Plant X
Request for Approval of Risk-Informed/Safety Based
Inservice Inspection Alternative for Class 1 and 2 Piping

Enclosure 1

ENCLOSURE 1
Plant X
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

Plant Site - Unit:	Plant X, Units 1 and 2
Interval - Dates:	First Interval Plant X, Unit 1: December 01, 2013 to November 30, 2023 Plant X, Unit 2: July 15, 2016 to July 14, 2016
Requested Date for Approval :	Approval is requested by March 01, 2013
ASME Code Components Affected:	All Class 1 and 2 piping welds – Examination Categories B-F, B-J, C-F-1, and C-F-2.
Applicable Code Edition and Addenda:	For Plant X, Unit 1 and Unit 2, the 1st Inservice Inspection (ISI) Interval will begin on December 1, 2013. The applicable Code of Record is the ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition with the 2000 Addenda (for Period 1). For Plant X Unit 2, the 1st Inservice Inspection Interval is currently scheduled to start on July 15, 2016. The applicable Code of Record for the 1st ISI Interval will be the latest edition and addenda of the ASME Code, Section XI, incorporated by reference into 10 CFR 50.55a on July 15, 2015.
Applicable Code Requirements:	For Unit 1, Period 1, the requirements from which an alternative is requested are specified in the ASME Code, Section XI, 1998 Edition with the 2000 Addenda, IWB-2500, Table IWB-2500-1, Examination Categories B-F and B-J; and in IWC-2500, Table IWC-2500-1, Examination Categories C-F-1 and C-F-2. For Periods 2 and 3, the requirement from which an alternative is requested is the same requirement of Unit 2, the 1st Inservice Inspection Interval. For Unit 2, the requirement from which an alternative is requested is the requirement of 10 CFR 50.55a(g)(4)(ii) that "Inservice examination of components …conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval...". This alternative is requested only for examination of Class 1 and 2 piping welds identified in Section 1 of this request.
Reason for 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.
Proposed Alternative and Basis for Use:	In lieu of the ASME Code requirements, Plant X proposes to use a RIS_B process as an alternate to the ASME Section XI 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 U.S. 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 as adapted by EPRI Topical Report 1021467 (Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidelines for Risk-Informed In-Service Inspection Programs)*. By using risk-insights to focus examinations on more important locations, while meeting the intent and principles of Regulatory Guides 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-2200, IWB-2420, IWB-2430, and IWB-2500 (Examination Categories B-F and B-J) and IWC-2200, IWC-2420, IWC-2430, and IWC-2500 (Examination Categories C-F-1 and C-F-2) is requested in accordance with 10 CFR 50.55a(a)(3)(i). A Plant X specific Template is attached that mirrors previous RIS_B submittals to the NRC. All other ASME Code, Section XI requirements for which relief was not

	specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.
Duration of Proposed Alternative:	For Plant X Unit 1, use of the proposed alternative is requested for the duration of the First Inservice Inspection Interval (currently scheduled from December 1, 2013 to November 30, 2023). For Plant X Unit 2, use of the proposed alternative is requested for the First Inservice Inspection Interval (currently scheduled to start on July 15, 2016 to July 14, 2026).
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 & 2.
References:	Vogtle Electric Generating Plant Safety Evaluation - See ADAMS Accession No. ML100610470. D. C. 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. North Anna Power Station (NAPS) Units 1 and 2 Safety Evaluation – See ADAMS Accession No. ML110050003.
Status:	Awaiting NRC approval.

TEMPLATE SUBMITTAL

APPLICATION OF ASME CODE CASE N-716

RISK-INFORMED/SAFETY-BASED (RIS_B) INSERVICE INSPECTION PROGRAM PLAN

ENCLOSURE 1
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Technical Acronyms/Definitions Used in the Template

CA	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
NV	Chemical Volume and Control System
DA	Data analysis
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
CF	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)

LOOP	Loss of Off-Site Power
LSS	Low Safety-Significant
MAAP	Modular Accident Analysis Program
MIC	Microbiologically-Influenced Corrosion
MOV	Motor Operated Valve
SM	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
NC	Reactor Coolant
NCP	Reactor Coolant Pump
NCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR, ND	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
RN	Nuclear 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
Plant X
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

Table of Contents

1. INTRODUCTION

Plant X Nuclear Station Units 1 and 2 (Plant X) are currently scheduled to commence commercial operation in 2013 and 2016, respectively and plan to implement the first 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. Plant X plans to implement a risk-informed/safety-based inservice inspection (RIS_B) program in the first ISI interval.

The ASME Section XI Code of record for the first ISI interval for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1, 2, 3, or Non-Class piping welds piping is as follow:

> For Plant X Unit 1, the Code of Record is the ASME Section XI 1998 Edition with the 2000 Addenda (for Period 1).
>
> For Plant X Unit 2, the Code of Record will be the latest edition and addenda of the ASME Code, Section XI, incorporated by reference into 10 CFR 50.55a on July 15, 2013.

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 Decisionmaking 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 pre-determined 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 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 as adapted by EPRI Topical Report (TR) 1021467 (Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidelines for Risk-Informed In-Service Inspection Programs)*.

As discussed in 1021467, there are some elements of the PRA that cannot be completed until the plant has gone operational (e.g. operating data). TR-1021467 provides guidance on what interim steps can be taken to assure a robust and stable ISI program. The Plant X PRA meets the guidance contained in TR-1021467 and will take the necessary steps to update the PRA and the RI-ISI program when that information becomes available (e.g. 10CFR50.71h).

A number of USNRC approved RI-ISI evaluations concluded external events are not likely to impact the consequence ranking. This position is further supported by Section 2 of TR-1021467 which concludes that quantification of these events will not change the conclusions derived from the RI-ISI process. As a result, there is no need to further consider these events.

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 program currently under development 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 (i.e., Class 1, 2 and 3 piping).

- The plant augmented inspection program for high energy line break has not been revised by this application. A separate evaluation and program is maintained in accordance with the risk-informed break exclusion region methodology (RI-BER) described in EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs*

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

- Plant X has conducted an evaluation in accordance with MRP-146, Revision 1 *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*, and these results have been incorporated into the RIS_B Program.

- N-770-1, Alloy 600, Alooy690???

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 3.1a (UNIT 1) and Table 3.1b (UNIT 2). 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.

(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. Class 3 nuclear service water system piping in the auxiliary feedwater pump room was identified as HSS due to CDF exceeding these 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.

A deviation to the EPRI RIS_B methodology has been implemented in the failure potential assessment for Plant X. 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

   ➢ $\Delta T > 50$ºF,

   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 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. The methodology used in the Plant X RIS_B application for assessing TASCS potential conforms to these updated criteria. Additionally, materials reliability program (MRP) MRP-146, Revision 1 guidance on the subject of TASCS was also incorporated into the Plant X 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:

  (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.

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

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected.

In contrast to a number of 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 3.3a (UNIT 1) and Table 3.3b (UNIT 2). 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)(5)		Class 2 Welds(2)		All Piping Welds(3)(4)	
	Total	Selected	Total	Selected	Total	Selected
1	782	83	3386	14	4168	97
2	765	80	3621	16	4386	96

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, 201 are HSS at Unit 1 and 205 are HSS at Unit 2; 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 nuclear service water system piping in the auxiliary feedwater pump room is defined as HSS and is included in the RIS_B Program.

(5) As described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment were removed from the RIS_B population totals in the above table prior to element selection.

3.3.1 Current Examinations

If this relief request were not approved, the deterministic ASME Section XI inspection methodology for ISI examination of piping welds per the 1998 Edition of ASME Section XI through the 2000 Addenda.

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. Evaluation of indications attributed to PWSCC and successive examinations of PWSCC indications will be performed in accordance with ASME Code Case N-770-1 or a subsequent NRC rule making. Finally, the evaluation will be documented in the corrective action program and the Owner submittals required by Section XI.

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 ASME Code Case N-770-1 or subsequent NRC rule makings.

3.3.4 Program Relief Requests

Consistent with previously approved RIS_B submittals, Plant X 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 Plant X 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 welds 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 current 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.

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 3.4a (UNIT 1) and Table 3.4b (UNIT 2)) for use in the change-in-risk assessment. Experience with previous industry RIS_B applications shows this to be conservative.

Plant X 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., Medium LOCA CCDP bounds the medium and small LOCA CCDPs).

Also, as described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment were removed from the RIS_B population prior to element selection and risk impact assessment.

CCDP and CLERP Values Based on Break Location					
Break Location Designation	**Estimated**		**Consequence Rank**	**Upper Bound**	
	CCDP	**CLERP**		**CCDP**	**CLERP**
LOCA	4.1E-03	4.1E-04	**HIGH**	4.1E-03	4.1E-04
RCPB pipe breaks that result in a loss of coolant accident - The highest CCDP for Medium LOCA (%ML) was used (0.1 margin used for CLERP). Unisolable RCPB piping of all sizes.					
PLOCA[1]	1.0E-05	1.0E-06	**MEDIUM**	1.0E-04	1.0E-05
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 Medium LOCA CCDP of 4.1E-3 and valve fail to close probability of <3E-3 (0.1 margin used for CLERP). Applies to piping between 1st and 2nd RCPB isolation valve.					
PPLOCA	<1.0E-06	<1.0E-07	**LOW**	1.0E-06	1.0E-07
Isolable or Potential LOCA with two valves (2 open valves or 2 closed valves) - RCPB pipe breaks that result in an isolable or potential LOCA beyond 2 valves - Calculated based on Medium LOCA CCDP of 4.1E-3 and failure of two MOVs to close <1E-4 (0.1 margin used for CLERP). Applies to piping between beyond the 2nd RCPB isolation valve inside containment.					
FLB	3E-05	3E-06	**MEDIUM**	1.0E-04	1.0E-05
Feedwater line breaks – bounding value used that envelopes %T6 and %T7 (0.1 margin used for CLERP).					
Class 2 LSS	1.0E-04	1.0E-05	**MEDIUM**	1.0E-04	1.0E-05
Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant - Estimated based on upper bound for Medium Consequence.					

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 (isolable LOCA) 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 3.4a (UNIT 1) and Table 3.4b (UNIT 2) present a summary of the RIS_B Program versus the deterministic interval program (note: inspections allocated on a prorated basis in anticipation of the final ISI program) 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.

Plant X Unit 1

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CA - Auxiliary Feedwater	1.30E-10	1.30E-11	1.30E-10	1.30E-11
CF - Feedwater	-1.50E-12	-1.50E-13	5.70E-12	5.70E-13
FW - Refueling Water	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KC - Component Cooling	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC - Reactor Coolant	-1.47E-08	-1.47E-09	-7.03E-09	-7.03E-10
ND - Residual Heat Removal	5.10E-10	5.10E-11	5.10E-10	5.10E-11
NF - Ice Condenser Refrig	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI - Safety Injection	3.62E-10	3.62E-11	3.62E-10	3.62E-11
NS - Containment Spray	2.00E-11	2.00E-12	2.00E-11	2.00E-12
NV - Chemical Volume & Control	3.85E-10	3.85E-11	3.87E-10	3.87E-11
RN - Nuclear Service Water	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RV - Cont Vent Cooling Water	1.00E-11	1.00E-12	1.00E-11	1.00E-12
SA - Auxiliary Steam	2.00E-11	2.00E-12	2.00E-11	2.00E-12
SM - Main Steam	1.60E-10	1.60E-11	1.60E-10	1.60E-11
SV - Main Steam Vent	3.00E-11	3.00E-12	3.00E-11	3.00E-12
VP - Cont Purge Vent	2.00E-11	2.00E-12	2.00E-11	2.00E-12
VQ - Cont Air Release & Addition	0.00E+00	0.00E+00	0.00E+00	0.00E+00
WL - Liquid Waste & Recycle	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	**-1.31E-08**	**-1.31E-09**	**-5.38E-09**	**-5.38E-10**

Plant X Unit 2

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CA - Auxiliary Feedwater	1.40E-10	1.40E-11	1.40E-10	1.40E-11
CF – Feedwater	1.20E-12	1.20E-13	8.40E-12	8.40E-13
FW - Refueling Water	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KC - Component Cooling	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC - Reactor Coolant	-1.59E-08	-1.59E-09	-7.59E-09	-7.59E-10
ND - Residual Heat Removal	5.30E-10	5.30E-11	5.30E-10	5.30E-11
NF - Ice Condenser Refrig	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI - Safety Injection	4.29E-10	4.29E-11	4.29E-10	4.29E-11

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
NS - Containment Spray	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NV - Chemical Volume & Control	5.05E-10	5.05E-11	5.07E-10	5.07E-11
RN - Nuclear Service Water	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RV - Cont Vent Cooling Water	1.00E-11	1.00E-12	1.00E-11	1.00E-12
SA - Auxiliary Steam	2.00E-11	2.00E-12	2.00E-11	2.00E-12
SM - Main Steam	1.90E-10	1.90E-11	1.90E-10	1.90E-11
SV - Main Steam Vent	3.00E-11	3.00E-12	3.00E-11	3.00E-12
VP - Cont Purge Vent	2.00E-11	2.00E-12	2.00E-11	2.00E-12
VQ - Cont Air Release & Addition	0.00E+00	0.00E+00	0.00E+00	0.00E+00
WL - Liquid Waste & Recycle	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	**-1.41E-08**	**-1.41E-09**	**-5.70E-09**	**-5.70E-10**

Note:
(1) The risk reduction associated with the HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the above tables.

As shown in Table 3.4a (UNIT 1) and Table 3.4b (UNIT 2), 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. Plant X identified Class 3 nuclear service water piping in the auxiliary feedwater pump 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) as well as incorporation of information as the plant transitions from the post construction phase (e.g. operating data, final set of deterministic ISI selections). 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).

At this time and consistent with the operating fleet's implementation of RIS_B programs, a number of preservice examinations have been conducting using the deterministic ASME Section XI requirements. Upon approval of this relief request, Plant X may optionally follow the rules contained in Section 3.0 of N-716. If so, welds classified HSS will require a preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of N-716. Welds classified as LSS will not require preservice inspection.

4. PROPOSED ISI PLAN CHANGE

Plant X Units 1 and 2 are anticipated to commence commercial operation in 2013 and 2016, respectively.

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 1998 Edition of Section XI program requirements for first interval in-scope piping is provided in Table 4a (UNIT 1) and Table 4b (UNIT 2).

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 Decisionmaking 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. ADAMS Accession No. ML072430005

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. See ADAMS Accession No. ML072620553.

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

Waterford-3 Safety Evaluation – See ADAMS Accession No. ML080980120.

Vogtle Electric Generating Plant Safety Evaluation - See ADAMS Accession No. ML100610470.

North Anna Power Station (NAPS) Units 1 and 2 Safety Evaluation – See ADAMS Accession No. ML110050003.

Supporting Onsite Documentation

“N716 Evaluation for Plant X Unit 1”

“N716 Evaluation for Plant X Unit 2”

“Degradation Mechanism Evaluation for Plant X Unit 1”

“Degradation Mechanism Evaluation for Plant X Unit 2”

Table 3.1a
Unit 1 Code Case N-716 Safety Significance Determination

System (1)	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
CA	159							✓
CF	119			✓			✓	
	13							✓
FW	136							✓
KC	8							✓
NC	377	✓					✓	
	24	✓	✓				✓	
	8	✓	✓		✓		✓	
ND	15	✓	✓				✓	
	2		✓				✓	
	485							✓
NF	2							✓
NI	142	✓					✓	
	105	✓	✓				✓	
	80		✓				✓	
	553							✓
NS	443							✓
NV	117	✓					✓	
	1031							✓
RN	17(2)							✓
RV	9							✓
SA	22							✓
SM	164							✓
SV	37							✓
VP	20							✓
VQ	78							✓
WL	8							✓
Summary Results for all Systems	636	✓					✓	
	144	✓	✓				✓	
	82		✓				✓	
	8	✓	✓		✓		✓	
	119			✓			✓	
	3185							✓

(1) System Scope:

CA - Auxiliary Feedwater
CF – Feedwater
FW - Refueling Water
KC - Component Cooling
NC - Reactor Coolant
ND - Residual Heat Removal
NF - Ice Condenser Refrigeration
NI - Safety Injection
NS - Containment Spray
NV - Chemical Volume & Control

RN - Nuclear Service Water
RV - Cont Vent Cooling Water
SA - Auxiliary Steam
SM - Main Steam
SV - Main Steam Vent
VP - Cont Purge Vent
VQ - Cont Air Release & Addition
WL - Liquid Waste & Recycle

(2) HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the Weld Count or Summary Results.

Table 3.1b
Unit 2 Code Case N-716 Safety Significance Determination

System (1)	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6 (2)	High	Low
CA	121							✓
CF	129			✓			✓	
	13							✓
FW	154							✓
KC	6							✓
NC	361	✓					✓	
	20	✓	✓				✓	
	8	✓	✓		✓		✓	
ND	17	✓	✓				✓	
	2		✓				✓	
	520							✓
NF	2							✓
NI	136	✓					✓	
	97	✓	✓				✓	
	74		✓				✓	
	573							✓
NS	464							✓
NV	126	✓					✓	
	1188							✓
RN	22(2)							✓
RV	10							✓
SA	22							✓
SM	185							✓
SV	35							✓
VP	20							✓
VQ	73							✓
WL	8							✓
Summary Results for all Systems	623	✓					✓	
	134	✓	✓				✓	
	76		✓				✓	
	8	✓	✓		✓		✓	
	129			✓			✓	
	3416							✓

(1) System Scope:

CA - Auxiliary Feedwater
CF – Feedwater
FW - Refueling Water
KC - Component Cooling
NC - Reactor Coolant
ND - Residual Heat Removal
NF - Ice Condenser Refrigeration
NI - Safety Injection
NS - Containment Spray

NV - Chemical Volume & Control
RN - Nuclear Service Water
RV - Cont Vent Cooling Water
SA - Auxiliary Steam
SM - Main Steam
SV - Main Steam Vent
VP - Cont Purge Vent
VQ - Cont Air Release & Addition
WL - Liquid Waste & Recycle

(2) HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the Weld Count or Summary Results.

ENCLOSURE 1
Plant X
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

Table 3.2
Failure Potential Assessment Summary

	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
System[(1)(2)]	**TASCS**	**TT**	**IGSCC**	**TGSCC**	**ECSCC**	**PWSCC**	**MIC**	**PIT**	**CC**	**E-C**	**FAC**
CA											
CF		✓									
FW											
KC											
NC	✓	✓				✓					
ND											
NF											
NI		✓	✓			✓					
NS											
NV		✓									
RN											
RV											
SA											
SM											
SV											
VP											
VQ											
WL											

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 CA, FW, KC, NF, RN, RV, SA, SM, SV, VP, VQ, WL in their entirety, as well as portions of the ND, NI and NV systems.

Table 3.3a: Unit 1 Code Case N716 Selections

System (1)	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CA		159	None					0
CF	8		TT					2
	111		None					10
		13	None					0
FW		136	None					0
KC		8	None					0
NC	2		TT,TASCS,PWSCC	✓	✓			2
	54		TT	✓	✓			14
	13		TT	✓				3
	25		TASCS,TT	✓	✓			7
	2		TASCS	✓	✓			1
	8		None	✓	✓		✓	5
	248		None	✓	✓			23
	27		None	✓				2
ND	15		None	✓				0
	2		None					0
		485	None					0
NF		2	None					0
NI	8		TT,IGSCC	✓				2
	30		IGSCC	✓				8
	205		None	✓				4
	80		None					2
		553	None					0
NS		443	None					0
NV	11		TT	✓				3
	20		None	✓	✓			9
	86		None	✓				0
		1031	None					0
RN	(2)	17	None					0
RV		9	None					0
SA		22	None					0
SM		164	None					0
SV		37	None					0
VP		20	None					0
VQ		78	None					0
WL		8	None					0
Summary Results All Systems	8		TT					2
	2		TT,TASCS,PWSCC	✓	✓			2
	54		TT	✓	✓			14
	24		TT	✓				6
	25		TT,TASCS	✓	✓			7
	2		TASCS	✓	✓			1
	8		TT,IGSCC	✓				2
	30		IGSCC	✓				8
	8		None	✓	✓		✓	5
	268		None	✓	✓			32
	333		None	✓				6
	193		None					12
		3185	None					0

System (1)	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
Totals	955	3185	None					97

Notes:

(1) Systems are described in Table 3.1

(2) HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the Weld Count or Summary Results

Table 3.3b: Unit 2 Code Case N716 Selections

System (1)	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CA		121	None					0
CF	8		TT					2
	121		None					11
		13	None					0
FW		154	None					0
KC		6	None					0
NC	2		TT,TASCS,PWSCC	✓	✓			1
	60		TT	✓	✓			15
	29		TASCS,TT	✓	✓			7
	2		TASCS	✓	✓			2
	8		None	✓	✓		✓	6
	245		None	✓	✓			24
	23		None	✓				0
ND	17		None	✓				0
	2		None					0
		520	None					0
NF		2	None					0
NI	8		TT,IGSCC	✓				2
	38		IGSCC	✓				10
	183		None	✓				0
	74		None					3
		573	None					0
NS		464	None					0
NV	10		TT	✓				3
	20		None	✓	✓			9
	96		None	✓				1
		1188	None					0
RN	(2)	22	None					0
RV		10	None					0
SA		22	None					0
SM		185	None					0
SV		35	None					0
VP		20	None					0
VQ		73	None					0
WL		8	None					0
Summary Results All Systems	8		TT					2
	2		TT,TASCS,PWSCC	✓	✓			1
	60		TT	✓	✓			15
	10		TT	✓	□			3
	29		TT,TASCS	✓	✓			7
	2		TASCS	✓	✓			2
	8		TT,IGSCC	✓				2
	38		IGSCC	✓				10
	8		None	✓	✓		✓	6
	265		None	✓	✓			33
	319		None	✓				1
	197		None					14
		3416	None					0
Totals	946	3416	None					96

Notes:

(1) Systems are described in Table 3.1
(2) HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the Weld Count or Summary Results

Table 3.4a Unit 1 Risk Impact Analysis Results

System (1)	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
CA Total	Low	Class 2 LSS		Assume Medium	13	0	-13	**1.30E-10**	**1.30E-10**	**1.30E-11**	**1.30E-11**
CF	High	FLB	TT	Medium	2	2	0	-2.40E-11	0.00E+00	-2.40E-12	0.00E+00
CF	High	FLB	None	Low	8	10	2	-1.00E-12	-1.00E-12	-1.00E-13	-1.00E-13
CF	Low	Class 2 LSS		Assume Medium	2	0	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
CF Total								**-5.00E-12**	**1.90E-11**	**-5.00E-13**	**1.90E-12**
FW Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
KC Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
NC	High	LOCA	TT,TASCS,PWSCC	Medium	2	2	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC	High	LOCA	TT	Medium	1	14	13	-1.01E-08	-5.33E-09	-1.01E-09	-5.33E-10
NC	High	PLOCA	TT	Medium	0	3	3	-5.40E-11	-3.00E-11	-5.40E-12	-3.00E-12
NC	High	LOCA	TASCS,TT	Medium	2	7	5	-4.67E-09	-2.05E-09	-4.67E-10	-2.05E-10
NC	High	LOCA	TASCS	Medium	0	1	1	-7.38E-10	-4.10E-10	-7.38E-11	-4.10E-11
NC	High	LOCA	None	Low	56	19	-37	7.59E-10	7.59E-10	7.59E-11	7.59E-11
NC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC	High	PPLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC Total								**-1.48E-08**	**-7.06E-09**	**-1.48E-09**	**-7.06E-10**
ND	High	PLOCA	None	Low	5	0	-5	2.50E-12	2.50E-12	2.50E-13	2.50E-13
ND	High	PPLOCA	None	Low	1	0	-1	5.00E-15	5.00E-15	5.00E-16	5.00E-16
ND	Low	Class 2 LSS		Assume Medium	51	0	-51	5.10E-10	5.10E-10	5.10E-11	5.10E-11
ND Total								**5.13E-10**	**5.13E-10**	**5.13E-11**	**5.13E-11**
NF Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
NI	High	PLOCA	TT,IGSCC	Medium	0	2	2	-2.00E-11	-2.00E-11	-2.00E-12	-2.00E-12
NI	High	PLOCA	IGSCC	Medium	11	8	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
NI	High	PLOCA	None	Low	26	4	-22	1.10E-11	1.10E-11	1.10E-12	1.10E-12
NI	High	PPLOCA	None	Low	48	2	-46	2.30E-13	2.30E-13	2.30E-14	2.30E-14
NI	Low	Class 2 LSS		Assume Medium	36	0	-36	3.60E-10	3.60E-10	3.60E-11	3.60E-11
NI Total								**3.81E-10**	**3.81E-10**	**3.81E-11**	**3.81E-11**
NS Total	Low	Class 2 LSS		Assume Medium	2	0	-2	**2.00E-11**	**2.00E-11**	**2.00E-12**	**2.00E-12**
NV	High	PLOCA	TT	Medium	0	3	3	-5.40E-11	-3.00E-11	-5.40E-12	-3.00E-12
NV	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NV	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NV	Low	Class 2 LSS		Assume Medium	39	0	-39	3.90E-10	3.90E-10	3.90E-11	3.90E-11

System (1)	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
NV Total								**3.36E-10**	**3.60E-10**	**3.36E-11**	**3.60E-11**
RN Total (7)	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
RV Total	Low	Class 2 LSS		Assume Medium	1	0	-1	**1.00E-11**	**1.00E-11**	**1.00E-12**	**1.00E-12**
SA Total	Low	Class 2 LSS		Assume Medium	2	0	-2	**2.00E-11**	**2.00E-11**	**2.00E-12**	**2.00E-12**
SM Total	Low	Class 2 LSS		Assume Medium	16	0	-16	**1.60E-10**	**1.60E-10**	**1.60E-11**	**1.60E-11**
SV Total	Low	Class 2 LSS		Assume Medium	3	0	-3	**3.00E-11**	**3.00E-11**	**3.00E-12**	**3.00E-12**
VP Total	Low	Class 2 LSS		Assume Medium	2	0	-2	**2.00E-11**	**2.00E-11**	**2.00E-12**	**2.00E-12**
VQ Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
WL Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Grand Total					**329**	**77**	**-252**	**-1.32E-08**	**-5.40E-09**	**-1.32E-09**	**-5.40E-10**

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).
6. As described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment are not included in the table.
7. The risk reduction associated with the HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the above table.

Table 3.4b Unit 2 Risk Impact Analysis Results

System (1)	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
CA Total	Low	Class 2 LSS		Assume Medium	14	0	-14	**1.40E-10**	**1.40E-10**	**1.40E-11**	**1.40E-11**
CF	High	FLB	TT	Medium	2	2	0	-2.40E-11	0.00E+00	-2.40E-12	0.00E+00
CF	High	FLB	None	Low	7	11	4	-2.00E-12	-2.00E-12	-2.00E-13	-2.00E-13
CF	Low	Class 2 LSS		Assume Medium	3	0	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
CF Total								**4.00E-12**	**2.80E-11**	**4.00E-13**	**2.80E-12**
FW Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
KC Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
NC	High	LOCA	TT,TASCS,PWSCC	Medium	2	1	-1	4.10E-10	4.10E-10	4.10E-11	4.10E-11
NC	High	LOCA	TT	Medium	0	15	15	-1.11E-08	-6.15E-09	-1.11E-09	-6.15E-10
NC	High	LOCA	TASCS,TT	Medium	3	7	4	-4.43E-09	-1.64E-09	-4.43E-10	-1.64E-10
NC	High	LOCA	TASCS	Medium	0	2	2	-1.48E-09	-8.20E-10	-1.48E-10	-8.20E-11
NC	High	LOCA	None	Low	49	19	-30	6.15E-10	6.15E-10	6.15E-11	6.15E-11
NC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC	High	PPLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NC Total								**-1.59E-08**	**-7.59E-09**	**-1.59E-09**	**-7.59E-10**
ND	High	PLOCA	None	Low	5	0	-5	2.50E-12	2.50E-12	2.50E-13	2.50E-13
ND	High	PPLOCA	None	Low	1	0	-1	5.00E-15	5.00E-15	5.00E-16	5.00E-16
ND	Low	Class 2 LSS		Assume Medium	53	0	-53	5.30E-10	5.30E-10	5.30E-11	5.30E-11
ND Total								**5.33E-10**	**5.33E-10**	**5.33E-11**	**5.33E-11**
NF Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
NI	High	PLOCA	TT,IGSCC	Medium	0	2	2	-2.00E-11	-2.00E-11	-2.00E-12	-2.00E-12
NI	High	PLOCA	IGSCC	Medium	8	8	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI	High	PLOCA	None	Low	18	0	-18	9.00E-12	9.00E-12	9.00E-13	9.00E-13
NI	High	PPLOCA	None	Low	53	3	-50	2.50E-13	2.50E-13	2.50E-14	2.50E-14
NI	Low	Class 2 LSS		Assume Medium	43	0	-43	4.30E-10	4.30E-10	4.30E-11	4.30E-11
NI Total								**4.19E-10**	**4.19E-10**	**4.19E-11**	**4.19E-11**
NS Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
NV	High	PLOCA	TT	Medium	0	3	3	-5.40E-11	-3.00E-11	-5.40E-12	-3.00E-12
NV	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NV	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NV	Low	Class 2 LSS		Assume Medium	51	0	-51	5.10E-10	5.10E-10	5.10E-11	5.10E-11
NV Total								**4.56E-10**	**4.80E-10**	**4.56E-11**	**4.80E-11**

System (1)	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
RN Total (7)	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
RV Total	Low	Class 2 LSS		Assume Medium	1	0	-1	**1.00E-11**	**1.00E-11**	**1.00E-12**	**1.00E-12**
SA Total	Low	Class 2 LSS		Assume Medium	2	0	-2	**2.00E-11**	**2.00E-11**	**2.00E-12**	**2.00E-12**
SM Total	Low	Class 2 LSS		Assume Medium	19	0	-19	**1.90E-10**	**1.90E-10**	**1.90E-11**	**1.90E-11**
SV Total	Low	Class 2 LSS		Assume Medium	3	0	-3	**3.00E-11**	**3.00E-11**	**3.00E-12**	**3.00E-12**
VP Total	Low	Class 2 LSS		Assume Medium	2	0	-2	**2.00E-11**	**2.00E-11**	**2.00E-12**	**2.00E-12**
VQ Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
WL Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Grand Total					**339**	**73**	**-266**	**-1.41E-08**	**-5.72E-09**	**-1.41E-09**	**-5.72E-10**

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).
6. As described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment are not included in the table.
7. The risk reduction associated with the HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room is not included in the above table.

Table 4a: Unit 1 Inspection Location Selections Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
CA		✓	Class 2 LSS		Assume Medium	C-F-1, C-F-2	159	13	0	0	NA
CF	✓		FLB	TT	Medium	C-F-2	8	2	0	2	NA
CF	✓		FLB	None	Low	C-F-2	111	8	0	10	NA
CF		✓	Class 2 LSS		Assume Medium	C-F-2	13	2	0	0	NA
FW		✓	Class 2 LSS		Assume Medium	C-F-1	136	0	0	0	NA
KC		✓	Class 2 LSS		Assume Medium	C-F-2	8	0	0	0	NA
NC	✓		LOCA	TT,TASCS,PWSCC	Medium	B-F	2	2	0	2	NA
NC	✓		LOCA	TT	Medium	B-J	54	1	15	14	NA
NC	✓		PLOCA	TT	Medium	B-J	13	0	2	3	NA
NC	✓		LOCA	TASCS,TT	Medium	B-J	25	2	6	7	NA
NC	✓		LOCA	TASCS	Medium	B-J	2	0	0	1	NA
NC	✓		LOCA	None	Low	B-F, B-J	256	56	32	19	9
NC	✓		PLOCA	None	Low	B-J	22	0	1	0	2
NC	✓		PPLOCA	None	Low	B-J	5	0	3	0	NA
ND	✓		PLOCA	None	Low	B-J	15	5	0	0	NA
ND	✓		PPLOCA	None	Low	C-F-1	2	1	0	0	NA
ND		✓	Class 2 LSS		Assume Medium	C-F-1	485	51	0	0	NA
NF		✓	Class 2 LSS		Assume Medium	C-F-1	2	0	0	0	NA
NI	✓		PLOCA	TT,IGSCC	Medium	B-J	8	0	8	2	NA
NI	✓		PLOCA	IGSCC	Medium	B-J	30	11	0	8	NA
NI	✓		PLOCA	None	Low	B-J	171	26	12	4	NA
NI	✓		PPLOCA	None	Low	C-F-1	114	48	0	2	NA
NI		✓	Class 2 LSS		Assume Medium	C-F-1	553	36	16	0	NA
NS		✓	Class 2 LSS		Assume Medium	C-F-1	443	2	0	0	NA
NV	✓		PLOCA	TT	Medium	B-J	11	0	4	3	NA
NV	✓		LOCA	None	Low	B-J	20	0	7	0	9
NV	✓		PLOCA	None	Low	B-J	86	0	23	0	NA
NV		✓	Class 2 LSS		Assume Medium	C-F-1	1031	39	40	0	NA
RN(5)		✓	Class 2 LSS		Assume Medium	C-F-2	17	0	0	0	NA
RV		✓	Class 2 LSS		Assume Medium	C-F-2	9	1	0	0	NA
SA		✓	Class 2 LSS		Assume Medium	C-F-2	22	2	0	0	NA
SM		✓	Class 2 LSS		Assume Medium	C-F-2	164	16	0	0	NA
SV		✓	Class 2 LSS		Assume Medium	C-F-2	37	3	0	0	NA

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
VP		✓	Class 2 LSS		Assume Medium	C-F-2	20	2	0	0	NA
VQ		✓	Class 2 LSS		Assume Medium	C-F-1	78	0	0	0	NA
WL		✓	Class 2 LSS		Assume Medium	C-F-1	8	0	0	0	NA
						Totals	**4140**	**329**	**169**	**77**	**20**

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 Plant X 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. Inspections allocated on a prorated basis in anticipation of the final ISI program.
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. Section XI
6. As described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment are not included in the table.
7. Inspection locations associated with the HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room are not included in the above table.

Table 4b: Unit 2 Inspection Location Selections Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
CA		✓	Class 2 LSS		Assume Medium	C-F-1, C-F-2	121	14	0	0	NA
CF	✓		FLB	TT	Medium	C-F-2	8	2	0	2	NA
CF	✓		FLB	None	Low	C-F-2	121	7	0	11	NA
CF		✓	Class 2 LSS		Assume Medium	C-F-2	13	3	0	0	NA
FW		✓	Class 2 LSS		Assume Medium	C-F-1	154	0	0	0	NA
KC		✓	Class 2 LSS		Assume Medium	C-F-2	6	0	0	0	NA
NC	✓		LOCA	TT,TASCS,PWSCC	Medium	B-F	2	2	0	1	NA
NC	✓		LOCA	TT	Medium	B-J	60	0	12	15	NA
NC	✓		LOCA	TASCS,TT	Medium	B-J	29	3	8	7	NA
NC	✓		LOCA	TASCS	Medium	B-J	2	0	2	2	NA
NC	✓		LOCA	None	Low	B-F, B-J	253	49	20	19	11
NC	✓		PLOCA	None	Low	B-J	18	0	3	0	NA
NC	✓		PPLOCA	None	Low	B-J	5	0	0	0	NA
ND	✓		PLOCA	None	Low	B-J	17	5	0	0	NA
ND	✓		PPLOCA	None	Low	C-F-1	2	1	0	0	NA
ND		✓	Class 2 LSS		Assume Medium	C-F-1	520	53	1	0	NA
NF	✓		Class 2 LSS		Assume Medium	C-F-1	2	0	0	0	NA
NI	✓		PLOCA	TT,IGSCC	Medium	B-J	8	0	7	2	NA
NI	✓		PLOCA,PPLOCA	IGSCC	Medium	B-J	38	8	1	8	2
NI	✓		PLOCA	None	Low	B-J	114	18	8	0	NA
NI	✓		PPLOCA	None	Low	C-F-1	143	53	6	3	NA
NI		✓	Class 2 LSS		Assume Medium	C-F-1	573	43	14	0	NA
NS		✓	Class 2 LSS		Assume Medium	C-F-1	464	0	0	0	NA
NV	✓		PLOCA	TT	Medium	B-J	10	0	4	3	NA
NV	✓		LOCA	None	Low	B-J	20	0	4	0	9
NV	✓		PLOCA	None	Low	B-J	96	0	26	0	1
NV		✓	Class 2 LSS		Assume Medium	C-F-1	1188	51	39	0	NA
RN(5)		✓	Class 2 LSS		Assume Medium	C-F-2	22	0	0	0	NA
RV		✓	Class 2 LSS		Assume Medium	C-F-2	10	1	0	0	NA
SA		✓	Class 2 LSS		Assume Medium	C-F-2	22	2	0	0	NA
SM		✓	Class 2 LSS		Assume Medium	C-F-2	185	19	0	0	NA
SV		✓	Class 2 LSS		Assume Medium	C-F-2	35	3	0	0	NA
VP		✓	Class 2 LSS		Assume Medium	C-F-2	20	2	0	0	NA

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
VQ		✓	Class 2 LSS		Assume Medium	C-F-1	73	0	0	0	NA
WL		✓	Class 2 LSS		Assume Medium	C-F-1	8	0	0	0	NA
						Totals	**4362**	**339**	**155**	**73**	**23**

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 Plant X 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. Inspections allocated on a prorated basis in anticipation of the final ISI program.
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. As described in Section 2.2, Alloy 82/182 welds susceptible to no degradation mechanism or PWSCC only per the RIS_B Program failure potential assessment are not included in the table.
6. Inspection locations associated with the HSS Class 3 nuclear service water system piping in the auxiliary feedwater pump room are not included in the above table.

ENCLOSURE 1
Plant X
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)