SCE&G®
A SCANA COMPANY

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August 12, 2016
NND-16-0240
10 CFR 50.55a

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

Subject: Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3
Docket Numbers 52-027 and 52-028
Request for Approval of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping

Pursuant to 10 CFR 50.55a(z)(1), South Carolina Electric & Gas Company (SCE&G), acting on behalf of itself and the South Carolina Public Service Authority (Santee Cooper), hereby requests NRC authorization to use an alternative to the requirements of Section XI, IWB-2500, of the ASME Boiler and Pressure Vessel (B&PV) Code, 2007 Edition through 2008 Addenda (code of record) for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3. The proposed alternative involves a request to use a risk-informed inservice inspection program for ASME Class 1 and 2 piping.

The details of the 10 CFR 50.55a(z)(1) request are contained in the enclosure to this letter. Approval is requested by June 30, 2017.

This letter contains no regulatory commitments. Should you have any questions, please contact Mr. Richard Troficanto at (803)941-9873.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 12th day of August 2016.

Sincerely,

April Rice
Manager
New Nuclear Licensing

BB/ARR/bb

Enclosure 1: Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1) – Request for Approval of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping

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NND-16-0240

Enclosure 1

Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)

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

(Enclosure 1 consists of 35 pages)

Plant Site - Unit:	VC Summer, Units 2 and 3
Interval - Dates:	It is expected that this alternative will be implemented at the start of the first inspection for both Unit 2 and Unit 3. The interval start dates have not yet been formally determined.
Requested Date for Approval :	Approval is requested by June 30, 2017.
ASME Code Components Affected:	All Class 1 and 2 piping welds – Examination Categories B-F (excluding component to component welds), B-J, C-F-1, and C-F-2.
Applicable Code Edition and Addenda:	The applicable Code of Record for each Unit will be determined 12 months prior to initial fuel load for each Unit per 10CFR50.55a. For the preservice inspection program the ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition through the 2008 Addenda is being used.
Applicable Code Requirements:	For Unit 2 and Unit 3, the requirements from which an alternative is requested are specified in the ASME Code, Section XI, 2007 Edition through the 2008 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.
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 components.
Proposed Alternative and Basis for Use:	In lieu of the ASME Code requirements, VC Summer 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-716-1, *Alternative Piping Classification and Examination Requirements,* Section XI, Division 1. Code Case N-716-1 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 (R.G.) 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 was concluded (e.g. References 6, 7, 9, 10 and 11 of Enclosure 1) 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-716-1 replaces a detailed evaluation of the safety significance of each pipe segment required by EPRI TR 112657, Rev. B-A with a predetermined 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). 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, excluding component to component welds, 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 VC Summer Unit 2 and Unit 3 specific submittal is attached that mirrors previous RIS_B submittals to the NRC with additional relevant information. All other ASME Code, Section XI requirements for which alternative was not specifically requested and approved in this alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.
Duration of Proposed Alternative:	For Unit 2 and Unit 3, use of the proposed alternative is requested for the duration of the First Inservice Inspection Interval.
Precedents:	Similar alternatives have been approved for Vogtle Electric Generating Plant 1 & 2, 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.

APPLICATION OF ASME CODE CASE N-716-1

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

Technical Acronyms/Definitions Used in the Template

ASME	American Society of Mechanical Engineers
BDS	Blowdown System
BER	Break Exclusion Region
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CIV	Containment Isolation Valve
CLERP	Conditional Large Early Release Probability
CVS	Chemical Volume and Control System
DM	Degradation Mechanism
E-C	Erosion-Cavitation
ECSCC	External Chloride Stress Corrosion Cracking
EPRI	Electric Power Research Institute
FAC	Flow-Accelerated Corrosion
F&O	Facts and Observations
FWS	Feedwater System
GL	Generic Letter
HELB	High Energy Line Break (synonymous with BER)
HSS	High Safety-Significant
IE	Initiating Events Analysis
IF	Internal Flooding
IFIV	Inside First Isolation Valve
IGSCC	Intergranular Stress Corrosion Cracking
ILOCA	Isolable Loss of Coolant Accident
IRWST	In Containment Refueling Water Storage Tank
ISI	Inservice Inspection
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LSS	Low Safety-Significant
MIC	Microbiologically-Influenced Corrosion
MOV	Motor Operated Valve
MRP	Materials Reliability Program
MSS	Main Steam System
NAPS	North Anna Power Station
NDE	Nondestructive Examination
NNS	Non-Nuclear Safety

Technical Acronyms/Definitions Used in the Template

NPS	Nominal Pipe Size
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PBF	Pressure Boundary Failure
PIT	Pitting
PLOCA	Potential Loss of Coolant Accident
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWRs	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PXS	Passive Core Cooling System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
R.G.	Regulatory Guide
RI-BER	Risk-Informed Break Exclusion Region
RI-ISI	Risk-Informed Inservice Inspection
RIS_B	Risk-Informed/Safety Based Inservice Inspection
RNS	Normal Residual Heat Removal System
RV	Reactor Pressure Vessel
SLB	Steam Line Break
SFW	Startup Feedwater System
SGTR	Steam Generator Tube Rupture
SR	Supporting Requirements
SXI	Section XI
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
TR	Technical Report
TT	Thermal Transients
Vol	Volumetric

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1. INTRODUCTION

VC Summer Units 2 and 3 (Summer) are currently scheduled to commence commercial operation in 2019 and 2020, 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. Summer 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 will be determined 12 months prior to initial fuel load for each unit per 10CFR50.55a.

The RIS_B process used in this submittal is based upon ASME Code Case N-716-1, *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 Risk Assessment (PRA) Quality

The methodology in Code Case N-716-1 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 predetermined population.

To that end, an EPRI Topical Report (1021467-A; Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidelines for Risk-Informed In-Service Inspection Programs) was developed and approved by the USNRC for identifying that portion of the plant-specific PRA (i.e. which supporting requirements (SRs)) that is needed to support the RI-ISI (RIS_B) application and for those portions of the PRA that are needed, what level of technical rigor (i.e. capability category) is needed.

As discussed in 1021467-A, there are some elements of the PRA that cannot be completed until the plant has gone operational (e.g. operating data). TR-1021467-A provides guidance on what interim steps can be taken to assure a robust and stable ISI program. With the exception of three SRs (HR-A2, HR-A3 and HR-G6), the Summer PRA meets the guidance contained in TR-1021467-A.

Given the small number of SRs not meeting 1021467-A guidance, the nature of these three SRs, and that N716-1 provides a substantive inspection population, it is expected that when these SRs are incorporated into the Unit 2 and 3 PRA, the RI-ISI inspection population will not be significantly impacted.

In addition, consistent with 1021467-A guidance, Units 2 and 3 will take the necessary steps to update the PRA and the RI-ISI program when that information becomes available (e.g. 10CFR50.71(h)(1) and 10CFR50.71(h)(2)). If there any new inspections as a result of these updates, they will be added to the RI-ISI program and conducted during the remainder of the inspection interval.

Finally, a number of USNRC approved RI-ISI evaluations concluded that external events are not likely to impact the RI-ISI results. This position is further supported by Section 2 of TR-1021467-A which concludes that quantification of these events will not change the conclusions derived from the RI-ISI (RIS_B) process. As a result, there is no need to further consider these events in the Unit 2 and 3 RI-ISI application.

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-716-1. The RIS_B Program will be substituted for the program currently under development for Class 1 and 2 piping (Examination Categories B-F, excluding component welds, 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 Applicability of ASME Code Case N716-1 to the AP1000 Design

In determining the applicability of N716-1 to the New Build fleet several consideration are of note as listed below and discussed in the following paragraphs.

- Scope of the program
- Determination of failure potential (material selection, operating conditions and operating experience)

Code Case N-716-1 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. As discussed above, the existing Summer PRA and future revisions to the PRA will ensure that any plant-specific piping not included in the pre-determined population of HSS components will be incorporated into the RIS_B program.

As to the predetermined scope of HSS components, Class 1 and Class 2 components can be discussed separately.

For Class 1, all components, including piping welds, are by definition HSS. Class 1 components, excluding piping welds, will be subject to existing deterministic SXI requirements. That is, no change. This assures that any New Build unique components will continue to meet deterministic SXI requirements. Only class 1 piping welds receive alternate treatment per N716-1 (e.g. element selection and NDE requirement)

For Class 2 components, the AP1000 design has eliminated, from the Class 2 pressure boundary, a number of components that are typically identified as Class 2 by the operating fleet (e.g. containment spray system). This is consistent with the N716-1 approach in that containment spray is also considered low safety significant. The AP1000 design also is consistent with the operating fleet in that portions of main steam, feedwater and residual heat removal system remain Class 2 and are HSS per N716-1. Any remaining Class 2 systems or portions of systems not HSS per N716-1 have been shown to contribute to less than 1E-06 CDF (1E-07 LERF) which is also consistent with experience on the operating fleet.

With respect to failure potential, a review of the operating characteristics and material selection of the New Build fleet as compared to the operating fleet was conducted.

Assessment of plant-specific operating characteristics are a fundamental component of the failure potential evaluation required by N716-1. The N716-1 process is more realistic and relevant with respect to identifying potential degradation prior to flaws propagating throughwall as compared to the deterministic SXI process that focuses on high design stress locations.

As discussed in Appendix A, the material selection process used by the New Build fleet reflects a robust assessment and understanding of the operating fleet experience. The materials selected are appropriate for the operating conditions (normal and abnormal) expected and have been previously analyzed using the N716-1 process at a large number plants for essentially identical conditions.

Additionally, the living program component of the N716-1 process provides for more real-time incorporation of operating events as compared to the deterministic SXI approach.

As such, the above considerations coupled with the N716-1 “inspection for cause” approach provides for a more robust and informed ISI program as compared to the deterministic SXI approach especially for plant designs such as the AP1000 which do not have any deterministic SXI experience.

2.3 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 being maintained in accordance with the risk-informed break exclusion region methodology (RI-BER) described in EPRI Report 1006937-A, *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.

- Summer is conducting 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.* The results of this effort will be incorporated into the RIS_B Program, as applicable.

- As Alloy 600/82/182 has been explicitly excluded from the Summer (AP1000) design, augmented inspections per ASME Code Case N-770-1 and 10CFR50.55a(g)(6)(ii)(F) [dated June 21, 2011] are not applicable.

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-716-1 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 (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 (Unit 2). <u>Unit 3 will be essentially identical.</u> 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-716-1 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.

(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 (RV) to the second isolation valve (i.e., farthest from the RV) 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 RV to the second isolation valve (i.e., farthest from the RV) 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-716-1, 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.

Low safety-significant (LSS) welds include all other Class 2, 3, or Non-Class welds.

3.2 Failure Potential Assessment

While the deterministic ASME SXI program selects inspection locations based on high design stress levels and structural discontinuities, operating experience has shown that well designed and engineered piping systems do not fail (e.g. flaw, indications, throughwall leakage) due to design basis events contained within the design stress reports. Rather, piping failures are due to conditions and stressors not contained within the design stress report.

As such, failure potential estimates for the Summer application were generated utilizing industry failure history, plant-specific as anticipated to be operated conditions, 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 Summer. 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 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 Summer 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 Summer RIS_B application.

3.3 Element and NDE Selection

Code Case N-716-1 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 RV) and the RV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (not applicable for Summer) 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-716-1 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 (Unit 2). Unit 3 will be essentially identical. 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 in the risk impact assessment.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		All Piping Welds(3)	
	Total	Selected	Total	Selected	Total	Selected
2	693	74	526	22	1314	104

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, 239 are HSS; the remaining are LSS.
(3) The total weld count and selections include some Class 3 BER piping. Also, 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.

As opposed to the deterministic SXI approach which is essentially a random inspection approach, N716-1 requires an “inspection for cause” approach be used for determining inspection locations and inspection requirements. For example, the deterministic SXI program for 4NPS and less, requires only an outside diameter (OD) surface exam. If some type of degradation were to be operative at this location (e.g. thermal fatigue), the deterministic SXI exam would only find the flaw after it had gone throughwall. While the N716-1 approach would require a volumetric exam (e.g. UT) of the inner 1/3 diameter which is much more effective (informed) at finding flaws prior to failure.

3.3.1 Current Examinations

If this alternative were not approved, the deterministic ASME Section XI inspection methodology for ISI examination of piping welds per the applicable Code of Record for each Unit as determined 12 months prior to initial fuel load for each Unit per 10CFR50.55a would be followed.

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-716-1. 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.

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-716-1. 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).

3.3.4 Proposed Alternative

Consistent with previously approved RIS_B submittals, Summer 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. Alternatives for those cases where greater than 90% coverage is not obtained will be submitted per the requirements of 10 CFR 50.55a(z).

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-716-1, 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-716-1, 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-716-1 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-716-1 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-716-1. 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.

Summer has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716-1 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).

Break Location	Estimated		Consequence Rank	Upper / Lower Bound		Description of Affected Piping
	CCDP	CLERP		CCDP	CLERP	
LOCA	3E-4	3E-05	HIGH	(U) 3E-04 (L) 1E-04	(U) 3E-05 (L) 1E-05	Unisolable RCPB piping of all sizes
The highest CCDP is Large LOCA, %LLOCA (0.1 margin used for CLERP)						
ILOCA [1]	<1E-06	<1E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping between 1st and 2nd normally open isolation valve inside containment (CVS letdown/charging)
Calculated based on Large LOCA CCDP of 3E-4 and valve fail to close probability of 3E-3 (0.1 margin used for CLERP)						
PLOCA [1]	<1E-06	<1E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Piping beyond the 1st normally closed isolation valve inside containment (CVS pressurizer spray, Accumulator discharge, PXS IRWST, RCS automatic depressurization, RNS hot leg suction and return)
Calculated based on Large LOCA CCDP of 3E-4 and valve rupture probability of <1E-3 (0.1 margin for CLERP).						
PPLOCA [1]	<1E-06	<1E-07	LOW	(U) 1E-06 (L) 1E-06	(U) 1E-07 (L) 1E-07	Piping beyond the 2nd normally closed isolation valve inside containment (RNS)
Calculated based on Large LOCA CCDP and failure of 2 normally closed valves (0.1 margin for CLERP).						
SLB	2E-06	2E-07	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	Secondary breaks, including BER scope, in the FWS, MSS, BDS and SFW systems
The bounding CCDP value conservatively used for all feedwater and steam line breaks, %SLBU, is for steam line breaks upstream of the MSIVs (0.1 margin for CLERP)						
LSS	1E-04	1E-05	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	All other Class 2 system piping designated as low safety significant
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-716-1 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.

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 (Unit 2) presents 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. Unit 3 will be essentially identical. 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 table, 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-716-1 are satisfied.

Summer Unit 2

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
BDS - Blowdown System	-2.40E-13	-2.40E-14	4.00E-13	4.00E-14
CVS - Chemical and Volume Control System	-8.80E-11	-8.80E-12	8.00E-11	8.00E-12
FWS - Feedwater System	-1.00E-14	-1.00E-15	-1.00E-14	-1.00E-15
MSS - Main Steam System	-1.00E-14	-1.00E-15	-1.00E-14	-1.00E-15
PXS - Passive Core Cooling System	2.53E-10	2.53E-11	7.33E-10	7.33E-11
RCS - Reactor Coolant System	1.46E-10	1.46E-11	6.86E-10	6.86E-11
RNS - Normal Residual Heat Removal System	-7.10E-12	-7.10E-13	3.00E-11	3.00E-12
SFW - Startup Feedwater	4.10E-13	4.10E-14	4.10E-13	4.10E-14
Total	**3.03E-10**	**3.03E-11**	**1.53E-09**	**1.53E-10**

As shown in Table 3.4 (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). Unit 3 will be essentially identical. 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-716-1 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-716-1, 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-716-1 requires that any piping segment or component 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.

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-716-1 will be prepared to implement and monitor the program. The new program will be implemented at the start of the first 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 will be conducted using the deterministic ASME Section XI requirements.

4. PROPOSED ISI PLAN CHANGE

Summer Units 2 and 3 are anticipated to commence commercial operation in 2019 and 2020, respectively.

A comparison between the RIS_B Program and the 2007 Edition of Section XI program requirements for first interval in-scope piping is provided in Table 4 (Unit 2). Unit 3 will be essentially identical.

5. REFERENCES/DOCUMENTATION

1. EPRI Report 1006937-A, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs.*
2. EPRI TR-112657, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A.
3. ASME Code Case N-716-1, *Alternative Piping Classification and Examination Requirements, Section XI Division 1.*
4. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.*
5. Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping.*
6. 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
7. 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.
8. EPRI Report 1021467-A Nondestructive Evaluation: *Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs.*
9. Waterford-3 Safety Evaluation – See ADAMS Accession No. ML080980120.
10. Vogtle Electric Generating Plant Safety Evaluation - See ADAMS Accession No. ML100610470.
11. North Anna Power Station (NAPS) Units 1 and 2 Safety Evaluation – See ADAMS Accession No. ML110050003.
12. REGULATORY GUIDE 1.147, *INSERVICE INSPECTION CODE CASE ACCEPTABILITY, ASME SECTION XI, DIVISION 1*, August 2014, Revision 17
13. Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*
14. Materials Reliability Program Report MRP-146, Revision 1*: Management of Thermal Fatigue in Normally Stagnant Non-Isolatable Reactor Coolant System Branch Lines.*
15. ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*

Supporting Onsite Documentation

"N716-1 Evaluation for Summer Units 2 and 3"

"Degradation Mechanism Evaluation for Summer Units 2 and 3"

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

System	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
BDS	26				✓		✓	
	32							✓
CVS	23	✓					✓	
	16				✓			
	32							✓
FWS	20			✓	✓		✓	
	21				✓			
	32			✓			✓	
MSS	101				✓		✓	
	36							✓
PXS	329	✓					✓	
	150							✓
RCS	300	✓					✓	
RNS	41	✓	✓				✓	
	87		✓				✓	
	20							✓
SFW	31				✓		✓	
	17							✓
Summary Results for all Systems	652	✓					✓	
	41	✓	✓					
	87		✓					
	32			✓			✓	
	20			✓	✓		✓	
	195				✓		✓	
	287							✓

(1) System Scope:

BDS - Blowdown System

CVS - Chemical and Volume Control System

FWS - Feedwater System

MSS - Main Steam System

PXS - Passive Core Cooling System

RCS - Reactor Coolant System

RNS - Normal Residual Heat Removal System

SFW - Startup Feedwater

Table 3.2
Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
BDS		✓									
CVS		✓									
FWS											
MSS											
PXS		✓									
RCS	✓	✓									
RNS	✓	✓									
SFW											

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 portions of the BDS, CVS, MSS, PXS, RNS and SFW systems.

Table 3.3: Unit 2 Code Case N716 Selections

System (1)	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
BDS	10		TT				✓	3
	16		None				✓	0
		32	N/A					0
CVS	11		TT	✓	✓			4
	6		TT	✓				0
	2		None	✓	✓			0
	4		None	✓				0
	16		None				✓	0
		32	N/A					0
FWS	41		None				✓	7
	32		None					0
MSS	101		None				✓	11
		36	N/A					0
PXS	27		TT	✓	✓			7
	237		None	✓	✓			26
	65		None	✓				0
		150	N/A					0
RCS	10		TASCS,TT	✓	✓			3
	1		TASCS	✓	✓			0
	18		TT	✓	✓			5
	241		None	✓	✓			23
	30		None	✓				0
RNS	1		TASCS	✓	✓			1
	10		TT	✓				5
	39		TT					6
	19		None	✓	✓			0
	11		None	✓				0
	48		None					0
		20	N/A					0
SFW	31		None				✓	3
		17	N/A					0
Summary Results All Systems	10		TASCS,TT	✓	✓			3
	2		TASCS	✓	✓			1
	56		TT	✓	✓			16
	16		TT	✓				5
	10		TT				✓	3
	39		TT					6
	499		None	✓	✓			49
	110		None	✓				0
	205		None				✓	21
	80		None					0
		287	N/A					0
Totals	1027	287						104

Notes:
(1) Systems are described in Table 3.1

Table 3.4 Unit 2 Risk Impact Analysis Results

System (1)	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI (2)	RIS_B(3)	Delta	w/POD	w/o POD	w/POD	w/o POD
BDS	High	SLB	TT	Medium	2	3	1	-4.20E-11	-1.00E-11	-4.20E-12	-1.00E-12
BDS	High	SLB	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BDS	Low	Class 2 LSS	N/A	Assume Medium	3	0	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
BDS Total								**-1.20E-11**	**2.00E-11**	**-1.20E-12**	**2.00E-12**
CVS	High	LOCA	TT	Medium	6	4	-2	-1.08E-10	6.00E-11	-1.08E-11	6.00E-12
CVS	High	ILOCA/PLOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVS	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVS	High	ILOCA/PLOCA	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
CVS	Low	Class 2 LSS	None	Low	2	0	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
CVS Total								**-1.07E-10**	**6.10E-11**	**-1.07E-11**	**6.10E-12**
FWS Total	High	SLB	None	Low	6	7	1	**-5.00E-13**	**-5.00E-13**	**-5.00E-14**	**-5.00E-14**
MSS	High	SLB	None	Low	10	11	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
MSS	Low	Class 2 LSS	N/A	Assume Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MSS Total								**-5.00E-13**	**-5.00E-13**	**-5.00E-14**	**-5.00E-14**
PXS	High	LOCA	TT	Medium	26	7	-19	9.00E-11	5.70E-10	9.00E-12	5.70E-11
PXS	High	LOCA	None	Low	61	26	-35	5.25E-11	5.25E-11	5.25E-12	5.25E-12
PXS	High	ILOCA/PLOCA	None	Low	4	0	-4	2.00E-12	2.00E-12	2.00E-13	2.00E-13
PXS	Low	Class 2 LSS	N/A	Assume Medium	11	0	-11	1.10E-10	1.10E-10	1.10E-11	1.10E-11
PXS Total								**1.45E-10**	**6.25E-10**	**1.45E-11**	**6.25E-11**
RCS	High	LOCA	TASCS,TT	Medium	10	3	-7	1.80E-11	2.10E-10	1.80E-12	2.10E-11
RCS	High	LOCA	TASCS	Medium	1	0	-1	1.80E-11	3.00E-11	1.80E-12	3.00E-12
RCS	High	LOCA	TT	Medium	18	5	-13	5.40E-11	3.90E-10	5.40E-12	3.90E-11
RCS	High	LOCA	None	Low	60	23	-37	5.55E-11	5.55E-11	5.55E-12	5.55E-12
RCS	High	ILOCA/PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RCS Total								**1.46E-10**	**6.86E-10**	**1.46E-11**	**6.86E-11**
RNS	High	LOCA	TASCS	Medium	1	1	0	-3.60E-11	0.00E+00	-3.60E-12	0.00E+00
RNS	High	ILOCA/PLOCA	TT	Medium	3	5	2	-7.20E-11	-2.00E-11	-7.20E-12	-2.00E-12
RNS	High	PPLOCA	TT	Medium	3	6	3	-9.00E-13	-3.00E-13	-9.00E-14	-3.00E-14
RNS	High	LOCA	None	Low	7	0	-7	1.05E-11	1.05E-11	1.05E-12	1.05E-12
RNS	High	ILOCA/PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RNS	High	PPLOCA	None	Low	4	0	-4	2.00E-14	2.00E-14	2.00E-15	2.00E-15
RNS	Low	Class 2 LSS	N/A	Assume Medium	2	0	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
RNS Total								**-7.84E-11**	**1.02E-11**	**-7.84E-12**	**1.02E-12**

Table 3.4 Unit 2 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
SFW	Low	Class 2 LSS	N/A	Assume Medium	2	0	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
SFW Total								**2.05E-11**	**2.05E-11**	**2.05E-12**	**2.05E-12**
Grand Total					**248**	**104**	**-144**	**1.12E-10**	**1.42E-09**	**1.12E-11**	**1.42E-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-716-1 (e.g., not part of the BER scope).

Table 4: Unit 2 Inspection Location Selections Comparison

System(1)	Safety Significance		Break Location	Failure Potential(4)		Code Category	Weld Count	Section XI		Code Case N716(3)	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other(2)
BDS	✓		SLB	TT	Medium	C-F-2/BER	10	2	0	3	0
BDS	✓		SLB	None	Low	C-F-2/BER	16	0	0	0	0
BDS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	32	3	0	0	0
CVS	✓		LOCA	TT	Medium	B-J	11	6	0	4	0
CVS	✓		ILOCA/PLOCA	TT	Medium	B-J	6	0	0	0	0
CVS	✓		LOCA	None	Low	B-J	2	0	0	0	0
CVS	✓		ILOCA/PLOCA	None	Low	B-J	4	0	0	0	0
CVS	✓		ILOCA/PLOCA	None	Low	C-F-1	16	2	0	0	0
CVS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	32	2	0	0	0
FWS	✓		SLB	None	Low	C-F-2/BER	73	6	0	7	0
MSS	✓		SLB	None	Low	C-F-2/BER	101	10	0	11	0
MSS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	36	0	0	0	0
PXS	✓		LOCA	TT	Medium	B-J	27	26	0	7	0
PXS	✓		LOCA	None	Low	B-F/B-J	237	61	0	26	0
PXS	✓		ILOCA/PLOCA	None	Low	B-J	65	4	0	0	0
PXS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	150	11	0	0	0
RCS	✓		LOCA	TASCS,TT	Medium	B-F/B-J	10	10	0	3	0
RCS	✓		LOCA	TASCS	Medium	B-F/B-J	1	1	0	0	0
RCS	✓		LOCA	TT	Medium	B-F/B-J	18	18	0	5	0
RCS	✓		LOCA	None	Low	B-F/B-J	241	60	0	23	0
RCS	✓		ILOCA/PLOCA	None	Low	B-F/B-J	30	0	0	0	0
RNS	✓		LOCA	TASCS	Medium	B-J	1	1	0	1	0
RNS	✓		ILOCA/PLOCA	TT	Medium	B-J	10	3	0	5	0
RNS	✓		PPLOCA	TT	Medium	C-F-1	39	3	0	6	0
RNS	✓		LOCA	None	Low	B-J	19	7	0	0	0
RNS	✓		ILOCA/PLOCA	None	Low	B-J	11	0	0	0	0
RNS	✓		PPLOCA	None	Low	C-F-1	48	4	0	0	0
RNS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	20	2	0	0	0
SFW	✓		SLB	None	Low	C-F-2/BER	31	4	0	3	0
SFW		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	17	2	0	0	0
						Totals	**1314**	**248**	**0**	**104**	**0**

Notes to Table 4

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-716-1. Code Case N-716-1 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 Summer 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”).

Appendix A

AP1000 MATERIALS REVIEW

Introduction

This task examines the performance history of a group of structural materials for new build PWRs, in particular the materials in the Westinghouse AP1000 design for ASME Code Class 1 and 2, Non-Class break exclusion region (BER) and some additional Non-Class system piping. As part of this effort, the materials identified in Table A-1 have been designated as materials requiring review in the new Westinghouse AP1000 designed PWR.

The purpose of this study is to document the use of these materials in operating BWRs or PWRs, where applicable, to demonstrate that sufficient evidence is available to confirm the performance of these materials under LWR operating conditions for substantial operating times in the nuclear power plant high temperature water environments.

As observed from Table A-1, the materials consist of three different groups of materials. The materials within these groups are as identified below:

I. Austenitic Stainless Steel

1. Type 304 L Stainless Steel
2. Type 316 LN Stainless Steel

II. Carbon Steel

3. SA-333 Grade 6 Carbon Steel

III. Low Alloy Steel

4. SA-335 Grade P11 Low Alloy Steel

Review of Austenitic Stainless Steels

The austenitic stainless steels, Type 304L, and Type 316L, have been used extensively in BWRs and PWRs for high temperature primary pressure boundary operation for most of the history of these plants. These materials are the low carbon forms of Type 304 and Type 316 stainless steel, and were incorporated into the nuclear power plants to provide margin against corrosion related problems such as general corrosion, pitting, intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC).

Type 316 LN stainless steel is a slight variation of Type 316 L stainless steel in which a small amount of nitrogen is added to the alloy to replace the carbon that is reduced from that in Type 316 stainless steel when fabricating Type 316 L stainless steel. As stated above, the reduced carbon level reduces the risk of pitting (in the case of Type 316 LN stainless steel) and stress corrosion cracking (SCC) in the nuclear power plant high temperature environment. The Type 316 LN alloy, is a slight variation of Type 316 Nuclear Grade (NG) stainless steel. Type 316 NG stainless steel has been used extensively in recirculation piping systems in BWRs as the preferred replacement to high carbon grades of austenitic stainless steel so as to obtain improved resistance to SCC while maintaining the strength of the corresponding higher carbon grades of austenitic stainless steel [1]. Type 316 LN stainless steel allows for slightly more carbon than does Type 316 NG stainless steel, thereby making the material more readily procured. Westinghouse has incorporated Type 316 LN alloy in the AP1000 design PWR, and the submittal for use of this material has been approved by the Nuclear Regulatory Commission (NRC) [2].

Review of SA-333 Grade 6 Carbon Steel

SA-333 Grade 6 carbon steel is an alloy that is used often in the nuclear industry. It essentially has the same composition and mechanical properties as SA-106 Grade B carbon steel. The differences relate to the fact that the SA-106 Grade B can contain unspecified minor elements such as chromium, copper, nickel, molybdenum and vanadium. These alloys are often added to SA-106 Grade B carbon steel for specific additional characteristics, such as improved FAC resistance or toughness. SA-333 Grade 6 does not allow these elements. However, because of the increased quality control of these elements, SA-333 Grade 6 carbon steel has guaranteed greater toughness at low temperatures (< -20^{0}F), than does SA-106 Grade B.

These alloys are both widely used in the nuclear industry and are quite often used interchangeably.

Review of SA-335 Grade P11

SA-335 Grade P11 low alloy steel, also known as 1¼ Cr- ½Mo low alloy steel has been used increasingly in nuclear power plant primary water piping since the mid 1980's. This alloy, and its companion alloy, SA-335 Grade P22 (2¼ Cr-1 Mo) low alloy steel, were originally specified as replacement materials to carbon steel related to erosion-corrosion damage in U. S. nuclear power plant carbon steel piping. The erosion-corrosion incidents were initially observed in 1978 and 1980 and a significant rupture occurred in 1986 in a PWR in a feedwater system pipe fabricated of carbon steel [3, 4]. As noted in Reference 4, the recommended replacement material for this phenomenon was identified as SA-335 Grade P22, or other low alloy steels, as were currently available for replacement. Several plants replaced with these low alloy steel components. Following these and other incidents of wall thinning and failure of carbon steel piping in feedwater systems, the NRC issued Bulletin 87-01 to all nuclear power plant licensees, to submit information concerning their programs for monitoring the thickness of pipe walls in high-energy single-phase and two-phase carbon steel piping systems [4]. Information Notice 88-17 summarized responses to the Bulletin 87-01.

EPRI was involved in these investigations developing the CHECWORKS™ code that EPRI utilized to develop a set of recommendations to help utility personnel design and implement a comprehensive FAC mitigation program [5]. This document presents a set of recommendations for an effective flow-accelerated corrosion program. These recommendations are the product of successful implementation of FAC inspection programs and experience of operating nuclear power plants. The essential ingredients for an effective FAC program have been presented in that document, including the steps that utilities should take to minimize the chances of experiencing a FAC-induced consequential leak or rupture.

One of the major features of that program is the recommendation in Section 4.2.2 of the Reference 5 report that "Based on laboratory and plant experience, the following systems can be safely excluded from further evaluation:

"Systems or portions of systems made of stainless-steel piping or low-alloy steel piping with nominal chromium content equal to or greater than 1.25 % (high content of FAC-resistant alloy). This exclusion pertains only to complete piping lines manufactured of FAC-resistant alloy."

Thus, SA-335 Grade P11 systems are now excluded from further evaluation for FAC resistance.

Summary

A review was performed of the list of materials identified for the Westinghouse AP1000 PWR design, identified in Table A-1, which was designed to document that sufficient evidence is available to confirm the performance of these materials under LWR operating conditions for substantial operating times in the nuclear power plant high temperature water environments.

The review demonstrated that all of these materials, or close analogs of these alloys, have seen extensive use in high temperature, high pressure components within the nuclear fleet and have performed successfully in these applications.

In addition, the NRC has endorsed the use of all of these alloys, specifically for the AP1000 in References 2 and 6.

References

1. "Alternative Alloys for BWR Pipe Applications", EPRI NP-2671-LD, October 1982.
2. U. S. NRC Response Subject: AP1000 Response to Request for Additional Information (SRP4.5.1), ML081550222, May 30, 2008.
3. U. S. NRC Bulletin NO. 87-01, "Thinning Of Pipe Walls in Nuclear Power Plants", July 9, 1987.
4. Information Notice 88-17, "Summary of Responses to NRC Bulletin 87-01, "Thinnning of Pipe Walls in Nuclear Power Plants", April 22, 1988
5. U. S. NRC NUREG-1344, "Erosion Corrosion Induced Pipe Wall Thinning In U. S. Nuclear Power Plants", April 1989.
6. EPRI 30020000563, Technical Report, "Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4), November 2013.
7. U. S. NRC Evaluation of Westinghouse Revision 16 and 17 to AP1000 Design Control Document to Add SA-335 Grade P11, page 3-49, ML103430502.

Table A-1

System Group	Subsystem	Material Selected
RCPB	Reactor Coolant System (RCS), including hot legs, cold legs, Automatic Depressurization System (ADS) lines, and pressurizer surge, spray and relief valve lines	ASME SA-312 GR TP316LN, Seamless, B36.10M (i.e., Type 316 austenitic stainless steel)
	Chemical and Volume Control System (CVS) purification loop lines, including letdown, charging and pressurizer auxiliary spray line	ASME SA-312 GR TP316LN, Seamless, B36.10M (i.e., Type 316 austenitic stainless steel)
	Normal Residual Heat Removal System (RNS), including the loop 2 hot leg take-off line and the return lines to the direct vessel injection (DVI) lines	ASME SA-312 GR TP316LN, Seamless, B36.10M (i.e., Type 316 austenitic stainless steel)
	Passive Core Cooling System (PXS), including take-off lines to the Passive Residual Heat Removal Heat Exchanger (PRHR HX) and Core Makeup Tanks (CMT), and return lines from the Accumulators, CMTs and in-containment refueling water storage tank (IRWST) to the DVI lines*	ASME SA-312 GR TP316LN, Seamless, B36.10M (i.e., Type 316 austenitic stainless steel)
SDC	The RNS take-off lines from valves V002A and V002B to the containment penetration, including a branch line from the IRWST and a branch line from the CVS	ASME SA-312 GR TP304L, Seamless, B36.10M (i.e., Type 304 austenitic stainless steel)
	The RNS return lines from the containment penetration to valves V015A and V015B, including a branch line to the IRWST and a branch line to the CVS	ASME SA-312 GR TP304L, Seamless, B36.10M (i.e., Type 304 austenitic stainless steel)
FW	Main feedwater lines	SA-335 Gr. P11 seamless ferritic alloy steel pipe and ASTM A-106 Gr. B seamless carbon steel pipe (Non Class 2 portion)
BER	Main steam lines, including lines to valves, MSIV bypass lines and drain lines	SA-335 Gr. P11 seamless ferritic alloy steel pipe)

Table A-1

System Group	Subsystem	Material Selected
	Main feedwater lines	SA-335 Gr. P11 seamless ferritic alloy steel pipe and ASTM A-106 Gr. B seamless carbon steel pipe (Non Class 2 portion)
	Startup feedwater lines	ASME SA-333 GR 6, B36.10M carbon steel piping
	Steam generator blowdown lines	ASME SA-335 GR P11, Seamless, B36.10M (i.e., SA-335 Gr. P11 seamless ferritic alloy steel pipe
	CVS makeup line between valve CVS-PL-V090 and valve CVS-PL-V091, passing through containment penetration C03	ASME SA-312, GR TP304L, B36.10M stainless steel piping