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NL-12-028

January 30, 2012

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

SUBJECT: Relief Request IP3-ISI-RR-05 For Fourth Ten-Year Inservice Inspection Interval
Indian Point Unit Number 3
Docket No. 50-286
License No. DPR-64

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (Entergy) is submitting Relief Request IP3-ISI-RR-05 (Enclosure 1) for the Indian Point Unit No. 3 (IP3) Fourth 10-year Inservice Inspection (ISI) Interval. The enclosed relief request is for the application of Code Case N-716, "Alternative Piping Classification and Examination Requirements", to implement a risk informed/safety based Inservice Inspection (ISI) as an alternative to the ASME Section XI Inservice Inspection requirements. The attached bases concludes this request provides an acceptable level of quality and safety. This relief is requested under the provisions of 10CFR 50.55a(a)(3)(i).

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Licensing Manager at 914-254-6710.

Sincerely,

RW/sp

Enclosure 1. Relief Request IP3-ISI-RR-05 Proposed Alternative to Use ASME Code Case N-716

cc: Mr. John P. Boska, Senior Project Manager, NRC NRR DORL
Mr. William Dean, Regional Administrator, NRC Region 1
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Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
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Enclosure 1 TO NL-12-028

RELIEF REQUEST IP3-ISI-RR-05

PROPOSED ALTERNATIVE TO USE ASME CODE CASE N-716

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Indian Point Unit 3
Fourth 10-Year ISI Interval
Relief Request No: IP3-ISI-RR-05
Proposed Alternative to Use ASME Code Case N-716 Alternative Piping Classification And Examination Requirements

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

1. ASME Code Components Affected

All Class 1 and 2 piping welds – Examination Categories B-F, B-J, C-F-1 and C-F-2.

2. Applicable Codes Edition and Addenda

The applicable Code edition and addenda is ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 2001 Edition through the 2003 addenda. In addition, as required by 10 CFR50.55a, piping ultrasonic examinations are performed per ASME Section XI, 2001 Edition, Appendix VIII, Performance Demonstration for Ultrasonic Examination Systems.

3. Applicable Code Requirements

For the current inservice inspection (ISI) program at Indian Point 3, IWB-2200 IWB-2420, IWB-2430 AND IWB-2500 provide the examination requirements for Category B-F and Category B-J welds. Similarly, IWC-2200, IWC-2420, IWC-2430 and IWC-2500 provide the examination requirements for Category C-F-1 and C-F-2 welds.

4. Reason for the Request

The objective of this submittal is to request the use of a risk-informed/safety based (RIS_B) ISI process for the inservice inspection of Class 1 and 2 piping.

5. Proposed Alternative and Basis for Use

In lieu of the existing Code requirements, Indian Point 3 proposes to use a RIS_B process as an alternate to the current ISI program for 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.

Code Case N-716 is founded, in large part, on the RI-ISI process described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev B-A, Revised Risk-Informed Inservice Inspection Evaluation Procedure, December 1999 (ADAMS Accession No. ML013470102) which was previously reviewed and approved by the US Nuclear Regulatory Commission (NRC).

In general, a risk-informed program replaces the number and locations of nondestructive examination (NDE) inspections based on ASME Code, Section XI requirements with the number and locations of these inspections based on the risk-informed guidelines. These processes result in a program consistent with the concept that, by focusing inspections on the most safety-

significant welds, the number of inspections can be reduced while at the same time maintaining protection of public health and safety.

NRC approved EPRI TR 112657, Rev B-A includes steps which, when successfully applied, satisfy the guidance provided in Regulatory Guide (RG) 1.174, An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis and RG 1.178, An Approach for Plant-Specific Risk-Informed Decision Making for Inservice Inspection of Piping. These steps are:
Scope definition
Consequence evaluation
Degradation mechanism evaluation
Piping segment definition
Risk categorization
Inspection/NDE selection
Risk impact assessment
Implementation monitoring and feedback

These same steps were also applied to this RIS_B process and it is concluded that this RIS_B process alternative also meets the intent and principles of Regulatory Guides 1.174 and 1.178.

In general, the methodology in Code Case N-716 replaces a detailed evaluation of the safety significance of each pipe segment required by EPRI TR 112657, Rev B-A with a generic population of high safety-significant segments, supplemented with a rigorous flooding analysis to identify any plant-specific high safety-significant segments (Class 1, 2, 3 or Non-Class). The flooding analysis was performed in accordance with Regulatory Guide 1.200 and ASME RA-Sb-2009, Standard for Probabilistic Risk Assessment for Nuclear Plant Applications.

By using risk-insights to focus examinations on more important locations, while meeting the intent and principles of Regulatory Guide 1.174 and 1.178, this proposed RIS_B program will continue to maintain an acceptable level of quality and safety. Additionally, all piping components, regardless of risk classification, will continue to receive ASME Code-required pressure testing, as part of the current ASME Code, Section XI program. Therefore, approval for this alternative to the requirements of IWB-220, IWB-2420, IWB-2430 and IWB-2500 (Examination Categories C-F-1 and C-F-2) is requested in accordance with 10CFR50.55a(a)(3)(i). An Indian Point Unit 3 specific relief request is attached that mirrors previous RIS_B submittals to the NRC.

6. Duration of Proposed Alternative

Through July 20, 2019

7. Precedents

Similar alternatives have been approved for Vogtle Electric Generating Plant, Donald C. Cook 1 and 2, Grand Gulf Nuclear Station, Waterford-3 and North Anna 1 and 2.

8. References

Vogtle Electric Generating Plant Safety Evaluation – see ADAMS Accession No. ML100610470. DC Cook Safety Evaluation – see ADAMS Accession No. ML072620553. Grand Gulf Nuclear Station Safety Evaluation – see ADAMS Accession No. ML072430005. Waterford-3 Safety Evaluation – see ADAMS Accession No. ML080980120.

Technical Acronyms/Definitions Used in the Template

AC	Alternating Current
AF	Auxiliary Feedwater
AS	Accident Sequence Analysis
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
BER	Break Exclusion Region
CAFTA	Computer-Aided Fault Tree Analysis
CC	PRA abbreviation for Capacity Category
CC	Crevice Corrosion
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CIV	Containment Isolation Valve
Class 2 LSS	Class 2 Pipe Break in LSS Piping
CLERP	Conditional Large Early Release Probability
CV	Chemical Volume and Control System
DA	Data analysis
DC	Direct Current
DM	Degradation Mechanism
E-C	Erosion-Corrosion
ECSCC	External Chloride Stress Corrosion Cracking
EOOS	Equipment Out of Service
FAC	Flow-Accelerated Corrosion
F&O	Facts and Observations
FLB	Feedwater Line Break
FT	Fault tree
FW	Feedwater
HELB	High Energy Line Break (synonymous with BER)
HEP	Human Error Probability
HFE	Human Failure Event
HR	Human Reliability
HRA	Human Reliability Analysis
HSS	High Safety-Significant
IE	Initiating Events Analysis
IF	Internal Flooding
IFIV	Inside First Isolation Valve
IGSSC	Intergranular Stress Corrosion Cracking
ILOCA	Isolable Loss of Coolant Accident
IPE	Individual Plant Evaluation
LE	LERF Analysis
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident

Technical Acronyms/Definitions Used in the Template (Continued)

LOSP	Loss of Off-Site Power
LSS	Low Safety-Significant
MAAP	Modular Accident Analysis Program
MIC	Microbiologically-Influenced Corrosion
MOV	Motor Operated Valve
MS	Main Steam
MU	Model Update
NDE	Nondestructive Examination
NNS	Non-Nuclear Safety
NPS	Nominal Pipe Size
PBF	Pressure Boundary Failure
PIT	Pitting
PLOCA	Potential Loss of Coolant Accident
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWSCC	Primary Water SCC
QU	Quantification
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR, RH	Residual Heat Removal
RI-BER	Risk-Informed Break Exclusion Region
RI-ISI	Risk-Informed Inservice Inspection
RIS_B	Risk-Informed/Safety Based Inservice Inspection
RM	Risk Management
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SC	Success Criteria
SDC	Shutdown Cooling
SLB	Steam Line Break
SGTR	Steam Generator Tube Rupture
SSC	Systems, Structures, and Components
SR	Supporting Requirements
SW	Service Water
SXI	Section XI
SY	Systems Analysis
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
TR	Technical Report
TT	Thermal Transients
Vol	Volumetric

ENCLOSURE 1
INDIAN POINT UNIT 3 RIS_B PROGRAM TEMPLATE
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

Table of Contents

1. INTRODUCTION

Indian Point Unit 3 (IP3) is currently in the fourth 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. Indian Point Unit 3 plans to implement a risk-informed/safety-based inservice inspection (RIS_B) program in the first Period of the fourth ISI interval. The fourth ISI interval began on July 21, 2009.

The ASME Section XI Code of record for the fourth ISI interval is the 2001 Edition through the 2003 Addenda for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components.

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*, which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*.

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*, and Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decision making Inservice Inspection of Piping*. Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Safety Assessment (PSA) Quality

The methodology in Code Case N-716 provides for examination of a generic population of high safety significant (HSS) segments, supplemented with a rigorous flooding analysis to identify if any plant-specific HSS segments need to be added. Satisfying the requirement for the plant-specific analysis requires confidence that the flooding PRA is capable of successfully identifying any significant flooding contributors that are not identified in the generic population.

The Indian Point Unit 3 PRA is based on a detailed model of the plant that was originally developed for the Individual Plant Examination (IPE) and Individual Plant Examination for External Events (IPEEE) projects. The IP3 internal events PRA model has been upgraded since the original IPE to meet the guidance of RG 1.200 Rev 2 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," as well as the American Society of Mechanical Engineers and American National Standard (ASME/ANS) PRA Standard RA-Sa-2009.

A formal, PWROG-sponsored industry peer review of the upgraded internal events model was completed in December 2010. The peer review utilized the process described in Nuclear Energy Institute document NEI 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard," January 2005, and the ASME/ANS PRA Standard. This review confirmed that the PRA model met the requirements of RG 1.200, Revision 2, and ASME/ANS RA-Sa–2009. There were 11 findings identified by the peer review team.

Attachment 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 application.
The IP3 PRA technical capability evaluations and the maintenance and update processes described above and Attachment A provide a robust basis for concluding that the IP3 PRA model is suitable for use in the risk-informed process used for this application.

External Events are addressed in Parts 4 through 9 of the ASME/ANS standard. The EPRI Report 1021467 proposes a qualitative treatment of the risk from fire events and from events that impose extreme loads on piping systems. The NRC Safety Evaluation concurred in the TR conclusion that challenges from fire events are expected to be less frequent and not significantly different than challenges caused by the random occurrence of internal initiating events. The NRC SE also concluded that additional analysis of extreme loading events are not needed and will not change the conclusion derived from the RI-ISI program.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 currently contain requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components.

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be substituted for the current program 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 alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected.

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 with the RIS_B application scope (e.g., Class 1 and 2 piping).

- A plant augmented inspection program has been implemented in response to NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*. This program was updated in response to MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*. The thermal fatigue concern addressed was explicitly considered in the application of the RIS_B process and is subsumed by the RIS_B Program.

- The plant augmented inspection program for flow accelerated corrosion (FAC) per 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.

- Since the issuance of the NRC safety evaluation for EPRI TR 112657, Rev. B-A, several instances of primary water stress corrosion cracking (PWSCC) of unmitigated Alloy 82/182 welds has occurred at pressurized water reactors. For Indian Point Unit 3, the unmitigated Alloy 82/182 Category B-F dissimilar metal welds (greater than NPS 1) subject to PWSCC are the three RPV hot leg nozzle to safe-end welds and the three cold leg nozzle to safe-end welds. The Steam Generator dissimilar metal welds are not subject to PWSCC because the welds are Alloy 52/152, and all of the pressurizer dissimilar metal welds (and the adjacent stainless steel welds) greater than 1" Nominal Pipe Size (NPS) have been overlaid with Full Structural Weld Overlays (FSWOL). All of the overlaid welds have been removed from the risk-informed program and will be examined in accordance with the requirements set forth in the NRC safety evaluation for the weld overlays.

  Even though Code Case N-716 only considers the RPV hot leg nozzle Alloy 82/182 weld locations to be susceptible to PWSCC, Indian Point Unit 3 has selected 4 welds to be ultrasonically examined for PWSCC within the scope of Code Case N-716. Code Case N-716 requires the examination of these welds every ten years. However, the examination frequency for these eight welds is currently based on the frequencies established by the requirements of Materials Reliability Program (MRP)-139, Revision 1. MRP-139 currently requires that the unmitigated hot legs be examined on a five year frequency and the unmitigated cold legs be examined on a six year frequency. These frequencies are subject to change based on factors such as industry experience and issuance of NRC rule making. The RIS_B Program will not be used to eliminate any MRP-139 or regulatory requirements. Indian Point Unit 3 plans to manage Alloy 82/182 welds per the requirements of Code Case N-770-1 once the program has been formally implemented in 2013.

  Per Code Case N-716 (Table 1, Item No. 1.15, *Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC)*, selected butt welds are subject to volumetric examination. Per Note 3 of Table 1, the examination includes essentially 100% of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Areas with acceptable limited examinations (coverage less or equal to 90%), and their bases, shall be documented and submitted for relief per the requirements of 10 CFR 50.55a(g)(5)(iv).

3. RISK-INFORMED/SAFETY-BASED ISI PROCESS

The process used to develop the RIS_B Program conformed to the methodology described in Code Case N-716 and consisted of the following steps:

- Safety Significance Determination (see Section 3.1)
- Failure Potential Assessment (see Section 3.2)
- Element and NDE Selection (see Section 3.3)

- Risk Impact Assessment (see Section 3.4)
- Implementation Program (see Section 3.5)
- Feedback Loop (see Section 3.6)

Each of these six steps is discussed below:

3.1 Safety Significance Determination

The systems assessed in the RIS_B Program are provided in Table 3.1. The piping and instrumentation diagrams and additional plant information, including the existing plant ISI Program were used to define the piping system boundaries. Per Code Case N-716 requirements, piping welds are assigned safety-significance categories, which are then used to determine the examination treatment requirements. High safety-significant (HSS) welds are determined in accordance with the requirements below. Low safety-significant (LSS) welds include all other Class 2, 3, or Non-Class welds.

(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);

(2) Applicable portions of the shutdown cooling pressure boundary function. That is, 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 [> 4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the steam generator to the outer containment isolation valve;

(4) Piping within the break exclusion region (BER) greater than 4" NPS for high-energy piping systems as defined by the Owner. Per Code Case N-716, this may include Class 3 or Non-Class piping. There is no BER augmented program at Indian Point Unit 3.

(5) Any piping segment whose contribution to Core Damage Frequency (CDF) is greater than 1E-06 [and per NRC feedback on the Grand Gulf and D. C. Cook RIS_B applications 1E-07 for Large Early Release Frequency (LERF)] based upon a plant-specific PSA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping. Service water piping in the 480 Volt Switchgear Room was identified as HSS due to CDF exceeding 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 NRC approved EPRI TR-112657 (i.e., the EPRI RI-ISI methodology), with the exception of the deviation discussed below.

Table 3.2 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

As previously approved for Indian Point Unit 3 during last interval, a deviation to the EPRI RIS_B methodology has been implemented in the failure potential assessment. Table 3-16 of EPRI TR-112657 contains the following criteria for assessing the potential for Thermal Stratification, Cycling, and Striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 include:

1. The potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids; or

2. The potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids; or

3. The potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid; or

4. The potential exists for two phase (steam/water) flow; or

5. The potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow;

   AND

   ➢ $\Delta T > 50°F$,

   AND

   ➢ Richardson Number > 4 (this value predicts the potential buoyancy of a stratified flow)

These criteria, based on meeting a high cycle fatigue endurance limit with the actual ΔT assumed equal to the greatest potential ΔT for the transient, will identify locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS, where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology, that would allow consideration of fatigue severity, is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

- **Turbulent Penetration TASCS**

Turbulent penetration is a swirling vertical flow structure in a branch line induced by high velocity flow in the connected piping. It typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will tend to keep the line filled with hot water. If there is in-leakage of cold water, a cold stratified layer of water may be formed and significant top-to-bottom ΔTs may occur in the horizontal portion of the branch line. Interaction with the swirling motion from turbulent penetration may cause a periodic axial motion of the cold layer. Therefore, TASCS is considered for these configurations.

For similar upward sloping branch lines, if there is no potential for in-leakage, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore, TASCS is not considered for these no in-leakage configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

- **Low flow TASCS**

In some situations, the transient startup of a system (e.g., shutdown cooling suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

- **Valve leakage TASCS**

Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection Heating TASCS**

Similarly, there sometimes exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However, since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. Consideration of cycle severity was used in previous NRC approved RIS_B program submittals for D. C. Cook, Grand Gulf Nuclear Station, Waterford-3, and the Vogtle Electric Generating Plant as well as Indian Point Unit 3 during the past interval. The methodology used in the Indian Point Unit 3 RIS_B application for assessing TASCS potential conforms to these updated criteria. Additionally, materials reliability program (MRP) MRP-146 guidance on the subject of TASCS was also incorporated into the Indian Point Unit 3 RIS_B application.

3.3 Element and NDE Selection

Code Case N-716 and lessons learned from the Grand Gulf and DC Cook RIS_B applications provided 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 (for Indian Point Unit 3, because there are limited IFIV welds present in the RH and SI systems due to the fact that most branch lines are classified as RC out to the first isolation valve, the overall IFIV 2/3 requirement must be satisfied by selecting RC system welds in lieu of normal system-specific selections.):

   (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 inside first isolation valve (IFIV) (i.e., isolation valve closest to the RPV) and the RPV (for Indian Point Unit 3, because there are limited IFIV welds present in the RH and SI systems due to the fact that most branch lines are classified as RC out to the first isolation valve, the overall IFIV 2/3 requirement must be satisfied by selecting RC system welds in lieu of normal system-specific selections.).

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (not applicable for Indian Point Unit 3) shall be selected.

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected (not applicable to Indian Point Unit 3).

In contrast to a number of traditional 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% of the HSS welds be chosen. A brief summary of the number of welds and the number selected 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. Only those RIS_B inspection locations that receive a volumetric examination are included.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		All Piping Welds(3)(4)	
	Total	Selected	Total	Selected	Total	Selected
3	631	60	1111	30	1742	90

Notes:
(1) Includes all Category B-F and B-J locations. All Class 1 piping weld locations are HSS.
(2) Includes all Category C-F-1 and C-F-2 locations. Of the Class 2 piping weld locations, 292 are HSS; the remaining are LSS.
(3) Regardless of safety significance, Class 1, 2, and 3 ASME Section XI in-scope piping components will continue to be pressure tested as required by the ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the pressure test program that remains unaffected by the RIS_B Program.
(4) Class 3 Service water piping in the 480 Volt Switchgear Room was identified as HSS and is included in the RIS_B Program.

3.3.1 Current Examinations

Indian Point Unit 3 is currently using the NRC previously approved application using EPRI-TR 112657B-A.

3.3.2 Successive Examinations

If indications are detected during RIS_B ultrasonic examinations, they will be evaluated per IWB-3514 (Class 1) or IWC-3514 (Class 2) to determine their acceptability. Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3600 or IWC-3600, as appropriate. As part of this evaluation, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. If the flaw is acceptable for continued service, successive examinations will be scheduled per Section 6 of Code Case N-716. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000, applicable ASME Section XI Code Cases, or NRC approved alternatives. The IWB-3600 analytical evaluation will be submitted to the NRC. Finally, the evaluation will be documented in the corrective action program and the Owner submittals required by Section XI. Evaluation of indications attributed to PWSCC and successive examinations of PWSCC indications will be performed in accordance with MRP-139 or a subsequent NRC rule making.

3.3.3 Scope Expansion

If the nature and type of the flaw is service-induced, then welds subject to the same type of postulated degradation mechanism will be selected and examined per Section 6 of Code Case N-716. The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those 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 need be performed if there are no additional elements identified as being susceptible to the same root cause conditions. The need for extensive root cause analysis beyond that required for the IWB-3600 analytical 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).

Scope expansion for flaws characterized as PWSCC will be conducted in accordance with MRP-139 or subsequent NRC rule makings.

3.3.4 Program Relief Requests

Consistent with previously approved RIS_B submittals, Indian Point Unit 3 will calculate coverage and use additional examinations or techniques in the same manner it has for traditional 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 the examinations are performed. Relief requests for those cases where greater than 90% coverage is not obtained will be submitted per the requirements of 10 CFR 50.55a(g)(5)(iv).

No Indian Point Unit 3 relief requests are being withdrawn due to the RIS_B application.

3.4 Risk Impact Assessment

The RIS_B Program development has been conducted in accordance with Regulatory Guide 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 were proposed for each system. The changes included 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

Code Case N-716 has adopted the NRC approved 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 Regulatory Guides 1.174 and 1.178. Section 3.7.2 of EPRI TR-112657 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, Conditional Core Damage Probability (CCDP)/Conditional Large Early Release Probability (CLERP) values of 1E-4/1E-5 were conservatively used. 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 risk-informed ISI (RI-ISI) methodology. As such, the goal is to determine CCDPs/CLERPs threshold values. For example, the threshold values between High and Medium consequence categories is 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; the change-in-risk evaluation would not require updating.

The updated internal flooding PRA was also reviewed to ensure that there is no LSS Class 2 piping with a CCDP/CLERP greater than 1E-4/1E-5. This review identified some piping in the RHR and SI systems located outside of containment with a CCDP greater than 1E-4. As a result, all LSS welds in these systems are conservatively assigned CCDP/CLERP equal to 1.2E-2 / 1.2E-3.

With respect to assigning failure potentials for LSS piping, the criteria are defined in Table 3 of Code Case N-716. That is, those locations identified as susceptible to FAC 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, unless they have an identified potential for water hammer loads. In such cases, they will be assigned a high failure potential. Finally, 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. LSS piping may be susceptible to FAC; however, the examination for FAC is performed per the FAC program. This review was conducted similar to that done for a traditional RI-ISI application. Thus, 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. Experience with previous industry RIS_B applications shows this to be conservative.

Indian Point Unit 3 has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716 that is consistent with the "Simplified Risk Quantification 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 the table below. Consistent with the EPRI 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., Large LOCA CCDP bounds the medium and small LOCA CCDPs).

<table>
<tr><th colspan="6">CCDP and CLERP Values Based on Break Location</th></tr>
<tr><th rowspan="2">Break Location Designation</th><th colspan="2">Estimated</th><th rowspan="2">Consequence Rank</th><th colspan="2">Upper Bound</th></tr>
<tr><th>CCDP</th><th>CLERP</th><th>CCDP</th><th>CLERP</th></tr>
<tr><td>LOCA</td><td>8E-03</td><td>8E-04</td><td>HIGH</td><td>8E-03</td><td>8E-04</td></tr>
<tr><td colspan="6">RCPB pipe breaks that result in a loss of coolant accident - The highest CCDP for Large LOCA, IE-A, was used (0.1 margin used for CLERP). Unisolable RCPB piping of all sizes.</td></tr>
<tr><td>PLOCA[1][2]</td><td>3E-05</td><td>3E-06</td><td>MEDIUM</td><td>1E-04</td><td>1E-05</td></tr>
<tr><td colspan="6">Isolable or Potential LOCA (1 open valve or 1 closed valve) inside containment - RCPB pipe breaks that result in an isolable or potential LOCA - Calculated based on Large LOCA CCDP of 8E-3 and valve fail to close probability of ~3E-3 (0.1 margin used for CLERP). Between 1st and 2nd isolation valve inside drywell.</td></tr>
<tr><td>PPLOCA[1]</td><td><1E-5</td><td><1E-06</td><td>MEDIUM</td><td>1E-04</td><td>1E-05</td></tr>
<tr><td colspan="6">Potential LOCA (2 closed valves) inside containment – Based on failure of two normally closed valves in series from the ISLOCA analysis. Applies to RHR shutdown cooling suction and discharge paths. Although the CCDP is less than 1E-6, 1E-5 is used as a bounding value in consideration of RHR operation during shutdown.</td></tr>
<tr><td>FB</td><td><1E-05</td><td><1E-06</td><td>MEDIUM</td><td>1E-04</td><td>1E-05</td></tr>
<tr><td colspan="6">Feedwater breaks based on bounding value for IE-T4, T5U, T5D and IE-FLD-AF-1 (0.1 margin used for CLERP)</td></tr>
<tr><td>Class 2 LSS</td><td>1E-04</td><td>1E-05</td><td>MEDIUM</td><td>1E-04</td><td>1E-05</td></tr>
<tr><td colspan="6">Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant except for AF and SI - Estimated based on upper bound for Medium Consequence.</td></tr>
<tr><td>Class 2 LSS SIS</td><td>1.2E-02</td><td>1.2E-03</td><td>MEDIUM</td><td>1.2E-02</td><td>1.2E-03</td></tr>
<tr><td colspan="6">Class 2 pipe break with internal flooding CCDP > 1E-4. The 1.2E-2 value is conservatively applied to all RHR and SI piping (only certain pipe sections apply on El 32 of PAB from IE-FLD-PB-31).</td></tr>
</table>

1. The PRA does not explicitly model potential and isolable LOCA events, because such events are subsumed by the LOCA initiators in the PRA. That is, the frequency of a LOCA in this limited piping downstream of the first RCPB isolation valve times the probability that the valve fails is a small contributor to the total LOCA frequency. The N-716 methodology must evaluate these segments individually; thus, it is necessary to estimate their contribution. This is estimated by taking the LOCA CCDP and multiplying it by the valve failure probability.

2. PLOCA is identified and used in the quantification of both ILOCA and PLOCA

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_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. 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 an increased POD from application of the RIS_B approach.

Table 3.4 presents a summary of the RIS_B Program versus the ASME Section XI 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, was 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 damage 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" (the implementation of the RIS_B program) 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 the following tables, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and that the acceptance criteria of Regulatory Guide 1.174 and Code Case N-716 are satisfied.

System	With POD Credit		Without POD Credit	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
CH - Chemical Volume & Control	-8.65E-09	-8.65E-10	-4.80E-09	-4.80E-10
FW – Feedwater	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC - Reactor Coolant	-1.14E-08	-1.14E-09	-3.68E-09	-3.68E-10
RHR - Residual Heat Removal	1.43E-08	1.43E-09	1.43E-08	1.43E-09
SI - Safety Injection	1.34E-08	1.34E-09	1.79E-08	1.79E-09
CS - Containment Spray	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS - Main Steam	1.80E-10	1.80E-11	1.80E-10	1.80E-11
Total	**7.89E-09**	**7.89E-10**	**2.39E-08**	**2.39E-09**

As shown in Table 3.4, new RIS_B locations were selected such that the RIS_B selections exceed the Section XI selections for certain categories (Delta column has a positive 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, a conservative sensitivity was conducted where the RIS_B selections were set equal to the 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 10 CFR 50.55a 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 terminal end locations, structural discontinuities, and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this methodology 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; that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure 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. Secondly, a generic assessment of high-consequence sites has been determined by Code Case N-716, 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 1E-07 for LERF) be included in the scope of the application. Indian Point Unit 3 identified Class 3 service water piping in the 480 Volt Switchgear Room as HSS.

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.

3.5 Implementation

Upon approval of the RIS_B Program, procedures that comply with the guidelines described in Code Case N-716 will be prepared to implement and monitor the program. The new program will be implemented during the fourth ISI interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements.

Existing ASME Section XI program implementing procedures will be retained and modified to address the RIS_B process, as appropriate.

3.6 Feedback (Monitoring)

The RIS_B Program is a living program that is required to be monitored continuously 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 NDE results, a review of site failure information from the corrective action program, and a review of industry failure information from industry operating experience (OE). 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 identification of HSS piping locations selected for examination is maintained. As a minimum, this review will be conducted on an ASME period basis. In addition, more frequent adjustment may be required as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

If an adverse condition, such as an unacceptable flaw is detected during examinations, the adverse condition will be addressed by the corrective action program and procedures. The following are appropriate actions to be taken:

A. Identify (Examination results conclude there is an unacceptable flaw).
B. Characterize (Determine if regulatory reporting is required and assess if an immediate safety or operation impact exists).
C. Evaluate (Determine the cause and extent of the condition identified and develop a corrective action plan or plans).
D. Decide (make a decision to implement the corrective action plan).
E. Implement (complete the work necessary to correct the problem and prevent recurrence).
F. Monitor (through the audit process ensure that the RIS_B program has been updated based on the completed corrective action).
G. Trend (Identify conditions that are significant based on accumulation of similar issues).

For preservice examinations, Indian Point Unit 3 will follow the rules contained in Section 3.0 of N-716. Welds classified HSS require a preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of N-716. Welds classified as LSS do not require preservice inspection.

4. PROPOSED ISI PLAN CHANGE

Indian Point Unit 3 is currently in the first period of the fourth ISI interval and plans to implement this RIS_B submittal for the entire fourth interval. The traditional ASME Section XI weld selections and inspections are being implemented until approval. In anticipation of the approval of this RIS_B submittal, any traditional ASME Section XI selected welds that require examination during the 1st Period prior to approval will also meet the examination requirements of Table 1 of Code Case N-716. After approval of the RIS_B submittal, those welds in the RIS_B scope that were examined during the 1st period that also met Table 1 requirements may be credited toward the RIS_B requirements for the Period.

As discussed in Section 2.2, implementation of the RIS_B program will not alter any PWSCC examination requirements for the Alloy 82/182 examinations.

A comparison between the RIS_B Program and the previous Section XI program requirements for in-scope piping is provided in Table 4. For Class 1 piping welds, this includes inspections conducted for the 2nd interval (prior to the N578 application in the 3rd interval) and for Class 2 piping welds, this included inspections conducted for the 3rd interval. In addition, service water piping in the 480 Volt Switchgear Room was identified as high safety significant and is included in the RIS_B Program. Ten percent of the welds will be inspected during the interval. No degradation mechanism was identified for this piping, but a wall thickness type of volumetric exam will be conducted since this is considered most relevant to service water systems.

5. REFERENCES/DOCUMENTATION

EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs.*

EPRI TR-112657, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A.

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

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

Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decision making Inservice Inspection of Piping.*

Regulatory Guide 1.200, Rev 2 *An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities.*

USNRC 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.

USNRC 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.

EPRI Report 1021467 Nondestructive Evaluation: *Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs.*

Supporting Onsite Documentation

Structural Integrity Report 0800767.302, Rev 0 “N-716 Evaluation for Indian Point Unit 3”

RELIEF REQUEST IP3-ISI-RR-05 ENCLOSURE 1
INDIAN POINT UNIT 3 RIS_B PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

Table 3.1
Code Case N-716 Safety Significance Determination

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
CH	69	✓					✓	
FW	64			✓			✓	
RC	254	✓					✓	
	4	✓	✓				✓	
RHR	20	✓	✓				✓	
	228	□	✓				✓	
	171							✓
SI	234	✓					✓	
	50	✓	✓				✓	
	391							✓
CS	74							✓
MS	183							✓
Summary Results for all Systems	557	✓					✓	
	74	✓	✓				✓	
	228	□	✓				✓	
	64			✓			✓	
	819							✓
TOTAL	1742							

(1) System Scope:
CH = Chemical Volume and Control System
FW = Main Feedwater
RC = Reactor Coolant
RHR = Residual Heat Removal
SI = Safety Injection
CS = Containment Spray
MS = Main Steam

(2) Service water piping in the 480 volt switchgear room is included in the HSS scope

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**
CH	✓	✓									
FW											
RC	✓	✓				✓					
RHR											
SI		✓	✓								
CS											
MS											

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 CS and MS in its entirety, as well as portions of the RHR and SI systems.

Table 3.3: Code Case N716 Selections

System	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CH	7		TT	✓	✓			3
	7		TT	✓				1
	3		TASCS	✓	✓			3
	42		None	✓	✓			0
	10		None	✓				0
FW	64		None					7
RC	13		TT	✓	✓			1
	9		TT,TASCS	✓	✓			7
	3		TASCS	✓	✓			2
	4		PWSCC	✓	✓			4
	192		None	✓	✓			12
	37		None	✓				0
RHR	7		None	✓	✓			2
	13		None	✓				0
	228		None					23
		171	Assumed None					0
SI	4		TT,IGSCC	✓				1
	14		IGSCC	✓				4
	20		TT	✓	✓			5
	70		None	✓	✓			15
	176		None	✓				4
		391	Assumed None					0
CS		74	Assumed None					0
MS		183	Assumed None					0
Summary Results All Systems	40		TT	✓	✓			9
	7		TT	✓				1
	9		TT,TASCS	✓	✓			7
	6		TASCS	✓	✓			5
	4		PWSCC	✓	✓			4
	4		TT,IGSCC	✓				1
	14		IGSCC	✓				4
	311		None	✓	✓			29
	236		None	✓				4
	292		None					30
		819	None					0
Totals	**923**	**819**						**94**

Note: Systems are described in Table 3.1

Table 3.4: Risk Impact Analysis Results

System	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
CH	High	LOCA	TT	Medium	0	3	3	-4.32E-09	-2.40E-09	-4.32E-10	-2.40E-10
CH	High	LOCA	TASCS	Medium	0	3	3	-4.32E-09	-2.40E-09	-4.32E-10	-2.40E-10
CH	High	PLOCA	TT	Medium	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
CH	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH Total								**-8.66E-09**	**-4.81E-09**	**-8.66E-10**	**-4.81E-10**
FW	High	FB	None	Low	7	7	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FW Total								**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
RC	High	LOCA	TT	Medium	2	1	-1	-4.80E-10	8.00E-10	-4.80E-11	8.00E-11
RC	High	LOCA	TT,TASCS	Medium	2	7	5	-9.12E-09	-4.00E-09	-9.12E-10	-4.00E-10
RC	High	LOCA	TASCS	Medium	0	2	2	-2.88E-09	-1.60E-09	-2.88E-10	-1.60E-10
RC	High	LOCA	PWSCC	Medium	4	4	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC	High	LOCA	None	Low	40	12	-28	1.12E-09	1.12E-09	1.12E-10	1.12E-10
RC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC Total								**-1.14E-08**	**-3.68E-09**	**-1.14E-09**	**-3.68E-10**
RH	High	LOCA	None	Low	0	2	2	-8.00E-11	-8.00E-11	-8.00E-12	-8.00E-12
RH	High	PLOCA	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
RH	High	PPLOCA	None	Low	28	23	-5	2.50E-12	2.50E-12	2.50E-13	2.50E-13
RH	Low	Class 2 LSS SIS		Assume Medium	12	0	-12	1.44E-08	1.44E-08	1.44E-09	1.44E-09
RH Total								**1.43E-08**	**1.43E-08**	**1.43E-09**	**1.43E-09**
SI	High	LOCA	TT	Medium	4	5	1	-5.28E-09	-8.00E-10	-5.28E-10	-8.00E-11
SI	High	PLOCA	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
SI	High	PLOCA	IGSCC	Medium	0	4	4	-4.00E-11	-4.00E-11	-4.00E-12	-4.00E-12
SI	High	LOCA	None	Low	2	15	13	-5.20E-10	-5.20E-10	-5.20E-11	-5.20E-11
SI	High	PLOCA	None	Low	13	0	-13	6.50E-12	6.50E-12	6.50E-13	6.50E-13
SI	Low	Class 2 LSS SIS		Assume Medium	16	0	-16	1.92E-08	1.92E-08	1.92E-09	1.92E-09
SI Total								**1.34E-08**	**1.78E-08**	**1.34E-09**	**1.78E-09**
CS Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
MS Total	Low	Class 2 LSS		Assume Medium	18	0	-18	**1.80E-10**	**1.80E-10**	**1.80E-11**	**1.80E-11**
Grand Total					**151**	**90**	**-61**	**7.84E-09**	**2.39E-08**	**7.84E-10**	**2.39E-09**

Notes

1. Systems are described in Table 3.1

2. Only those ASME Section XI Code inspection locations that received a volumetric 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. Only those RIS_B inspection locations that receive a volumetric examination are included in the count. Locations subjected to VT2 only are not credited in the count for risk impact assessment.
4. The failure potential rank for high safety significant (HSS) locations is 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")
5. The "LSS" designation is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criteria of Section 2(a) of N-716 (e.g., not part of the BER scope).

Table 4: Inspection Location Selections Comparison

System	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
CH	✓		LOCA	TT	Medium	B-J	7	0	1	3	NA
CH	✓		LOCA	TASCS	Medium	B-J	3	0	0	3	NA
CH	✓		PLOCA	TT	Medium	B-J	7	0	0	1	NA
CH	✓		LOCA	None	Low	B-J	42	0	12	0	NA
CH	✓		PLOCA	None	Low	B-J	10	0	6	0	NA
FW	✓		FB	None	Low	C-F-2	64	7	0	7	NA
RC	✓		LOCA	TT	Medium	B-F, B-J	13	2	1	1	NA
RC	✓		LOCA	TT,TASCS	Medium	B-J	9	2	0	7	NA
RC	✓		LOCA	TASCS	Medium	B-J	3	0	1	2	NA
RC	✓		LOCA	PWSCC	Medium	B-F	4	4	0	4	NA
RC	✓		LOCA	None	Low	B-F, B-J	192	40	35	12	NA
RC	✓		PLOCA	None	Low	B-J	37	0	5	0	NA
RH	✓		LOCA	None	Low	B-J	7	0	0	2	NA
RH	✓		PLOCA	None	Low	B-J	13	3	0	0	NA
RH	✓		PPLOCA	None	Low	C-F-1	228	28	0	23	NA
RH		✓	Class 2 LSS SIS		Assume Medium	C-F-1	171	12	1	0	NA
SI	✓		LOCA	TT	Medium	B-J	20	4	0	5	NA
SI	✓		PLOCA	TT, IGSCC	Medium	B-J	4	0	0	1	NA
SI	✓		PLOCA	IGSCC	Medium	B-J	14	0	2	4	NA
SI	✓		LOCA	None	Low	B-J	70	2	6	15	NA
SI	✓		PLOCA	None	Low	B-J	176	13	59	0	4
SI		✓	Class 2 LSS IF		Assume Medium	C-F-1	391	16	19	0	NA
CS		✓	Class 2 LSS IF		Assume Medium	C-F-1	74	0	0	0	NA
MS		✓	Class 2 LSS		Assume Medium	C-F-2	183	18	3	0	NA
						Total	1742	151	151	90	4

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. This option is not applicable for the Indian Point Unit 3 RIS_B application. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals and to indicate when RIS_B selections will receive a VT-2 examination (these are not credited in risk impact assessment).
3. The failure potential rank for high safety significant (HSS) locations is 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").

The Indian Point 3 (IP3) Probabilistic Risk Assessment (PRA) model used for this application [Reference 1] is the most recent evaluation of the IP3 risk profile for internal event challenges. The IP3 PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause failure events. The PRA model quantification process used for the IP3 PRA is based on the event tree and fault tree methodology, which is a well-known methodology in the industry.

Entergy employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Entergy nuclear power plants. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the IP3 PRA model.

PRA Maintenance and Update

The Entergy risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the Entergy fleet procedure EN-DC-151, "PSA Maintenance and Update" [Reference 2]. This procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Entergy nuclear power plants. In addition, the procedure also defines the process for implementing regularly scheduled and interim PRA model updates, and for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.). To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. Potential PRA model changes resulting from these reviews are entered into the Model Change Request (MCR) database, and a determination is made regarding the significance of the change with respect to current PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years, and
- Industry standards, experience, and technologies are periodically reviewed to ensure that any changes are appropriately incorporated into the models.

In addition, following each periodic PRA model update, Entergy performs a self assessment to assure that the PRA quality and expectations for all current applications are met. The Entergy PRA maintenance and update procedure requires updating of all risk informed applications that may have been impacted by the update.

Regulatory Guide 1.200 PWROG Peer Review of the IP3 Internal Events PRA Model

The IP3 PRA internal events model went through a Regulatory Guide 1.200 PWR Owners Group peer review in December 2010. The NEI 05-04 process [Reference 3], the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard

[Reference 4], and Regulatory Guide 1.200, Rev. 2 [Reference 5]) were used for the peer review.

The 2010 IP3 PRA Peer Review was a full-scope review of all the Technical Elements of the internal events, at-power PRA:

- Initiating Events Analysis (IE)
- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Internal Flooding (IF)
- Quantification (QU)
- LERF Analysis (LE)
- Maintenance and Update Process (MU)

During the IP3 PRA model Peer Review, the technical elements identified above were assessed with respect to Capability Category II criteria to better focus the Supporting Requirement assessments. The ASME/ANS PRA Standard has 326 individual Supporting Requirements. Eleven (11) of the ASME/ANS PRA Standard Supporting Requirements are not applicable to IP3 (e.g., BWR related, multi-site related). Of the 315 ASME/ANS PRA Standard Supporting Requirements applicable to the IP3 PRA model, approximately 97% were satisfied at Capability Category II criteria or greater. The Facts and Observations (F&Os) for the IP3 PRA peer review are provided in the report, entitled, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 3 Probabilistic Risk Assessment" [Reference 6]. Of the 68 Facts and Observations (F&Os) generated by the Peer Review Team, 11 were considered Findings, 52 were Suggestions, and five were Best Practices.

As a result of the Regulatory Guide 1.200 PWROG peer review, all the abovementioned F&Os (other than best practices) have been identified as potential improvements to the IP3 PRA model and are tracked in the Entergy Model Change Request (MCR) database. Table A-1 contains the findings resulting from the peer review, including the status of the resolution for each finding and the potential impact of each finding on this application. In summary, a majority of the findings were related to documentation and have no material impact. Resolution of the peer review findings is expected to have a minor impact on the model and its quantitative results and will have a negligible, if any, impact on the conclusions of this application.

In resolving the peer review findings, several additional internal flooding sources were identified as not being addressed in the original internal flooding analysis report. Most of those sources involved fire protection piping, but they also included auxiliary component cooling water (ACCW) piping in the fan house and short sections of component cooling water (CCW) piping in a pipe chase in the foyer outside the charging pump rooms. These additional sources are described in more detail in Table A-2, including their expected impact on this application.

It should be noted that, while the model documentation has been revised to resolve most of the documentation related findings, since the revised documents will be formally issued with the final update package, those findings are considered resolved but will not be considered

closed until the final revised model and report, addressing the peer review findings, are formally released, which is expected later this year.

External Events

External Events are addressed in Parts 4 through 9 of the ASME/ANS standard. The EPRI Topical Report (TR) [Reference 7] proposes a qualitative treatment of the risk from fire events and from events that impose extreme loads on piping systems. The NRC Safety Evaluation concurred in the TR conclusion that challenges from fire events are expected to be less frequent and not significantly different than challenges caused by the random occurrence of internal initiating events. The NRC SE also concluded that additional analysis of extreme loading events are not needed and will not change the conclusion derived from the RI-ISI program.

Summary

The IP3 PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the IP3 PRA model is suitable for use in the risk-informed process used for this application.

<u>References</u>

[1] Engineering Report, IP3-RPT-10-00023, Rev.0, "Indian Point Unit 3 Probabilistic Safety Assessment (PSA)", November 2010.

[2] Entergy Fleet Procedure EN-DC-151, Revision 2, "PSA Maintenance and Update", January 2011.

[3] NEI 05-04, Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard, Nuclear Energy Institute, Rev. 2, November 2008.

[4] American Society of Mechanical Engineers/American Nuclear Society, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-Sa-2009), February 2009.

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

[6] PWR Owners Group LTR-RAM-I-11-055, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 3 Probabilistic Risk Assessment," October 2011.

[7] EPRI Technical Report 1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs", July 2011.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
1-11	Appendix C1 of IP-RPT-10-00023, Rev. 0 provides a high to medium level summary of the flood scenarios, and provides greater depth in some areas. Analysis details available to the peer review team such as flooding calculations, were not sufficient to support upgrades and would have to be obtained or reproduced for future model changes. The documentation also lacks in reference to quantification input documentation (initiator specific flag files) (This F&O originated from SR IFSN-B1)	IFSN-B1	Analysis details available to the peer review team such as flooding calculations, were not sufficient to support upgrades and would have to be obtained or reproduced for future model changes. The documentation also lacks in reference to quantification input documentation (initiator specific flag files)	Provide required documentation	**Resolved - No impact** This is a documentation issue that would impact future model updates and upgrades. The backup spreadsheets used for flooding rates and frequency calculations have been obtained as well as the software used for flood level calculations, instructions for use of this software and material that supports its application. This additional documentation will be included in the final model documentation package. Initiator specific flag files exist but were not included in either the internal flooding or quantification notebooks but are contained in the electronic files to be included in the model update documentation package. These flag files will be added to the internal flooding notebook as well.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
1-12	The walkdown notes in Appendix A of IP-RPT-10-00023, Rev. 0, Appendix C.1 note the general location of each SSC with respect to its room and elevation as well as its submergence height. Some additional general locational information is sometimes identified in Section 4.2 of IP-RPT-10-00023, Rev. 0, Appendix C.1. For example, it may state that a flood source may impact one but not both trains of equipment; specifics are not given as to why both cannot be impacted (e.g., shielding, curbs, etc.), but the information implies the impact of spatial information. There is no specific physical location information related to spray type failures found in the documentation. SSCs are only identified locationally by their flood area and elevation. It cannot be determined which SSCs in any area are susceptible to spray from any specific spray source. (This F&O originated from SR IFSN-A5)	IFSN-A5	There is no specific physical location information found in the documentation for SSCs other than flood area and elevation. Therefore, it cannot be determined which SSCs in any area are susceptible to spray from any specific spray source. In the scenario development it identifies which equipment is impacted by spray, but it cannot be determined how that information was obtained or if it is correct.	For SSCs susceptible to spray failure (also see F&O 2-3), ensure sufficient relational location information between the target SSC and spray sources are provided so that a determination can be made as to whether the SSCs can be damaged by each potential spray source.	**Resolved – No Impact** Additional discussion has been added to the walkdown Appendix to support the spray impacts included in the model. This includes reference to environmental qualification documents where these were used as a basis for stating that equipment would not be vulnerable to spray damage. A conservative separation criterion of 30 feet was used in examining the potential for spray impacts in the analysis. The composite piping and general arrangement drawings were scrutinized to ascertain whether equipment could be sprayed should a line or other piece of equipment rupture. The text of the report has been changed to note this. Providing additional specific location information within the model documentation will be considered to support future updates but is considered a documentation enhancement issue with no expected impact on the analysis.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
3-7	The effects of the flood on PSFs were not specifically addressed in the HRA analysis. (This F&O originated from SR IFQU-A6)	IFQU-A6	Limited flooding-related human actions are included in the HRA discussion in Appendix H, but there is no mention of any affects of the flood on PSFs.	Discuss flood effects on PSFs and make adjustments to the HRA analysis if needed.	**Open – Minor Impact (Increase)** No short term isolation actions were credited in the flooding analysis. The only significant field actions credited in the internal events model that could be impacted by the plant conditions associated with flooding are alignment of alternate cooling to the charging pumps on loss of CCW and operator actions associated with locally operating the turbine-driven AFW (TDAFW) pump. Major flood scenarios in these areas would fail the components involved, thus rendering any impact on operator actions moot. Lesser flood or spray scenarios could affect operator actions. With respect to the need to locally align the TDAFW Pump, this action is only required coincident with a station blackout, or a substantial number of other failures. Since the combined frequency of the flooding events that could impact this action is approximately 1E-5/yr, it is reasonable to conclude that such scenarios would be well below the criteria for low safety significance and would have no significant impact on the application. With respect to the operator action to align city water to the charging pumps, since flooding initiator IE-FLD-PB-8 has the same impact as assuming failure of this operator action (i.e. both CCW and backup city water to the charging pumps are lost) the effect of any HRA impacts on the flood scenarios in this area were bounded by assuming the operator actions were precluded by the flood event and comparing the impact to this existing flooding initiator. The frequency of a failure of the CCW piping in the charging pump foyer that would require operator alignment of city water and could impact that action is approximately 2.3E-6 per year. Existing flood initiator IE-FLD-PB8 has a flood frequency of 2.76E-6 and CDF and LERF contributions of 5.85 E-7/yr and 1E-9/yr, respectively. Since both of these impacts are below the criteria for low safety significance, the impact of assuming that the operator action is precluded for breaks in the charging pump foyer would be similar.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
6-6	Supporting requirement IFSO-A4 is intentionally not met as stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Section 3.3: 'The one supporting requirement of the ASME standard that we have made no attempt to meet is IF-B2: "for each potential source of flooding, identify the mechanisms that would result in a flooding release". In this analysis, no distinction was made between the various causes of floods because the rupture frequencies used included all floods." (This F&O originated from SR IFSO-A4)	IFSO-A4	This supporting requirement is intentionally not met as stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Section 3.3: 'The one supporting requirement of the ASME standard that we have made no attempt to meet is IF-B2: "for each potential source of flooding, identify the mechanisms that would result in a flooding release". In this analysis, no distinction was made between the various causes of floods because the rupture frequencies used included all floods."	Identify the flooding mechanisms that would result in a release for each potential source of flooding to meet the SR.	**Resolved - No impact** The intent of the statement in the report was to acknowledge that the EPRI data used for the analysis included all rupture mechanisms that contribute to piping system failures and to note there are no readily available data that would allow us to distinguish between different release mechanisms. The identification of specific causes of failure is therefore a documentation issue. The only contributor not included in the EPRI data is human induced flooding events. Since no applicable generic data exists related to human induced events, plant specific condition reports were reviewed for applicable events (none were identified) and discussions were held with plant operations personnel. Based on those discussions, activities that could challenge system integrity such as large scale movements of water and plant modifications are typically performed during outages and would not constitute significant contributors to flooding risk. Nonetheless, the model documentation has been modified to specifically discuss both failure mechanisms and the conclusions of these human induced failure evaluations.
6-7	As stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Table 3.3.1.1 for IFSO-A5, maximum flow rate resulting from a guillotine rupture is determined and used, instead of identifying the characteristic of release for different failure mechanism. (This F&O originated from SR IFSO-A5)	IFSO-A5	As stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Table 3.3.1.1 for IFSO-A5, maximum flow rate resulting from a guillotine rupture is determined and used, instead of identifying the characteristic of release for different failure mechanism. This is in contrary to the SR.	Identify the characteristic of release for each source and its identified failure mechanism.	**Resolved – No Impact** No impact. We consider this a documentation issue. While the table mentioned in the finding did state that a maximum flow rate resulting from a guillotine rupture was determined, it also noted that the frequency of this and lesser releases were calculated. A range of release sizes consistent with the available EPRI pipe rupture frequency data were, in fact, considered and a flow rate and frequency of occurrence derived for each. By this means, the size and frequency of possible releases were matched as required for the quantitative determination of the consequences of internal flooding. The text in the report has been modified to clarify this matter. Additional information regarding the pressures and temperatures of the ruptured systems has also been added to the documentation.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
6-8	IP-RPT-10-00023, Rev. 0, Appendix C1, *Section 4.1.3 states that the potential* flood sources were identified by walkdowns and the examination of drawings, and listed in Appendix A, Plant Walkdown. However, Appendix A does not provide adequate information on flood source as (1) some flood areas are not *included in the walkdown such as* 3PAB41-1A, 43-60A, 46-73A, 55-63A, 3FH72-B, 3FH80-A, etc.; (2) Appendix A has stressed that the walkdown notes do NOT provide a definitive listing of all equipment and lines or other flood sources. Also other fluid sources have not been considered in the analysis. (This F&O originated from SR IFSO-A1)	IFSO-A1	IP-RPT-10-00023, Rev. 0, Appendix C1, Section 4.1.3 states that the potential flood sources were identified by walkdowns and the examination of drawings, and listed in Appendix A, Plant Walkdown. However, Appendix A does not provide adequate information on flood source as (1) some flood areas are not included in the walkdown such as 3PAB41-1A, 43-60A, 46-73A, 55-63A, 3FH72-B, 3FH80-A, etc.; (2) Appendix A has stressed that the walkdown notes do NOT provide a definitive listing of all equipment and lines or other flood sources. Also other fluid sources have not been considered in the analysis.	Identify the potential sources of flooding for each flood area per the standard. Perform and document walkdowns for missed flood areas. If these areas cannot be walked down for operational or health reasons, other methods of obtaining this data (e.g., plant drawings, operator interviews, etc.) should be employed and *documented.* Prepare an integrated list of the internal flood source.	**Resolved – No Impact** All accessible fire areas were included in the plant walkdowns. Appendix A has been revised to include the areas that were omitted from the documentation, including those areas mentioned in the finding. The statement in the introduction to the walkdown notes was intended only to acknowledge that there might be small bore, field run piping (less than 1 inch diameter) that were not shown on system drawings and would not have been confirmed by the walkdown. Such small bore pipes were not considered to be significant flood sources.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
6-11	IP-RPT-10-00023, Rev. 0, Appendix C, Section 4.1.3, which is the section in the main report for flood sources, just refers Appendix A, Plant Walkdown for the information. There is no list of the internal flood sources in the analysis that may facilitate PRA applications, upgrades, and peer review. (This F&O originated from SR IFSO-B1)	IFSO-B1	There is no list of the internal flood sources in the analysis that may facilitate PRA applications, upgrades, and peer review. It could facilitate applications, update and review if sources were identified in the main report.	Prepare an integrated list of the internal flood source.	**Resolved - No impact** This is documentation issue. A list of internal flooding sources has been developed and will be included in a new Table 4.2.1.1 in the final update report. This table identifies all the flooding sources in each area and identifies adjacent or lower areas through which floodwater might propagate.
6-12	IP-RPT-10-00023, Rev. 0, Appendix C identifies applicable flood sources in its Appendix A, Plant Walkdown, which is not *adequate for process documentation* purpose. For example, the walkdown notes stressed that they do NOT provide a definitive listing of all equipment and lines or other flood sources; there is no list of sources to be examined. (This F&O originated from SR IFSO-B2)	IFSO-B2	IP-RPT-10-00023, Rev. 0, Appendix C identifies applicable flood sources in its Appendix A, Plant Walkdown, which is not adequate for process documentation purpose. For example, the walkdown notes stressed that they do NOT provide a definitive listing of all equipment and lines or other flood sources; there is no list of sources to be examined.	Provide adequate documentation on the process used to identify applicable flood sources	**Resolved - No impact** Although Section 3.1.2 previously described the process for identifying flooding sources, additional description has been added to that section and an additional table (Table 4.2.1.1) has been added, which provides additional detail describing the sources in each flood zone. In any case, this is an issue of enhanced documentation and does not impact this application. The statement in the introduction to the walkdown notes was intended only to acknowledge that there might be small bore, field run piping (less than 1 inch diameter) that were not shown on system drawings and would not have been confirmed by the walkdown. Such small bore pipes were not considered to be significant flood sources.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
6-1	The justification/statement that the CST inventory is sufficient for AFW for 24 hrs should be enhanced. (This F&O originated from SR SC-B1)	SC-B1	The justification/ statement that the CST inventory is sufficient for AFW for 24 hrs should be enhanced. IP-RPT-10-00023, Rev. 0, Appendix B, Section B1.3.1.3.2 states early that CST inventory is sufficient for 24 hrs while later reveals that the MAAP analysis shows insufficient CST inventory with statement that alignment to the city water supply may be required. An informal calculation with the minimum flow requirement in EOP concludes that "it would seem that there is enough inventory in the CST to allow the AFW system to operate for 24 hours". Then in IP-RPT-10-0023, Section Insights states that 'As the normal CST inventory is sufficient to supply the AFW pumps for the 24-hour mission time in the PSA', no credit is taken for the alternate suction path from city water supply.	Perform rigorous evaluation/justification of the CST inventory to support 24-hour AFW operation.	**Resolved - No impact** Plant design documentation supports the 24 mission time for the CST. In addition, as noted, CST inventory is typically maintained above the minimum inventory level, providing additional margin. Final model documentation will be modified to remove the apparent discrepancies.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
4-14	Failure modes and success criteria defined in Systems Analysis are consistent with the Data Analysis. This SR also asks for establishing consistent SSC boundaries between the system level analysis and the data analysis. Reviewed Appendix E6 and E27 of the systems notebooks and Appendix D for the Data Analysis. Below is a list of issues identified: 1. System notebooks do not define the component boundaries. The component boundaries are defined by the generic failure rate data source with limited discussions on plant-specific SSC features and modeling considerations. 2. The guidance document Appendix D0 Section 5.10 states 'Assure the component boundaries established in the generic data match those defined in the PSA model. Make adjustments or justify differences'. Also, Attachment 4, Section 3.0 of the same document states that CCF boundaries are dictated by the fault tree modeling. However, the component boundaries defined for failure rate and CCF data do not match. The justification for using the data that way is that it is the conservative to do so. It is true that this approach is conservative for Emergency Diesel Generators, but it may not be conservative for other cases like batteries and battery chargers where CCF of output breakers are not modeled.	DA-A2	Based on the documents reviewed and the issues identified, component boundaries are not consistent among failure rate, CCF and unavailability data. Plant-specific features need to be considered for boundary definitions. It is possible to ensure that the inconsistent boundary definitions result in conservative results, but realistic rather than conservative results is ideal. CCF events tend to dominate system level cutsets and conservative CCF basic event values may mask other important components in a system.	As described in Sections 5.10 and 6.3.11 of Appendix D0, assure component boundaries defined in failure rate and CCF data match the PSA model. Assure the boundaries used in the test and maintenance data is consistent with the PSA model. Make adjustments or provide justification for any mismatch identified. Review plant-specific CCF experience for consistency to meet SY DA-D6 requirements.	**Open - No significant impact** This is a documentation issue. The model documentation will be revised to provide sufficient detail to show that all system and component boundaries in the current model meet or are conservative when compared to the way the generic databases define the boundaries. Resolution will only impact future model updates and upgrades if the apparent discrepancies in the generic boundary definitions are resolved or change. Any future impact is not likely to be significant. Note that Battery Charger input and output breakers are included in the generic database boundary definition for common cause failures whereas the input breakers are not clearly identified to be included in the generic independent failure rate. The PSA model does not include common cause failure of the input or output breakers but does conservatively include independent failure of the input breakers due to specific modeling considerations.

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
4-14	*(continued)* 3. Sections 1.2 and 1.4 of Appendix D1 state that the data analysis package is consistent with the system analysis. However, as discussed in Item number 1 above, systems analysis only defines the system boundary and not the component boundaries within the system. 4. Boundaries of the test and maintenance unavailability events are not specifically discussed, but seem to be same as the boundaries for the failure rates. Data from the Maintenance Rule program is used for this case, but it is not clear if the system and component boundaries considered in this program is consistent with the PSA model boundaries. Section 6.3.11 of Appendix D0 discusses this issue, but there is no evidence that the analysis done in Appendix D1 considered boundaries applies to routine test and maintenance practices at IP3. (This F&O originated from SR DA-A2)				

Table A-1
Summary of Industry Peer Review Findings for the IP3 Internal Events PRA Model Update

Finding	Finding Description	Assoc SR	Basis for Significance	Review Team Suggested Resolution	Status / N716 Impact
1-15	The initiating event frequencies are not weighted by the fraction of time the plant is at power. Section 10.9 of Appendix A0 provides guidance to account for plant availability in initiating event calculations. Section 4.0 of Appendix A1 states that the availability factor for the data update period was calculated. However, the calculated value is not incorporated into the initiating event or final CDF results. (This F&O originated from SR IE-C5)	IE-C5	The initiating event frequencies are not weighted by the fraction of time the plant is at power.	Include the plant availability factor in the calculation of initiating event frequencies.	**Open - No significant impact** While we agree that the wording in the SR itself indicates that weighting should be done, the ASME standard acknowledges that the SR wording is somewhat unclear by providing a lengthy and detailed note of explanation (i.e. Note 1 of the SR). Entergy believes that the annual average model, which Note (1) acknowledges should not include the weighting factors, is the appropriate baseline model in the absence of an all modes model. We do agree, as the standard states, that an all modes model should account for the time in each operating state. Since we do not have an all modes model at this time and we believe that tying risk values to plant availability without an all modes model can potentially provide inappropriate risk insights to non-PSA personnel, in that it does not apply any risk to other operating states, we believe that at the least, our current model meets the SR, when taken in concert with the associated Note 1. In any case, the current approach provides, at most, a slightly conservative result in comparison to use of the stipulated weighting approach and would have no significant impact on this application.

Table A-2 – Additional Flood Sources and Impact

Location	Source	N716 Impact
Primary Auxiliary Building – 55 ft Elevation	Component Cooling Water (CCW) lines 148 and 149 in a pipe chase adjacent to the charging pump room foyer	As noted in the N716 impact discussion for Finding 3-7 in Table A-1 above, failures of these lines can potentially impact the ability to align alternate cooling to the charging pumps in addition to the loss of CCW, similar to the impact of existing flooding initiator IE-FLD-PB-8. The frequency of such a failure of the CCW piping is approximately 2.3E-6 per year. Existing flood initiator IE-FLD-PB-8 has a flood frequency of 2.76E-6 and CDF and LERF contributions of 5.85 E-7/yr and 1E-9/yr, respectively. This is bounding for this additional impact since some potential would still exist for the operator to successfully align alternate cooling to the charging pumps. Therefore, since the impact of assuming that the operator action is completely precluded for breaks in the charging pump foyer would still be below the 1E-6/yr and 1E-7/yr criteria for low safety significance, this additional source is not expected to impact this application.
Primary Auxiliary Building – 55 ft Elevation	Primary Water line 393 in a pipe chase adjacent to the charging pump room foyer	The loss of primary water has no significant impact on the risk model and, in any case, has a failure frequency of 4.1E-7/yr, which is already below the CDF threshold for low safety significance.
Primary Auxiliary Building – 46 ft Elevation	Fire Protection line traversing the upper electrical penetration area	No damage or plant transient is predicted from this scenario due to the large duct and drains in the floor of the electrical tunnel.
Primary Auxiliary Building – 34 ft Elevation	City water system in the lower electrical tunnel, flood zone 3PAB34-7A	The rupture of the city water system in the lower electrical tunnel will not result in any spray damage since the city water line is surrounded by a guard tube.
Primary Auxiliary Building – 43 ft Elevation and 34 ft Elevation	Fire protection system in both the upper and lower electrical tunnels	The fire protection system in both the upper and lower electrical tunnels are dry-pipe pre-action systems.
Primary Auxiliary Building – 34 ft Elevation	Fire protection line in the boron injection tank room.	No damage or plant transient is predicted from this scenario due to the large duct and drains in the floor of the electrical tunnel.
Fan House – 67 ft Elevation	Component Cooling Water (CCW) lines on the fan house mezzanine associated with auxiliary component cooling water pumps 31, 32, 33 and 34	The impact of the failure of these lines may result in a loss of CCW event but has no other consequential impacts. Since the frequency of such a failure is less than 2E-6/yr and the conditional core damage probability (CCDP) following a Loss of CCW initiating event is 1.6E-3, the contribution of such a failure would be several orders of magnitude below the 1E-6/yr threshold for low safety significance.
Fuel Storage Building – 55 ft Elevation	Liquid waste lines enter and traverse the fuel storage building.	This flood scenario will neither require a plant shutdown nor damage safety related equipment.

Table A-2 – Additional Flood Sources and Impact

Location	Source	N716 Impact
Pipe Bridge – 41 ft Elevation Feedwater Building – 43 ft Elevation and 18 ft Elevation	Fire Protection lines traverse the pipe bridge between the Turbine Building and the Feedwater Building, the main boiler feedwater area on the 18 ft Elevation, and the main steam and feedwater valve area on the 43 ft Elevation.	The only flooding scenario of consequence would be a catastrophic failure of this fire protection line. Although there are multiple egress paths, the bounding impact of such a failure would involve all the flood water entering the turbine building through the openings in the west wall of the pipe bridge. Such a failure would have to continue at maximum flow for well over an hour before building up on the turbine floor sufficiently to challenge the normal offsite power busses. Since the flood frequency for this event is approximately 1.4E-5/yr and the onsite EDGs would remain unaffected by this event, it can be concluded that the CDF contribution would be well below the 1E-6 threshold for low safety significance and would not significantly impact this application.
Feedwater Building – 18 ft Elevation	Fire protection lines in the AFW Pump room	An evaluation of the fire protection system in the IP3 AFW Pump room, done as part of the fire suppression analysis performed for IP3 in response to GI-57, concluded that the AFW pump motors would not be impacted by spray from an inadvertent actuation or rupture of the fire protection piping in that room. Although the AFW pump motors were not specifically qualified for the chemical spray associated with a DBA, the qualification testing did impose HELB conditions, including a period of immersion. It is therefore expected that the AFW pumps will operate successfully should they be subjected to spray following a fire protection system failure. A catastrophic failure of the fire protection system will not result in submergence of the AFW pumps since the AFP room has a door flap designed to relieve such an inflow. Such a failure of fire protection in this area would not lead to a plant transient and operators would not be expected to manually trip the plant or perform a controlled shutdown prior to assessing any impacts. It is therefore concluded that this additional scenario would not impact this application.