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NINE MILE POINT
NUCLEAR STATION

P.O. Box 63
Lycoming, NY 13093

October 25, 2012

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

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 2; Docket No. 50-410

Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) for Implementation of a Risk-Informed, Safety-Based Inservice Inspection Program Based on ASME Code Case N-716

In accordance with 10 CFR 50.55a(a)(3)(i), Nine Mile Point Nuclear Station, LLC (NMPNS) requests authorization to implement a risk-informed, safety-based inservice inspection (ISI) program based on American Society of Mechanical Engineers (ASME) Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1," as documented in the enclosed 10 CFR 50.55a Request Number 2ISI-011, Rev. 00. The information provided in the enclosed request demonstrates that the proposed alternative provides an acceptable level of quality and safety.

NMPNS plans to implement the proposed alternative during the third ten-year ISI interval, which began on April 5, 2008 and is scheduled to end on April 4, 2018, and requests NRC approval by October 31, 2013 to facilitate planning for the 2014 refueling outage and for remainder of the third ten-year ISI interval.

This letter contains no new regulatory commitments. Should you have any questions regarding the information in this submittal, please contact John J. Dosa, Director Licensing, at (315) 349-5219.

Very truly yours,

Paul M. Swift
Manager Engineering Services

PMS/DEV

A047
NRR

Enclosure: Nine Mile Point Nuclear Station, Unit 2 – Third Inservice Inspection Interval, 10 CFR 50.55a Request Number 2ISI-011, Rev. 00

cc: Regional Administrator, Region I, NRC
Project Manager, NRC
Resident Inspector, NRC

ENCLOSURE

NINE MILE POINT NUCLEAR STATION, UNIT 2

THIRD INSERVICE INSPECTION INTERVAL

10 CFR 50.55a REQUEST NUMBER 2ISI-011, REV. 00

Nine Mile Point Nuclear Station, LLC
October 25, 2012

Nine Mile Point Nuclear Station, Unit 2
Third Inservice Inspection Interval
10 CFR 50.55a Request Number 2ISI-011 Rev. 00

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

A. COMPONENT IDENTIFICATION

System: Various Class 1 and 2 Systems

Class: Quality Groups A, and B (ASME Code Class 1, and 2)

Components Affected: All Class 1 and 2 Piping Welds – Examination Categories B-F, B-J, C-F-1, and C-F-2

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, 2004 Edition, No Addenda, Examination Tables IWB-2500-1 and IWC-2500-1, Examination Categories B-F, B-J, C-F-1, C-F-2 must receive inservice inspection during each successive 120-month (ten-year) interval.

The Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 requires Intergranular Stress Corrosion Cracking (IGSCC) Category A welds to be examined over the 10-year interval in accordance with the staff positions on schedule, methods, personnel and sample expansion.

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 modified by BWRVIP-75-A. Table 1 below reflects these requirements.

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 an alternative to the requirements of the ASME B&PVC, 2004 Edition, No 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.

NMPNS also requests an alternative to GL 88-01 staff positions, as modified by BWRVIP-75-A, on schedule, methods, personnel and sample expansion for Examination Category A welds (resistant materials) only.

NMPNS also requests authorization to use ASME Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1," for risk-informed / safety-based insights.

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 2 using risk-informed and safety based (RIS_B) insights.

The objective of the inservice inspection (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 the General Design Criteria and 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, "Revised Risk-Informed In-service Inspection Evaluation Procedure" (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 inservice 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.

EPRI provided support in the development of this submittal.

As stated within the EPRI-TR, no changes to the augmented inspection programs for Flow Accelerated Corrosion (FAC) or Intergranular Stress Corrosion Cracking (IGSCC) GL 88-01 (as modified by BWRVIP-75-A) Categories B through G welds are being made in the proposed RIS_B inspection program. The proposed RIS_B program will supersede augmented inspection programs for IGSCC resistant 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, 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 inservice 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 RIS_B inspection program is an alternative to the ASME Code Section XI inservice 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 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.

NMPNS plans to complete the current Third Ten-Year ISI Interval by implementing the ASME Code Case N-716 based RIS_B program during the Second and Third Inspection Periods. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met for the current Interval, which began on April 5, 2008 and is scheduled to end on April 4, 2018.

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 current ASME Section XI pressure testing program.

F. PRECEDENTS

NRC Safety Evaluation for Seabrook Station, Unit 1, Relief For Alternative 3AR-1, Use of a Risk-Informed, Safety-Based Inservice Inspection Program, dated June 21, 2012 (ML121320552).

NRC Safety Evaluation for Millstone Power Station, Unit No. 2, Issuance of Relief Request RR-04-11 Regarding Risk-Informed Inservice Inspection Program, dated March 27, 2012 (ML120800433).

NRC Safety Evaluation for Joseph M. Farley Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based Inservice Inspection Alternative for Class 1 and Class 2 Piping Welds, dated January 18, 2012 (ML12012A135).

NRC Safety Evaluation for River Bend Station, Unit 1, Relief for Alternative RBS-ISI-013, Use of a Risk-Informed, Safety-Based Inservice Inspection Program, dated June 30, 2010 (ML101730157).

NRC Safety Evaluation for Nine Mile Point Nuclear Station, Unit No. 1, Request for Alternative 1ISI-003, Request to Use ASME Code Case N-716 Associated with the Fourth 10-Year Inservice Inspection Interval, dated March 15, 2010 (ML100700034).

G. ATTACHMENTS

Attachment 1, Summary Submittal (Template), Application of ASME Code Case N-716, Risk-Informed / Safety-Based Inservice Inspection Program

H. REFERENCES

Refer to Attachment 1.

ATTACHMENT 1

SUMMARY SUBMITTAL (TEMPLATE)
APPLICATION OF ASME CODE CASE N-716

RISK-INFORMED / SAFETY-BASED
INSERVICE INSPECTION PROGRAM

REQUEST FOR ALTERNATIVE
2ISI-011, Rev. 00

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1 Introduction

Nine Mile Point Nuclear Station, Unit 2 (NMP2) is currently in the Third Ten-Year Inservice Inspection (ISI) Interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for inspection Program B. Nine Mile Point Nuclear Station, LLC (NMPNS) plans to complete the current (Third) ISI Interval by implementing a Risk-Informed Safety-Based (RIS_B) Program based on ASME Code Case N-716 during the Second and Third Inspection Periods of the Third Interval. The NMP2 Third Interval began on April 5, 2008 and is scheduled to end on April 4, 2018. The Second Inspection Period (of the Third Interval) began on April 5, 2011.

The ASME Section XI Code of record for the Third ISI Interval is the 2004 Edition, no Addenda, for Examination Categories B-F, B-J, C-F-1, and C-F-2, and Generic Letter (GL) 88-01 IGSCC resistant Category A Class 1 and 2 piping components. In the Second ISI Inspection Period of the Second Interval, NMPNS implemented a Risk-Informed ISI (RI-ISI) Program based on ASME Code Case N-578. NRC approval to adopt the Code Case N-578 alternative was documented in a letter dated May 31, 2001 (ML011420195). NMPNS requested and received approval to continue using the Code Case N-578 alternative in the current (Third) Interval, as documented in NRC letter dated December 1, 2008 (ML083190494). The delta-risk evaluations for both the approved Code Case N-578 program and the proposed Code Case N-716 program are based on a comparison to a traditional program based on ASME Section XI 1989 Edition requirements. The 1989 Edition of ASME Section XI was the code of record during development of the initial Code Case N-578 RI-ISI submittal.

The objective of this submittal is to provide the information required to support the NMPNS request to use an alternate RIS_B process for the inservice inspection of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1," 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, "Revised Risk-Informed In-service Inspection Evaluation Procedure" (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 (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 2), and RG 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for In-service Inspection of Piping" (Reference 3). Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Risk Assessment (PRA) Quality

The NMP2 PRA (Reference 4) is based on a detailed model of the plant that was originally developed from the NMP2 Individual Plant Examination (IPE) and NMP2 Individual Plant Examination for External Events (IPEEE) projects. The original model was reviewed by the NRC and underwent Boiling Water Reactor Owner's Group (BWROG) certification. NRC reviews of the IPE and IPEEE are documented in the NRC Staff evaluations on the IPE dated August 18, 1994 (TAC No M74437) and the IPEEE dated August 12, 1998 (TAC No M83646). The NRC concluded that the NMP2 process is capable of identifying the most likely severe accidents and no significant impacts on the PRA were identified.

The NMP2 PRA has since been upgraded. It is a Level 2, at-power model that includes both internal and external events. A major upgrade of the internal events portion of the model to meet the guidance of RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference

7), as well as the American Society of Mechanical Engineers and American National Standard (ASME/ANS) PRA Standard RA-Sa-2009, was completed in July 2009. A formal, BWROG-sponsored industry peer review of the upgraded internal events model was completed in August 2009. The peer review utilized the process described in Nuclear Energy Institute document NEI 05-04, "Process for Performing PRA Peer Reviews Using the ASME PRA Standard," January 2005, and the ASME/ANS PRA Standard. This review to ASME Capability Category II requirements confirmed that the PRA model met the guidance of RG 1.200, Revision 1, and ASME/ANS RA-Sa–2009. There were 18 findings identified by the peer review team. Appendix A contains a summary of these findings, including the status of the resolution for each finding and the potential impact of each finding on this RIS_B application. In summary, a majority of the findings were related to documentation that has no material impact on the results of this application. Resolution of the peer review findings to date has had a minor impact on the model and its quantitative results. Assessment of the remaining open peer review findings has determined that required model changes would result in minor reductions in model quantification results and, therefore, would have a negligible, if any, impact on the conclusions of this application.

Section 2 of EPRI TR 1021467-A, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs," concludes that quantification of external events will not change the conclusions derived from the RI-ISI process. As a result, there is no need to further consider external events.

The latest revision of the PRA model (Reference 4), which takes into account the NMP2 extended power uprate that was implemented during the spring 2012 refueling outage, was used in the development of the RIS-B evaluation.

Based on the above, NMPNS believes that the current PRA model, used in the RIS_B evaluation, has an acceptable quality to support this application.

2 Proposed Alternative to Current Inservice 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 inservice examination of piping welds, utilizing nondestructive examination (NDE) methods as amended by the application of Code Case N-578.

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 will remain unaffected by the proposed RIS_B program.

2.2 Augmented Programs

The impact of the RIS_B application on the various plant augmented inspection programs listed below were considered. This section documents only those plant augmented inspection programs that address common piping within the RIS_B application scope (e.g., Class 1 and 2 piping).

- The original plant augmented inspection program for high-energy line breaks, implemented in accordance with the NMP2 Updated Safety Analysis Report (USAR), was revised in accordance with the risk-informed break exclusion region methodology (RI-

BER) described in EPRI TR 1006937, Revision 0-A, "Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs." EPRI TR 1006937 was approved by the NRC in 2002. The results of the RI-BER application demonstrated that the volumetric examination requirement for this scope of piping could be reduced from 100% to approximately 12%. As a result, a minimum of 12% of the BER population will be examined during the course of each ten-year interval, which exceeds the 10% requirement imposed by Code Case N-716. NMP2 was a pilot plant for the Risk-Informed BER application.

- The NMP2 augmented inspection program for intergranular stress corrosion cracking (IGSCC) per GL 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," as modified by BWRVIP-75-A (Reference 15), is relied upon to manage this damage mechanism. GL 88-01 specifies the examination extent and frequency requirements for austenitic stainless steel welds classified as Categories A through G, depending on their susceptibility to IGSCC. In accordance with EPRI TR 112657, piping welds identified as Category A are considered resistant to IGSCC and are assigned a low failure potential provided no other damage mechanisms are present. Consequently, the weld examinations identified as Category A inspection locations are subsumed by the RIS_B Program. The existing NMP2 augmented inspection program for the other piping welds susceptible to IGSCC (Categories D and E) remains unaffected by the RIS_B Program submittal.

- The plant augmented inspection program for flow-accelerated corrosion (FAC) per Generic Letter (GL) 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RIS_B Program.

3 Risk-Informed / Safety-Based Inservice Inspection Process

The process used for the development of the RIS_B program conformed to the methodology described in Code Case N-716. The process applied involves the following steps:

- Safety Significance Determination
- Failure Potential Assessment
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

3.1 <u>Safety Significance Determination</u>

The systems assessed in the RIS_B Program are provided in Table 3.1. Piping and instrumentation diagrams and additional plant information, including the existing plant ISI Program, were used to define the system 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) of 10CFR50.55a.

(2) Applicable portions of the shutdown cooling pressure boundary function; i.e., 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., 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., 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 (greater than 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 NMP2, which is a boiling water reactor (BWR).

(4) Piping within the BER (greater than 4 inch NPS) for high-energy piping systems as defined by the owner. This may include Class 3 or non-class piping. As discussed in Section 2.2, NMP2 has a plant specific BER Program.

(5) Any piping segment, including segments subsumed into internal event initiating events, whose contribution to core damage frequency (CDF) is greater than 1E-06 (or 1E-07 for large early release frequency (LERF)) 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.

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 traditional 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 and lessons learned from the RIS_B pilot applications provide 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., the 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 of containment (e.g., portions of the main feedwater system in BWRs) shall be selected.

(5) A minimum of 10% of the welds within the 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%, this application results in selection of greater than 10% of the Class 1 welds. 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]		Class 3 and Non-class Welds[3]		All Piping Welds[4]	
Total	Selected	Total	Selected	Total	Selected	Total	Selected
997	109	1390	1	6	0	2393	110

Notes

(1) Includes all Category B-F and B-J locations. All 997 Class 1 piping weld locations are HSS.
(2) Includes all Category C-F-1 and C-F-2 locations. Of the 1390 Class 2 piping weld locations, 1383 weld locations are LSS and the remaining 7 are HSS (BER) welds in the ICS and MSS systems (system abbreviations are defined in Section 7).
(3) There are 4 Class 3 BER welds in the WCS system and 2 non-class BER welds in the FWS system.
(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. VT-2 visual examinations are scheduled in accordance with the pressure test program, which remains unaffected by the RIS_B Program.

3.3.1 Successive and Additional Examinations

RIS_B examinations will be performed to the requirements specified within Table 1 of Code Case N-716. The RIS_B program will determine, through an engineering evaluation, the root cause of any unacceptable flaw or relevant condition (exceeding the Code Case N-716 Table 1 acceptance standards) determined to be service-related (e.g., fatigue, wall loss, IGSCC, etc.) found during examination. The flaw evaluation, performed in accordance with ASME Section XI, IWB-3600, will account for responsible service conditions and degradation mechanisms to determine whether the element(s) will still perform their intended safety function during subsequent operation. If the flaw is acceptable for continued service, successive examinations will be scheduled per Section 6 of Code Case N-716. Elements not meeting this requirement will be repaired, replaced, or analyzed in accordance with the applicable ASME Code Edition and Addenda as identified in the ISI Program including approved Code Cases and alternatives.

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). ASME Section XI, IWB-3134(b) and IWB-3144(b) require submission of the analytical evaluation to the NRC. In addition, the evaluation will be documented in the Corrective Action Program and the Owner submittals required by ASME Section XI or approved alternatives.

The evaluation will include whether other elements on the segment or additional segments are subject to the same root cause and degradation mechanism. Additional examinations will be performed, to the requirements of Section 6 of Code Case N-716, on these elements up to a number equivalent to the number of elements requiring examinations on the segment or segments initially examined during the current outage. If unacceptable flaws are determined to be service related 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 service related root cause conditions or degradation mechanism.

3.3.2 Program Relief Requests

Consistent with previously approved RI-ISI submittals, NMPNS will calculate coverage and use additional examinations or techniques in the same manner as for traditional Section XI examinations and previous RI-ISI 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. In instances where a location may be found that does not meet >90 percent coverage (limited examination), an evaluation will be performed to ensure that the impact of the limited examination is acceptable as required by Footnote 3 to Table 1 of Code Case N-716. This evaluation will be completed as part of the periodic program update required by Section 7 of Code Case N-716.

A relief request will be submitted for all limited examinations per the guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

Request for Alternative 2ISI-007 pertaining to the application of Code Case N-578 will be withdrawn for use at NMP2 upon NRC approval of this RIS_B Program submittal.

3.4 Risk Impact Assessment

The RIS_B Program has been conducted in accordance with RG 1.174 and the requirements of Code Case N-716, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

This evaluation categorized segments as high safety significant or low safety significant in accordance with 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. As an example, for locations subject to thermal fatigue, inspection locations have an expanded volume and the examination is focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

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 requirements of RGs 1.174 and 1.178. The EPRI criterion requires that the 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-4 and 1E-5, respectively, were conservatively used except for the high pressure core spray (CSH) and reactor core isolation cooling (ICS) systems, where the CCDP for suction piping off the suppression pool had a higher CCDP. 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 are 1E-4 (CCDP) / 1E-5 (CLERP) and between Medium and Low consequence categories are 1E-6 (CCDP) / 1E-7 (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-5 to 3E-5 due to an update, it will remain below the 1E-4 threshold value and the change-in-risk evaluation would not require updating.

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 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 that verified that the LSS piping was not susceptible to water hammer, as documented in Reference 5. LSS piping may be susceptible to FAC; however, the susceptibility evaluation and examination for FAC is governed by the site FAC program. This review was conducted similar to that done for a traditional RI-ISI application. 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 a medium failure potential ("Assume Medium" in Table 3.4) for use in the change-in-risk assessment. Experience with previous industry RI-ISI applications shows this to be conservative.

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 loss of coolant accident (LOCA) in the CSH piping for NMP2 bounds large and small LOCA initiating events as well as other medium LOCA events).

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 1E-08 per Code Case N-716. Piping locations identified as medium failure potential have a likelihood of 2E-07 per Code Case N-716. 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 POD from application of the RIS_B approach.

Table 3.4 presents a summary of the change-in-risk (delta risk) for the proposed RIS_B Program versus ASME Section XI Code program requirements on a "per system" basis. The impact of FAC is not accounted for in Table 3.4 because the FAC degradation mechanism is addressed via the site augmented FAC 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, using estimated CCDP/CLERP values, has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and satisfies the acceptance criteria of RG 1.174 and Code Case N-716.

The Inspection selections for the original ASME Section XI program, the proposed Code Case N-716 program and the difference between those selections, are contained in Table 3.4 under the column headings SXI, RIS_B, and Delta respectively. The risk impact (change-in-risk) analysis included changes made to the original ASME Section XI inspections as a result of implementing Code Case N-716 and the results are displayed in the Delta column as either no change (represented by 0), an increase (represented by a positive number) or a decrease (represented by a negative number). To show that the use of a conservative upper bound CCDP/CLERP does not result in an optimistic calculation with regard to meeting the acceptance criteria (Code Case N-716 Section 5), a conservative sensitivity was conducted where the RIS_B selections were set equal to the ASME Section XI selections (Delta changed from positive number to zero). The acceptance criteria are met when the number of RIS_B selections is not allowed to exceed Section XI.

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 pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As referenced in Section 2.3 of EPRI TR 112657 and depicted in the Summary of the ASME White Paper 92-01-01, Revision 1, "Evaluation of In-service Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds" (Reference 8), this method has been ineffective in identifying leaks or failures. EPRI TR 112657 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 leaks 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 or LERF of greater than 1E-7 be included in the scope of the application. NMP2 did not identify any such piping (as documented in References 4 and 5).

All locations within the Class 1, 2, and 3 pressure boundaries will continue to be pressure tested in accordance with ASME Section XI Code requirements, regardless of their safety significance.

4 Implementation and Monitoring Program

4.1 Implementation

Upon approval of the proposed RIS_B Program, appropriate procedures and/or revisions to the existing inspection program that implement the guidelines described in EPRI TR 112657 and/or Code Case N-716 will be completed to implement and monitor the program. The new program will be integrated into the existing and subsequent ASME Section XI inservice inspection intervals. No changes to the Technical Specifications or Updated 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 implementation of the Code Case N-716 prescribed examination methods, details of the Code Case N-716 prescribed acceptance standards, pressure testing, corrective measures, documentation requirements, reporting requirements, and quality control requirements. Existing ASME Section XI program implementation documents will be retained and modified to address the RIS_B process.

4.2 Feedback (Monitoring)

The RIS_B Program is a living program that is required to be monitored periodically for changes that could impact the basis for which welds are selected for examination. Monitoring encompasses numerous facets, including the review of changes to the plant configuration, changes to operations that could affect the degradation assessment, a review of NMP2 NDE results, a review of site failure information from the NMPNS corrective action program, and a review of industry failure information from industry operating experience. Also included is a review of PRA changes for their impact on the RIS_B program. These reviews provide a feedback loop such that new relevant information is obtained that will ensure that the appropriate locations selected for examination are maintained. As a minimum, this review will be conducted on an ASME period basis. In addition, more frequent adjustment may be required as identified by NRC Bulletins or Generic Letters, or by industry and plant-specific feedback. Periodic updates will meet the guideline recommendations contained within Nuclear Energy Institute (NEI) 04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems." Changes will be reflected, as appropriate, in the future 10-Year inspection plan and schedule submittals as required by ASME Section XI, IWA-1400(c).

If a flaw or relevant condition is detected during examination, this adverse condition will be addressed by the corrective action program and procedures, and the ISI Program Plan. The following are appropriate actions to be taken:

(1) Identify - Examination results conclude there is an unacceptable flaw.
(2) Characterize - Determine if regulatory reporting is required and assess if an immediate safety or operation impact exists.
(3) Evaluate - Determine the cause and extent of the condition identified and develop a corrective action plan or plans.
(4) Decide - Make a decision to implement the corrective action plan.
(5) Implement - Complete the work necessary to correct the problem and prevent recurrence.
(6) Monitor - Ensure that the RIS_B program has been updated based on the completed corrective action.
(7) Trend - Identify conditions that are significant based on accumulation of similar issues.

For preservice examinations, NMPNS will follow the rules contained in Section 3.0 of Code Case N-716. Welds classified as HSS require a preservice 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 preservice inspection.

5 Proposed Inservice Inspection Program Plan Change

A comparison between the RIS_B Program and ASME Section XI inspection program requirements for in-scope piping is provided in Table 5.1.

NMP2 is currently in the Second Period of the Third Ten-Year ISI Interval. NMPNS plans to complete the current (Third) ISI Interval by implementing a Code Case N-716 based Risk-Informed Safety-Based Program during the Second Inspection Period of the Third Interval. The NMP2 Third Interval began on April 5, 2008. The Second Inspection Period (of the Third Interval) began on April 5, 2011 and includes the 2012 and 2014 refueling outages (RFO13 and RFO14). NMP2 has completed the First Period examinations as defined in the current ISI Program Plan including the approved Code Case N-578 alternatives satisfying ASME Section XI percentage requirements. In anticipation of implementation of the Code Case N-716 RIS_B program, weld exams have been rescheduled within the current Second ISI Period exam schedule (RFO13 and RFO14). Upon approval of this RIS_B submittal, NMPNS will remove the exams (moved from RFO13 to RFO14) from the RFO14 schedule to make the Second Inspection Period consistent with the proposed Code Case N-716 RIS_B exam schedule. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met for the current Interval.

As discussed in previous sections, implementation of the RIS_B program will not alter the augmented examination requirements for FAC, GL 88-01 (IGSCC) welds, or RI-BER welds in high-energy piping.

6 References/Documentation

1. Electric Power Research Institute Topical Report 112657, Revised Risk-Informed In-service Inspection Evaluation Procedure, Revision B-A, dated December 1999

2. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

3. Regulatory Guide 1.178, An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping

4. NMP2 PRA (Model 10U2Model) 09152011 EPU

5. CNG-NMP2-ISI-003-RI-001, ASME Code Case N-716 Evaluation – Nine Mile Point Unit 2 dated January 2012, Revision 00, prepared by J.H. Moody Consulting, Inc.

6. ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI, Division 1

7. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1, January 2007

8. Summary of the ASME White Paper 92-01-01, Revision 1, Evaluation of In-service Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds, dated July 1995

9. NER-2A-025, NMP2 RI-ISI BER Evaluation

10. Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, January 25, 1988

11. NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Revision 2, January 1988

12. Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning, dated May 2, 1989

13. Electric Power Research Institute Topical Report 1021467-A, Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs, June 2012

14. Nuclear Energy Institute (NEI) 04-05, Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems, April 2004

15. BWRVIP-75-A, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules, Final Report, October 2005 (EPRI Report 1012621)

7 List of Acronyms and System Abbreviations

Acronyms

BER	Break Exclusion Region
BWROG	Boiling Water Reactor Owners Group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
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
HELBCUU	High Energy Line Break - Cleanup System
HSS	High Safety Significant
IGSCC	Intergranular Stress Corrosion Cracking
ILOCA-OC	Isolable Loss of Coolant Accident - Outside Containment
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination External Events
ISI	Inservice Inspection
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LSS	Low Safety Significant
MIC	Microbiologically-Influenced Corrosion
MLOCAHS	Medium Loss of Coolant Accident - CSH System
NDE	Non-destructive Examination
NNS	Non-nuclear Safety
NPS	Nominal Pipe Size
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
RIS-BER	Risk-informed Break Exclusion Region
SDC	Shutdown Cooling
SP	Suction Piping
RPV	Reactor Pressure Vessel
SXI	ASME Section XI
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
TT	Thermal Transients

System Abbreviations

ASS	Auxiliary Steam
CSH	High Pressure Core Spray
CSL	Low Pressure Core Spray
DER	Drywell Equipment Drains
FWS	Feedwater
ICS	Reactor Core Isolation Cooling
ISC	Nuclear Boiler and Process Instrumentation
MSS	Main Steam
RCS	Reactor Recirculation
RDS	Control Rod Drive (CRD) Scram Discharge Volume
RHS	Residual Heat Removal
RPV	Reactor Pressure Vessel
SLS	Standby Liquid Control
WCS	Reactor Water Cleanup

Table 3.1
Code Case N-716 Safety Significance Determination

System[1]	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6[2]	High	Low
ASS	4							✓
CSH	21	✓					✓	
	164							✓
CSL	19	✓					✓	
	117							✓
DER	2	✓					✓	
FWS	72	✓					✓	
	27	✓			✓		✓	
	2				✓		✓	
ICS	65	✓					✓	
	5	✓			✓		✓	
	3				✓		✓	
	207							✓
ISC	19	✓					✓	
MSS	209	✓					✓	
	44	✓			✓		✓	
	4				✓		✓	
	90							✓
RCS	106	✓					✓	
RDS	2	✓					✓	
	76							✓
RHS	86	✓					✓	
	78	✓	✓				✓	
	725							✓
RPV	34	✓					✓	
SLS	50	✓					✓	
WCS	112	✓					✓	
	46	✓			✓		✓	
	4				✓		✓	
SUMMARY RESULTS FOR ALL SYSTEMS	797	✓					✓	
	122	✓			✓		✓	
	78	✓	✓				✓	
	13				✓		✓	
	1383							✓
TOTALS	**2393**						**1010**	**1383**

Notes:
(1) System abbreviations are defined in Section 7.
(2) Piping is also evaluated for impact on LERF > 1E-7.

Table 3.2
Failure Potential Assessment Summary

	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
System[1]	**TASCS**	**TT**	**IGSCC**	**TGSCC**	**ECSCC**	**PWSCC**	**MIC**	**PIT**	**CC**	**E-C**	**FAC**
ASS[3]											
CSH[3]	✓										
CSL[3]	✓										
DER											
FWS	✓										(2)
ICS[3]	✓										
ISC	✓										
MSS[3]	✓										
RCS			✓								
RDS[3]											
RHS[3]	✓									✓	
RPV			✓								
SLS	✓										
WCS	✓		✓								(2)

Notes:

(1) System abbreviations are defined in Section 7.

(2) The FAC Program has previously identified areas for inspection in these systems, but this has no impact on this application.

(3) A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the ASS in its entirety, as well as portions of the CSH, CSL, ICS, MSS, RDS, and RHS 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)	RCPB (OC)	BER	
ASS		4	None					0
CSH	11		TASCS	✓	✓			2
CSH	8		None	✓				0
CSH	2		None	✓		✓		1
CSH		164	None					0
CSL	8		TASCS	✓	✓			2
CSL	9		None	✓				0
CSL	2		None	✓		✓		1
CSL		117	None					0
DER	2		None	✓				1
FWS	25		TASCS	✓	✓			3
FWS	6		TASCS	✓	✓		✓	5
FWS	3		TASCS	✓			✓	0
FWS	12		TASCS	✓		✓	✓	2
FWS	47		None	✓	✓			0
FWS	6		None	✓		✓	✓	1
FWS	2		None				✓	0
ICS	9		TASCS	✓	✓			3
ICS	4		TASCS	✓				1
ICS	14		None	✓	✓			1
ICS	30		None	✓				0
ICS	2		None	✓	✓		✓	1
ICS	2		None	✓			✓	0
ICS	8		None	✓		✓		1
ICS	1		None	✓		✓	✓	1
ICS	3		None				✓	0
ICS		207	None					0
ISC	6		TASCS	✓	✓			2
ISC	13		None	✓	✓			0
MSS	10		TASCS	✓	✓			3
MSS	181		None	✓	✓			6
MSS	18		None	✓	✓		✓	9
MSS	8		None	✓				0
MSS	6		None	✓			✓	1
MSS	10		None	✓		✓		0
MSS	20		None	✓		✓	✓	7
MSS	4		None				✓	0
MSS		90	None					0
RCS	1		IGSCC	✓	✓			1
RCS	105		None	✓	✓			10
RDS	2		None	✓	✓			1
RDS		76	None					0
RHS	22		TASCS	✓	✓			6
RHS	4		EC	✓		✓		1
RHS	46		None	✓	✓			6

Table 3.3
Code Case N-716 Element Selections

System[(1)]	Weld Count		N-716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
RHS	68		None	✓				0
RHS	24		None	✓		✓		4
RHS		725	None					0
RPV	30		IGSCC	✓	✓			4
RPV	4		None	✓	✓			0
SLS	10		TASCS	✓	✓			4
SLS	26		None	✓				0
SLS	14		None	✓		✓		2
WCS	8		TASCS,IGSCC	✓	✓			5
WCS	10		IGSCC	✓	✓			3
WCS	11		TASCS	✓	✓			2
WCS	29		TASCS	✓		✓	✓	5
WCS	75		None	✓	✓			1
WCS	1		None	✓	✓		✓	1
WCS	8		None	✓				0
WCS	2		None	✓			✓	0
WCS	14		None	✓		✓	✓	0
WCS	4		None				✓	0
Summary Results All Systems	112		TASCS	✓	✓			27
	6		TASCS	✓	✓		✓	5
	3		TASCS	✓			✓	0
	41		TASCS	✓		✓	✓	7
	4		TASCS	✓				1
	8		TASCS,IGSCC	✓	✓			5
	41		IGSCC	✓	✓			8
	4		EC	✓		✓		1
	487		None	✓	✓			25
	21		None	✓	✓		✓	11
	159		None	✓				1
	10		None	✓			✓	1
	60		None	✓		✓		9
	41		None	✓		✓	✓	9
	13		None				✓	0
		1383	—					0
Totals	**1010**	**1383**						**110**

Note:

(1) System abbreviations are defined in Section 7.

Table 3.4
Risk Impact Analysis Results

System[1]	Safety Significance	Break Location	Failure Potential[3]		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI[2]	RIS_B[4]	Delta	w/POD	w/o POD	w/POD	w/o POD
ASS Total	Low	Class 2 LSS		Assume Medium	1	0	-1	1.00E-11	1.00E-11	1.00E-12	1.00E-12
CSH	High	LOCA	TASCS	Medium	3	2	-1	-5.40E-11	3.00E-11	-5.40E-12	3.00E-12
CSH	High	PLOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
CSH	High	PLOCA-OC	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSH	Low	Class 2 SP		Assume Medium	13	0	-13	2.60E-08	2.60E-08	2.60E-09	2.60E-09
CSH Total								**2.59E-08**	**2.60E-08**	**2.59E-09**	**2.60E-09**
CSL	High	LOCA	TASCS	Medium	3	2	-1	-5.40E-11	3.00E-11	-5.40E-12	3.00E-12
CSL	High	PLOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
CSL	High	PLOCA-OC	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSL	Low	Class 2 LSS		Assume Medium	10	0	-10	1.00E-10	1.00E-10	1.00E-11	1.00E-11
CSL Total								**4.65E-11**	**1.31E-10**	**4.65E-12**	**1.31E-11**
DER	High	PLOCA	None	Low	0	1	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
DER Total								**-5.00E-13**	**-5.00E-13**	**-5.00E-14**	**-5.00E-14**
FW	High	LOCA	TASCS	Medium	10	8	-2	-2.52E-10	6.00E-11	-2.52E-11	6.00E-12
FW	High	PLOCA	TASCS	Medium	2	0	-2	1.20E-11	2.00E-11	1.20E-12	2.00E-12
FW	High	PLOCA-OC	TASCS	Medium	4	2	-2	-3.60E-11	6.00E-11	-6.00E-12	1.00E-11
FW	High	LOCA	None	Low	2	0	-2	3.00E-12	3.00E-12	3.00E-13	3.00E-13
FW	High	PLOCA-OC	None	Low	4	1	-3	4.50E-12	4.50E-12	7.50E-13	7.50E-13
FW Total								**-2.69E-10**	**1.48E-10**	**-2.90E-11**	**1.91E-11**
ICS	High	LOCA	TASCS	Medium	3	3	0	-1.08E-10	0.00E+00	-1.08E-11	0.00E+00
ICS	High	PLOCA	TASCS	Medium	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
ICS	High	LOCA	None	Low	4	2	-2	3.00E-12	3.00E-12	3.00E-13	3.00E-13
ICS	High	PLOCA	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
ICS	High	PLOCA-OC	None	Low	3	2	-1	1.50E-12	1.50E-12	2.50E-13	2.50E-13
ICS	Low	Class 2 SP		Assume Medium	12	0	-12	2.40E-08	2.40E-08	2.40E-09	2.40E-09
ICS Total								**2.39E-08**	**2.40E-08**	**2.39E-09**	**2.40E-09**
ISC	High	LOCA	TASCS	Medium	0	2	2	-1.08E-10	-6.00E-11	-1.08E-11	-6.00E-12

Table 3.4
Risk Impact Analysis Results

System[(1)]	Safety Significance	Break Location	Failure Potential[(3)]		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI[(2)]	RIS_B[(4)]	Delta	w/POD	w/o POD	w/POD	w/o POD
ISC	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ISC Total								**-1.08E-10**	**-6.00E-11**	**-1.08E-11**	**-6.00E-12**
MSS	High	LOCA	TASCS	Medium	0	3	3	-1.62E-10	-9.00E-11	-1.62E-11	-9.00E-12
MSS	High	LOCA	None	Low	46	13	-33	4.95E-11	4.95E-11	4.95E-12	4.95E-12
MSS	High	PLOCA	None	Low	7	1	-6	3.00E-12	3.00E-12	3.00E-13	3.00E-13
MSS	High	PLOCA-OC	None	Low	21	7	-14	2.10E-11	2.10E-11	3.50E-12	3.50E-12
MSS	Low	Class 2 LSS		Assume Medium	5	0	-5	5.00E-11	5.00E-11	5.00E-12	5.00E-12
MS Total								**-3.85E-11**	**3.35E-11**	**-2.45E-12**	**4.75E-12**
RCS	High	LOCA	IGSCC	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RCS	High	LOCA	None	Low	26	10	-16	2.40E-11	2.40E-11	2.40E-12	2.40E-12
RCS Total								**2.40E-11**	**2.40E-11**	**2.40E-12**	**2.40E-12**
RDS	High	LOCA	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RDS	Low	Class 2 LSS		Assume Medium	6	0	-6	6.00E-11	6.00E-11	6.00E-12	6.00E-12
RDS Total								**6.00E-11**	**6.00E-11**	**6.00E-12**	**6.00E-12**
RHR	High	LOCA	TASCS	Medium	9	6	-3	-1.62E-10	9.00E-11	-1.62E-11	9.00E-12
RHR	High	PLOCA-OC	E-C	Medium	2	1	-1	3.00E-11	3.00E-11	5.00E-12	5.00E-12
RHR	High	LOCA	None	Low	15	6	-9	1.35E-11	1.35E-11	1.35E-12	1.35E-12
RHR	High	PLOCA	None	Low	6	0	-6	3.00E-12	3.00E-12	3.00E-13	3.00E-13
RHR	High	PLOCA-OC	None	Low	4	4	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RHR	Low	Class 2 LSS		Assume Medium	62	0	-62	6.20E-10	6.20E-10	6.20E-11	6.20E-11
RHR Total								**5.05E-10**	**7.57E-10**	**5.25E-11**	**7.77E-11**
RPV	High	LOCA	IGSCC	Medium	30	4	-26	3.90E-11	3.90E-11	3.90E-12	3.90E-12
RPV	High	LOCA	None	Low	2	0	-2	3.00E-12	3.00E-12	3.00E-13	3.00E-13
RPV Total								**4.20E-11**	**4.20E-11**	**4.20E-12**	**4.20E-12**
SLS	High	LOCA	TASCS	Medium	0	4	4	-2.16E-10	-1.20E-10	-2.16E-11	-1.20E-11
SLS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SLS	High	PLOCA-OC	None	Low	0	2	2	-3.00E-12	-3.00E-12	-5.00E-13	-5.00E-13

Table 3.4
Risk Impact Analysis Results

System(1)	Safety Significance	Break Location	Failure Potential(3)		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI(2)	RIS_B(4)	Delta	w/POD	w/o POD	w/POD	w/o POD
SLS Total								**-2.19E-10**	**-1.23E-10**	**-2.21E-11**	**-1.25E-11**
WCS	High	LOCA	TASCS,IGSCC	Medium	0	5	5	-1.50E-10	-1.50E-10	-2.50E-11	-2.50E-11
WCS	High	LOCA	IGSCC	Medium	4	3	-1	3.00E-11	3.00E-11	5.00E-12	5.00E-12
WCS	High	LOCA	TASCS	Medium	0	2	2	-1.08E-10	-6.00E-11	-1.08E-11	-6.00E-12
WCS	High	PLOCA-OC	TASCS	Medium	15	5	-10	0.00E+00	3.00E-10	0.00E+00	5.00E-11
WCS	High	LOCA	None	Low	4	2	-2	3.00E-12	3.00E-12	3.00E-13	3.00E-13
WCS	High	PLOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
WCS	High	PLOCA-OC	None	Low	4	0	-4	6.00E-12	6.00E-12	1.00E-12	1.00E-12
WCS Total								**-2.19E-10**	**1.30E-10**	**-2.95E-11**	**2.54E-11**
Grand Total					**351**	**108**	**-243**	**4.97E-08**	**5.12E-08**	**4.97E-09**	**5.14E-09**

Notes:

(1) System abbreviations are defined in Section 7.

(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 CCDP	Estimated CLERP	Consequence Rank	Upper / Lower Bound CCDP	Upper / Lower Bound CLERP	Description of Affected Piping
LOCA	3E-04	3E-05	HIGH	(U) 3E-04 (L) 1E-04	(U) 3E-05 (L) 1E-05	Unisolable RCPB piping of all sizes
The highest CCDP for Medium LOCA in CSH (MLOCAHS) was used (0.1 margin used for CLERP)						
ILOCA(1)	1E-06	1E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping between 1st and 2nd normally open isolation valve inside containment (WCS, ICS, MSS, FWS)
Calculated based on MLOCAHS CCDP of 3E-4 and valve fail to close probability of 3E-3 (0.1 margin used for CLERP)						
PLOCA	1E-06	1E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping between 1st and 2nd normally closed isolation valve inside containment (RHS, CSL, CSH, SLS)
Calculated based on MLOCAHS CCDP of 3E-4 and valve rupture probability of <1E-3 (0.1 margin used for CLERP)						
ILOCA-OC(2)	5E-05	5E-05	HIGH	(U) 3E-04 (L) 1E-04	(U) 5E-05 (L) 1E-05	Piping between penetration and outside containment isolation valve with normally open isolation valve inside containment (WCS, ICS, MSS, FWS)
Isolable LOCA outside containment CCDP based on initiating event HELBCUU CCDP of 5E-5 (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 (RHS, CSL, CSH, SLS)
Potential LOCA outside containment CCDP based on valve rupture probability <1E-3 and CCDP for ISLOCA <1E-2 (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 for ICS and CSH suction from suppression pool
Estimated based on upper bound for Medium Consequence						
Class 2 SP	2E-02	2E-03	HIGH	(U) 2E-02 (L) 1E-04	(U) 2E-03 (L) 1E-05	Class 2 ICS and CSH suction lines from the suppression pool designated as low safety significant.
ICS and CSL suction piping from the suppression pool although low frequency and low risk has a CCDP ~2E-2 and CLERP ~2E-3 (0.1 margin used for CLERP) (3)						

Notes:
(1) All welds located inside containment and beyond the first isolation valve are designated as PLOCA whether normally closed or normally open auto closed.
(2) All welds located outside containment and beyond the first isolation valve are designated as PLOCA-OC whether normally closed or normally open auto closed. Quantification is conservatively based on ILOCA-OC CCDP and CLERP.
(3) All ICS and CSH Class 2 LSS welds were conservatively assigned the "Class 2 SP" CCDP and CLERP in the risk impact quantification.

Table 3.6 NMP2 Risk Impact Results				
System	**With POD Credit**		**Without POD Credit**	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
ASS - Auxiliary Steam	1.00E-11	1.00E-12	1.00E-11	1.00E-12
CSH - High Pressure Core Spray	2.59E-08	2.59E-09	2.60E-08	2.60E-09
CSL - Low Pressure Core Spray	4.65E-11	4.65E-12	1.31E-10	1.31E-11
DER - Drywell Equipment Drain	-5.00E-13	-5.00E-14	-5.00E-13	-5.00E-14
FWS - Feedwater	-2.69E-10	-2.90E-11	1.48E-10	1.91E-11
ICS - Reactor Core Isolation Cooling	2.39E-08	2.39E-09	2.40E-08	2.40E-09
ISC - Nuclear Boiler and Process Instrument Lines	-1.08E-10	-1.08E-11	-6.00E-11	-6.00E-12
MSS - Main Steam	-3.85E-11	-2.45E-12	3.35E-11	4.75E-12
RCS - Reactor Recirculation	2.40E-11	2.40E-12	2.40E-11	2.40E-12
RDS - Control Rod Drive Scram Discharge Volume	6.00E-11	6.00E-12	6.00E-11	6.00E-12
RHS - Residual Heat Removal	5.05E-10	5.25E-11	7.57E-10	7.77E-11
RPV - Reactor Pressure Vessel	4.20E-11	4.20E-12	4.20E-11	4.20E-12
SLS - Standby Liquid Control	-2.19E-10	-2.21E-11	-1.23E-10	-1.25E-11
WCS - Reactor Water Cleanup	-2.19E-10	-2.95E-11	1.30E-10	2.54E-11
Total	**4.97E-08**	**4.96E-09**	**5.12E-08**	**5.14E-09**

Table 5.1
Inspection Location Selection Comparisons Between ASME Section XI 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			Volume	Surface	RIS_B	Other[(2)]
ASS		✓	LSS	n/a	Assume Medium	C-F-2	4	1		0	NA
CSH	✓		LOCA	TASCS	Medium	B-J	11	3		2	NA
CSH	✓		PLOCA	None	Low	B-J	8	1		0	NA
CSH	✓		PLOCA-OC	None	Low	B-J	2	1		1	NA
CSH		✓	LSS	n/a	Assume Medium	C-F-1, C-F-2	164	13	1	0	NA
CSL	✓		LOCA	TASCS	Medium	B-J	8	3		2	NA
CSL	✓		PLOCA	None	Low	B-J	9	1		0	NA
CSL	✓		PLOCA-OC	None	Low	B-J	2	1		1	NA
CSL		✓	LSS	n/a	Assume Medium	C-F-1, C-F-2	117	10		0	NA
DER	✓		PLOCA	None	Low	B-J	2	0		1	NA
FW	✓		LOCA	TASCS	Medium	B-J	31	10		8	NA
FW	✓		PLOCA	TASCS	Medium	B-J	3	2		0	NA
FW	✓		PLOCA-OC	TASCS	Medium	B-J	12	4		2	NA
FW	✓		LOCA	None	Low	B-J	47	2		0	NA
FW	✓		PLOCA-OC	None	Low	B-J, CL4	8	4		1	NA
ICS	✓		LOCA	TASCS	Medium	B-J	9	3		3	NA
ICS	✓		PLOCA	TASCS	Medium	B-J	4	0		1	NA
ICS	✓		LOCA	None	Low	B-J	16	4		2	NA
ICS	✓		PLOCA	None	Low	B-J	32	2		0	NA
ICS	✓		PLOCA-OC	None	Low	B-J, C-F-2	12	3	1	2	NA
ICS		✓	LSS	n/a	Assume Medium	C-F-1, C-F-2	207	12		0	NA
ISC	✓		LOCA	TASCS	Medium	B-J	6	0	1	2	NA
ISC	✓		LOCA	None	Low	B-F, B-J	13	0	11	0	NA
MSS	✓		LOCA	TASCS	Medium	B-J	10	0	2	3	NA
MSS	✓		LOCA	None	Low	B-J	199	46	15	13	2 VT-2
MSS	✓		PLOCA	None	Low	B-J	14	7		1	NA
MSS	✓		PLOCA-OC	None	Low	B-J, C-F-2	34	21		7	NA
MSS		✓	LSS	n/a	Assume Medium	C-F-2	90	5		0	NA
RCS	✓		LOCA	IGSCC	Medium	B-J	1	1		1	NA
RCS	✓		LOCA	None	Low	B-J	105	26		10	NA

Table 5.1
Inspection Location Selection Comparisons Between ASME Section XI 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			Volume	Surface	RIS_B	Other(2)
RDS	✓		LOCA	None	Low	B-J	2	1		1	NA
RDS		✓	LSS	n/a	Assume Medium	C-F-2	76	6		0	NA
RHR	✓		LOCA	TASCS	Medium	B-J	22	9		6	NA
RHR	✓		PLOCA-OC	E-C	Medium	B-J	4	2		1	NA
RHR	✓		LOCA	None	Low	B-J	46	15		6	NA
RHR	✓		PLOCA	None	Low	B-J	68	6	1	0	NA
RHR	✓		PLOCA-OC	None	Low	B-J	24	4		4	NA
RHR		✓	LSS	n/a	Assume Medium	C-F-1, C-F-2	725	62	3	0	NA
RPV	✓		LOCA	IGSCC	Medium	B-F	30	30		4	NA
RPV	✓		LOCA	None	Low	B-F, B-J	4	2	1	0	NA
SLS	✓		LOCA	TASCS	Medium	B-J	10	0	4	4	NA
SLS	✓		PLOCA	None	Low	B-J	26	0	1	0	NA
SLS	✓		PLOCA-OC	None	Low	B-J	14	0	1	2	NA
WCS	✓		LOCA	TASCS,IGSCC	Medium	B-J	8	0		5	NA
WCS	✓		LOCA	IGSCC	Medium	B-J	10	4		3	NA
WCS	✓		LOCA	TASCS	Medium	B-J	11	0	3	2	NA
WCS	✓		PLOCA-OC	TASCS	Medium	B-J	29	15		5	NA
WCS	✓		LOCA	None	Low	B-J	76	4		2	NA
WCS	✓		PLOCA	None	Low	B-J	10	1		0	NA
WCS	✓		PLOCA-OC	None	Low	B-J, CL3	18	4		0	NA

Table 5.1
Inspection Location Selection Comparisons Between ASME Section XI and Code Case N-716

Notes:

(1) System abbreviations are defined in Section 7.

(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.

(3) The failure potential rank for high safety significant (HSS) locations is then assigned as "High", "Medium", or "Low" depending upon potential susceptibility 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 A

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE NMP2 INTERNAL EVENTS PRA MODEL UPDATE

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
1-1	Demands from causes other than surveillance tests were not included in the collection of plant-specific data. (This Finding originated from Supporting Requirement (SR) DA-C6)	DA-C6 DA-C7	SR requires all types of demands be counted or estimated.	Include demands from the four causes listed in the SR. Perhaps use Mitigating System Performance Indicator (MSPI) estimates for MSPI components because that program includes all demands (except post maintenance test).	**Open - Insignificant Impact** This was looked at during the Unit 1 update and considered again during the Unit 2 update. It is slightly conservative and not considered significant to estimate using surveillance procedures. Note that MSPI no longer counts actual events.
1-2	Maintenance Rule unavailability data were used, which include unavailability during plant shutdowns if that component is required to be operable. SR states that only at power unavailability should be used. NUREG/CR-6890 Vol. 2, Table A-2, data indicate that DG unavailability during shutdown is 5 to 10 times higher than during power operation. (This Finding originated from SR DA-C-13)	DA-C13	SR specifically says to include UA events only occurring while the plant is at power.	Either exclude Maintenance Rule unavailability data while the plant is shut down, or provide more justification why using such data does not significantly affect the results if only at power unavailability were to be used.	**Closed - Minor Impact (Reduction)** Section 3 of the Data Analysis (DA) Notebook and the model were updated with a maintenance unavailability calculation that does not include unavailability during non-power operation.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
1-9	The selection of a failure probability of 1.0E-4 for the low-pressure system component(s) rupturing given exposure to RCS pressure and temperature is optimistic given the information provided in the referenced NUREG/CR-5603. (This Finding originated from SR LE-D4)	LE-D4	More realistic failure probabilities of 0.1 or 0.01 would increase the frequencies of these ISLOCA sequences by a factor of 100 to 1000.	Reconsider the 1.0E-4 failure probability or provide detailed justification for such a low probability.	**Closed - Minor Impact (Increase)** Section 5 of the DA Notebook was revised to provide a more detailed evaluation of the NMP2 piping and heat exchanger fragilities. As a result, the probability of rupture was revised in the model, which varies for each system from 0.05 to 0.003.
1-11	Several spray events identified (for example, FDSWCB1 and FDSWCB2 in Table 5.1 of the Internal Flooding (IF) Notebook, use flood frequencies rather than spray frequencies from EPRI Report 1013141. There could be others. (This Finding originated from SR IFEV-A5)	IFEV-A5	Incorrect frequencies (too low) were used for these internal flood initiators.	Use the spray frequencies for these initiating events. Check other internal flooding initiators for correct type and frequency.	**Closed - Minor Impact (Increase)** Reviewed the IF Notebook Main Report and Appendix B for potential spray events and frequency. The following changes were required: (1) Initiators FDSWCB1, FDSWCB2 and FDSWCB5 were changed to spray frequency initiating events because there is no detection and no propagation from these rooms. (2) North Auxiliary Bay panel impact corrected in Appendix B (no PRA impact). (3) Sections 4.3, 4.5, 4.6 and 5.4 of the IF Notebook were updated to include the screened spray events where PRA equipment was affected.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
2-5	P. 2-7 of the DA Notebook states that a Bayesian analysis was not done when there are no plant-specific failures. This is unacceptable for Category II or Category III. The discussion justifying not performing such updates on p. 2-6 and 2-7 of the DA Notebook is misleading because of the very small failure probabilities involved in the example given. Based on NUREG/CR-6928 parameters for distributions with as few as 200 to 1000 demands, the posterior mean could drop by a factor of 2. (This Finding originated from SR DA-D1)	DA-D1	It is not acceptable to skip performing a Bayesian update when zero plant-specific failures are observed.	Perform Bayesian update when data is available and zero plant-specific failures are observed, or, alternatively, show that it is unlikely to get the required number of demands to significantly change the failure probability for specific equipment showing zero failures.	**Closed - Minor Impact (Decrease)** Section 2 of the DA Notebook and model were updated with Bayesian analysis for zero events down to failure rates on the order of 1E-3. The conservatism of not performing this update for lower failure rates is shown to be minor.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update					
Finding	**Finding Description**	**Assoc. SR**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Code Case N-716 Impact**
2-6	A critical test of the posterior that is suggested in this Supporting Requirement is: (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate. There is at least one case in which data is inconsistent—Motor Operated Valve (MOV) (lake) fails to open. There were 6 failures in 150 demands. The prior from NUREG/CR-6928 for MOV FTO/C has a mean of 1.07 E-3. The method from NUREG/CR-6823, Sections 6.2.3.5 & 6.3.3.4, describe a method for consistency evaluation that suggests that greater than or equal to 2 failures would be inconsistent and that another prior should be used. There is no documentation of any NMP2 analysis like this. (This Finding originated from SR DA-D4)	DA-D4	Consistency between the plant-specific data and the prior was not evaluated. A representative example of such an inconsistency is provided.	Perform recommended consistency analyses for all data.	**Closed - Minor Impact (Increase)** Section 2.7 of DA Notebook was updated to include a test of key distributions with documentation of methodology. A few distributions were identified as potentially inconsistent (prior versus posterior and plant data). As a result, the uncertainty in the prior distribution was increased to be more representative of plant data.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
2-9	Section 2.12 of the Service Water System (SWS) Notebook, which deals with Component Spatial Information, needs a small improvement. It is stated that SWS is credited for operation after containment failure, but no justification is given for why it would be available, given spatial effects from containment failure. (This Finding originated from SR SY-B8)	SY-B8	This is an isolated example of weakness in the treatment of spatial effects. They are treated well in other notebooks. However, treatment of spatial effects is a clear requirement of the Standard.	Provide discussion of effects on SWS of containment failure.	**Closed - Documentation Only** Section 2.12 of SY.04 was corrected to address the fact that SWS is not affected by containment failure.
2-11	The list of sources of uncertainty has been omitted from Section 3.5 of the 125 Vdc SY Notebook. (This Finding originated from SR SY-C3)	SY-C3	This is an isolated occurrence of failing to provide this information; however, requirements of the ASME Standard to list sources of uncertainty are clear.	Discuss sources of uncertainty in the 125 Vdc SY Notebook.	**Closed - Documentation Only** A potential important uncertainty is associated with battery life, which was added to the Notebook.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
2-16	This SR requires identification of contributors to CDF. To satisfy Category II (and III) requires including structures, systems, and components (SSCs) and operator actions that contribute to Initiating Event (IE) frequencies. These are not included for NMP2, so only Category I has been met. (This Finding originated from SR QU-D6)	QU-D6	Since Category II requires including SSCs and operator actions that contribute to IE frequencies, this is a finding.	Identify CDF contribution from SSCs and operator actions that contribute to IE frequencies.	**Closed - Documentation Only** Support system initiating event fault trees have been added to the model. The IE Notebook refers to this. SY.00 Notebook provides methodology. Applicable SY notebooks develop the models. ------------------------ **Open - Documentation Only** Equipment and operator contributions will be developed in the Quantification (QU) Notebook. The IE Notebook will be updated with correction factors.
3-5	At the time of the Peer Review, various PRA documentation notebooks were not signed by performers, reviewers, or approvers. (This Finding originated from SR MU-F1)	MU-F1	The lack of signatures was widespread throughout the PRA notebooks. The preparer, reviewer, and approver signatures normally imply that they have concurred with the statements made in the associated documentation.	Obtain signatures from the personnel who were designated preparer, reviewer, or approver. Add lines for signature dates. Ensure documentation (PRA notebooks) reflects proper revision number.	**Closed - Documentation Only** The Peer Review issuance of all notebooks has been signed and issued.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
3-6	The IF Notebook describes a plant feature important in mitigation of flooding that could disable Div 1 and Div 2 switchgear – "There is an open door that is held open by a latch, which actuates to close door on a fire alarm." (pg 4.1-6). This is cited throughout the IF notebook in multiple places. This design change has not actually been installed, but an interim measure to block the door open has been taken. (This Finding originated from SRIFSO-B1)	IFSO-B1	This feature has a significant impact on IF results. The IF Notebook and model should accurately reflect current plant configuration.	Revise documentation (and flooding model, if required) to accurately reflect current plant configuration.	**Closed - Documentation Only** The IF Notebook was revised to indicate that doors are currently held open by door stop and there is a future modification which will hold doors open by latch. This was a documentation issue only.
3-8	An important plant modification associated with an internal flood event that could disable Div I and II Switchgear is not entered into the Configuration Risk Management Program (CRMP) database.	MU-A1	This modification has a significant impact on core damage frequency, and tracking of the modification is required by this SR and CNG-CM-1.01-3003, "Probabilistic Risk Assessment Configuration Control."	Enter and track this issue in the CRMP database.	**Closed - Documentation Only** CRMP 376 issued. No impact on model or results.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
4-7	Several system notebooks do not have a completed system walk down. (This Finding originated from SR SY-A4)	SY-A4	There are only 3 systems.	Provide completed system walk down checklist for those systems in Appendix C.	**Closed - Documentation Only** Only 3 System Notebooks (Automatic Depressurization System, Vapor Suppression and Reactor Recirc) did not have documented walk downs (NA was included) and it is stated that they are in the Drywell (inaccessible).
5-2	Routine system alignments contributing to initiating event frequencies are not included. (This Finding originated from SR IE-A6)	IE-A6	Does not meet IE-A6 Category II requirements.	Include routine system alignments in the calculation of initiating event frequencies, where applicable.	**Closed - Documentation Only** Routine alignments are already included in the average initiating event frequency development. In addition, the addition of support system initiating event fault trees to the model (see Finding 2-16) adds some important alignments for these systems. ------------------------ **Open - Insignificant** It would be a significant effort to add the type of factors that are typically reserved for EOOS risk management modeling such as ½ scram testing, etc. This will have to wait until a plant reliability program is developed (e.g., scram, turbine trip risk).

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update

Finding	Finding Description	Assoc. SR	Basis for Significance	Peer Review Team Suggested Resolution	Code Case N-716 Impact
6-1	In some cases the assignment of a conservative screening human error probability (HEP) value may not have been appropriate given the risk significance of the operator action it represents. In particular, the use of a conservative screening value of 1E-02 assigned to the HEP ZHS05_HSROOMCOL, "Operator Fails to open HPCS ROOM Doors and HVAC Duct," may not have been appropriate given the risk significance of the HPCS room cooling support system. (This Finding originated from SR HR-G1)	HR-G1	Failure to perform a detailed analysis for the estimation of HEPs that represent significant human failure events (HFEs).	Identify risk-significant HFEs in the PRA model, and perform detailed analysis using appropriate human reliability analysis (HRA) methodology(ies).	**Closed - Documentation Only** Section 1 of HRA Notebook updated to explicitly identify HEPs based on screening, the basis for screening, and their importance. ------------------------ **Open - Insignificant** Detailed HRA will be considered in future updates as appropriate.

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update					
Finding	**Finding Description**	**Assoc. SR**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Code Case N-716 Impact**
6-4	The most significant operator action in terms of importance (RRW = 2, RAW = 11) is ZZOHX, "Failure to Recover Heat Removal before Containment Failure." There does not appear to be a detailed analysis of this operator action with regard to procedure availability and operator training (nor is justification given for omission), nor were shaping factors and sufficiency of manpower for performing this recovery action included in the evaluation which documents this recovery action. (This Finding originated from SR HR-H2)	HR-H2	Failure to satisfy HR-H2 criteria for Capability Category I/II/III for significant operator action.	Perform a review of all significant operator recovery actions, and ensure that a detailed analysis is presented which includes consideration of procedure availability and operator training (or justification given for omission), as well as consideration of the shaping factors and sufficiency of manpower for performing the recovery actions.	**Closed - Documentation Only** ZZOHX is not an operator action. The modeling of recovery term ZZOHX includes an operator action ZOH01, which is a direct dependency for operators performing containment heat removal. ZZOHX is an equipment recovery value for failure to recover loss of containment heat removal, given ZOH01 was previously successful. Agree that the basis for ZZOHX in Section 5 of the DA Notebook needs improvement and this has been updated. Also, sufficiency of manpower for actions required after one day is not considered an issue.
6-5	The Accident Sequence (AS) Notebook does not contain the event tree top event fault trees, which are necessary for understanding the accident sequence logic. (This Finding originated from SR AS-C1)	AS-C1	The AS analysis documentation does not provide sufficient information to facilitate PRA applications, upgrades, and peer review.	Revise the AS Notebook to include all applicable top-logic fault trees, and additional description in the notebook to explain the top event logic.	**Closed - Documentation Only** The final post Peer Review issuance of the AS Notebook has all the documentation in the AS Notebook as suggested versus external (facilitates review etc).

Appendix A - Summary of Industry Peer Review Findings for the NMP2 Internal Events PRA Model Update					
Finding	**Finding Description**	**Assoc. SR**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Code Case N-716 Impact**
6-10	Based on a review of the design features, detection and response section, this supporting requirement appears to have been met for the above areas except for the South Aux Service Bldg. (This Finding originated from SR IFSN-A14)	IFSN-A14	Table 4-14 indicates that the South Aux Service Bldg can be screened based upon the presence of flood detection. The NMP2 IF Notebook, Section 4.2.6, does not indicate that there is detection for this area. The responsible Constellation engineer corroborated this conclusion.	Revise Table 4-14 to change YES to NO under the column for Criteria #3 for the South Aux Service Bldg.	**Closed - Documentation Only** Footnote (1) was added to the "Yes" which states "There is no detection in the South Aux Service Building. However, there is no PRA equipment here, the piping is relatively small and there is reliable detection, isolation and significant time available when propagation occurs to Turbine and or Control buildings."