	C	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE121	C	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE122	C	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-HE11	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-HE12	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-HE13R	C	2	2	2 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		070-HE14R	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-HE15R	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		28	7	2 / 0	28.6 %	0 / 0	28.6 %	5 / 0	100.0 %
D1.20	PIPING WELDED ATTACHMENTS 100% of weld length, 10% sample, proportional to number connected to piping in each system										
		54.0	C	42	4	0 / 0	0.0 %	3 / 0	75.0 %	1 / 0	100.0 %
		70.0	C	35	4	0 / 0	0.0 %	2 / 0	50.0 %	2 / 0	100.0 %
		72.0	C	11	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		93.0	C	69	6	6 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		157	16	6 / 0	37.5 %	5 / 0	68.8 %	5 / 0	100.0 %
		Category Total:		185	23	8 / 0	34.8 %	5 / 0	56.5 %	10 / 0	100.0 %

D-B ALL PRESSURE RETAINING COMPONENTS

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
D2.10	Pressure Retaining Components System Leakage Test each inspection period										
		210.1	C	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		50.0	C	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		57.0	C	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		60.0	C	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		70.0	C	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		72.0	C	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		9	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		9	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

NINE MILE POINT NUCLEAR STATION, UNIT 1 FOURTH INSERVICE INSPECTION INTERVAL	CNG-NMP1-ISI-004
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APPENDIX D - CLASS 1, 2, 3 COMPONENT SUPPORT SUMMARY TABLES

ASME CODE CLASS 1, 2, 3 COMPONENT SUPPORT
SUMMARY TABLES
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F-A **Class 1 Piping Supports**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
F1.10A	25% of Class 1 One Directional Categorized to identify support types by component function A, B, C, etc										
		32.0	D	20	6	0 / 0	0.0 %	5 / 0	83.3 %	1 / 0	100.0 %
		33.1	D	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		33.2	D	4	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		37.0	D	5	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		40.0	D	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		42.1	D	4	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		44.1	D	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		37	13	1 / 0	7.7 %	7 / 0	61.5 %	5 / 0	100.0 %
F1.10B	25% of Class 1 Multidirectional Categorized to identify support types by component function A, B, C, etc,										
		01.0	D	2	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		02.0	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		31.0	D	2	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		33.0	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.2	D	2	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		33.3	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	D	4	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		38.0	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.0	D	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		42.1	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.1	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		22	6	1 / 0	16.7 %	2 / 0	50.0 %	3 / 0	100.0 %
F1.10C	25% of Class 1 Allows Thermal Movement Categorized to identify support types by component function A, B, C, etc,										
		01.0	D	6	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		31.0	D	14	4	0 / 0	0.0 %	2 / 0	50.0 %	2 / 0	100.0 %
		32.0	D	35	10	0 / 0	0.0 %	6 / 0	60.0 %	4 / 0	100.0 %
		33.0	D	2	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		33.2	D	2	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		33.3	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36.0	D	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		37.0	D	5	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		37.1	D	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		38.0	D	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		39.0	D	10	4	0 / 0	0.0 %	2 / 0	50.0 %	2 / 0	100.0 %
		40.0	D	12	5	0 / 0	0.0 %	3 / 0	60.0 %	2 / 0	100.0 %
		42.1	D	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		44.1	D	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		Item Total:		99	34	0 / 0	0.0 %	19 / 0	55.9 %	15 / 0	100.0 %
F1.20A	15% of Class 2 One Directional Categorized to identify support types by component function A, B, C, etc.										
		44.2	D	11	3	2 / 0	66.7 %	0 / 0	66.7 %	1 / 0	100.0 %
		80.0	D	35	8	7 / 0	87.5 %	0 / 0	87.5 %	1 / 0	100.0 %
		81.0	D	47	11	5 / 0	45.5 %	2 / 0	63.6 %	4 / 0	100.0 %
		93.0	D	21	3	1 / 0	33.3 %	2 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		114	25	15 / 0	60.0 %	4 / 0	76.0 %	6 / 0	100.0 %

F-A Class 2 Piping Supports

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	1st Period		2nd Period		3rd Period	
F1.20B	15% of Class 2 Multidirectional Categorized to identify support types by component function A, B, C, etc.										
		39.0	D	15	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		44.2	D	10	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		80.0	D	55	8	3 / 0	37.5 %	1 / 0	50.0 %	4 / 0	100.0 %
		81.0	D	44	7	1 / 0	14.3 %	2 / 0	42.9 %	4 / 0	100.0 %
		93.0	D	20	4	2 / 0	50.0 %	1 / 0	75.0 %	1 / 0	100.0 %
	Item Total:			144	22	7 / 0	31.8 %	4 / 0	50.0 %	11 / 0	100.0 %
F1.20C	15% of Class 2 Allows Thermal Movement Categorized to identify support types by component function A, B, C, etc.										
		39.0	D	27	6	3 / 0	50.0 %	1 / 0	66.7 %	2 / 0	100.0 %
		44.2	D	4	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		80.0	D	45	8	1 / 0	12.5 %	5 / 0	75.0 %	2 / 0	100.0 %
		81.0	D	18	5	1 / 0	20.0 %	2 / 0	60.0 %	2 / 0	100.0 %
	Item Total:			94	20	6 / 0	30.0 %	8 / 0	70.0 %	6 / 0	100.0 %
F1.30A	10% of Class 3 One Directional Categorized to identify types by component support function A, B, C, etc. within each system										
		54.0	D	42	5	1 / 0	20.0 %	3 / 0	80.0 %	1 / 0	100.0 %
		57.0	D	5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		70.0	D	136	14	0 / 0	0.0 %	2 / 0	14.3 %	12 / 0	100.0 %
		72.0	D	48	5	3 / 0	60.0 %	0 / 0	60.0 %	2 / 0	100.0 %
		93.0	D	74	7	3 / 0	42.9 %	3 / 0	85.7 %	1 / 0	100.0 %
	Item Total:			305	32	7 / 0	21.9 %	8 / 0	46.9 %	17 / 0	100.0 %
F1.30B	10% of Class 3 Multidirectional Categorized to identify types by component support function A, B, C, etc. within each system										
		54.0	D	39	4	0 / 0	0.0 %	3 / 0	75.0 %	1 / 0	100.0 %
		60.0	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		70.0	D	76	7	0 / 0	0.0 %	2 / 0	28.6 %	5 / 0	100.0 %
		72.0	D	8	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		93.0	D	62	8	6 / 0	75.0 %	2 / 0	100.0 %	0 / 0	100.0 %
	Item Total:			187	20	6 / 0	30.0 %	8 / 0	70.0 %	6 / 0	100.0 %
F1.30C	10% of Class 3 Allows Thermal Movement Categorized to identify types by component support function A, B, C, etc. within each system										
		54.0	D	4	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		70.0	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		93.0	D	4	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
	Item Total:			10	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %

F-A Supports Other Than Piping Supports

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
F1.40B	100% of the supports, For multiple components, only one of multiple components required Only one of the multiple components are required										
		00.1-RPV	D	9	9	2 / 0	22.2 %	4 / 0	66.7 %	3 / 0	100.0 %
		032-P187	D	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		032-P188	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P189	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P190	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P191	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		038-HE11	D	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		038-HE12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		038-HE13	D	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		039-HE111	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE112	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE121	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		039-HE122	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-F12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-HE11	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-HE12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-P11	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-P12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-T68	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054-T69	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		054F11	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-HE13R	D	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		070-HE14R	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-HE15R	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-P01	D	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		070-P02	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-P03	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		070-T126	D	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		072-P11	D	1	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		072-P12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE111	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE112	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE121	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-HE122	D	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		080-P111	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-P112	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		080-P121	D	1	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		080-P122	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		081-P111	D	2	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		081-P112	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		081-P121	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		081-P122	D	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		093-P111	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		093-P112	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		093-P121	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		093-P122	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		58	20	6 / 0	30.0 %	5 / 0	55.0 %	9 / 0	100.0 %

F-A Supports Other Than Piping Supports

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
F1.40C	100% of the supports, For multiple components, only one of multiple components required Only one of the multiple omponents are required										
		032-P11	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P12	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P13	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P14	D	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		032-P15	D	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		5	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		Category Total:		1075	195	50 / 0	25.6 %	67 / 0	60.0 %	78 / 0	100.0 %

APPENDIX E - IGSCC 88-01 SUMMARY TABLES

IGSCC GENERIC LETTER 88-01
SUMMARY TABLES

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GL-A Resistant Materials

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
G-L-A1	USNRC Generic Letter 88-01, Supplement 1, Class 1, UT 25% every 10 years (at least 12% in 6 years, incorporated under RI-ISI Program										
		32.0	E	92	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.0	E	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	E	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	E	20	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		116	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
G-L-A2	USNRC Generic Letter 88-01, Supplement 1, Class 1 VT-2 25% every 10 years (at least 12% in 6 years), incorporated under RI-ISI Program										
		33.0	E	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	E	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		39.0	E	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
G-L-A3	USNRC Generic Letter 88-01, Supplement 1, Class 2, UT 25% every 10 years (at least 12% in 6 years), incorporated under RI-ISI Program										
		39.0	E	12	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		132	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

GL-D **Non-resistant Materials; No Stress Improvement**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
G-L-D1	USNRC Generic Letter 88-01, Supplement 1, Class 1 UT At least 100% every 3 refueling cycles, 100% every 6 years BWRVIP-75										
		32.0-N1A	E	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		32.0-N1B	E	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		32.0-N1C	E	1	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		32.0-N1D	E	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		32.0-N1E	E	1	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		32.0-N2A	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		32.0-N2B	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		32.0-N2C	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		32.0-N2D	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		32.0-N2E	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		33.0	E	2	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		33.2	E	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		37.0	E	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		38.0	E	12	22	7 / 0	31.8 %	5 / 0	54.5 %	10 / 0	100.0 %
		39.0	E	22	44	18 / 0	40.9 %	4 / 0	50.0 %	22 / 0	100.0 %
		39.0-N5A	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		39.0-N5B	E	1	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		40.0	E	70	134	35 / 0	26.1 %	32 / 0	50.0 %	67 / 0	100.0 %
		40.0-N6A	E	1	2	1 / 0	50.0 %	0 / 0	50.0 %	1 / 0	100.0 %
		40.0-N6B	E	1	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		Item Total:		122	229	64 / 0	27.9 %	55 / 0	52.0 %	110 / 0	100.0 %
G-L-D2	USNRC Generic Letter 88-01, Supplement 1, Class 2, UT At least 100% every (3) refueling outages, 100% every 6 years BWRVIP-75										
		39.0	E	31	62	2 / 0	3.2 %	29 / 0	50.0 %	31 / 0	100.0 %
		Category Total:		153	291	66 / 0	22.7 %	84 / 0	51.5 %	141 / 0	100.0 %

GL-E **Known crack, but reinforced by weld overlay**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
G-L-E1	Once every two refueling cycles 25% every 10 years, BWRVIP-75										
		33.1	E	2	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		Category Total:		2	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %

GL-F **Known Cracks, accepted by analysis**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
G-L-F	Limited for additional service without repair, Examine each outage. Every refueling outage, or once per period per IWB-2420(b)										
		32.0	E	5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Category Total:			5	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

GL-G **Non-Resistant; No SI treatment;**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
G-L-G1	USNRC Generic Letter 88-01, Supplement 1, Class 1, VT-2 Each refueling outage, in conjunction with system pressure test										
		33.2	E	1	5	1 / 0	20.0 %	2 / 0	60.0 %	2 / 0	100.0 %
		33.3	E	1	5	1 / 0	20.0 %	2 / 0	60.0 %	2 / 0	100.0 %
		37.0	E	1	5	1 / 0	20.0 %	2 / 0	60.0 %	2 / 0	100.0 %
		38.0	E	4	20	4 / 0	20.0 %	8 / 0	60.0 %	8 / 0	100.0 %
		39.0	E	6	30	6 / 0	20.0 %	12 / 0	60.0 %	12 / 0	100.0 %
		40.0	E	6	30	6 / 0	20.0 %	12 / 0	60.0 %	12 / 0	100.0 %
		Item Total:		19	95	19 / 0	20.0 %	38 / 0	60.0 %	38 / 0	100.0 %
		Category Total:		19	95	19 / 0	20.0 %	38 / 0	60.0 %	38 / 0	100.0 %

GL-S **GL88-01 Supp. 1 sample of RWCU welds**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed 1st Period		2nd Period		3rd Period	
G-L-S	Scheduled RWCU welds outside CIVs each RFO No further exam required										
		33.0	E	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.1	E	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
G-L-S1	USNRC Generic Letter 88-01, Supplement 1, Class 1, UT At least 10% of RWCU outboard of CIVs, Each RFO	33.1	E	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
G-L-S2	USNRC Generic Letter 88-01, Supplement 1, Class 1, UT At least 10% of RWCU outboard of CIVs, each RFO	33.1	E	23	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
	Category Total:			30	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

APPENDIX F - RISK-INFORMED SUMMARY TABLES

ALTERNATIVE RISK-INFORMED SUMMARY TABLES

Pages 36 thru 38

Systems subject to examination requirements of this Appendix are identified in the Table below:

System Descriptions	System Abbreviations	System Identification Number
Reactor Pressure Vessel	RPV	00.0, 09.0
Main Steam System	MSS	01.0
Feedwater System	FWS	31.0
Reactor Recirculation System	RR	32.0
Reactor Water Cleanup System	RWCU	33.0, 33.2, 37.0, 37.1
Reactor Vessel Instrumentation	RXVI	36.0
Shutdown Cooling System	SDC	38.0
Emergency Cooling System	ECS	39.0
Core Spray System	CSS	40.0, 40.1, 81.0, 81.1
Liquid Poison System	LPS	42.1
Control Rod Drive System	CRDS	44.1, 44.2
Reactor Building Closed Loop Cooling System	RBCLC	70.0
Containment Spray System	CTN-SP	80.0, 93.0, 93.1

R-A **Risk-Informed Piping Examinations**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
R1.11	Elements Subject to Thermal Fatigue Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 1										
		01.0	F	10	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		32.0	F	15	4	0 / 0	0.0 %	0 / 0	0.0 %	4 / 0	100.0 %
		33.0	F	2	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		33.2	F	13	2	0 / 0	0.0 %	0 / 0	0.0 %	2 / 0	100.0 %
		38.0	F	9	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		39.0	F	32	4	0 / 0	0.0 %	1 / 0	25.0 %	3 / 0	100.0 %
		42.1	F	8	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		89	18	0 / 0	0.0 %	5 / 0	27.8 %	13 / 0	100.0 %
R1.14	Elements Subject to Crevice Corrosion Cracking Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 1										
		31.0	F	6	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		40.0	F	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	F	3	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		Item Total:		11	3	0 / 0	0.0 %	1 / 0	33.3 %	2 / 0	100.0 %
R1.16	Elements Subject to Intergranular Stress Corrosion Cracking(IGSCC) Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 1										
		32.0	F	15	15	0 / 0	0.0 %	6 / 0	40.0 %	9 / 0	100.0 %
		33.0	F	2	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		33.2	F	2	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	F	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.0	F	1	1	0 / 0	0.0 %	1 / 0	100.0 %	0 / 0	100.0 %
		38.0	F	17	2	0 / 0	0.0 %	1 / 0	50.0 %	1 / 0	100.0 %
		39.0	F	30	3	0 / 0	0.0 %	0 / 0	0.0 %	3 / 0	100.0 %
		40.0	F	74	11	10 / 0	90.9 %	1 / 0	100.0 %	0 / 0	100.0 %
		Item Total:		142	33	10 / 0	30.3 %	10 / 0	60.6 %	13 / 0	100.0 %
R1.18	Elements Subject to Flow Accelerated Corrosion(FAC) Code Case N-578-1, Unit 2, Code Case N-716, Unit 1 in accordance with FAC program										
		31.0	F	7	2	1 / 0	50.0 %	1 / 0	100.0 %	0 / 0	100.0 %
R1.20	Elements not Subject to a Damage Mechanism Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 1										
		01.0	F	56	5	2 / 0	40.0 %	2 / 0	80.0 %	1 / 0	100.0 %
		31.0	F	38	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		32.0	F	158	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.0	F	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.2	F	8	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		33.3	F	3	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		36.0	F	39	4	0 / 0	0.0 %	0 / 0	0.0 %	4 / 0	100.0 %
		37.0	F	32	6	1 / 0	16.7 %	3 / 0	66.7 %	2 / 0	100.0 %
		37.1	F	19	15	3 / 0	20.0 %	6 / 0	60.0 %	6 / 0	100.0 %
		39.0	F	14	1	1 / 0	100.0 %	0 / 0	100.0 %	0 / 0	100.0 %
		40.0	F	4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.1	F	16	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		42.1	F	10	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.1	F	16	2	0 / 0	0.0 %	2 / 0	100.0 %	0 / 0	100.0 %
		44.3	F	5	1	0 / 0	0.0 %	0 / 0	0.0 %	1 / 0	100.0 %
		Item Total:		421	35	8 / 0	22.9 %	13 / 0	60.0 %	14 / 0	100.0 %

R-A Risk-Informed Piping Examinations

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	Number of Components Scheduled / Completed					
						1st Period		2nd Period		3rd Period	
R2.16	Elements Subject to Intergranular Stress Corrosion Cracking(IGSCC) Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 2										
		39.0	F	31	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
R2.20	Elements Not Subject to a Damage Mechanism Code Case N-578-1, Unit 2, Code Case N-716, Unit 1, Class 2										
		39.0	F	44	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		44.2	F	60	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		80.0	F	306	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		81.0	F	182	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		81.1	F	60	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		93.0	F	8	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		93.1	F	49	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		709	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		1410	91	19 / 0	20.9 %	30 / 0	53.8 %	42 / 0	100.0 %

R-B **Risk-Informed Piping Examinations**

Item Number	Item Description	Zone	Tbl	No of Comp	No. Req.	1st Period		2nd Period		3rd Period	
R1.21	Socket-Welds requiring VT-2 examination each refueling outage VT-2 exam in conjunction with system pressure test, CC N-578-1, Unit 2; Code Case N-716 Unit 1										
		01.0	F	4	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		32.0	F	30	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		37.1	F	8	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		40.1	F	1	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Item Total:		43	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %
		Category Total:		43	0	0 / 0	0.0 %	0 / 0	0.0 %	0 / 0	0.0 %

Number of Components Scheduled / Completed

APPENDIX G - ASME SECTION XI CODE BOUNDARY DIAGRAMS

The NMP1 ASME Section XI Code Boundary Classification Diagrams identifying ASME Code Class 1, 2 and 3 system boundaries are provided in the following diagrams as listed below:

CODE BOUNDARY DIAGRAM LISTING	
Boundary Diagram Number	**System Title**
F-63002-C	Main Steam and HP Turbine
F-63003-C	Condensate Flow
F-63005-C	High Pressure FW Flow
F-63006-C	Drywell and Torus Isolation Valves
F-63007-C	Reactor Core Spray
F-63008-C	Spent Fuel Storage Pool Filter & Cooling
F-63009-C	Reactor Cleanup System
F-63011-C	Instrument Air
F-63012-C	Reactor Containment Spray
F-63013-C	Reactor Building Heat and Cooling
F-63014-C	Drywell & Torus Inert Gas & Cooling
F-63015-C	Reactor vessel Instrumentation
F-63016-C	Control Rod Drive
F-63017-C	Emergency Cooling System
F-63018-C	Reactor Shutdown Cooling
F-63019-C	Reactor Liquid Poison System
F-63020-C	Reactor Recirculation Loops
F-63021-C	Turbine Building Heating & Cooling
F-63022-C	Service Water, Closed Loop Cooling
F-63026-C	Diesel Generator Air, Water, Oil & Fuel
F-63027-C	Service Water

CODE BOUNDARY DIAGRAM LISTING	
Boundary Diagram Number	**System Title**
F-63035-C	Resin Transfer Regeneration
F-63036-C	Sealing Water
F-63041-C	Sampling
F-63045-C	Waste Disposal
F-63046-C	Air Conditioning
F-63047-C	Heating, Ventilating & Air Conditioning
F-63048-C	Condensate Transfer

APPENDIX H - RELIEF REQUESTS / ALTERNATIVES

H.0 RELIEF REQUESTS / ALTERNATIVES

During the First, Second and Third Ten-Year In-service Inspection Intervals, there were cases where component configuration and/or interference prevented the code required volume or surface area from being examined in it's entirety. In each case where such limitations were encountered, the details were documented on a Request for Relief or Alternate and submitted to the Nuclear Regulatory Commission as required by Title 10, Part 50, Section 55a of the Code of Federal Regulations for review and approval.

H.1 NMPNS has determined based on a detailed review that previous granted Requests for Relief or Alternates that certain limitations still exist and therefore, will require re-approval for the Fourth In-service Inspection Interval. Those requests for relief or Alternates on components which remain applicable for the Fourth In-service Inspection Interval have been provided within this Appendix H.

Appendix H includes a listing that provides the identification and current status of each Request for Relief or Alternate submitted to the USNRC, and are an integral part of the current Inspection Plan and Schedule.

Note: Examination volume or surface area that cannot be examined due to interference by another component or part geometry, a reduction in examination coverage on any weld will be considered acceptable provided the reduction in coverage for that weld is less than 10%. Subject of ASME Code Case N-460 Examination volume or surface area interference that does not meet the coverage requirements of Code Case N-460, will be documented in the form of a Request for Relief per 10 CFR 50.55a (g)(4)(iv).

In cases where parts of the required examination areas cannot by effectively examined because of a combination of component design or current inspection technique limitations, NMPNS will continue to evaluate the development of new or improved examination techniques with the intent of applying these techniques where a practical improvement on the examination can be achieved.

H.2 USNRC Staff Position on Impractically Based Relief Requests

As a result of the NMPNS Unit 2 Ten-Year Program Update submittal to the NRC staff, they expressed a position that Request for Relief for impractically-based requests be submitted following attempts being made to perform the subject examinations during the inspection interval, but no later than 12-month following the end of the 10-year inspection interval. At the request of NMPNS the NRC staff provided within the Safety Evaluation the following direction going forward.

The NRC staff concurs with the licensee that RRs based on impractically should be submitted for NRC review after attempts have been made to perform the subject examinations. Additionally, these requests are to be submitted to and approved by the NRC not later than 12-months after the end of the associated 10-year ISI interval pursuant to 10 CFR 50.55a(g)(5)(iv). Reference NRC SE dated December 1, 2008, (TAC NOS. MD7688 and MD7690)

SUMMARY STATUS OF CLASS 1, 2 AND 3 REQUESTS FOR RELIEF OR ALTERNATES

Relief Req. No.	System or Component	Exam Cat.	Exam Item No.	Summary of Request for Relief or Alternate	Proposed Alternatives	Relief Request Status^4
1ISI-001A	Reactor Pressure Vessel	B-A	B1.11 B1.12	Relief is requested for permanent relief from examining RPV Circ. Welds	NMPNS proposes to perform 2 to 3% of the circumferential welds that intersect long welds for the extended period of operation	Submitted 9/16/08
1ISI-002	Reactor Pressure Vessel	N/A	N/A	Relief is requested to use Roll-Expansion of Class 1 CRD Bottom Head Penetrations	Utilize ASME approved Code Case N-730	Submitted 8/29/08
1ISI-003	ASME Code Class 1 and 2 Piping Systems	B-F B-J C-F-1 C-F-2 GL-A	All	Relief is requested from Tables IWB-2500-1 and IWC-2500-1 of ASME Section XI 2001 Edition through 2003 Addenda	NMPNS proposes to implement the Risk-Informed / Safety-based In-service Inspection Program	

<u>Standard Note Legend</u>

Note 1

Nine Mile Point Nuclear Station, Unit 1
Fourth Inservice Inspection Interval
10 CFR 50.55a Request Number 1ISI-001A

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

A. COMPONENT IDENTIFICATION

System: Reactor Pressure Vessel

Class: Quality Group A, ASME Code Class 1

Component Description: Volumetric Examination of all Pressure Retaining Reactor Pressure Vessel Shell Circumferential Welds

Components Affected:

Circumferential Welds	Description	Code Category	Code Item Number
RVWD-100	Circumferential Shell Weld	B-A	B1.11
RVWD-101	Circumferential Shell Weld	B-A	B1.11
RVWD-137	Circumferential Shell Weld	B-A	B1.11
RVWD-138	Bottom Head to Shell Weld	B-A	B1.11

B. APPLICABLE CODE REQUIREMENTS

The applicable ASME Code, Section XI, for the Nine Mile Point Nuclear Station (NMPNS), Unit 1 (NMP1), Fourth 10-Year Interval, In-service Inspection Program is the 2001 Edition through 2003 Addenda. The fourth 10-year interval will begin on August 23, 2009, concurrent with the NMP1 license renewal period of extended operation.

In accordance with the provisions of 10 CFR 50.55a, Codes and Standards, paragraph 10 CFR 50.55a(a)(3)(i), Nine Mile Point Nuclear Station, LLC, (NMPNS) requests permanent relief for the NMP1 license renewal period of extended operation, from the requirements of ASME Code, Section XI, Sub article IWB-2500, Table IWB-2500-1, Volumetric Examination of Examination Category B-A, Pressure Retaining Welds in Reactor Vessel, Examination Item Number B1.11, Circumferential Shell Welds. See Figure 1 for weld locations.

C. REASON FOR REQUEST FOR RELIEF

The technical basis providing justification for the permanent elimination of the examination requirement of the RPV shell circumference welds is contained in BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations", (Reference 1). In the report, the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) concluded that the probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. The NRC staff conducted an independent risk-informed, probabilistic fracture mechanics assessment (PFMA) of the analysis contained in BWRVIP-05 (Reference 1), and the results are documented in the NRC's final safety evaluation of the BWRVIP-05 report (Reference 2). This assessment concluded that the probability of failure of the BWR RPV circumferential welds is orders of magnitude lower than that of the axial shell welds and the added risk caused by not inspecting the circumferential welds is negligible. Additionally, the NRC assessment demonstrated that inspection of BWR RPV circumferential welds does not measurably affect the probability of failure. Therefore, NMPNS has determined that the proposed alternative described below provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS

Proposed Alternative

In accordance with 10 CFR 50.55a(a)(3)(i), and consistent with information contained in NRC Generic Letter 98-05, (Reference 4) and in the NRC safety evaluation for BWRVIP-74-A (Reference 10) NMPNS will implement the following alternate provisions for the subject weld examinations.

The failure frequency for ASME Code Section XI, Table IWB-2500-1 Examination Category, B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," is sufficiently low to justify their elimination from the in-service inspection (ISI) requirement of 10 CFR 50.55a (g) based on the NRC Safety Evaluation. (Reference 2)

The ISI examination requirements of the ASME Code Section XI, Table IWB-2500-1 Examination Category B-A, Item No. B1.12, "Reactor Pressure Vessel Shell Longitudinal Welds," shall be performed, to the extent possible, and shall include inspection of the RPV Shell Circumferential Welds only at the intersection of these welds with the longitudinal welds, or approximately 2 to 3 percent of the RPV shell circumferential welds. The proposed alternative for volumetric examination of the RPV shell welds includes performing an examination, from the external outside diameter (OD) surface or where access is practical from the internal inside diameter (ID) surface of the RPV to the maximum extent possible. The examination of the remaining accessible portions of the RPV circumferential shell welds will be permanently deferred for the life of the original license and the license renewal period of extended operation.

The procedures for these examinations shall be qualified such that flaws relevant to the RPV integrity can be reliably detected and sized, and the personnel implementing these procedures shall be qualified in the use of these procedures. Qualification and examination will be completed in accordance with the 2001 Edition through 2003 Addenda of ASME Section XI, Appendix VIII as modified by the Performance Demonstration Initiative (PDI) and 10 CFR 50.55(a), "Codes and Standards."

Basis for Relief

The technical basis providing justification for the permanent elimination of the examination requirement of the RPV shell circumference welds is contained in a report (BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" - Reference 1), that was transmitted to the NRC in September 1995 and supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13 and December 18, 1997, and January 13, 1998. The NRC staff conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05 as documented in the final safety evaluation of the BWRVIP-05 report (Reference 2) and the supplement to final safety evaluation (Reference 3). This assessment concluded that the probability of failure of the BWR RPV circumferential welds is orders of magnitude lower than that of the axial shell welds and the added risk caused by not inspecting the circumferential welds is negligible. Additionally, the NRC assessment demonstrated that inspection of BWR RPV circumferential welds does not measurably affect the probability of failure.

The NRC issued Generic Letter 98-05, (Reference 4), permitting BWR licensees to request permanent relief from the in-service inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV shell circumferential welds, ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11. The NRC stated in that BWR licensees may request permanent relief for the remaining current license period by demonstrating that:

(1) At the expiration of their license the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, safety evaluation (Criterion 1), and

(2) Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, safety evaluation (Criterion 2).

For the original operating license period, the NRC authorized the alternative allowing permanent relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV circumferential welds by letter dated April 7, 1999 (Reference 13).

This request also demonstrates that the safety criteria specified in BWRVIP-74-A (Reference 9) and the associated October 18, 2001 safety evaluation (Reference 10) will continue to be met for the license renewal period of extended operation.

BWRVIP-74-A (Reference 9) provides generic guidelines intended to present the appropriate inspection and flaw evaluation recommendations to assure safety function integrity of the RPV components during both the current operating term and the license renewal term. The NRC staff's review of BWRVIP-74 was provided by safety evaluation (SE) dated October 18, 2001 (Reference 10), which concluded that Appendix E of the July 28, 1998 SE for BWRVIP-05 conservatively evaluated BWR RPVs to 64 effective full power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of an additional 20-year license renewal period. Therefore, the staff's analysis provided a technical basis for relief from the current ISI requirements of the ASME Code Section XI for volumetric examination of the circumferential welds as they may apply for the license renewal period. The October 18, 2001 SE further stated that to obtain relief, each licensee will have to demonstrate that:

(1) At the end of the renewal period, the circumferential welds will satisfy the limiting conditional failure probabilities for circumferential welds in Appendix E of the NRC staff's July 28, 1998 SE for BWRVIP-05, and

(2) They have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998 SE for BWRVIP-05.

Criterion (1) – Conditional Failure Probability

Demonstrate that at the expiration of the license (initial and renewed), the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the July 28, 1998 Safety Evaluation.

Response

In order to demonstrate that the circumferential welds satisfy the July 28, 1998 NRC safety evaluation limiting conditional failure probabilities, a comparison of the chemistry values and the predicted fluence at the end of the original license period can be made. Note that the NMP1 current license period is equivalent to 28 EFPY. However, for the purpose of the inspection relief for the initial 40-year license, NMP1 used values for 32 EFPY to compare against the NRC 32 EFPY values presented in the SER. In addition, failure probabilities are also calculated for NMP1 at 46 EFPY, which corresponds to the end of the license renewal period of extended operation. For the license renewal period, it is more appropriate to compare the change in failure probabilities since the NRC analysis did not consider the effects of the license renewal period (added fluence and crack growth). In this evaluation, the change in risk is the governing factor determined by the difference between the probability of failure at the end of the license renewal period (46 EFPY) and the end of the original license period (28 EFPY). The NMP1 request for relief for the original license period is given in Reference 11, and the NRC's authorization is documented in Reference 13.

For the original license period, Table 1 illustrates that NMP1 has conservatism in comparison to NRC Final Evaluation of BWRVIP-05 Limiting Plant Specific Analysis (comparing 32 EFPY values). The chemistry factor, adjustment for reference temperature (Δ RT_{NDT}), and mean RT_{NDT}, are calculated consistent with the guidelines of NRC Regulatory Guide 1.99, Rev. 2 (Reference 16). The data presented for NMP1 in the BWRVIP response to the NRC Request for Additional Information (RAI) on BWRVIP-05 is also shown in Table 1. The maximum Cu% and Ni% variability from the most current data available is also bounded. The fluence values in Table 1 for 28 EFPY and 46 EFPY (from Reference 15) bound the highest fluence beltline circumferential weld, and were calculated using methods that are in accordance with Regulatory Guide 1.190 (Reference 17) and have been previously reviewed and approved by the NRC (References 5 and 6).

Table 1: Comparison of Input Parameters for NRC Staff Assessment and BWRVIP Methodology

Parameter Description	Nine Mile Point 1 (Circumferential Weld)			NRC Staff Assessment for 32 EFPY (Ref. 2) (Circumferential Weld)		Nine Mile Point 1 (Axial Weld)	
	Using BWRVIP Methodology			Safety Evaluation "VIP"	Safety Evaluation "CEOG"	Using BWRVIP Methodology	
	28 EFPY**	32 EFPY*	46 EFPY**	32 EFPY	32 EFPY	28 EFPY**	46 EFPY**
Fluence, n/cm^2	1.16 x10^{18}	2.21 x10^{18}	1.67 x10^{18}	2 x10^{18}	2 x10^{18}	1.65 x10^{18}	2.49 x10^{18}
Initial RT$_{NDT}$ °F	-50	-50	-50	0	0	-50	-50
Chemistry Factor	99.9	112	99.9	151.7	172.2	97.6	97.6
Cu%	0.214	0.22	0.214	0.13	0.183	0.214	0.214
Ni%	0.076	0.20	0.076	0.71	0.704	0.046	0.046
Δ RT$_{NDT}$ °F	44.7	66.5	52.7	86.4	98.1	51.2	60.8
Mean ART °F	-5.3	16.5	2.7	86.4	98.1	1.2	10.8

* From Reference 11. Note that the weld heat number 1248 chemistry used for the 32 EFPY calculations was revised for the 28 EFPY and 46 EFPY calculations based on the resolution of the NMP1 surveillance capsule weld identity as discussed in Reference 14.

** From Reference 7, 14, and 15.

As shown in Table 1, the impact of irradiation results in a lower plant specific mean RT$_{NDT}$ for the NMP1 circumferential weld material as compared to that for any of the NRC's plant-specific analyses which were performed for the Combustion Engineering (CE) fabricated RPVs with the highest adjusted reference temperatures. Comparison of the NMP1 specific data and the data used in the NRC Final Safety Evaluation indicates that the combined effects of the Ni% and Cu% on the Chemistry Factor, which is by itself bounded by the NRC Independent Assessment, and the initial RT$_{NDT}$. Therefore, the limiting plant-specific conditional probability of failure P(FIE), determined by the NRC, bounds the NMP1 case through the projected end of the original license period.

Thus, for the original license period, the BWRVIP specific results relative to NMP1 as presented in BWRVIP-05 and subsequent RAI responses are consistent with those in the NRC Independent Assessment. Both analyses conclude that the failure probability associated with the circumferential welds is extremely small, and is orders of magnitude less than that for axial welds. Therefore, the NMP1 circumferential weld satisfies, at the end of the original license period, the limiting conditional failure probability for circumferential welds stated in the NRC's July 28, 1998, Safety Evaluation. Note that the discussion above is applicable for the original license period.

For the license renewal period, an NMP1 specific probabilistic fracture mechanics (PFM) evaluation was performed with the VIPER Program (Reference 8) using the data under the column "Using BWRVIP Methodology" in Table 1 for 28 EFPY and 46 EFPY (end of license renewal period). This evaluation was performed using the VIPER probabilistic fracture mechanics program that was developed as part of the BWRVIP-05 (Reference 1) effort. The same low temperature over pressure (LTOP) event parameters (Temperature = 88°F, Pressure = 1150 psi) used in the BWRVIP-05 effort were used in this NMP1 specific calculation. Using the BWRVIP methodology the conditional probability of failure for the NMP1 circumferential weld was found to be less than $1x10^{-7}$ for 28 EFPY and 46 EFPY (no failures predicted in 10^{7} trials for both 28 EFPY and 46 EFPY). The BWRVIP frequency of over-pressurization was determined to be $1x10^{-3}$/yr. This gives a total probability of failure for NMP1 of less than $2.5x10^{-12}$/yr for the circumferential welds for 28 EFPY (40 years) and 46 EFPY (60 years) of operation. Note that the failure probabilities are reported to be the same for 28 EFPY and 46 EFPY since there were no failures in 10^{7} Monte Carlo trial simulations. The 46 EFPY includes higher fluence and considers crack growth for 18 EFPY beyond the original license period (28 EFPY).

For NMP1 axial welds with the data shown in Table 1 under the column "Using BWRVIP Methodology," the total probability is $<2.5x10^{-11}$/yr for both 28 EFPY and 46 EFPY, as no failures were predicted for either the original license period or the license renewal period.

The fact that no failures occurred through the initial license period and license renewal period shows that the reliability of the NMP1 RPV is extremely high. Most importantly, the results show that the increase in failure probability due to the license renewal period is essentially negligible. In addition, the reliability of the circumferential welds is likely much higher than calculated because of lower stress (axial stress is one half the hoop stress) and lower chemistry factors.

These calculations have been performed conservatively assuming a constant fluence at all weld locations equal to the peak fluence. For example, the maximum fluence anywhere in the beltline circumferential weld is assumed to exist throughout the circumferential weld. The peak fluence at any axial weld is assumed to exist at all axial weld locations. In reality, the fluence varies both circumferentially and axially. If analysis were performed considering these fluence variations, the resulting probability of failures would be lower than calculated using the peak fluence at all weld locations.

Thus, the BWRVIP-05 NMP1 specific results as determined using the BWRVIP-05 methodology and subsequent BWRVIP responses to NRC RAIs are consistent with those in the NRC Independent Assessment. Both analyses conclude that the failure probability associated with circumferential welds is extremely small. In addition, due to the NMP1 specific conditions, the failure probability for the axial welds is also extremely small. Most importantly, the increase in failure probability due to operation during the license renewal period is extremely small since no failures were predicted even through the license renewal period. Thus, it is concluded that the NMP1 circumferential weld satisfies, at the end of the license renewal period, the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998 safety evaluation.

Criterion (2) – Limiting the Frequency of Cold Over-pressure Events

Demonstrate licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, safety evaluation.

Response

The NRC indicated that the potential for, and consequences of, non-design basis events not discussed in the BWRVIP-05 report should be addressed. In particular, the NRC stated that non-design basis, low temperature over-pressure transients (LTOP) transients, should be considered. The NRC further went on to describe several types of events that could be precursors to an LTOP event. The BWRVIP provided a response to this issue concluding that Condensate and Control Rod Drive (CRD) pumps could cause such a condition leading to an LTOP event. This was summarized in the NRC Safety Evaluation for BWRVIP-05 (Reference 2).

NMP1 has in place procedures which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown minimizing the likelihood of an RPV LTOP event. Additionally, these procedures are reinforced through the NMP1 reactor operator training program. The procedural controls and training provisions that will be used for the license renewal period of extended operation will be the same as those used for the original operating license period for NMP1 (see Reference 11 and 13).

The RPV leakage pressure test procedures used at NMP1 have sufficient procedural guidance to prevent LTOP events. The leakage test is performed at the conclusion of each refueling outage. These pressure tests are infrequently-performed, complex tasks, and the test procedures are controlled as Special Plant Evolutions. As such, a requirement is included in the procedures for an extensive pre-job briefing to be conducted with all essential personnel including Operations management. The briefing details the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications and finally the process in which the test would be aborted if plant systems responded in an adverse manner. Vessel pressure and temperature are required to be monitored throughout the tests to ensure compliance with the plant Technical Specification pressure-temperature curve. Also, the procedures require the designation of a "Principal Test Engineer" for the duration of the test who is a single point of accountability, responsible for the coordination of testing from initiation to closure, and maintaining operations and plant management cognizant of the test status.

With regard to inadvertent system injection resulting in an LTOP condition, the NMP1 high pressure make-up system, (i.e., the High Pressure Coolant Injection (HPCI)) and the normal Feedwater system are interconnected. The HPCI system is a mode of operation of the Condensate and Feedwater systems rather than an independent, stand alone system. The HPCI system utilizes two condensate pumps, two feedwater booster pumps, two motor-driven feedwater pumps, and an integrated control system. As such, the HPCI system contains only instrumentation and control components as its own dedicated equipment. HPCI initiation is prompted by the Reactor Protection System under the following conditions: (1) a turbine trip, or (2) low reactor water level. During shutdown of the unit, the associated booster and feedwater pumps in the system are secured in accordance with operating procedures. Equipment malfunction or inappropriate operational action would be necessary to cause inadvertent system operation.

During normal cold shutdown conditions, with the RPV head installed, RPV level and pressure are controlled with the CRD System, Condensate Feedwater System, and Reactor Water Cleanup (RWCU) systems using a “feed and bleed” process. The RPV is not taken solid during these times, and plant procedures require opening of the head vent valves after the reactor has been depressurized to approximately 15 psig.

The Liquid Poison System is another high pressure water source to the RPV; however, there are no means of automatic system activation. System injection requires an operator to manually reposition a key-locked control switch to start the system from the Control Room. The system may also be operated from a remote local test station. The only injection path to the RPV is through two explosive actuated injection valves that are interlocked with the key-locked switch in the Control Room. The injection rate for each pump is approximately 30 gpm, which would give the operator sufficient time to control reactor pressure. Local testing of the pumps uses demineralized water from a test tank and is a closed test loop.

Procedural controls are in place to respond to an unexplained rise in reactor pressure which could result from a spurious activation of an injection source. Actions specified include determination and isolation of the pressure source, verification of reactor head vents and/or MSIVs open and, as necessary, relieving reactor pressure using available plant equipment (e.g., electromagnetic relief valves, reactor water cleanup system and reactor bottom drain).

During normal cold shutdown conditions, reactor water level and temperature are maintained within established ranges in accordance with operating procedures. Procedures governing the conduct of operations require that the Control Room operators frequently monitor for indications and alarms to detect problems and abnormalities as early as possible. An Operations procedure also requires that the control room supervisor be notified immediately of any change or abnormality in plant indications and controls. Furthermore, reactor water level and temperature operating bands and changes thereto are established under the direction of the Shift Manager. Therefore, any deviations in reactor water level or temperature from a specified band will be identified and corrected. Finally, plant conditions and on-going activities are discussed during each shift turnover. This ensures that on-coming operators are cognizant of activities that could adversely affect reactor level, pressure, or temperature.

Plant specific procedures have been developed to provide operator guidance regarding compliance with the plant Technical Specifications and RPV pressure-temperature curve limits. Additionally, operators receive training on RPV brittle fracture and the relationship of these pressure-temperature curve limits.

During plant outages, NMP1 work control processes ensure that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. Work is coordinated through the Work Execution Center which provides an additional level of Operations oversight. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor safety during refueling outages. Expected plant responses and contingency actions to address unexpected conditions that may be encountered are required to be evaluated as stated in the administrative controls for risk management and management of outages.

As discussed above, NMP1 has implemented procedural controls and training to minimize the probability of an LTOP event. Accordingly, the above information and the supporting technical documentation contained in the BWRVIP-05 report and NRC Safety Evaluation provide a basis for excluding RPV circumferential welds from the augmented examination requirements of 10 CFR 50.55a(g) and ASME Section XI.

Summary

In summary, the NMP1 specific chemistry, and adjusted reference temperature (ART) were compared against the NRC's July 28, 1998, safety evaluation values for 32 EFPY. The NMP1 values were found to be bounded demonstrating that the NRC SE conclusions regarding failure probability have been satisfied. An NMP1 specific probabilistic fracture mechanics evaluation was performed to determine the probability of failure when subjected to an LTOP event during the license renewal period. In addition, it was confirmed that NMP1 has taken steps to reduce the potential for LTOP events through procedural controls and personnel training. An evaluation to identify the sources for increased pressure was also performed and found that the probability of a cold overpressure transient is considered to be less than or equal to that used in the NRC evaluation.

In effect the criterion in RG 1.174 regarding defense-in-depth, and safety margins are maintained and USNRC safety goals are not exceeded.

NMPNS has concluded that permanent deferral of the examination of the RPV circumferential shell welds for the license renewal period of extended operation and the reduced examination coverage of the circumferential welds is justified and presents an acceptable level of quality and safety to satisfy the requirements in accordance with 10 CFR 50.55a (a)(3)(i).

E. IMPLEMENTATION SCHEDULE

Pursuant to 10 CFR 50.55a(a)(3)(i), NMPNS requests permanent relief for the license renewal period of extended operation. NMPNS has demonstrated that the criteria specified in GL 98-05 (Reference 4) are met for the original operating license period, (NRC SE, dated 4/7/99, Reference 13) , and that the criteria of BWRVIP-74-A (Reference 9) are met for the license renewal period of extended operation. Therefore, the requested duration of the proposed alternative is justified.

F. PRECEDENTS

- Nine Mile Point Nuclear Station, Unit 2, NRC letter dated November 5, 2007 (TAC No. MD3696)
- Dresden Nuclear Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2; NRC letter dated March 23, 2005 (TAC Nos. MC2190, MC2191, MC2192, and MC2193)

G. ATTACHMENTS

None

H. REFERENCES

1. Electric Power Research Institute (EPRI) Proprietary Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation, BWRVIP -05," dated September 1995.

2. Letter, G. C. Lainas (NRC) to Carl Terry, BWRVIP Chairman, NRC Report "Final Safety Evaluation of the BWR Vessel Internals Project BWRVIP-05 Report," (TAC No. MA93925), Division of Engineering Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, dated July 28, 1998.

3. Letter, J. R. Strosnider (NRC) to Carl Terry, BWRVIP Chairman, NRC Report "Supplement to Final Safety Evaluation of BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. MA3395), Division of Engineering, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, dated March 7, 2000.

4. United States Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

5. Letter, P. S. Tam (NRC) to P. E. Katz (NMPNS), "Nine Mile Point Nuclear Station, Unit No. 1, Issuance of Amendment Re: Pressure-Temperature Limit Curves" (TAC Nos. MB6687), dated October 27, 2003.

6. Letter, P. S. Tam (NRC) to P. E. Katz (NMPNS), "Nine Mile Point Nuclear Station, Unit No. 2, "Issuance of Amendment Re: Pressure-Temperature Limit Curves" (TAC No. MC0331), dated January 27, 2004.

7. Structural Integrity Associates Calculation 0800297.300.RA, Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts.

8. Viper Computer Code, Version 1.2, Structural Integrity Associates, January 1998.

9. BWR Vessel Internals Project, BWRVIP-74-A, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal, June 2003.

10, Letter, C. I. Grimes (NRC) to C. Terry, BWRVIP Chairman, "Acceptance for Referencing of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)" and Appendix A, "Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," dated October 18, 2001.

11. Letter, R. B. Abbott (NMPC) to Document Control Desk (NRC), Nine Mile Point Unit 1, Proposed Alternatives for Examination of Reactor Pressure Vessel Shell Welds, dated December 10, 1998.

12. Structural Integrity Associates Report No. SIR-06-512, "Technical Justification for Elimination of Nine Mile Point Unit 1 Reactor Pressure Vessel Circumferential Weld Inspections for the License Renewal Term," dated August 21, 2008, Rev. 1.

13. Letter, S. S. Bajwa (NRC) to J. H. Mueller (NMPC), Alternatives for Examination of Reactor Pressure Vessel Shell Welds, Nine Mile Point Nuclear Station, Unit 1 (TAC No. MA4383), dated April 7, 1999.

14. Letter, R. B. Abbott (NMPC) to Document Control Desk (NRC), Request for Additional Information Regarding Reactor Pressure Vessel Structural Integrity at Nine Mile Point Nuclear Station Unit 1 (TAC No. MA1200), dated September 4, 1998.

15. Report Number NMP-405778, Neutron Transport Analysis for Nine Mile Point Unit 1, MPM Technologies, May 2006.

16. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Rev. 2, May 1988.

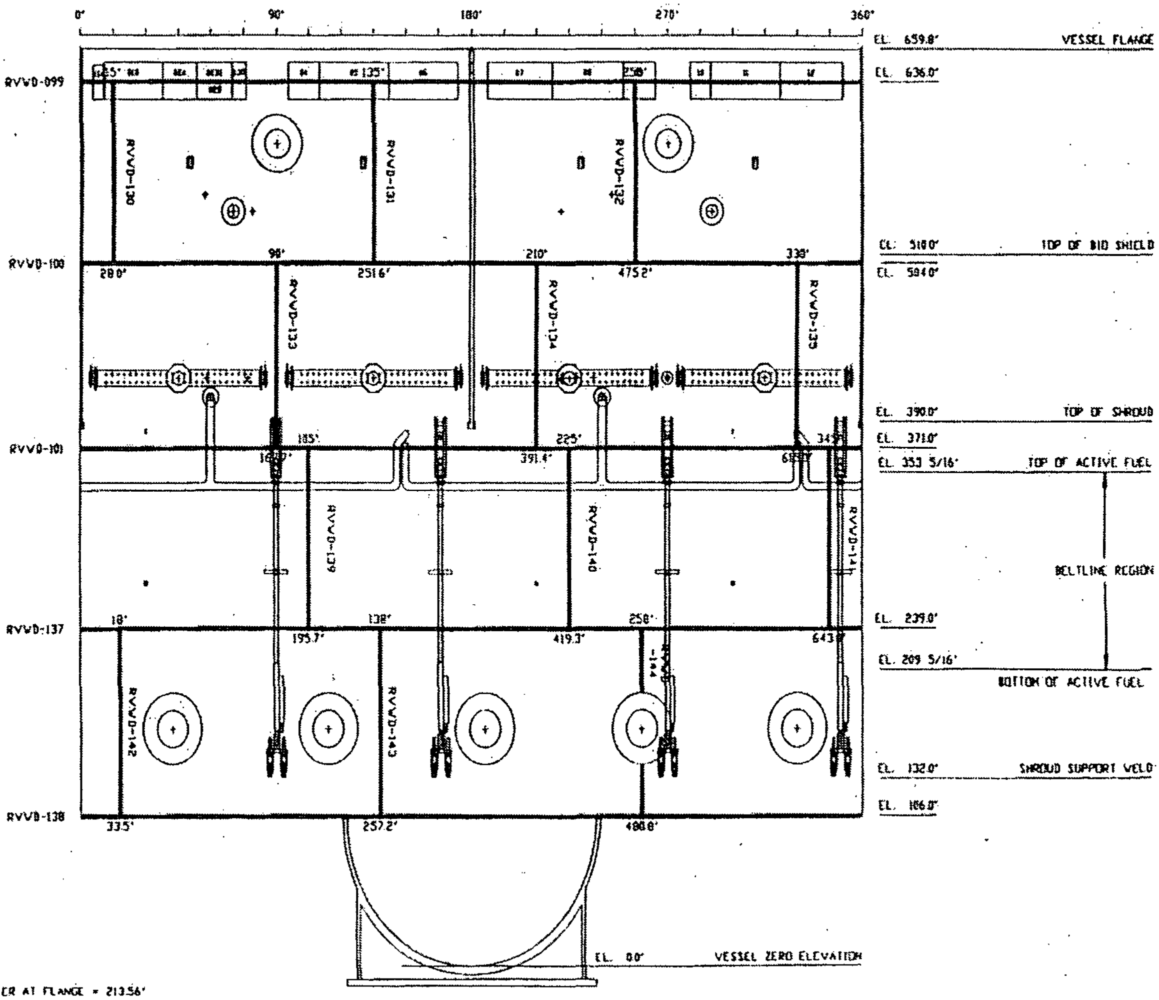
17. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

Nine Mile Point Nuclear Station, Unit 1
Fourth Inservice Inspection Interval
10 CFR 50.55a Request Number 1ISI-001A

REACTOR VESSEL SHELL WELD LOCATIONS

Figure 1

NINE MILE POINT - UNIT 1



VESSEL DIAMETER AT FLANGE = 213.56"

AZIMUTHS = 1.8637 INCHES PER DEGREE

GE NUCLEAR ENERGY	NINE MILE POINT - UNIT 1	VESSEL ID ROLLOUT	SCALE: NONE	DWG. NM1-0001	REV. 1

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii)

A. COMPONENT IDENTIFICATION:

System:	Control Rod Drive (CRD) Bottom Head Penetrations
Code Class:	Quality Group A, ASME Code Class 1
Description:	Use of Variation of Code Case N-730, "Roll-Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWR's, Section XI, Division 1"
Components Affected:	Reactor Vessel Control Rod Drive (CRD) Bottom Head Penetrations

B. APPLICABLE CODE REQUIREMENTS:

The applicable ASME Code, Section XI, for the Nine Mile Point Unit 1 (NMP1) fourth 10-year in-service inspection interval as currently referenced in 10CFR50.55a is the 2001 Edition through the 2003 Addenda. The fourth 10 year interval will begin on August 23, 2009, concurrent with the NMP1 license renewal period of extended operation.

ASME Code Case N-730, "Roll-Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs, Section XI, Division 1," is applicable from the 1989 Edition through the 2004 Edition with the 2006 Addenda of ASME Section XI. All references provided in ASME Code Case N-730 to the ASME Code apply to 2004 Edition. (Reference 1)

ASME Section XI, 2001 Edition through 2003 Addenda, IWA-4000, "Repair/Replacement Activities," requires that all repair and replacement be performed in accordance with the provisions of IWA-4000. Additionally, IWB-3142, "Acceptance," provides acceptance criteria for components, which includes removal of the relevant condition.

C. REASON FOR REQUEST FOR RELIEF:

The NRC staff informed Nine Mile Point Nuclear Station, LLC (NMPNS) during the License Renewal Application process that the staff did not consider the currently approved plant specific roll repair process to be acceptable for permanent repair (Reference 2). As a result, NMPNS has committed to implement ASME Code Case N-730 during the license renewal period commencing August 23, 2009. Although ASME has approved Code Case N-730 and NRC ASME participants' comments were incorporated into the final version of the Code Case, the Code Case has not yet been approved for general use by the NRC and added to Regulatory Guide (RG) 1.147 (Reference 3). Hence, NMPNS is requesting relief to use Code Case N-730 at NMP1.

Pursuant to 10 CFR 50.55a(a)(3)(i), NMPNS requests to use ASME Code Case N-730, as an alternative to the requirements of ASME Section XI, applicable edition and addenda, IWA-4000 ("Repair/Replacement Activities") and IWB-3142 ("Acceptance") for the repair of Control Rod Drive Bottom Head Penetrations utilizing the mechanical roll expansion technique to eliminate leakage from Class 1 CRD bottom head penetrations. NMPNS has determined that the proposed alternative described below provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

Additionally, NMPNS requests approval to vary from paragraph 6.6 of Code Case N-730, which requires a VT-2 visual examination in conjunction with a system leakage test in accordance with IWB-5000 for CRD housings subjected to roll expansion following completion of the post-roll-expansion UT examination. The justification below demonstrates that compliance with the specified code case requirement would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety per the requirements of 10 CFR 50.55a(a)(3)(ii).

D. BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS:

1A. Proposed Alternative:

In accordance with 10 CFR 50.55a(a)(3)(i), NMPNS requests the use of ASME Approved Code Case N-730, "Roll-Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs, Section XI, Division 1," for repair of CRD housing penetrations. Currently, NMPNS has roll expanded thirty-three (33) CRD housings as approved by the NRC as an alternate to ASME Section XI pursuant to 10 CFR 50.55a(a)(3) in a safety evaluation (Reference 4). NMPNS is requesting approval of the code case as an alternative permanent repair for any additional CRD housing penetrations that may exhibit leakage during the License Renewal Period commencing August 23, 2009 or for the re-roll expansion of any previously roll expanded CRD housing that exhibits a repeat occurrence of leakage.

Additionally, NMPNS will perform future volumetric (UT) and system leakage tests on the thirty-three existing roll-expansion repairs in accordance with requirements of ASME Code Case N-730 to provide added assurance of CRD housing structural integrity and a zero leakage acceptance criteria, respectively.

1B. Basis for Relief:

The technical basis for the Code Case is provided in the Code Case N-730 Technical Basis Report, XGEN-2005-10, Revision 3, "Technical Basis for ASME Code Case N-730 Roll-Expansion of Class 1 Control Rod Drive (CRD) Bottom Head Penetrations in BWRs" (Reference 1).

The CRD penetrations in a BWR are located on the bottom head of the vessel. The NMP1 BWR CRD penetrations used a stub tube to which the CRD housing is welded. The CRD housing is approximately 6 inches in diameter and is made of Type 304 stainless steel. The CRD housing encloses the control rod drives. There are 129 CRD penetrations. The use of the stub tube allows the stainless steel housing to be welded to the stub tube after post-weld heat treatment (PWHT) of the vessel. Like the CRD housing, the stub tube was also made of Type 304 stainless and was welded to the bottom head prior to PWHT. The subsequent vessel PWHT caused furnace sensitization of the stub tube making it susceptible to Intergranular stress corrosion cracking (IGSCC) with exposure to the high temperature BWR water environment. The stub tubes are attached to the vessel bottom head by inconel 182 welds which are also susceptible to IGSCC. Cracking in either the stub tube or the inconel 182 welds would typically be identified by lower plenum in-vessel visual inspections or by leakage identified on the CRD housing during the under vessel leakage inspections. A stub tube is depicted in Figure 1.

NMP1 has CRD penetrations that have leaked due to stress corrosion cracking in the furnace sensitized stainless steel stub tubes and/or potentially in the stub tube-to-vessel attachment welds. Code Case N-730 applies to these locations. A roll-expansion repair method to stop the leakage has been implemented at NMP1 since 1984 (Reference 4).

Leakage from the CRD penetration cracking prior to roll repair expansion has been minimal with the maximum measured leak rate being approximately 80 drops/min. The observed leakage prior to roll expansion has been a small fraction of system make-up capability. Following roll repair, a zero leakage condition was observed in all cases. To date 33 CRD penetrations have been roll-expanded to a nominal 4% wall thinning. Of the 33 roll-expanded CRD penetrations, only one of the penetrations (50-19) has been re-roll expanded to 6% wall thinning due to repeat occurrence of leakage. Zero leakage has been observed at this penetration since it was last roll expanded in

2005. NMPNS implemented Hydrogen Water Chemistry and Noble Metals Chemical Addition (HWC/NMCA) in 2000. To date, only one new CRD penetration has exhibited leakage following commencement of HWC/NMCA. That was in early 2001 indicating the crack likely originated prior to HWC/NMCA.

The inspection results demonstrate that NMPNS can effectively and repeatedly roll-expand CRD bottom head penetrations to achieve a zero leakage condition and that HWC/NMCA has been an effective method to mitigate further cracking and leakage. As such, the NMPNS roll repair history along with the XGEN-2005-10 technical basis provide adequate basis for the proposed alternative to use ASME Code Case N-730 for future roll repairs, if required, at NMPNS.

2A. Proposed Alternative:

NMPNS proposes to follow Code Case N-730, paragraph 6.6 with regard to performing the VT-2 visual examination for leakage from previously roll expanded CRD housing penetrations in accordance with IWB-5000. In the future, if a new CRD housing penetration leak is detected during the leakage test required by IWB-5000, then during that outage the CRD housing would be roll expanded in accordance with Code Case N-730. However, as an alternative to the rated test pressure required by IWB-5000, NMPNS proposes to perform the post-roll expansion VT-2 examination at approximately 900 psig prior to returning the Unit to full power operation.

2B. Basis for Relief:

Historically, CRD stub tube leakage is detected at the end of a refueling outage during the performance of the non-nuclear heat-up ASME Class 1 System Leakage Test performed using an external pressure source prior to returning the Unit to service. Under the current roll repair program, NMPNS would be required to disassemble the CRD housing support grid (shoot-out steel), TIP tubing, remove the CRD mechanism and CRD thermal sleeve, perform the roll repair and post-roll UT prior to plant startup from the outage. This historically has extended a refueling outage by at least one day. After completion of the post-roll repair UT, NMPNS has performed the post-maintenance test (PMT) leakage examination of the roll repaired penetration at cold static head conditions and repeats the leakage examination during the final drywell entry during plant startup typically with Reactor Pressure Vessel (RPV) pressure at approximately 900 psig.

ASME Code Case N-730 step 6.6 requires the rolled penetration to be visually (VT-2) examined for leakage in conjunction with a system leakage test in accordance with ASME XI 2004 edition, IWB-5000. IWB-5000 requires that the system leakage test be conducted at a pressure not less than the pressure corresponding to 100% rated power. Assuming the leakage is first detected during the performance of the non-nuclear heat-up ASME Class 1 System Leakage Test at the end of a refueling outage, this code case requirement would mandate that after performance of the roll expansion, NMPNS either: 1) repeat the non-nuclear heat-up ASME Class 1 System Leakage Test using an external pressure source; or 2) attempt to perform the leakage exam at power at approximately 1030 psig.

In accordance with 10CFR50.55a(a)(3)(ii), NMPNS proposes to provide an acceptable alternative which is to perform the PMT VT-2 visual examination at approximately 900 psig during the final drywell entry prior to power ascension, which is consistent with NMP1's past 20-year practice. Justification for this position follows:

i) NMPNS has roll expanded 33 CRD penetrations to date. The Post- Maintenance Test (PMT) visual leakage examinations were performed during plant startup at reactor pressures between 500-900 psig. No evidence of leakage was identified during these examinations. During the non-nuclear heat-up ASME Class 1 System Leakage Test performed immediately following the outage in which the CRD penetration was roll expanded and prior to the unit's return to service, no evidence of leakage was detected. This confirms that if the non-nuclear heat-up ASME Class 1 System Leakage Test was repeated following each roll repair during the outage in which the leakage was detected and corrected, the leakage would have been zero. These inspection results demonstrate that NMPNS can effectively and repeatedly roll expand penetrations to achieve a zero leakage condition, such that repeating the non-nuclear heat-up ASME XI Class 1 leakage test in the same outage is not necessary.

ii) Inspections of CRD stub tubes for leakage during a system leakage test at the nominal operating pressure (~1030 psig) associated with 100% rated reactor power cannot be performed during a normal plant startup, due to the excessive temperature and radiological exposure conditions to which the NDE examiners performing the visual VT-2 examinations would be exposed in the primary containment. Furthermore, NMP1 Technical Specification 3.3.1 requires the primary containment atmosphere oxygen concentration to be reduced to less than four percent by volume within 24 hours of the reactor being placed in the run mode, thereby also potentially requiring the NDE examiners to wear breathing apparatus.

iii) The CRD penetrations are not isolable from the reactor vessel. The entire primary system will have to be pressurized in order to perform the non-nuclear heat-up Section XI system leakage test. During implementation of the non-nuclear heat-up ASME Class 1 System Leakage Test extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the primary system to establish the necessary test pressure during plant outage conditions, without the withdrawal of control rods. This is done in order to perform the necessary system leakage test without exposing the examiners to the excessive temperature and radiological exposure conditions described in the paragraph above. This special test usually takes one full day of plant outage time, and the additional valve lineups and system reconfigurations necessary to support this special test impose an additional challenge to the affected systems. A normal plant startup then occurs after completion and subsequent recovery from the test procedure at which time the visual VT-2 examination of the roll repaired penetration can be performed at approximately 900 psig. Therefore, repeating the non-nuclear heat-up ASME Class 1 System Leakage Test after a roll repair is not necessary.

iv) During a normal plant startup, operators are required to perform final drywell entry walk-down inspections of the primary containment at approximately 900 psig. The roll-expanded penetration can be visually inspected by a qualified visual VT-2 examiner during these final drywell entry walk-down inspections at approximately 900 psig. During the next refueling outage, the roll-expanded connections are re-examined for signs of leakage during the non-nuclear heat-up ASME XI in-service leak test at a pressure not less than the pressure corresponding to 100% rated power.

v) CRD penetration leakage rates are expected to be proportional to the square root of the test pressure ratio as noted in Table 1, Note 1, of Reference 4. Using this correlation, a 25 drop per minute leak detected at 1030 psig would be approximately a 23 drop per minute leak at 900 psig. Furthermore, the sensitivity of leakage detection is not expected to decrease due to temperature because the reactor temperature difference between 900 psig and rated pressure (1030 psig) is less than 20°F. Therefore, leakage identified at nominal operating conditions should be detectable at 900 psig.

vi) The roll repair process has been qualified in mockups and tested for leakage at pressures much greater than that required by IWB-5221(a) in the mock-up qualification. The roll expanded mockups were hydro-tested 1.8 times greater than the nominal operating pressure and no evidence of leakage was detected.

vii) This position is essentially consistent with the BWRVIP-146 position (Reference 5), section 4.3, that indicates repeating the non-nuclear heat-up ASME Class 1 System Leakage Test is not essential and instead performance of the leakage exam at a minimum of 90% normal operating pressure is sufficient to detect leakage.

E. IMPLEMENTATION SCHEDULE

Pursuant to 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii), NMPNS requests relief for the NMP1 License Renewal Period from August 23, 2009 to August 22, 2029. NMPNS has demonstrated that CRD Bottom Head Penetrations that have been repaired by roll-expansion can achieve a zero leakage condition. Therefore, the requested duration of the proposed alternative is justified.

F. PRECEDENT:

NRC Safety Evaluation, Oyster Creek Nuclear Generating Station – Relief Request, Application of Draft ASME Code Case N-730, TAC No. MD1070, October 6, 2006

G. ATTACHMENTS:

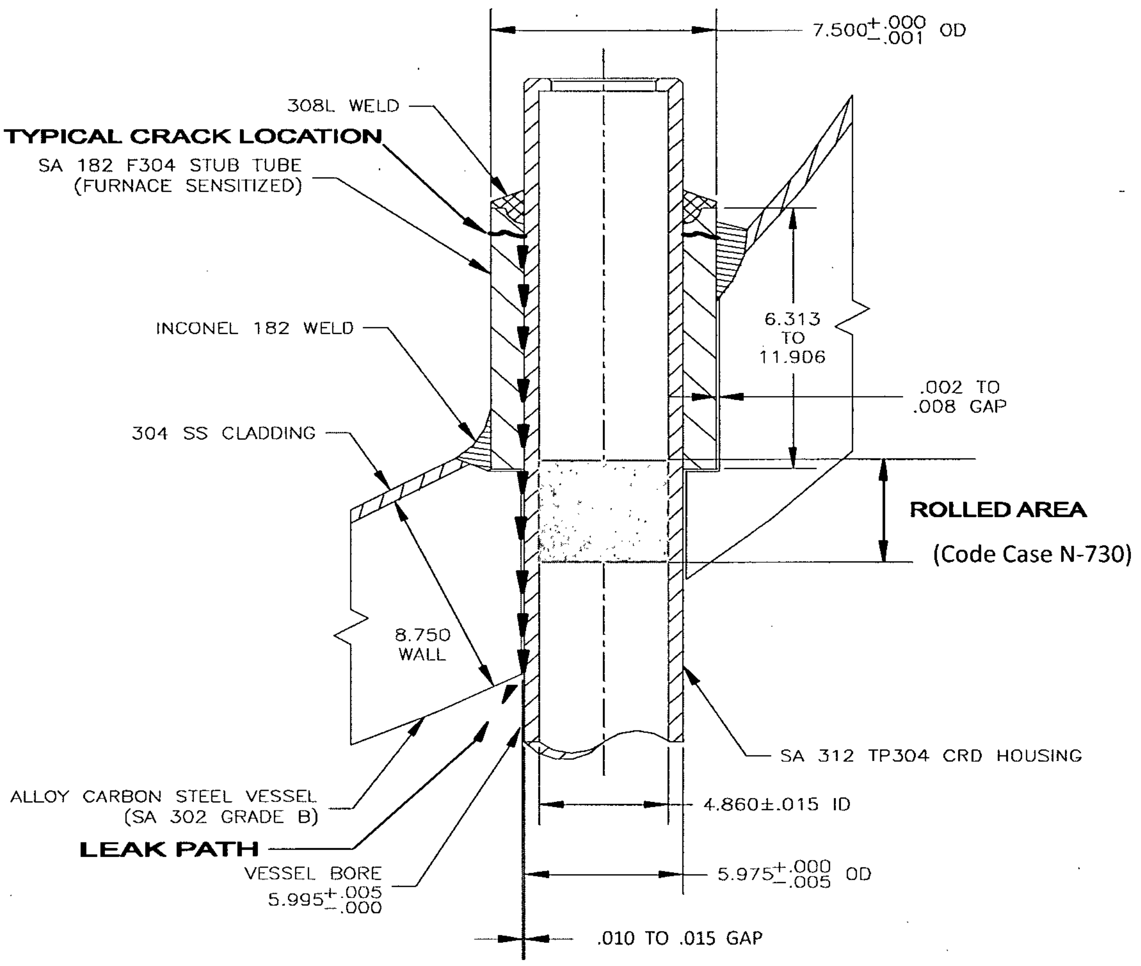
None.

H. REFERENCES:

1. Report XGEN-2005-10, Revision 3: "Technical Basis for ASME Code Case N-730 Roll-Expansion of Class 1 Control Rod Drive (CRD) Bottom Head Penetrations in BWRs," dated August 2006.
2. Letter from N. B. Le (NRC) to J. A. Spina (NMPNS), "Request for Additional Information for the Review of Nine Mile Point Nuclear Station, Units 1 and 2, Amended License Renewal Application (TAC Nos. MC3272 and MC3273)," dated November 2, 2005.
3. NRC Regulatory Guide 1.147, Revision 15, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated October 2007.
4. Letter from R. Auluck (NRC) to C. V. Mangan (NMPC), "Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) (TAC 61181)," dated March 25, 1987.
5. BWRVIP-146, "BWR Vessel and Internals Project Technical Basis for ASME Code Case N-730, Roll-Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs," dated November 2005.

Figure 1

Stub Tube (Typical)



Nine Mile Point Nuclear Station, Unit 1
Fourth In-service Inspection Interval
10 CFR 50.55a Request Number 1ISI-003

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

A. COMPONENT IDENTIFICATION

System:	Various Systems
Class:	Quality Groups A and B, (ASME Code Class 1 and 2)
Component Description:	Piping Circumferential Welds
Components Affected:	

B. APPLICABLE CODE REQUIREMENTS

Pursuant to 10 CFR 50.55a(g), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Section XI, 2001 Edition through 2003 Addenda, Examination Tables IWB-2500-1 and IWC-2500-1, Examination Categories B-F, B-J, C-F-1, C-F-2 must receive in-service inspection (ISI) during each successive 120-month (ten-year) interval.

In addition, Intergranular Stress Corrosion Cracking (IGSCC) Category A welds are examined over the 10-year interval in accordance with the NRC guidance on schedule, methods, personnel and sample expansion outlined in NRC Generic Letter (GL) 88-01.

The required examinations in each Examination Category shall be completed during each successive inspection interval in accordance with Inspection Program B, Tables IWB-2412-1 and IWC-2412-1 and GL 88-01 guidelines, as defined in Table 1 below.

Table 1
ASME Section XI and GL 88-01 Examination Requirements

ASME Code Class	Examination Category	Types of Welds	Examination Methods	Percentage Requirements
1	B-F	Dissimilar Metal Welds	Volumetric and Surface or Surface	100% Required
1	B-J	Piping Welds	Volumetric and Surface or Surface	25% Required
1	GL-A	Resistant Material	Volumetric	25% Required
2	C-F-1	Piping Welds	Volumetric and Surface or Surface	7.5% Required
2	C-F-2	Piping Welds	Volumetric and Surface or Surface	7.5% Required

C. REASON FOR REQUEST FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(i), Nine Mile Point Nuclear Station, LLC (NMPNS) requests NRC authorization of an alternative to the ASME B&PVC, 2001 Edition through 2003 Addenda of Section XI, Division 1, Tables IWB-2500-1 and IWC-2500-1, Examination Categories B-F, B-J, C-F-1 and C-F-2 requirements.

NMPNS also requests approval to use ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI, Division 1 for Risk-Informed / Safety-Based insights.

Consistent with EPRI-TR-112657, Revision B-A , IGSCC Category A welds are integrated into the proposed risk-informed, safety-based ISI program.

D. BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS

The basis for this request for alternative is to document the application of ASME Code Case N-716 to Class 1 and 2 piping systems at Nine Mile Point Nuclear Station Unit 1 using risk-informed and safety based (RIS_B) insights.

The objective of the ISI program is to identify service-induced degradation that might lead to pipe leaks and ruptures, thereby meeting, in part, the requirements set forth in 10 CFR 50.55a. ISI programs are intended to address all piping locations that are subject to degradation. Incorporating risk insights into ISI programs can focus examinations on the more important locations and reduce personnel exposure, while at the same time maintaining or improving the public health and safety.

Electric Power Research Institute (EPRI) Topical Report (TR) EPRI-TR-112657, Revision B-A (hereafter referred to as EPRI-TR) was submitted for NRC review by letter dated July 29, 1999. The NRC review documented in a Safety Evaluation dated October 28, 1999, concluded that the EPRI-TR was acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the EPRI-TR and the associated NRC safety evaluation.

In addition, the NRC staff concluded that the proposed risk-informed in-service inspection program (RI-ISI) as described in the EPRI-TR, 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.

As stated within the EPRI-TR, no changes to the augmented inspection programs for Flow Accelerated Corrosion (FAC) or Intergranular Stress Corrosion Cracking (IGSCC) Categories B through G welds are being made in the proposed RIS_B program. The proposed RIS_B program will supersede augmented inspection programs for IGSCC Category A welds.

In addition to development of the Proposed Risk-Informed ISI program utilizing the EPRI methodology, NMPNS will convert from implementing ASME Code Case N-578-1, Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1, to the implementation of ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI, Division 1, which was approved by ASME on April 19, 2006.

As a result of the above insights, more efficient and technically sound means for selecting and scheduling in-service examinations of piping can be achieved, which will provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

E. IMPLEMENTATION SCHEDULE

In accordance with 10 CFR 50.55a(a)(3)(i), the proposed Risk-Informed Inspection Program (RIS_B) change is an alternative to the ASME Code Section XI in-service inspection requirements for piping with regard to the number of inspections, locations of inspections, and methods of inspections as summarized in Attachment 1 of this request.

NMPNS proposes to implement the alternative Risk-Informed / Safety-Based (RIS_B) inspection plan and schedule in accordance with ASME Code Case N-716, utilizing the EPRI methodology applied to plant specific ASME Code Class 1 and 2 piping in accordance with the EPRI-TR and Regulatory Guide 1.178.

All examinations required by the alternative risk-informed methodology will be accomplished by the end of the Fourth Ten-Year ISI Interval that is currently scheduled for completion on August 22, 2019. The Fourth Ten-Year ISI Interval is scheduled to begin on August 23, 2009.

System pressure tests and visual examination of piping structural elements will continue to be performed on all Class 1, 2 and 3 systems in accordance with the ASME Section XI pressure testing program.

F. PRECEDENTS

NRC Safety Evaluation for Grand Gulf Nuclear Station Unit 1, Request for Alternative GG-ISI-002-Implement Risk-Informed ISI based on ASME Code Case N-716, dated September 21, 2007.

NRC Safety Evaluation for DC Cook Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based ISI program for Class 1 and 2 Piping Welds, dated September 28, 2007.

NRC Safety Evaluation for Waterford Steam Electric Station, Unit 3, Request for Alternative W3-ISI-005, Request to Use ASME Code Case N-716, dated April 28, 2008.

G. ATTACHMENTS

Attachment 1, Application of ASME Code Case N-716, Risk-Informed / Safety-Based Inservice Inspection Program

H. REFERENCES

a. ASME Section XI Code Case N-716, Alternative Piping Classification and Examination Requirements, Approved April 19, 2006

b. EPRI TR-112657, Revised Risk-Informed In-service Inspection Evaluation Procedure Final Report, Revision B-A, December 1999

c. Nine Mile Point Report NER-1A-020, Rev. 0, NMP1 RI-ISI Degradation Evaluation

d. Nine Mile Point Letter Report NMP1-RI-ISI-01-003-SE, Rev. 1, Service History Review

e. Risk-Informed ISI Program Update Nine Mile Point Units 1 and 2, December 2005

f. NRC Regulatory Guide 1.178, An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping, Revision 1, September 2003

REQUEST FOR ALTERNATIVE
1ISI-003

ATTACHMENT 1

APPLICATION OF ASME CODE CASE N-716 RISK-INFORMED / SAFETY-BASED INSERVICE INSPECTION PROGRAM

Table of Contents

1 INTRODUCTION

The Nine Mile Point Nuclear Station, Unit 1 (NMP1) Third In-service Inspection Interval as defined by ASME Section XI, Inspection Program B, will end on August 22, 2009. In-service examinations for the third 10-year interval were performed on Code Class 1 and 2 piping in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, 1989 Edition, No Addenda, as required by Title 10, Code of Federal Regulations (CFR), Part 50, Section 50.55a, and the NRC-approved risk-informed in-service inspection (RI-ISI) program (Reference 1).

The ASME Section XI Code of record for the fourth in-service inspection interval is the 2001 Edition through 2003 Addenda for Examination Categories B-F, B-J, C-F-1, C-F-2 and IGSCC Category A Class 1 and 2 piping components.

The objective of this submittal is to provide the information required to support the Nine Mile Point Nuclear Station, LLC (NMPNS) request to use an alternate risk-informed, safety-based (RIS_B) process for the in-service inspection of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI Division 1 (Reference 2), which is founded in large part on the RI-ISI process as described in the Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Revision B-A , titled Revised Risk-Informed In-service Inspection Evaluation Procedure (Reference 3). The NRC previously approved a RI-ISI Program for NMP1 based on Reference 3 and ASME Code Case N-578-1 (see Reference 1).

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (Reference 4), and Regulatory Guide 1.178, An Approach for Plant-Specific Risk-Informed Decisionmaking for In-service Inspection of Piping (Reference 5). Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Risk Assessment (PRA) Quality

The original Individual Plant Examination (IPE), which was reviewed by the NRC and BWROG Certification Process, screened most pipe break initiators. These initial reviews did not identify significant issues with the original internal flooding screening analyses. NRC reviews of the IPE and Individual Plant Examination External Events (IPEEE) are documented in the NRC Staff Evaluations on the IPE dated April 2, 1996 (Reference 6) and on the IPEEE dated July 18, 2000 (Reference 7). The NRC concluded that the NMP1 IPE and IPEEE processes were capable of identifying the most likely severe accidents and severe accident vulnerabilities.

The NMP1 PRA has since been updated to meet ASME PRA Standard RA-Sb-2005 (Reference 8) and Regulatory Guide (RG) 1.200, Revision 1 (Reference 9). The updated PRA model meets the Capability Category II supporting requirements (SRs) and combined Category II and Category III SRs where both requirements are equivalent (e.g., SR IF-D5a). Based on this updated PRA, several internal floods and high energy line breaks (HELB) were added to the PRA model to meet the ASME PRA Standard. There are no internal floods or HELB initiating events with a Core Damage Frequency (CDF) ≥ 1E-06/yr or a Large Early Release Frequency (LERF) ≥ 1E-07 year.

An industry peer review of the updated PRA model was conducted in February 2008. The peer review utilized the process described in Nuclear Energy Institute document NEI-05-04 (Reference 10) and the ASME PRA Standard, including consideration of the NRC staff positions provided in Appendix A of RG 1.200, Revision 1. A summary of the peer review findings and the impact of those findings on the PRA model have previously been submitted to the NRC by NMPNS letter dated December 4, 2008 (Reference 11). Four (4) of the peer review findings were related to internal flooding supporting requirements (finding numbers IF-D5a-01, IF-C8-01, IF-C3-01, and IF-E5a-01). The peer review findings were related primarily to documentation, have since been incorporated, and had an insignificant impact on the PRA results. The risk impact assessment performed for this RIS_B program application was based on the final post-peer-review PRA update, which incorporated the resolutions of these findings.

2 PROPOSED ALTERNATIVE TO CURRENT IN-SERVICE INSPECTION PROGRAMS

2.1 ASME Section XI

ASME Section XI, Tables IWB-2500-1 and IWC-2500-1, Examination Categories B-F, B-J, C-F-1, and C-F-2 currently provide the requirements for in-service examination of piping welds, utilizing nondestructive examination (NDE) methods.

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be implemented as an alternative for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by providing an acceptable level of quality and safety. Non-related portions of the ASME Section XI Code, (inspection intervals, acceptance criteria for evaluation of flaws, expansion criteria for flaws discovered, and qualification of examination techniques and personnel) are essentially unaffected by the proposed RIS_B program.

2.2 Augmented Programs

NMPNS, together with the Boiling Water Reactor (BWR) Vessel & Internals Project and EPRI, has investigated operating experience and material performances with respect to the BWR fleet and intergranular stress corrosion cracking (IGSCC) issues. Consistent with the EPRI-TR, Category A, Generic Letter 88-01(Reference 12) and NUREG-0313, Rev. 2 (Reference 13) welds are integrated into the proposed alternative RIS_B program. As such, the NMP1 response to Generic Letter 88-01 and its supplement remains unchanged for IGSCC Categories B through G for the fourth ISI interval. Another augmented inspection program, Generic Letter 89-08 Flow Accelerated Corrosion Program (FAC) (Reference 14), is credited in the proposed RIS_B program but is not affected or changed by the proposed RIS_B program. Any other existing augmented inspection programs are unaffected by this submittal.

3 RISK-INFORMED / SAFETY-BASED IN-SERVICE INSPECTION PROCESS

The process used to develop the RIS_B program conformed to the methodology described in Code Case N-716, including lessons learned from the pilot applications, and consisted of the following steps:

- Safety Significance Determination
- Failure Potential Assessment
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

There were no deviations to the process described in Code Case N-716.

3.1 Safety Significance Determination

The systems included in the RIS_B Program are provided in Table 3.1. Piping and instrumentation diagrams and additional plant information were used to define the system piping boundaries.

Per Code Case N-716 requirements, piping welds are assigned safety-significance categories, which are used to determine the treatment requirements. High safety-significant (HSS) welds are determined in accordance with the requirements below.

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii). For NMP1 this includes all Class 1 piping welds in the ISI Program.

(2) Applicable portions of the shutdown cooling system pressure boundary function. This includes Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling flow path either:

   a. As part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., isolation valve farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or

   b. Other systems or portions of systems from the RPV to the second isolation valve (i.e., isolation valve farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds.

(3) That portion of the Class 2 feedwater system [>4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the steam generator to the outer containment isolation valve. This does not apply to NMP1, which is a BWR.

(4) Piping within the break exclusion region (BER), (>NPS 4) for high-energy piping systems as defined by the owner.

(5) Any piping segment whose contribution to CDF is greater than 1E-06 based upon a plant-specific PRA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping. Based on lessons learned from pilot plant applications, a LERF greater than 1E-07 is also used to determine high safety significance.

Low Safety Significance (LSS) is applied to all remaining Class 2, 3 and Non-Class piping welds that are not determined to be HSS based on the above criteria.

3.2 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in EPRI-TR-112657 (i.e., the EPRI RI-ISI methodology).

Table 3.2 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

3.3 Element and NDE Selection

Code Case N-716 provides criteria for identifying the number and location of required examinations. Ten percent of the HSS welds shall be selected for examination as follows:

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:

   a. A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.

   b. If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.

   c. If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the RCPB welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the first isolation valve (i.e., isolation valve closest to the RPV) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (e.g., portions of the main feedwater system in BWRs) shall be selected.

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected.

In contrast to a number of RI-ISI Program applications where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen. A brief summary is provided below, and the results of the selections are presented in Table 3.3. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

	Class 1 Welds (1)		Class 2 Welds (2)		NSS Welds (3)		All Piping Welds (4)	
Unit	Total	Selected	Total	Selected	Total	Selected	Total	Selected
1	629	69	740	0	0	0	1369	69

Notes

(1) Includes all Category B-F and B-J locations. All 629 Class 1 piping weld locations are HSS.
(2) Includes all Category C-F-1 and C-F-2 locations. Of the 740 Class 2 piping weld locations, all are LSS.
(3) There are no non-nuclear safety (NNS) piping weld locations that are HSS.
(4) Regardless of safety significance, Class 1, 2 and 3 piping components will continue to be pressure tested as required by the ASME Code, Section XI Program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RIS_B Program.

3.3.1 Additional Examinations

If the flaw is original construction or otherwise acceptable, Code rules do not require any additional inspections. Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500 and/or IWB-3600. As part of performing an evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. The process for ordinary flaws is to perform the evaluation using ASME Section XI. If the flaw meets the criteria, then it is noted and appropriate successive examinations scheduled. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined. If the flaw is found unacceptable for continued service, it will be repaired in accordance with IWA-4000 and/or applicable ASME Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation will be dependent on practical considerations (i.e., the practicality of performing additional NDE or removing the flaw for further evaluation during the outage). The NRC staff is involved in the process at several points. For preemptive weld overlays, a relief request in accordance with 10 CFR 50.55a(a)(3) is usually required for design and installation. Should a flaw be discovered during an examination, a notification in accordance with 10 CFR 50.72 or 10 CFR 50.73 may be required. IWB-3600 requires the evaluation to be submitted to the NRC staff. Finally, the required documentation will be included in the corrective action program and the Owner submittals required by Section XI.

The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause and degradation mechanism. Additional examinations will be performed on the elements with the same root cause conditions or degradation mechanisms. The additional examinations will include HSS elements up to a number equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions or degradation mechanism.

3.3.2 Program Relief Requests

Alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations such as inaccessibility or radiation exposure hazard.

An attempt has been made to select RIS_B locations for examination such that a minimum of >90 percent coverage (i.e., ASME Code Case N-460 (Reference 15) criteria) is attainable. However, some limitations will not be known until the examination is performed, since some locations may be examined for the first time by the specified techniques.

Weld 38-WD-007, located on the Shutdown Cooling System, is inaccessible due to its location inside a containment penetration. No other selection can be made as all four (4) items from which a selection may be made are inaccessible. Structural integrity of these welds was previously addressed in request for relief ISI-12 (granted by NRC letter dated May 31, 2001 - Reference 16), and requires a visual VT-2 examination each refueling outage. The visual (VT-2) examination will continue to be performed during the fourth inspection interval.

In instances where locations at the time of the examination fail to meet the >90% coverage requirement, the process outlined in 10 CFR 50.55a will be followed.

Consistent with traditional techniques, NMPNS will calculate coverage and use additional examinations or techniques in the same manner as ASME Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until that time. Relief requests will be submitted per the requirements of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

3.4 Risk Impact Assessment

The RIS_B Program has been developed in accordance with Regulatory Guide 1.174 and the requirements of ASME Code Case N-716. The the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current analyses.

This evaluation categorized segments as high safety significant or low safety significant in accordance with ASME Code Case N-716, and then determined what inspection changes are proposed for each system. The changes include changing the number and location of inspections and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

ASME Code Case N-716 has adopted the EPRI TR-112657 process for risk impact analyses whereby limits are imposed to ensure that the change in risk of implementing the RIS_B Program meets the guidance of Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively.

For LSS welds the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) values of 1E-04 and 1E-05, respectively, were conservatively used except for the emergency cooling system, which was found to have CCDP and CLERP value slightly greater than 1E-04. The rationale for using these values is that the change-in-risk evaluation process of Code Case N-716 is similar to that of the EPRI RI-ISI methodology. As such, the goal is to determine CCDP and CLERP threshold values. For example, the threshold values between High and Medium consequence categories are1E-04 (CCDP) / 1E-05 (CLERP) and between Medium and Low consequence categories are 1E-06 (CCDP) / 1E-07 (CLERP) from the EPRI RI-ISI Risk Matrix. Using these threshold values streamlines the change-in-risk evaluation as well as stabilizes the update process. For example, if a CCDP changes from 1E-05 to 3E-05 due to an update, it will remain below the 1E-04 threshold value and the change-in-risk evaluation would not require updating.

The above 1E-04 (CCDP) and 1E-05 (CLERP) values were checked against the NMP1 PRA and the internal flooding study which was performed to ASME RA-Sb-2005 capability category II requirements and Regulatory Guide 1.200, Revision 1. CCDP and CLERP values for in-scope LSS Class 2 piping previously being inspected are less than 1E-04 (CCDP) and 1E-05 (CLERP) except for the emergency cooling system. Therefore, the 0.1 conditional LERF value used in the calculations is considered reasonable. The values are consistent with and conservatively above any CCDP value obtained for NMP1 in-scope Class 2 piping, and the CLERP value is appropriately scaled.

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of Code Case N-716. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion or stress corrosion cracking are assigned to a medium failure potential and those locations that are identified as not susceptible to degradation are assigned a low failure potential.

In order to streamline the risk impact assessment, a review was conducted to verify that the LSS piping was not susceptible to FAC or water hammer. This review was conducted similar to that done for a traditional RI-ISI application. Thus for NMP1, the high failure potential category is not applicable to LSS piping. In lieu of conducting a formal degradation mechanism evaluation for all LSS piping (e.g., to determine if thermal fatigue is applicable), these locations were conservatively assigned to the Medium failure potential ("Assume Medium" in Table 3.4) for use in the change-in-risk assessment.

NMPNS has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716 that is consistent with the "Simplified Risk Qualification Method" described in Section 3.7 of EPRI TR-112657. The analysis estimates the net change in risk due to the positive and negative influences of adding and removing locations from the inspection program.

The CCDP and CLERP values used to assess risk impact were estimated based on pipe break location. Based on these estimated values, a corresponding consequence rank was assigned per the requirements of EPRI TR-112657 and upper bound threshold values were used as provided in Table 3.5. Consistent with the EPRI risk-informed methodology, the upper bound for all break locations that fall within the high consequence rank range was based on the highest CCDP value obtained (e.g., Medium LOCA for NMP1 bounds large and small LOCA).

The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_0 and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_0$.These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to increased Probability of Detection (POD) from application of the RIS_B approach.

Table 3.4 presents a summary of the RIS_B Program versus 1989 Edition with no Addenda ASME Section XI Code program requirements on a "per system" basis. The presence of FAC was adjusted for in the quantitative analysis by excluding its impact on the failure potential rank. The exclusion of the impact of FAC on the failure potential rank, and therefore in the determination of the change in risk, is performed because FAC is a damage mechanism managed by a separate, independent plant augmented inspection program.

The RIS_B Program credits and relies upon this plant augmented inspection program to manage this degradation mechanism. The plant FAC Program will continue to determine where and when examinations shall be performed. Hence, since the number of FAC examination locations remains the same "before" and "after" and no delta exists, there is no need to include the impact of FAC in the performance of the risk impact analysis.

As indicated in Table 3.6, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and satisfies the acceptance criteria of Regulatory Guide 1.174 and ASME Code Case N-716.

3.4.2 Defense-In-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for selecting inspection locations is based upon structural discontinuity and stress analysis results. As referenced in Section 2.3 of the EPRI-TR and depicted in the Summary of the ASME White Paper 92-01-01 Rev. 1, Evaluation of In-service Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds (Reference 17), this method has been ineffective in identifying leaks or failures. The EPRI-TR and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; a determination of each location's susceptibility to degradation and an independent assessment of the consequence of the piping failure. These two ingredients assure that defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Second, a generic assessment of high-consequence sites has been determined by Code Case N-716 as supplemented by plant-specific evaluations, thereby requiring a minimum threshold of inspection for important piping whose failure would result in a LOCA or BER break. Finally, Code Case N-716 requires that any piping on a plant-specific basis that has a contribution to CDF of greater than 1E-06 be included in the scope of the application. NMP1 did not identify any such piping.

All locations within the Class 1, 2, and 3 pressure boundaries will continue to be pressure tested in accordance with the Code, regardless of its safety significance.

4 IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the proposed RIS_B Program, appropriate procedures and/or revisions to the existing inspection program that implement the guidelines described in the EPRI-TR-112657 and/or Code Case N-716 will be prepared to implement and monitor the program. The new program will be integrated into the fourth 10-year ASME Section XI in-service inspection interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for the alternative RIS_B Program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance standards, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementation documents will be retained and modified to address the RIS_B process, as appropriate.

The monitoring and corrective action program will contain the following elements:

A. Identification of the condition

B. Characterization of the condition

C. Evaluation of the condition, to include

  1. Determination of the cause and extent of the condition identified

  2. Develop a corrective action plan or plans

D. Decide

E. Implement

F. Monitor

G. Trend

The RIS_B Program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant (HSS) piping locations. As a minimum, this review will be conducted on an ASME period basis. In addition, significant changes may require more frequent adjustment as identified by NRC Bulletins, Generic Letters, or by industry and plant- specific feedback.

For pre-service examinations, NMPNS will follow the rules contained in Section 3.0 of Code Case N-716. Welds classified HSS require pre-service inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of Code Case N-716. Welds classified as LSS do not require pre-service inspection.

5 PROPOSED IN-SERVICE INSPECTION PROGRAM PLAN CHANGE

A comparison between the RIS_B Program and the 1989 Edition with no Addenda ASME Section XI inspection program requirements for in-scope piping is given in Table 5.1.

Intersecting longitudinal seam welds of RI-ISI selected elements will continue to be examined.

NMPNS intends to start implementing the RIS_B Program during the fourth in-service inspection interval and will implement 100% of the inspection locations selected for examination per the RIS_B Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

6 REFERENCES/DOCUMENTATION

1. Letter from R. J. Laufer (NRC) to J. T. Conway (NMPNS) dated September 4, 2002, Nine Mile Point Nuclear Station, Unit No. 1 - Risk-Informed Inservice Inspection Program (TAC No. MB4085).

2. ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI Division 1.

3. Electric Power Research Institute Topical Report-112657, Revised Risk-Informed In-service Inspection Evaluation Procedure, Revision B-A, dated December 1999.

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

5. NRC Regulatory Guide 1.178, An Approach for Plant -Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping, Revision 1, September 2003.

6. Letter from D. S. Hood (NRC) to B. R. Sylvia (NMPC) dated April 2, 1996, NRC Staff's Evaluation of the Nine Mile Point Nuclear Station Unit No. 1 Individual Plant Examination (IPE) Submittal (TAC No. M74436).

7. Letter from P. S. Tam (NRC) to J. H. Mueller (NMPC) dated July 18, 2000, Nine Mile Point Unit 1 – Individual Plant Examination of External Events (TAC No. M83645).

8. ASME RA-Sb-2005, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to ASME RA-S-2002, December 30, 2005.

9. NRC Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1, January 2007.

10. NEI-05-04, Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard, January 2005.

11. Letter from K. J. Polson (NMPNS) to Document Control Desk (NRC) dated December 4, 2008, License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval – Response to NRC Request for Additional Information (TAC No. MD9453).

12. Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, dated January 25, 1988.

13. NUREG-0313, Revision 2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, dated January 1988.

14. Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning, dated May 2, 1989.

15. ASME Code Case N-460, Alternative Examination Coverage for Class 1 and Class 2 Welds, Section XI, Division 1, approved by ASME on February 14, 2003, and approved by Regulatory Guide 1.147, Revision 15, dated October 2007.

16. Letter from R. P. Correia (NRC) to J. H. Mueller (NMPC) dated May 31, 2001, Nine Mile Point Nuclear Station, Unit No. 1 - Inservice Inspection Relief Request ISI-12 (TAC No. MA9662).

17. Summary of the ASME White Paper 92-01-01 Rev. 1, Evaluation of In-service Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds, dated July 1995.

18. Nine Mile Point Report NER-1A-020, Rev. 0, NMP1 RI-ISI Degradation Evaluation.

19. CNG-NMP1-004-021, ASME Code Case N-716 Evaluation of Nine Mile Point Unit 1, Revision 00.

7 LIST OF ACRONYMS

BER	Break Exclusion Region
CC	Crevice Corrosion
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CLERP	Conditional Large Early Release Probability
DM	Degradation Mechanism
ECSCC	External Chloride Stress Corrosion Cracking
E-C	Erosion-Cavitation
FAC	Flow-Accelerated Corrosion
FWLOCA-OC	Feedwater Unisolable Break - Outside Containment
HELB	High Energy Line Break
HELBCU	High Energy Line Break-Cleanup Line
HELBECR	High Energy Line Break-Emergency Cooling Condensate Return Line
HELBECS	High Energy Line Break-Emergency Cooling Steam Supply Line
HELBFWD	High Energy Line Break-Feedwater Line
HSS	High Safety Significant
IGSCC	Intergranular Stress Corrosion Cracking
ILOCA	Isolable Loss of Coolant Accident
ILOCA-OC	Isolable Loss of Coolant Accident - Outside Containment
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination External Events
ISLOCA	Interfacing Systems Loss of Coolant Accident
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LSS	Low Safety Significant
MIC	Microbiologically-Influenced Corrosion
MLOCA	Medium Loss of Coolant Accident
MLOCAW	Medium Water Loss of Coolant Accident
NDE	Non-destructive Examination
NNS	Non-nuclear Safety
PBF	Pressure Boundary Failure
PIT	Pitting
PLOCA	Potential Loss of Coolant Accident
PLOCA-OC	Potential Loss of Coolant Accident - Outside Containment
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PWSCC	Primary Water Stress Corrosion Cracking
RCPB	Reactor Coolant Pressure Boundary
RCPB^{IFIV}	Reactor Coolant Pressure Boundary Inside First Isolation Valve
RCPB^{OC}	Reactor Coolant Pressure Boundary Outside Containment
RI-ISI	Risk-Informed Inservice Inspection
RIS_B	Risk-Informed / Safety-Based Inservice Inspection
RPV	Reactor Pressure Vessel
Sec.XI	ASME Section XI
Sur	Surface
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
TT	Thermal Transients
Vol/Sur	Volumetric and Surface

Table 3.1
Code Case N-716 Safety Significance Determination

System (1) Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-06	High	Low
CRD	21	✓					✓	
CS	94	✓					✓	
	242							✓
EC	50	✓					✓	
	75							✓
FW	50	✓					✓	
INST	39	✓					✓	
LP	21	✓					✓	
MS	66	✓					✓	
RD	22	✓					✓	
RHV	30	✓					✓	
RR	187	✓					✓	
RWCU	33	✓					✓	
SDC	16	✓	✓				✓	
CTN	363							✓
SDV	60							✓
Summary Results For All Systems	16	✓	✓				✓	
	613	✓					✓	
	740							✓
TOTALS	**1369**						**629**	**740**

Notes

1. System Description Abbreviations:

CRD - Control Rod Drive Injection Path
CS - Core Spray
EC - Emergency Cooling
FW - Feedwater
INST - Instrument Connections
LP - Liquid Poison
MS - Main Steam
RD - Reactor Vessel Drain
RHV - Reactor Head Vent
RR - Reactor Recirculation
RWCU - Reactor Water Cleanup
SDC - Shutdown Cooling
CTN - Containment Spray
SDV - Scram Discharge Volume

Table 3.2
Failure Potential Assessment Summary

System(1)	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CRD									✓		
CS(2)			✓								
EC(2)	✓		✓								
FW									✓		✓
INST											
LP	✓										
MS	✓										
RD											
RHV			✓								
RR	✓		✓								
RWCU	✓		✓								
SDC	✓		✓								
CNT(2)											
SDV(2)											

Notes

1. Systems are described in Table 3.1.
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the CTN and SDV systems in their entirety, as well as portions of the CS and EC systems.

Table 3.3
Code Case N-716 Element Selections

System(1)	Weld Count		N-716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB^{IFIV} (2)	RCPB^{OC} (3)	BER	
CRD	1		CC	✓	✓			1
CRD	20		None	✓	✓	✓		2
CS	74		None (IGSCC)	✓	✓	✓		11
CS	20		None	✓	✓	✓		0
CS		242	n/a					0
EC	4		None (IGSCC)	✓	✓			0
EC	6		TASCS	✓	✓			1
EC	26		TASCS, None (IGSCC)	✓				3
EC	14		None	✓	✓	✓		1
EC		75	n/a					0
FW	5		CC	✓	✓			2
FW	6		None (FAC)	✓	✓			2
FW	1		CC, None (FAC)	✓	✓			0
FW	38		None	✓	✓	✓		1
INST	39		None	✓	✓			4
LP	8		TASCS	✓	✓			2
LP	13		None	✓	✓			1
MS	10		TASCS	✓	✓	✓		2
MS	56		None	✓	✓	✓		5
RD	22		None	✓	✓			3
RHV	1		None (IGSCC)	✓	✓			1
RHV	29		None	✓	✓			2
RR	15		None (IGSCC)	✓	✓			15
RR	15		TASCS	✓	✓			4
RR	157		None	✓	✓			0
RWCU	4		None (IGSCC)	✓	✓			0
RWCU	14		TASCS	✓	✓			3
RWCU	1		TASCS, None (IGSCC)	✓	✓			1
RWCU	14		None	✓	✓			0
SDC	7		None (IGSCC)					0
SDC	9		TASCS, None (IGSCC)					2
CTN		363	n/a					0
SDV		60	n/a					0

Table 3.3
Code Case N-716 Element Selections

System(1)	Weld Count		N-716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB^{IFIV} (2)	RCPB^{OC} (3)	BER	
Summary for all Systems	6		CC	✓	✓			3
	1		CC, None (FAC)	✓	✓			0
	53		TASCS	✓	✓	✓		12
	36		TASCS, None (IGSCC)	✓	✓			6
	6		None (FAC)	✓	✓			2
	104		None (IGSCC)	✓	✓	✓		27
	422		None	✓	✓	✓		19
		740	n/a					0
Totals	**629**	**740**						**69**

Notes

1. Systems are described in Table 3.1
2. Reactor coolant pressure boundary welds that are located between the first isolation valve and the reactor vessel.
3. Reactor coolant pressure boundary welds that are located outside the primary containment.

Table 3.4
Risk Impact Analysis Results

System(1)	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank(3)	Sec. XI(2)	RIS_B (4)	Delta	w/POD	w/o POD	w/POD	w/o POD
CRD	High	LOCA	CC	Medium	0	1	1	-2.00E-10	-2.00E-10	-2.00E-11	-2.00E-11
CRD	High	LOCA	None	Low	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
CRD	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CRD	High	ILOCA-OC	None	Low	0	1	1	-1.50E-11	-1.50E-11	-1.50E-11	-1.50E-11
CRD Total								**-2.25E-10**	**-2.25E-10**	**-3.60E-11**	**-3.60E-11**
CS	High	LOCA	None (IGSCC)	Low (Medium)	8	7	-1	1.00E-11	1.00E-11	1.00E-12	1.00E-12
CS	High	PLOCA	None (IGSCC)	Low (Medium)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CS	High	PLOCA-OC	None (IGSCC)	Low (Medium)	0	4	4	-2.00E-12	-2.00E-12	-2.00E-13	-2.00E-13
CS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CS	High	PLOCA-OC	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CS	Low	Class 2 LSS	n/a	Assume Medium	21	0	-21	2.10E-10	2.10E-10	2.10E-11	2.10E-11
CS Total								**2.18E-10**	**2.18E-10**	**2.18E-11**	**2.18E-11**
EC	High	LOCA	None (IGSCC)	Low (Medium)	2	0	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
EC	High	LOCA	TASCS	Medium	0	1	1	-3.60E-10	-2.00E-10	-3.60E-11	-2.00E-11
EC	High	ILOCA	TASCS	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
EC	High	ILOCA	TASCS, None (IGSCC)	Medium	0	3	3	-3.00E-11	-3.00E-11	-3.00E-12	-3.00E-12
EC	High	LOCA	None	Low	3	0	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
EC	High	ILOCA-OC	None	Low	0	1	1	-1.50E-11	-1.50E-11	-1.50E-11	-1.50E-11
EC	Low	Class 2 EC	None (IGSCC)	Assume Medium	10	0	-10	2.00E-09	2.00E-09	2.00E-09	2.00E-09
EC	Low	Class 2 EC	n/a	Assume Medium	20	0	-20	4.00E-09	4.00E-09	4.00E-09	4.00E-09
EC Total								**5.65E-09**	**5.81E-09**	**5.95E-09**	**5.97E-09**
FW	High	LOCA	CC	Medium	4	2	-2	4.00E-10	4.00E-10	4.00E-11	4.00E-11
FW	High	LOCA	None (FAC)	Low (High)	1	2	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
FW	High	LOCA	CC, None (FAC)	Medium (High)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FW	High	LOCA	None	Low	5	0	-5	5.00E-11	5.00E-11	5.00E-12	5.00E-12
FW	High	ILOCA-OC	None	Low	2	1	-1	1.50E-11	1.50E-11	1.50E-11	1.50E-11
FW	High	FWLOCA-OC	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FW Total								**4.55E-10**	**4.55E-10**	**5.90E-11**	**5.90E-11**
INST	High	LOCA	None	Low	18	4	-14	**1.40E-10**	**1.40E-10**	**1.40E-11**	**1.40E-11**

Table 3.4
Risk Impact Analysis Results

System(1)	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank(3)	Sec. XI(2)	RIS_B (4)	Delta	w/POD	w/o POD	w/POD	w/o POD
LP	High	LOCA	TASCS	Medium	0	2	2	-7.20E-10	-4.00E-10	-7.20E-11	-4.00E-11
LP	High	LOCA	None	Low	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
LP	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
LP Total								**-7.30E-10**	**-4.10E-10**	**-7.30E-11**	**-4.10E-11**
MS	High	ILOCA-OC	TASCS	Medium	0	2	2	-1.08E-09	-6.00E-10	-1.08E-09	-6.00E-10
MS	High	LOCA	None	Low	11	5	-6	6.00E-11	6.00E-11	6.00E-12	6.00E-12
MS	High	ILOCA	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
MS	High	ILOCA-OC	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS Total								**-1.02E-09**	**-5.39E-10**	**-1.07E-09**	**-5.94E-10**
RD	High	LOCA	None	Low	0	2	2	-2.00E-11	-2.00E-11	-2.00E-12	-2.00E-12
RD	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RD Total								**-2.00E-11**	**-2.00E-11**	**-2.00E-12**	**-2.00E-12**
RHV	High	LOCA	None (IGSCC)	Low (Medium)	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RHV	High	LOCA	None	Low	1	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RHV	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RHV Total								**-2.00E-11**	**-2.00E-11**	**-2.00E-12**	**-2.00E-12**
RR	High	LOCA	None (IGSCC)	Low (Medium)	15	15	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RR	High	LOCA	TASCS	Medium	2	4	2	-1.20E-09	-4.00E-10	-1.20E-10	-4.00E-11
RR	High	LOCA	None	Low	33	0	-33	3.30E-10	3.30E-10	3.30E-11	3.30E-11
RR Total								**-8.70E-10**	**-7.00E-11**	**-8.70E-11**	**-7.00E-12**
RWCU	High	LOCA	None (IGSCC)	Low (Medium)	1	0	-1	0.00E+00	1.00E-11	1.00E-12	1.00E-12
RWCU	High	ILOOCA	None (IGSCC)	Low (Medium)	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
RWCU	High	LOCA	TASCS	Medium	5	3	-2	-4.80E-10	4.00E-10	-4.80E-11	4.00E-11
RWCU	High	LOCA	TASCS, None (IGSCC)	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RWCU	High	LOCA	None	Low	3	0	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
RWCU	High	ILOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
RWCU Total								**-4.49E-10**	**4.41E-10**	**-4.39E-11**	**4.41E-11**
SDC	High	PLOCA	None (IGSCC)	Low (Medium)	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
SDC	High	PLOCA	TASCS, None (IGSCC)	Medium	1	2	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
SDC Total								**-9.00E-12**	**-9.00E-12**	**-9.00E-13**	**-9.00E-13**

Table 3.4
Risk Impact Analysis Results

System(1)	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank(3)	Sec. XI(2)	RIS_B (4)	Delta	w/POD	w/o POD	w/POD	w/o POD
CTN	Low	Class 2 LSS	n/a	Assume Medium	27	0	-27	**2.70E-10**	**2.70E-10**	**2.70E-11**	**2.70E-11**
SDV	Low	Class 2 LSS	n/a	Assume Medium	6	0	-6	**6.00E-11**	**6.00E-11**	**6.00E-12**	**6.00E-12**
GRAND TOTAL								**3.45E-09**	**6.10E-09**	**4.76E-09**	**5.46E-09**

Notes

1. Systems are described in Table 3.1.
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. The failure potential rank for high safety significant (HSS) locations is then assigned as “High”, “Medium”, or “Low” depending upon potential susceptibly to the various types of degradation. Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”).
4. Only those RIS_B inspection locations that receive a volumetric examination are included in the count. Inspection locations receiving VT2 exams per Code Case N-716 were not considered.

Table 3.5
CCDP and CLERP Values Based on Break Location

Break Location	Estimated		Consequence Rank	Upper / Lower Bound		Description of Affected Piping
	CCDP	CLERP		CCDP	CLERP	
LOCA	2E-03	2E-04	HIGH	(U) 2E-03 (L) 1E-04	(U) 2E-04 (L) 1E-05	Unisolable RCPB piping of all sizes
The highest CCDP for Medium Water LOCA (MLOCAW) was used (0.1 margin used for CLERP)						
ILOCA	6E-06	6E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping between 1st and 2nd normally open isolation valves
Calculated based on MLOCAW CCDP of 2E-03 and valve fail to close probability of 3E-03 (0.1 margin used for CLERP)						
PLOCA	2E-06	2E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping between 1st and 2nd normally closed isolation valves
Calculated based on MLOCA CCDP of 2E-3 and valve rupture probability of <1E-03 (0.1 margin used for CLERP)						
FWLOCA–OC	3E-02	3E-02	HIGH	(U) 3E-02 (L) 1E-04	(U) 3E-02 (L) 1E-05	Feedwater unisolable break outside containment
Based on unisolated feedwater break outside containment (HELBFWD) CCDP of 3E-02 (CCDP = CLERP)						
ILOCA-OC	3E-03	3E-03	HIGH	(U) 3E-03 (L) 1E-04	(U) 3E-03 (L) 1E-05	Piping between penetration and outside containment isolation valve with normally open isolation valve inside containment
Isolable LOCA outside containment CCDP based on initiating event HELBCU CCDP of 3E-02 (CCDP = CLERP)						
PLOCA-OC	1E-05	1E-05	MEDIUM	(U) 1E-04 (L) 1E-06	U) 1E-05 (L) 1E-07	Piping between penetration and outside containment isolation valve with normally closed isolation valve inside containment
Potential LOCA outside containment CCDP based on valve rupture probability <1E-03 and CCDP for ISLOCA <1E-02 (CCDP = CLERP)						
Class 2 LSS	1E-04	1E-05	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	All other Class 2 system piping designated as low safety significant except emergency cooling system
Estimated based on upper bound for Medium Consequence						
Class 2 EC	3.4E-04	3.4E-04	MEDIUM	(U) 2E-03 (L) 2E-03	(U) 2E-03 (L) 2E-03	Class 2 emergency cooling system piping designated as low safety significant
Estimated based on upper bound for Medium Consequence HELBECR and HELBECS CCDP and CLERP ~3.4E-04						

Table 3.6
NMP1 Risk Impact Results

System[1]	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CRD	-2.25E-10	-3.60E-11	-2.25E-10	-3.60E-11
CS	2.18E-10	2.18E-11	2.18E-10	2.18E-11
EC	5.65E-09	5.95E-09	5.81E-09	5.97E-09
FW	4.55E-10	5.90E-11	4.55E-10	5.90E-11
INST	1.40E-10	1.40E-11	1.40E-10	1.40E-11
LP	-7.30E-10	-7.30E-11	-4.10E-10	-4.10E-11
MS	-1.02E-09	-1.07E-09	-5.39E-10	-5.94E-10
RD	-2.00E-11	-2.00E-12	-2.00E-11	-2.00E-12
RHV	-2.00E-11	-2.00E-12	-2.00E-11	-2.00E-12
RR	-8.70E-10	-8.70E-11	-7.00E-11	-7.00E-12
RWCU	-4.49E-10	-4.39E-11	4.41E-10	4.41E-11
SDC	-9.00E-12	-9.00E-13	-9.00E-12	-9.00E-13
CTN	2.70E-10	2.70E-11	2.70E-10	2.70E-11
SDV	6.00E-11	6.00E-12	6.00E-11	6.00E-12
Total	**3.45E-09**	**4.76E-09**	**6.10E-09**	**5.46E-09**

Note

1. Systems are described in Table 3.1.

Table 5.1
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System(1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank			Vol/Sur	Sur Only	RIS_B	Other(2)
CRD	✓		LOCA	CC	Medium	B-F	1	0	1	1	-
CRD	✓		LOCA	None	Low	B-J	17	0	7	1	-
CRD	✓		ILOCA	None	Low	B-J	2	0	0	0	-
CRD	✓		ILOCA-OC	None	Low	B-J	1	0	1	1	-
CS	✓		LOCA	None (IGSCC)	Low (Medium)	B-F,B-J	32	8	0	7	-
CS	✓		PLOCA	None (IGSCC)	Low (Medium)	B-J	26	0	0	0	-
CS	✓		PLOCA-OC	None (IGSCC)	Low (Medium)	B-J	16	0	0	4	-
CS	✓		PLOCA	None	Low	B-J	2	0	0	0	-
CS	✓		PLOCA-OC	None	Low	B-J	18	0	3	0	-
CS		✓	Class 2 LSS	n/a	Assume Medium	C-F-2	242	21	0	0	-
EC	✓		LOCA	None (IGSCC)	Low (Medium)	B-F,B-J	4	2	0	0	-
EC	✓		LOCA	TASCS	Medium	B-J	4	0	0	1	-
EC	✓		ILOCA	TASCS	Medium	B-J	2	0	0	0	-
EC	✓		ILOCA	TASCS, None (IGSCC)	Medium	B-J	26	0	0	3	-
EC	✓		LOCA	None	Low	B-J	12	3	0	0	-
EC	✓		ILOCA-OC	None	Low	B-J	2	0	0	1	-
EC		✓	Class 2 EC	None (IGSCC)	Assume Medium	C-F-1	31	10	0	0	-
EC		✓	Class 2 EC	n/a	Assume Medium	C-F-1	44	20	0	0	-
FW	✓		LOCA	CC	Medium	B-J	5	4	0	2	-
FW	✓		LOCA	None (FAC)	Low (High)	B-J	6	1	0	2	-
FW	✓		LOCA	CC, None (FAC)	Medium (High)	B-J	1	0	0	0	-
FW	✓		LOCA	None	Low	B-J	32	5	0	0	-
FW	✓		ILOCA-OC	None	Low	B-J	4	2	0	1	-
FW	✓		FWLOCA-OC	None	Low	B-J	2	0	0	0	-
INST	✓		LOCA	None	Low	B-F,B-J	39	18	12	4	-
LP	✓		LOCA	TASCS	Medium	B-J	8	0	3	2	-
LP	✓		LOCA	None	Low	B-F,B-J	4	0	1	1	-
LP	✓		PLOCA	None	Low	B-J	9	0	2	0	-

Table 5.1
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System(1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank			Vol/Sur	Sur Only	RIS_B	Other(2)
MS	✓		ILOCA-OC	TASCS	Medium	B-J	10	0	0	2	-
MS	✓		LOCA	None	Low	B-J	47	11	5	5	-
MS	✓		ILOCA	None	Low	B-J	6	2	0	0	-
MS	✓		ILOCA-OC	None	Low	B-J	3	0	1	0	-
RD	✓		LOCA	None	Low	B-J	20	0	6	2	1 VT2
RD	✓		PLOCA	None	Low	B-J	2	0	0	0	-
RHV	✓		LOCA	None (IGSCC)	Low (Medium)	B-J	1	0	0	1	-
RHV	✓		LOCA	None	Low	B-F,B-J	17	1	5	2	-
RHV	✓		PLOCA	None	Low	B-J	12	0	3	0	-
RR	✓		LOCA	None (IGSCC)	Low (Medium)	B-J	15	15	0	15	-
RR	✓		LOCA	TASCS	Medium	B-J	15	2	0	4	-
RR	✓		LOCA	None	Low	B-F,B-J	157	33	18	0	-
RWCU	✓		LOCA	None (IGSCC)	Low (Medium)	B-F	1	1	0	0	-
RWCU	✓		ILOOCA	None (IGSCC)	Low (Medium)	B-F,B-J	3	1	0	0	-
RWCU	✓		LOCA	TASCS	Medium	B-J	14	5	0	3	-
RWCU	✓		LOCA	TASCS, None (IGSCC)	Medium	B-J	1	1	0	1	-
RWCU	✓		LOCA	None	Low	B-J	8	3	1	0	-
RWCU	✓		ILOCA	None	Low	B-J	6	1	0	0	-
SDC	✓		PLOCA	None (IGSCC)	Low (Medium)	B-J	7	2	0	0	-
SDC	✓		PLOCA	TASCS, None (IGSCC)	Medium	B-J	9	1	0	2	-
CTN		✓	Class 2 LSS	n/a	Assume Medium	C-F-2	363	27	2	0	-
SDV		✓	Class 2 LSS	n/a	Assume Medium	C-F-2	60	6	0	0	-

Notes

1. Systems are described in Table 3.1
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 4 of Code Case N-716. Code Case N-716 allows the existing plant augmented inspection program for IGSCC (Categories B through G) in a BWR to be credited toward the 10% requirement. Also, this column is used to denote those welds 2-inch and smaller that will receive a VT2 exam. Confirm that VT2 schedule requirements are consistent with N-716 requirements.
3. The failure potential rank for high safety significant (HSS) locations is then assigned as "High", "Medium", or "Low" depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").

APPENDIX I - ASME CODE CASES

I.0 ASME Code Cases

I.1 ASME Section XI Code Cases Used in the Preparation of ISI Program

ASME Code Cases approved through Regulatory Guide 1.147 may be proposed for revision to the inspection plan. Specific Code Cases used in the preparation of the inspection plan are identified below.

I.2 ASME Section XI Code Case Listing

The following ASME Code Cases have been reviewed for possible use during the Fourth Inservice Inspection Interval. Code Cases used during the inspection interval shall be identified with the specific component within the Inspection Plan and Schedule.

ASME SECTION XI CODE CASE LISTING			
CODE CASE NUMBER	TITLE	RG 1.147 Revision	IMPLEMENTED IN PROGRAM
N-416-3	Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2 and 3, Section XI, Division 1.	Rev. 15	No [1]
N-432-1	Repair Welding Using Automatic or Machine Gas Tungsten-Arc Welding (GTAW) Temper Bead Technique, Section XI, Division 1.	Rev. 15	No
N-460	Alternative Examination Coverage for Class 1 and Class 2 Welds, Section XI, Division 1	Rev. 15	Yes
N-491-2	Rules for Examination of Class 1, 2, 3, and MC Component Supports of Light Water Cooled Power Plants, Section XI, Division 1	Rev. 15	No
N-494-3	Pipe Specific Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 ferritic Piping that exceed the Acceptance Standards of IWB-3514.2 and in Class 1 Austenitic Piping that exceed the Acceptance Standards of IWB-3514.3, Section XI, Division 1	Rev. 15	No
N-496-2	Helical Coil Threaded Inserts, Section XI, Division 1	Rev. 15	No
N-504-3	Alternative Rules for Repair of Class 1, 2 and 3 Austenitic Stainless Steel Piping, Section XI, Division 1	Rev. 15	No
Conditions: 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.			
N-508-3	Rotation of Serviced Snubbers and Pressure Relief Valves for	Rev. 15	No [2]

<table>
<tr><th colspan="4">ASME SECTION XI CODE CASE LISTING</th></tr>
<tr><th>CODE CASE NUMBER</th><th>TITLE</th><th>RG 1.147 Revision</th><th>IMPLEMENTED IN PROGRAM</th></tr>
<tr><td></td><td>the Purpose of Testing, Section XI, Division 1</td><td></td><td></td></tr>
<tr><td>N-513-2</td><td>Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping, Section XI, Division 1</td><td>Rev. 15</td><td>No</td></tr>
<tr><td>N-516-3</td><td>Underwater Welding, Section XI, Division 1</td><td>Rev. 15</td><td>No</td></tr>
<tr><td colspan="4">Conditions: Licensees must obtain NRC 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.</td></tr>
<tr><td>N-522</td><td>Pressure Testing of Containment Penetration Piping, Section XI, Division 1</td><td>Rev. 15</td><td>No[1]</td></tr>
<tr><td>N-526</td><td>Alternative Requirements for Successive Inspection of Class 1 and 2 Vessels, Section XI, Division 1</td><td>Rev. 15</td><td>Yes</td></tr>
<tr><td>N-532-4</td><td>Repair/Replacement Activity Documentation Requirements and Inservice Summary Report Preparation and Submission, Section XI, Division 1.</td><td>Rev. 15</td><td>Yes</td></tr>
<tr><td>N-534</td><td>Alternative Requirements for Pneumatic Pressure Testing, Section XI, Division 1</td><td>Rev. 15</td><td>No[1]</td></tr>
<tr><td>N-537</td><td>Location of Ultrasonic Depth Sizing Flaws, Section XI, Division 1.</td><td>Rev. 15</td><td>No</td></tr>
<tr><td>N-554-3</td><td>Alternative Requirements for Reconciliation of Replacement Items and additions of New Systems, Section XI, Division 1.</td><td>Rev. 15</td><td>No</td></tr>
<tr><td>N-566-2</td><td>Corrective Action for Leakage Identified at Bolted Connections, Section XI, Division 1</td><td>Rev. 15</td><td>No[1]</td></tr>
<tr><td>N-573</td><td>Transfer of Procedure Qualification Records Between Owners, Section XI, Division 1.</td><td>Rev. 15</td><td>No</td></tr>
<tr><td>N-583</td><td>Annual Training Alternative, Section XI, Division 1</td><td>Rev. 15</td><td>No</td></tr>
<tr><td colspan="4">Conditions: (1) Supplemental practice shall be performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks.
(2) The training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.</td></tr>
<tr><td>N-586-1</td><td>Alternative Additional Examination Requirements for Classes 1, 2 and 3 Piping, Components and Supports, Section XI, Division 1</td><td>Rev. 15</td><td>No</td></tr>
<tr><td>N-606-1</td><td>Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique for BWR CRD Housing/Stub Tube Repairs, Section XI, Division 1</td><td>Rev 15</td><td>No</td></tr>
</table>

ASME SECTION XI CODE CASE LISTING

CODE CASE NUMBER	TITLE	RG 1.147 Revision	IMPLEMENTED IN PROGRAM
Conditions: Prior to welding, an examination or verification must be performed to ensure proper preparation of the base metal, and that the surface is properly contoured so that as acceptable weld can be produced. The surfaces to be welded, and surfaces adjacent to the weld, are to be free from contaminates, such as rust, moisture, grease, and other foreign material or any other condition that would prevent proper welding and adversely affect the quality or strength of the weld. This verification is to be required in the welding procedures.			
N-613-1	Ultrasonic Examination of 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), Section XI, Division 1.	Rev. 15	Yes
N-617	Alternative Examination Distribution Requirements for Table IWC-2500-1, Examination Category C-G, Pressure Retaining Welds in Pumps and Valves, Section XI, Division 1	Rev. 15	No
N-623	Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel, Section XI, Division 1	Rev. 15	Yes
N-624	Successive Inspections, Section XI, Division 1	Rev. 15	Yes
N-638-1	Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1	Rev.15	No
Conditions: UT volumetric examinations shall be performed with personnel and procedures qualified for the repaired volume and qualified by demonstration using representative samples which contain construction type flaws. The acceptance criteria of NB-5330 in the 1998 Edition through 2000 Addenda of Section III (Ref. 7) apply to all flaws identified within the repaired volume.			
N-640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1	Rev. 15	Thru Technical Specification Amendment
N-648-1	Alternative Requirements For Inner Radius Examination of Class 1 Reactor Vessel Nozzles, Section XI, Division 1	Rev.15	Yes
Condition: 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 assumptions on the flaw aspect ration. The provisions of Table IWB-2500-1, Examination B-D, continue to apply except that, in place of examination volumes, the surface 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-716	Alternative Piping Classification and Examination Requirements, Section XI, Division 1	N/A	Yes See Relief 1ISI-003
N-730	Roll Expansion of Class 1 Control Rod Drive Bottom Head Penetration in BWR's, Section XI, Division 1	N/A	Yes See Relief 1ISI-002

1 See Pressure Test Program
2 See Snubber Examination and Testing Program

APPENDIX J - CLASS 1, 2 AND 3 WELD/SUPPORT LOCATION MAPS

APPENDIX J
CLASS 1, 2 and 3 WELD/SUPPORT LOCATION MAPS

DRAWING NUMBER	SYSTEM	TITLE
C-26830-C	01	Main Steam
C-26831-C	02	Main Steam
C-26832-C	03	Main Steam
C-27165-C	05	Emergency Condenser Vent
C-27165-C	05.1	Emergency Condenser Vent
C-27165-C	05.2	Emergency Condenser Vent
C-26839-C	31	High Pressure Reactor Feedwater
C-26846-C	32	Reactor Recirculation
C-26852-C	33	Reactor Clean UP
C-26852-C	33.1	Reactor Clean UP
C-26839-C	33.2	Reactor Clean UP
C-26839-C	33.3	Reactor Clean UP
C-26851-C	34	Reactor Head Spray
F-45028-C	36	Reactor Instrumentation
C-26852-C	37	Reactor Drain
C-45060-C	37	Reactor Head Vent
C-26852-C	37.1	Reactor Drain
C-26847-C	38	Reactor Shutdown Cooling
C-26843-C	39	Emergency Condenser
C-26844-C	40	Reactor Core Spray
C-35682-C	40.1	Reactor Core Spray
C-27168-C	41	Liquid Poison

APPENDIX J
CLASS 1, 2 and 3 WELD/SUPPORT LOCATION MAPS

DRAWING NUMBER	SYSTEM	TITLE
C-27168-C	42	Liquid Poison
C-27168-C	42.1	Liquid Poison
C-26849-C	44.1	Control Rod Drive Hydraulic
C-34124-C	44.2	Control Rod Drive Hydraulic
C-27171-C	54	Fuel Pool Cooling
C-27178-C	66	Main Steam Line
C-26855-C	70	Reactor Building Closed Loop Cooling
C-39816-C	70	Closed Loop Cooling
C-27179-C	72	Service Water System
F-45236-C	72	Service Water System
C-26848-C	80	Reactor Containment Spray
C-27181-C	80.1	Reactor Containment Spray
C-26845-C	81	Reactor Core Spray
C-26845-C	81.1	Reactor Core Spray
C-27183-C	93	Containment Spray Cooling Raw Water
F-45183-C , Sheet 1	N/A	Weld Map Index
F-45183-C , Sheet 2	01	Main Steam
F-45183-C , Sheet 2A	01	Main Steam
F-45183-C , Sheet 2B	01	Main Steam
F-45183-C , Sheet 3	02	Main Steam
F-45183-C , Sheet 3A	02	Main Steam
F-45183-C , Sheet 4	03	Main Steam Turbine By Pass to Condenser
F-45183-C , Sheet 5	31	Reactor Feedwater
F-45183-C , Sheet 5A	31	Reactor Feedwater
F-45183-C , Sheet 6	34	Reactor Head Spray

APPENDIX J
CLASS 1, 2 and 3 WELD/SUPPORT LOCATION MAPS

DRAWING NUMBER	SYSTEM	TITLE
F-45183-C , Sheet 7	32	Reactor Recirculation
F-45183-C , Sheet 7A	32	Reactor Recirculation
F-45183-C , Sheet 7B	32	Reactor Recirculation Pump Discharge Instrumentation
F-45183-C , Sheet 7C	32	Reactor Recirculation Pump Instrumentation
F-45183-C , Sheet 7D	36	Reactor Recirculation Pump Seal Cavity Instrumentation
F-45183-C , Sheet 8	36	Reactor Instrumentation Level Protection System
F-45183-C , Sheet 8A	36	Reactor Instrumentation Level Control Range
F-45183-C , Sheet 8B	36	Reactor Instrumentation Level Control Range
F-45183-C , Sheet 8C	36	Reactor Instrumentation Level Protection System
F-45183-C , Sheet 8D	36	Reactor Instrumentation Wide Range Level
F-45183-C , Sheet 8E	36	Reactor Instrumentation Seal Leak Detection
F-45183-C , Sheet 8F	36	Reactor Instrumentation Level Triple Low
F-45183-C , Sheet 8G	36	Reactor Instrumentation Level Triple Low
F-45183-C , Sheet 8H	36	Reactor Instrumentation Emergency Condenser Steam
F-45183-C , Sheet 8J	36	Reactor Instrumentation Emergency Condenser Steam
F-45183-C , Sheet 9	37, 37.1	Reactor Head Vent and Drain
F-45183-C , Sheet 10	38	Reactor Shutdown Cooling
F-45183-C , Sheet 10A	38	Reactor Shutdown Cooling
F-45183-C , Sheet 11	39	Emergency Condenser Steam Supply
F-45183-C , Sheet 11A	39	Emergency Condenser Steam Return
F-45183-C , Sheet 11B	39	Emergency Condenser Drain and Overflow Lines
F-45183-C , Sheet 12	40, 40.1	Reactor Core Spray
F-45183-C , Sheet 12A	40,40.1	Reactor Core Spray
F-45183-C , Sheet 13	80,80.1	Reactor Core Spray
F-45183-C , Sheet 13A	80,80.1	Reactor Core Spray

APPENDIX J
CLASS 1, 2 and 3 WELD/SUPPORT LOCATION MAPS

DRAWING NUMBER	SYSTEM	TITLE
F-45183-C , Sheet 13B	80,80.1	Reactor Core Spray
F-45183-C , Sheet 13C	80,80.1	Reactor Core Spray
F-45183-C , Sheet 14	42.1	Liquid Poison
F-45183-C , Sheet 15	80	Reactor Containment Spray
F-45183-C , Sheet 15A	80	Reactor Containment Spray
F-45183-C , Sheet 15B	80	Reactor Containment Spray
F-45183-C , Sheet 15C	80	Reactor Containment Spray
F-45183-C , Sheet 15D	80	Reactor Containment Spray
F-45183-C , Sheet 16	93,93.1	Reactor Containment Spray
F-45183-C , Sheet 16A	93,93.1	Reactor Containment Spray
F-45183-C , Sheet 16B	93,93.1	Reactor Containment Spray
F-45183-C , Sheet 16C	93,93.1	Reactor Containment Spray
F-45183-C , Sheet 17	44.2	Control Rod Drive
F-45183-C , Sheet 17A	44.2	Control Rod Drive
F-45183-C , Sheet 18	44.1	Control Rod Drive
F-45183-C , Sheet 19	44.3	Control Rod Drive
F-45183-C , Sheet 20	33,33.2,33.3	Reactor Clean Up
F-45183-C , Sheet 21	27	Reactor Head Spray
F-45183-C , Sheet 22	05,05.1,05.2	Emergency Condenser Vent
F-45183-C , Sheet 22A	05,05.1,05.2	Emergency Condenser Vent
F-45183-C , Sheet 23	23	Main Steam Line Drain
F-45183-C , Sheet 24	66	Main Steam Power Operated Relief Valve Discharge
F-45183-C , Sheet 24A	66	Main Steam Power Operated Relief Valve Discharge
F-45183-C , Sheet 25	80.1	Reactor Containment Spray
F-45183-C , Sheet 25A	80.1	Reactor Containment Spray

APPENDIX J
CLASS 1, 2 and 3 WELD/SUPPORT LOCATION MAPS

DRAWING NUMBER	SYSTEM	TITLE
F-45183-C , Sheet 26	41,42	Liquid Poison
F-45183-C , Sheet 28	00.0	Reactor Vessel
F-45183-C , Sheet 28A	00.0	Reactor Vessel