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SOUTHERN COMPANY
Energy to Serve Your World

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

Vogtle Electric Generating Plant
Request for Approval of Risk-Informed/Safety Based
Inservice Inspection Alternative for Class 1 And 2 Piping
VEGP-ISI-ALT-02, Version 1

Ladies and Gentlemen:

Pursuant to the requirements of 10 CFR 50.55a (a) (3) (i), Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electric Generating Plant (VEGP) Units I and 2, requests authorization to implement Risk-Informed/Safety Based Inservice Inspection (RIS_B ISI) alternative VEGP-ISI-ALT-02. This alternative will be used in lieu of the existing ASME Section XI Code Category B-F, B-J, C-F-1, and C-F-2 requirements for examination of Class 1 and 2 piping welds. This alternative, which is described in Enclosure I to this letter, has been developed in accordance with Code Case N-716, "Alternative Piping Classification and Examination Requirements."

SNC plans to implement the proposed alternative during the third 10-year inservice inspection interval that began on May 31, 2007. Implementation details are provided in the alternative. To facilitate the NRC's review, this alternative contains a template format modeled after previous submittals that the NRC has approved and a detailed evaluation of the PRA adequacy, including a gap analysis performed against Regulatory Guide 1.200. SNC requests approval of the RIS_B ISI Program by February 26, 2010, to facilitate planning for the remainder of the inspection interval.

Sincerely,

Mark J Ajluni

M. J. Ajluni
Manager, Nuclear Licensing

MJA/TAH/daj

Enclosure: 1. Inservice Inspection Alternative for Class 1 And 2 Piping
VEGP-ISI-ALT-02, Version 1

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Vogtle Electric Generating Plant
Request for Approval of Risk-Informed/Safety Based
Inservice Inspection Alternative for Class 1 And 2 Piping
VEGP-ISI-ALT-02, Version 1

Enclosure 1

Inservice Inspection Alternative for Class 1 And 2 Piping
VEGP-ISI-ALT-02, Version 1

ENCLOSURE 1
VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)
VEGP-ISI-ALT-02, VERSION 1.0

Plant Site-Unit:	Vogtle Electric Generating Plant, Units 1 and 2 (VEGP-1&2).
Interval Dates:	Third ISI Interval – May 31, 2007 through May 30, 2017.
Requested Date for Approval :	Approval is requested by February 26, 2010.
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:	The applicable Code edition and addenda is ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant components,” 2001 Edition with 2003 addenda. In addition, as required by 10 CFR 50.55a, piping ultrasonic examinations are performed per ASME Section XI, 2001 Edition, Appendix VIII, “Performance Demonstration for Ultrasonic Examination Systems.”
Applicable Code Requirements:	For the current inservice inspection (ISI) program at VEGP-1&2, IWB-2200 IWB-2420, IWB-2430, and IWB-2500 provide the examination requirements for Category B-F and Category B-J welds. Similarly, IWC-2200, IWC-2420, IWC-2430, and IWC-2500 provide the examination requirements for Category C-F-1 and C-F-2 welds.
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 existing Code requirements, Southern Nuclear Operating company (SNC) proposes to use a RIS_B process as an alternate to the current ISI program for Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, “Alternative Piping Classification and Examination Requirements, Section XI Division 1”. Code Case N-716 is founded, in large part, on the risk-informed ISI (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, followed by a screening flooding analysis to identify any plant-specific high safety-significant segments (Class 1, 2, 3, or Non-Class). The screening flooding analysis was performed in accordance with Regulatory Guide 1.200, Revision 1 and the flooding analysis described in Section 4.5.7 of ASME RA-Sb-2005, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to ASME RA-S-2002. (The screening did not identify any plant-specific high safety-significant segments). By using risk-insights to focus examinations on more important examination locations, while meeting the intent and principles of Regulatory Guides 1.174 and 1.178, this proposed RIS_B 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 detailed Template is

	attached that mirrors previous RIS_B submittals to the NRC.
Duration of Proposed Alternative:	Through May 30, 2017.
Precedents:	Similar alternatives have been approved for Donald C. Cook 1 and 2, Grand Gulf Nuclear Station, and Waterford-3.
References:	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.
Status:	Awaiting NRC approval.

TEMPLATE SUBMITTAL

APPLICATION OF ASME CODE CASE N-716

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

Technical Acronyms/Definitions Used in the Template

AC	Alternating Current
AFW	Auxiliary Feedwater
AOV	Air Operated Valve
AOVLOCA	LOCA Isolated by an Air Operated Valve
ARV	Atmospheric Relief Valve
ASME	American Society of Mechanical Engineers
ATWT	Anticipated Transient Without Trip
BER	Break Exclusion Region
BL-PRA	Base Line PRA
CAFTA	Computer-Aided Fault Tree Analysis
CC	Crevice Corrosion
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCPs	Centrifugal Charging Pumps
CDF	Core Damage Frequency
CIV	Containment Isolation Valve
Class 2 LSS	Class 2 Pipe Break in LSS Piping
CLERP	Conditional Large Early Release Probability
CS	Containment Spray
CST	Condensate Storage Tank
CVCS	Chemical Volume and Control System
DG	Diesel Generator
DM	Degradation Mechanism
E-C	Erosion-Corrosion
ECCS	Emergency core Cooling Systems
ECSCC	External Chloride Stress Corrosion Cracking
EDG	Emergency Diesel Generator
FAC	Flow-Accelerated Corrosion
F&O	Facts and Observations
FT	Fault tree
FW	Feedwater
HEP	Human Error Probability
HFE	Human Failure Event
HRA	Human Reliability Analysis
HSS	High Safety-Significant
HX	Heat Exchanger
IE	Initiating Event
IF	Internal Flooding
IFIV	Inside First Isolation Valve
IGSSC	Intergranular Stress Corrosion Cracking
ILOCA	Isolable Loss of Coolant Accident
IPE	Individual Plant Evaluation
IPLOCA	ILOCA or PLOCA Occurs During Operation/Standby
ISLOCA	Inter-system LOCA
LERF	Large Early Release Frequency
LERF-CFE	LERF - Containment Failure Early

Technical Acronyms/Definitions Used in the Template (Continued)

LERF-ISO	LERF- Isolation Failure
LOCA	Loss Of Coolant Accident
LSS	Low Safety-Significant
MAAP	Modular Accident Analysis Program
MGL	Multiple Greek Letter
MIC	Microbiologically-Influenced Corrosion
MOV	Motor Operated Valve
MR	Maintenance Rule
MS	Main Steam
MSPI	Mitigating Systems Performance Indicator
MV	Manual Valve
MVLOCA	LOCA Isolated by a Manual Valve
NDE	Nondestructive Examination
NNS	Non-Nuclear Safety
NPS	Nominal Pipe Size
NSCW	Nuclear Service Cooling Water
OA	Operator Action
OC	Outside Containment
PBF	Pressure Boundary Failure
PIT	Pitting
PLOCA	Potential Loss of Coolant Accident
PLOCASD	Potential LOCA in SDC Suction Piping
PLOCASD2	PLOCASD Between the Second MOV and the Containment Penetration
POD	Probability of Detection
PORV	Power Operated Relief Valve
PPLOCA	Potential LOCA in Class 2 Piping Requiring Failure of Two Check Valves in Series
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PWR: FW	Pressurized Water Reactor: Feedwater
PWROG	Pressurized Water Reactor Owner’s Group
PWSCC	Primary Water SCC
PZR	Pressurizer
RWST	Refueling Water Storage Tank
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RHR	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
SAIC	Science Applications International Corporation
SAMA	Severe Accident Management Alternatives
SBO	Station Blackout
SDC	Shutdown Cooling

Technical Acronyms/Definitions Used in the Template (Continued)

SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIP	Safety Injection Pump
SSBI	Main Steam or Feedwater Break inside the Outer CIV
SSBO	Main Steam or Feedwater Break Beyond the Outer CIV
SSC	Systems, Structures, and Components
SI	Safety Injection
Sur	Surface
SV	Safety Valve
SXI	Section XI
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
TR	Technical Report
TT	Thermal Transients
Vol	Volumetric
WOG	Westinghouse Owner's Group

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1. INTRODUCTION

Vogtle Electric Generating Plant Units 1 and 2 (VEGP 1&2) is currently in the third 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. VEGP 1&2 plans to implement a risk-informed/safety-based inservice inspection (RIS_B) program in the first inspection period of the third ISI interval. The third interval commenced in May 31, 2007 for VEGP Units 1 and 2.

The ASME Section XI code of record for the third ISI interval at VEGP is the 2001 Edition with 2003 Addenda for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components.

The RIS_B process used in this submittal is based upon ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*, which is founded in large part on the RIS_B 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 VEGP PRA has been demonstrated to be adequate for this application. The history and development of the PRA is described in further detail in Attachment A. As described in Attachment A, a complete re-analysis of internal flooding events has been completed to the ASME Standard and Regulatory Guide 1.200, Revision 1. In addition, the internal flooding PRA was reviewed by an independent contractor to confirm compliance with these standards. The PRA, as a whole, has undergone several updates to maintain the model current with the plant design and operation. All Westinghouse Owner's Group (WOG) peer review "B" findings from a peer review conducted in 2001 (there were no "A" findings for the VEGP PRA) were addressed in the Revision 3 PRA model. The Revision 3 model was reviewed by internal reviewers. Additionally, as a part of the mitigating system performance indicator (MSPI) scoping and implementation, the Revision 3 model was partially reviewed by selected NRC region staff, as well as industry peers. A gap analysis of the Revision 3 model versus the ASME Standard and Regulatory Guide 1.200 was performed by an external contractor. The evaluation of the gaps, applicable to this submittal, are included in Attachment A.

The PRA model for internal events (except internal flooding) used for the RIS_B evaluation was the Vogtle PRA L2UP model. The Vogtle PRA L2UP model includes an upgraded level 1 internal event PRA model and a level 2 PRA model. The upgraded level 1 PRA model included in the VEGP L2UP model was based on the VEGP Level 1 PRA model Revision 3. The upgraded level 2 PRA model included in the L2UP model was based on new PWROG methodology (WCAP-16341-P), which was intended to develop an ASME PRA standard Capability Category II level 2 PRA model. The Vogtle PRA L2UP model was used for the Vogtle Severe Accident Management Alternatives (SAMA) Analysis for the VEGP license renewal submitted in 2007.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

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

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be substituted for the current program for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected.

2.2 Augmented Programs

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

- The plant augmented inspection program for high-energy line breaks outside containment, implemented in accordance with VEGP Final Safety Analysis Report (FSAR) Section 6.6 and Technical Specification 5.5.16, has not been revised 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*. Therefore, 100% of these welds will continue to be examined per the VEGP Final Safety Analysis Report (FSAR) Section 6.6 and Technical Specification 5.5.16 requirements. It is the intention of Vogtle to implement the RI-BER program later during the third ISI interval.

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

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

- Since the issuance of the NRC safety evaluation for EPRI TR 112657, Rev. B-A , several instances of primary water stress corrosion cracking of Alloy 82/182 welds has occurred at pressurized water reactors. For examination of these welds, a plant augmented inspection program is already being implemented at VEGP in response to MRP-139, *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines*. The requirements of MRP-139 are used for the inspection and management of Primary Water Stress Corrosion Cracking (PWSCC) susceptible welds and will supplement the RIS_B Program selection process. The RIS_B Program will not be used to eliminate any MRP-139 requirements.

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 3.1.b (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, but all BER piping at VEGP is Class 2.

(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. No piping segments with a contribution to CDF greater than 1E-06 (1E-07 for LERF) were identified.

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 RIS_B 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 VEGP. Table 3-16 of EPRI TR-112657 contains the following criteria for assessing the potential for Thermal Stratification, Cycling, and Striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 include:

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

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

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

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

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

   AND

   ➢ $\Delta T > 50°F$,

   AND

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

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

➢ **Turbulent Penetration TASCS**

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

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

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

- **Low flow TASCS**

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

- **Valve leakage TASCS**

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

- **Convection Heating TASCS**

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

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. Consideration of cycle severity was used in previous NRC approved RIS_B program submittals for D. C. Cook, Grand Gulf Nuclear Station, and Waterford-3. The methodology used in the VEGP RIS_B application for assessing TASCS potential conforms to these updated criteria. Additionally, materials reliability program (MRP) MRP-146 guidance on the subject of TASCS was also incorporated into the VEGP RIS_B application. It should be noted that the NRC has granted approval for RIS_B relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (NRC letter dated September 28, 2001) and South Texas Project (NRC letter dated March 5, 2002).

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 Vogtle) shall be selected.

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

Currently, there are seventy-nine BER program welds at Vogtle 1 and eighty-four BER welds at Vogtle 2. A RI-BER program has not been implemented, so 100% of the population is currently being inspected.

In contrast to a number of RI-ISI program applications, where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% of the HSS welds be chosen. A brief summary 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.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		NNS Welds(3)		All Piping Welds(4)	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected
1	902	102	1,997	34	0	0	2,899	136
2	948	106	1,916	35	0	0	2,864	141

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, 413 are HSS at Unit 1 and 418 are HSS at Unit 2; the remaining are LSS.
(3) There are no HSS Class 3 or non-nuclear safety (NNS) piping weld locations.
(4) 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.

3.3.1 Current Examinations

VEGP 1&2 is currently using the traditional ASME Section XI inspection methodology for ISI examination of piping welds. However, in anticipation of the approval of this RIS_B submittal, welds being examined using the traditional Section XI methodology also meets the examination requirements of Table 1 of Code Case N-716. Therefore, after approval of the RIS_B submittal, those welds that have already been examined during the 3rd Interval that are selected by the RIS_B process, will be credited toward the RIS_B requirements.

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 either ASME Code Section XI, IWB-3600 or IWC-3600, as appropriate. As part of this evaluation, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. If the flaw is acceptable for continued service, successive examinations will be scheduled per Section 6 of Code Case N-716. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000, applicable ASME Section XI Code Cases, or NRC approved alternatives. The IWB-3600 analytical evaluation will be submitted to the NRC. Finally, the evaluation will be documented in the corrective action program and the Owner submittals required by Section XI.

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).

3.3.4 Program Relief Requests

Consistent with previously approved RIS_B submittals, SNC 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 guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval

No VEGP relief requests are being withdrawn due to the RIS_B application.

3.4 Risk Impact Assessment

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

This evaluation categorized segments as high safety significant or low safety significant in accordance with Code Case N-716, and then determined what inspection changes were proposed for each system. The changes included changing the number and location of inspections, and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

Code Case N-716 has adopted the NRC approved EPRI TR-112657 process for risk impact analyses, whereby limits are imposed to ensure that the change in risk of implementing the RIS_B Program meets the requirements of Regulatory Guides 1.174 and 1.178. Section 3.7.2 of EPRI TR-112657 requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively.

For LSS welds, Conditional Core Damage Probability (CCDP)/Conditional Large Early Release Probability (CLERP) values of 1E-4/1E-5 were conservatively used. The rationale for using these values is that the change-in-risk evaluation process of N-716 is similar to that of the EPRI RI-ISI methodology. As such, the goal is to determine CCDPs/CLERPs threshold values. For example, the threshold values between High and Medium consequence categories is 1E-4 (CCDP)/1E-5 (CLERP) and between Medium and Low consequence categories are 1E-6 (CCDP)/1E-7 (CLERP) from the EPRI RI-ISI Risk Matrix. Using these threshold values streamlines the change-in-risk evaluation as well as stabilizes the update process. For example, if a CCDP changes from 1E-5 to 3E-5 due to an update, it will remain below the 1E-4 threshold value; the change-in-risk evaluation would not require updating.

The updated internal flooding PRA was also reviewed to ensure that there is no 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 the Code Case. 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 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 3.4-1a and Table 3.4-1b) for use in the change-in-risk assessment. Experience with previous industry RI-ISI applications shows this to be conservative.

VEGP 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 RI-ISI 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 for VEGP).

CCDP and CLERP Values Based on Break Location					
Break Location Designation	**Estimated**		**Consequence Rank**	**Upper Bound**	
	CCDP	**CLERP**		**CCDP**	**CLERP**
LOCA	2E-02	2E-03	**HIGH**	2E-02	2E-03
A LOCA is a RCPB pipe break that results in a loss of coolant accident – The highest CCDP for a Large LOCA was used (0.1 margin was used for CLERP)					
ILOCA(1) (2)	2E-05	2E-06	**MEDIUM**	1E-04	1E-05
An ILOCA is a pipe break that results in an isolable LOCA – Calculated based on Large LOCA CCDP of 2E-2 and a valve fail to close probability of ~1E-3 (0.1 margin used for CLERP)					
PLOCA(1) (2)	2E-05	2E-06	**MEDIUM**	1E-04	1E-05
A PLOCA is a RCPB pipe break that results in a potential LOCA – Calculated based on Large LOCA CCDP of 2E-2 and a valve rupture probability of ~1E-3 (0.1 margin used for CLERP)					
PLOCASD(1) (3)	2E-05	2E-06	**MEDIUM**	1E-04	1E-05
A PLOCASD is a RCPB pipe break that occurs in shutdown cooling suction piping resulting in a potential LOCA at power and an isolable LOCA during shutdown. LOCA CCDP and MOV failure on demand is judged to be appropriate for lines inside containment (0.1 margin used for CLERP)					
AOVLOCA(1)	4E-06	4E-07	**MEDIUM**	1E-04	1E-05
An AOVLOCA is a RCPB pipe break that results in an isolable LOCA with an air operated valve (AOV) – Calculated based on Large LOCA CCDP of 2E-2 and AOV fail to close probability of ~2E-4 (0.1 margin used for CLERP)					
MVLOCA(1)	4E-06	4E-07	**MEDIUM**	1E-04	1E-05
A MVLOCA is a RCPB pipe break that results in a potential LOCA with a manual valve (MV) – Calculated based on Large LOCA CCDP of 2E-2 and valve rupture probability of ~2E-4 (0.1 margin used for CLERP)					
SSBI	3E-05	3E-06	**MEDIUM**	1E-04	1E-05
An SSBI is a main steam or feedwater break inside the outer containment isolation valve – obtained from PRA (0.1 margin used for CLERP)					
SSBO	2E-06	2E-07	**MEDIUM**	1E-04	1E-05
An SSBO is a main steam or feedwater break beyond the outer containment isolation valve outside containment – obtained from PRA (0.1 margin used for CLERP)					
PPLOCA(1)	<1E-06	<1E-07	**MEDIUM**	1E-04	1E-05
A PPLOCA is a potential LOCA in Class 2 piping that requires two check valves in series to cause a rupture – based on Large LOCA CCDP of 2E-2 and 2 valve ruptures <1E-6 (0.1 margin used for CLERP). Medium was assumed rather than low because these lines support multiple cold leg injection paths.					
Class 2 LSS	1E-04	1E-05	**MEDIUM**	1E-04	1E-05
Class 2 LSS - Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant – Estimated based on upper bound for Medium Consequence					

Notes
1. The VEGP 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. IPLOCA is used as a designator when the pipe break can occur during system operation or standby.

3. PLOCASD2 is used for piping beyond second MOV on the SDC hot leg suction lines between the valve and the containment penetration. The same CCDP and CLERP are used.

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 3.4-1a (Unit 1) and Table 3.4-1b (Unit 2) presents a summary of the RIS_B Program versus the 1989 ASME Section XI Code Edition program requirements on a “per system” basis for the second interval. 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.

VEGP Unit 1 Risk Impact Summary				
System	**With POD Credit**		**Without POD Credit**	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
Auxiliary Feedwater	-2.75E-10	-2.75E-11	-9.85E-11	-9.85E-12
Chemical & Volume Control	-7.43E-09	-7.43E-10	-4.21E-09	-4.21E-10
Main Feedwater	5.00E-11	5.00E-12	9.00E-11	9.00E-12
Main Steam	8.95E-11	8.95E-12	8.95E-11	8.95E-12
Reactor Coolant	-5.14E-08	-5.14E-09	-9.00E-09	-9.00E-10
Residual Heat Removal	3.69E-10	3.69E-11	3.69E-10	3.69E-11
Safety Injection	-4.32E-08	-4.32E-09	-2.40E-08	-2.40E-09
Containment Spray	1.90E-10	1.90E-11	1.90E-10	1.90E-11
Total	**-1.02E-07**	**-1.02E-08**	**-3.66E-08**	**-3.66E-09**

VEGP Unit 2 Risk Impact Summary				
System	**With POD Credit**		**Without POD Credit**	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
Auxiliary Feedwater	-2.64E-10	-2.64E-11	-5.95E-11	-5.95E-12
Chemical & Volume Control	-7.43E-09	-7.43E-10	-4.21E-09	-4.21E-10
Main Feedwater	7.50E-12	7.50E-13	3.95E-11	3.95E-12
Main Steam	9.95E-11	9.95E-12	9.95E-11	9.95E-12
Reactor Coolant	-3.45E-08	-3.45E-09	2.30E-09	2.30E-10
Residual Heat Removal	2.79E-10	2.79E-11	2.79E-10	2.79E-11
Safety Injection	-4.35E-08	-4.35E-09	-2.43E-08	-2.43E-09
Containment Spray	1.70E-10	1.70E-11	1.70E-10	1.70E-11
Total	**-8.52E-08**	**-8.52E-09**	**-2.57E-08**	**-2.57E-09**

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 method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; 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. VEGP did not identify any such piping.

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

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 third 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 Vogtle NDE results, a review of site failure information from the Vogtle corrective action program, and a review of industry failure information from industry operating experience (OE). Also included is a review of PRA changes for their impact on the RIS_B program. These reviews provide a feedback loop such that new relevant information is obtained that will ensure that the appropriate identification of HSS piping locations selected for examination is maintained. As a minimum, this review will be conducted on an ASME period basis. In addition, more frequent adjustment may be required as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

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

A. Identify (Examination results conclude there is an unacceptable flaw).

B. Characterize (Determine if regulatory reporting is required and assess if an immediate safety or operation impact exists).

C. Evaluate (Determine the cause and extent of the condition identified and develop a corrective action plan or plans).

D. Decide (make a decision to implement the corrective action plan).

E. Implement (complete the work necessary to correct the problem and prevent recurrence).

F. Monitor (through the audit process ensure that the RIS_B program has been updated based on the completed corrective action).

G. Trend (Identify conditions that are significant based on accumulation of similar issues).

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

4. PROPOSED ISI PLAN CHANGE

VEGP 1&2 is currently in the first period of the third inspection interval and is using the traditional ASME Section XI inspection methodology for ISI examination of piping welds. At least 16% of the ASME Section XI piping examinations will be performed by the end of the first period of the third inspection interval to ensure compliance with the traditional ASME Section XI inspection methodology.

In anticipation of the approval of this RIS_B submittal, welds that are being examined using the traditional ASME Section XI methodology also meet the examination requirements of Table 1 of Code Case N-716. After approval of the RIS_B submittal, those welds that were examined during the third inspection interval, which are selected by the RIS_B process, will be credited toward the RIS_B requirements.

During the second and third ISI periods, the remainder of the inspection locations selected for examination per the RIS_B Program will be examined. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

A comparison between the RIS_B Program and the ASME Section XI 1989 Code Edition program requirements for 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 1 "*An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities*."

USNRC Safety Evaluation for Grand Gulf Nuclear Station Unit 1, Request for Alternative GG-ISI-002-Implement Risk-Informed ISI based on ASME Code Case N-716, dated September 21, 2007

USNRC Safety Evaluation for DC Cook Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based ISI program for Class 1 and 2 Piping Welds, dated September 28, 2007

Supporting Onsite Documentation

Structural Integrity Calculation 0800472.302 "N-716 Evaluation for Vogtle Units 1 and 2" Rev 0

Structural Integrity Calculation 0800472.301 "Degradation Mechanism Evaluation for Vogtle Units 1 & 2" Rev 0

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

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
RC	49	✓	✓				✓	
	252	✓					✓	
CVCS	87	✓					✓	
	310							✓
SI	126	✓	✓				✓	
	388	✓					✓	
	98		✓				✓	
	462							✓
RHR	6		✓				✓	
	401							✓
AFW	178			✓			✓	
FW	27				✓		✓	
	52			✓			✓	
	35							✓
MS	52				✓		✓	
	160							✓
CS	216							✓
SUMMARY RESULTS FOR ALL SYSTEMS	175	✓	✓				✓	
	727	✓					✓	
	104		✓				✓	
	79				✓		✓	
	230			✓			✓	
	1584							✓
TOTALS	**2899**							

AFW = Auxiliary Feedwater portion of main feedwater
CS = Containment Spray
CVCS – Chemical Volume and Control System
FW = Main Feedwater
MS = Main Steam
RC = Reactor Coolant
RHR = Residual Heat Removal
SI = Safety Injection
SDC = Shutdown Cooling

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

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
RC	51	✓	✓				✓	
	283	✓					✓	
CVCS	100	✓					✓	
	329							✓
SI	118	✓	✓				✓	
	404	✓					✓	
	90		✓				✓	
	432							✓
RHR	6		✓				✓	
	399							✓
AFW	182			✓			✓	
FW	31				✓		✓	
	48			✓			✓	
	28							✓
MS	53				✓		✓	
	106							✓
CS	204							✓
SUMMARY RESULTS FOR ALL SYSTEMS	169	✓	✓				✓	
	787	✓					✓	
	96		✓				✓	
	84				✓		✓	
	230			✓			✓	
	1498							✓
TOTALS	**2864**							

AFW = Auxiliary Feedwater portion of main feedwater
CS = Containment Spray
CVCS – Chemical Volume and Control System
FW = Main Feedwater
MS = Main Steam
RC = Reactor Coolant
RHR = Residual Heat Removal
SI = Safety Injection
SDC = Shutdown Cooling

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
RC	✓	✓				✓					
CVCS[2]		✓									
SI[2]	✓	✓	✓								
RHR[2]											
AFW		✓									
FW[2]		✓									
MS[2]											
CS[2]											

Notes
1. Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the CS system in its entirety, as well as portions of the CVCS, SI, RHR, FW and MS systems.

Table 3.3a
VEGP-1 Code Case N-716 Element Selections

System	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
AFW	138		TT					18
AFW	40		None					0
CVCS	9		TT	✓	✓			2
CVCS	6		TT	✓				2
CVCS	62		None	✓	✓			5
CVCS	10		None	✓				0
CVCS		310						0
FW	12		TT					3
FW	27		None				✓	5
FW	40		None					0
FW		35						0
MS	52		None				✓	6
MS		160						0
RC	4		PWSCC	✓	✓			4
RC	8		TASCS	✓	✓			8
RC	12		TASCS,TT	✓	✓			6
RC	23		TT	✓	✓			6
RC	207		None	✓	✓			5
RC	47		None	✓				2
RHR	6		None					2
RHR		401						0
SI	10		IGSCC	✓				3
SI	12		TASCS,TT	✓	✓			12
SI	8		TT	✓	✓			4
SI	4		TT, IGSCC	✓				1
SI	42		None	✓	✓			26
SI	438		None	✓				16
SI	98		None					0
SI		462						0
CS		216						0
Summary Results All Systems	40		TT	✓	✓			12
	6		TT	✓				2
	150		TT					21
	4		PWSCC	✓	✓			4
	8		TASCS	✓	✓			8
	24		TASCS,TT	✓	✓			18
	10		IGSCC	✓				3
	4		TT, IGSCC	✓				1
	311		None	✓	✓			36
	495		None	✓				18
	184		None					2
	79		None				✓	11
		1584						0
Totals	**1315**	**1584**						**136**

Note
Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).

Table 3.3b
VEGP-2 Code Case N-716 Element Selections

System	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
AFW	141		TT					19
AFW	41		None					0
CVCS	9		TT	✓	✓			2
CVCS	6		TT	✓				2
CVCS	75		None	✓	✓			6
CVCS	10		None	✓				0
CVCS		329						0
FW	12		TT					3
FW	31		None				✓	5
FW	36		None					0
FW		28						0
MS	53		None				✓	6
MS		106						0
RC	4		PWSCC	✓	✓			4
RC	8		TASCS	✓	✓			4
RC	13		TASCS,TT	✓	✓			6
RC	26		TT	✓	✓			6
RC	235		None	✓	✓			10
RC	48		None	✓				4
RHR	6		None					2
RHR		399						0
SI	10		IGSCC	✓				3
SI	12		TASCS,TT	✓	✓			12
SI	8		TT	✓	✓			4
SI	4		TT, IGSCC	✓				1
SI	42		None	✓	✓			27
SI	438		None	✓				15
SI	98		None					0
SI		432						0
CS		204						0
Summary Results All Systems	43		TT	✓	✓			12
	6		TT	✓				2
	153		TT					22
	4		PWSCC	✓	✓			4
	8		TASCS	✓	✓			4
	25		TASCS,TT	✓	✓			18
	10		IGSCC	✓				3
	4		TT, IGSCC	✓				1
	352		None	✓	✓			43
	496		None	✓				19
	181		None					2
	84		None				✓	11
		1498						0
Totals	**1366**	**1498**						**141**

Note
Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).

Table 3.4-1a
VEGP-1 Risk Impact Analysis Results

System (1)	Safety Significance	Break Location (5)	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank (4)	SXI (2)	RIS_B (3)	Delta	w/POD	w/o POD	w/POD	w/o POD
AFW	High	SSBI	TT	Medium	8	18	10	-2.76E-10	-1.00E-10	-2.76E-11	-1.00E-11
AFW	High	SSBI	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
AFW Total								**-2.75E-10**	**-9.85E-11**	**-2.75E-11**	**-9.85E-12**
CVCS	High	LOCA	TT	Medium	0	2	2	-7.20E-09	-4.00E-09	-7.20E-10	-4.00E-10
CVCS	High	IPLOCA	TT	Medium	0	2	2	-3.60E-11	-2.00E-11	-3.60E-12	-2.00E-12
CVCS	High	AOVLOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	High	LOCA	None	Low	0	5	5	-5.00E-10	-5.00E-10	-5.00E-11	-5.00E-11
CVCS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	Low	LSS		Assume Medium	31	0	-31	3.10E-10	3.10E-10	3.10E-11	3.10E-11
CVCS Total								**-7.43E-09**	**-4.21E-09**	**-7.43E-10**	**-4.21E-10**
FW	High	SSBI	TT	Medium	4	3	-1	-3.00E-11	1.00E-11	-3.00E-12	1.00E-12
FW	High	SSBI	None	Low	1	5	4	-2.00E-12	-2.00E-12	-2.00E-13	-2.00E-13
FW	High	SSBO	None	Low	4	0	-4	2.00E-12	2.00E-12	2.00E-13	2.00E-13
FW	Low	LSS		Assume Medium	8	0	-8	8.00E-11	8.00E-11	8.00E-12	8.00E-12
FW Total								**5.00E-11**	**9.00E-11**	**5.00E-12**	**9.00E-12**
MS	High	SSBI	None	Low	5	6	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
MS	High	SSBO	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS	Low	LSS		Assume Medium	9	0	-9	9.00E-11	9.00E-11	9.00E-12	9.00E-12
MS Total								**8.95E-11**	**8.95E-11**	**8.95E-12**	**8.95E-12**
RC	High	LOCA	PWSCC	Medium	4	4	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC	High	LOCA	TASCS	Medium	0	8	8	-2.88E-08	-1.60E-08	-2.88E-09	-1.60E-09
RC	High	LOCA	TASCS,TT	Medium	10	6	-4	-9.60E-09	8.00E-09	-9.60E-10	8.00E-10
RC	High	LOCA	TT	Medium	3	6	3	-1.80E-08	-6.00E-09	-1.80E-09	-6.00E-10
RC	High	LOCA	None	Low	55	5	-50	5.00E-09	5.00E-09	5.00E-10	5.00E-10
RC	High	PLOCASD	None	Low	0	2	2	-1.00E-12	-1.00E-12	-1.00E-13	-1.00E-13

Table 3.4-1a
VEGP-1 Risk Impact Analysis Results

System (1)	Safety Significance	Break Location (5)	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank (4)	SXI (2)	RIS_B (3)	Delta	w/POD	w/o POD	w/POD	w/o POD
RC	High	MVLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC Total								**-5.14E-08**	**-9.00E-09**	**-5.14E-09**	**-9.00E-10**
RHR	High	PLOCASD2	None	Low	0	2	2	-1.00E-12	-1.00E-12	-1.00E-13	-1.00E-13
RHR	Low	LSS		Assume Medium	37	0	-37	3.70E-10	3.70E-10	3.70E-11	3.70E-11
RHR Total								**3.69E-10**	**3.69E-10**	**3.69E-11**	**3.69E-11**
SI	High	PLOCA	IGSCC	Medium	6	3	-3	3.00E-11	3.00E-11	3.00E-12	3.00E-12
SI	High	LOCA	TASCS,TT	Medium	0	8	8	-2.88E-08	-1.60E-08	-2.88E-09	-1.60E-09
SI	High	LOCA	TT	Medium	0	4	4	-1.44E-08	-8.00E-09	-1.44E-09	-8.00E-10
SI	High	PLOCA	TT, IGSCC	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	LOCA	None	Low	22	26	4	-4.00E-10	-4.00E-10	-4.00E-11	-4.00E-11
SI	High	PLOCA	None	Low	18	8	-10	5.00E-12	5.00E-12	5.00E-13	5.00E-13
SI	High	PPLOCA	None	Low	5	8	3	-1.50E-12	-1.50E-12	-1.50E-13	-1.50E-13
SI	Low	LSS		Assume Medium	37	0	-37	3.70E-10	3.70E-10	3.70E-11	3.70E-11
SI Total								**-4.32E-08**	**-2.40E-08**	**-4.32E-09**	**-2.40E-09**
CS Total	Low	LSS		Assume Medium	19	0	-19	**1.90E-10**	**1.90E-10**	**1.90E-11**	**1.90E-11**
Grand Total					**289**	**131**		**-1.02E-07**	**-3.66E-08**	**-1.02E-08**	**-3.66E-09**

Notes

1. Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. Only those RIS_B inspection locations that receive a volumetric examination are included in the count. In section locations subjected to VT2 only are not credited in count for risk impact assessment.
4. The failure potential rank for high safety significant (HSS) locations is then assigned as “High”, “Medium”, or “Low” depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”)
5. The “LSS” designation in Table 3.4-1a (Unit 1) and Table 3.4-1b (Unit 2) is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criteria of Section 2(a) of N-716 (e.g., not part of the BER scope).

Table 3.4-1b
VEGP-2 Risk Impact Analysis Results

System (1)	Safety Significance	Break Location (5)	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank (4)	SXI (2)	RIS_B (3)	Delta	w/POD	w/o POD	w/POD	w/o POD
AFW	High	SSBI	TT	Medium	13	19	6	-2.64E-10	-6.00E-11	-2.64E-11	-6.00E-12
AFW	High	SSBI	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
AFW Total								**-2.64E-10**	**-5.95E-11**	**-2.64E-11**	**-5.95E-12**
CVCS	High	LOCA	TT	Medium	0	2	2	-7.20E-09	-4.00E-09	-7.20E-10	-4.00E-10
CVCS	High	IPLOCA	TT	Medium	0	2	2	-3.60E-11	-2.00E-11	-3.60E-12	-2.00E-12
CVCS	High	AOVLOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	High	LOCA	None	Low	0	6	6	-6.00E-10	-6.00E-10	-6.00E-11	-6.00E-11
CVCS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	Low	LSS		Assume Medium	41	0	-41	4.10E-10	4.10E-10	4.10E-11	4.10E-11
CVCS Total								**-7.43E-09**	**-4.21E-09**	**-7.43E-10**	**-4.21E-10**
FW	High	SSBI	TT	Medium	2	3	1	-4.20E-11	-1.00E-11	-4.20E-12	-1.00E-12
FW	High	SSBI	None	Low	4	5	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
FW	High	SSBO	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FW	Low	LSS		Assume Medium	5	0	-5	5.00E-11	5.00E-11	5.00E-12	5.00E-12
FW Total								**7.50E-12**	**3.95E-11**	**7.50E-13**	**3.95E-12**
MS	High	SSBI	None	Low	3	6	3	-1.50E-12	-1.50E-12	-1.50E-13	-1.50E-13
MS	High	SSBO	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
MS	Low	LSS		Assume Medium	10	0	-10	1.00E-10	1.00E-10	1.00E-11	1.00E-11
MS Total								**9.95E-11**	**9.95E-11**	**9.95E-12**	**9.95E-12**
RC	High	LOCA	PWSCC	Medium	4	4	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC	High	LOCA	TASCS	Medium	0	4	4	-1.44E-08	-8.00E-09	-1.44E-09	-8.00E-10
RC	High	LOCA	TASCS,TT	Medium	12	6	-6	-7.20E-09	1.20E-08	-7.20E-10	1.20E-09
RC	High	LOCA	TT	Medium	2	6	4	-1.92E-08	-8.00E-09	-1.92E-09	-8.00E-10
RC	High	LOCA	None	Low	73	10	-63	6.30E-09	6.30E-09	6.30E-10	6.30E-10
RC	High	PLOCASD	None	Low	1	2	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
RC	High	MVLOCA	None	Low	0	2	2	-1.00E-12	-1.00E-12	-1.00E-13	-1.00E-13

Table 3.4-1b
VEGP-2 Risk Impact Analysis Results

System (1)	Safety Significance	Break Location (5)	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank (4)	SXI (2)	RIS_B (3)	Delta	w/POD	w/o POD	w/POD	w/o POD
RC Total								**-3.45E-08**	**2.30E-09**	**-3.45E-09**	**2.30E-10**
RHR	High	PLOCASD2	None	Low	0	2	2	-1.00E-12	-1.00E-12	-1.00E-13	-1.00E-13
RHR	Low	LSS		Assume Medium	28	0	-28	2.80E-10	2.80E-10	2.80E-11	2.80E-11
RHR Total								**2.79E-10**	**2.79E-10**	**2.79E-11**	**2.79E-11**
SI	High	PLOCA	IGSCC	Medium	5	3	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
SI	High	LOCA	TASCS,TT	Medium	0	8	8	-2.88E-08	-1.60E-08	-2.88E-09	-1.60E-09
SI	High	LOCA	TT	Medium	0	4	4	-1.44E-08	-8.00E-09	-1.44E-09	-8.00E-10
SI	High	PLOCA	TT, IGSCC	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	LOCA	None	Low	20	27	7	-7.00E-10	-7.00E-10	-7.00E-11	-7.00E-11
SI	High	PLOCA	None	Low	16	7	-9	4.50E-12	4.50E-12	4.50E-13	4.50E-13
SI	High	PPLOCA	None	Low	4	8	4	-2.00E-12	-2.00E-12	-2.00E-13	-2.00E-13
SI	Low	LSS		Assume Medium	35	0	-35	3.50E-10	3.50E-10	3.50E-11	3.50E-11
SI Total								**-4.35E-08**	**-2.43E-08**	**-4.35E-09**	**-2.43E-09**
CS Total	Low	LSS		Assume Medium	17	0	-17	**1.70E-10**	**1.70E-10**	**1.70E-11**	**1.70E-11**
Grand Total					**298**	**136**		**-8.52E-08**	**-2.57E-08**	**-8.52E-09**	**-2.57E-09**

Notes

1. Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. Only those RIS_B inspection locations that receive a volumetric examination are included in the count. In section locations subjected to VT2 only are not credited in count for risk impact assessment.
4. The failure potential rank for high safety significant (HSS) locations is then assigned as “High”, “Medium”, or “Low” depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”)
5. The “LSS” designation in Table 3.4-1a (Unit 1) and Table 3.4-1b (Unit 2) is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criteria of Section 2(a) of N-716 (e.g., not part of the BER scope).

Table 4a
VEGP-1 Inspection Location Selection Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank (3)			Vol	Surface	RIS_B	Other (2)
AFW	✓		SSBI	TT	Medium	C-F-2	138	8	0	18	NA
AFW	✓		SSBI	None	Low	C-F-2	40	3	0	0	NA
CVCS	✓		LOCA	TT	Medium	B-J	9	0	4	2	NA
CVCS	✓		IPLOCA	TT	Medium	B-J	4	0	4	2	NA
CVCS	✓		AOVLOCA	TT	Medium	B-J	2	0	0	0	NA
CVCS	✓		LOCA	None	Low	B-J	62	0	29	5	NA
CVCS	✓		PLOCA	None	Low	B-J	2	0	0	0	NA
CVCS	✓		ILOCA	None	Low	B-J	8	0	0	0	NA
CVCS		✓	LSS	N/A	Assume Medium	B-J	310	31	2	0	NA
FW	✓		SSBI	TT	Medium	C-F-2	12	4	0	3	NA
FW	✓		SSBI	None	Low	C-F-2	56	1	0	5	NA
FW	✓		SSBO	None	Low	C-F-2	11	4	0	0	NA
FW	□	✓	LSS	N/A	Assume Medium	C-F-2	35	8	0	0	NA
MS	✓		SSBI	None	Low	C-F-2	44	5	0	6	NA
MS	✓		SSBO	None	Low	C-F-2	8	0	0	0	NA
MS		✓	LSS	N/A	Assume Medium	C-F-2	160	9	0	0	NA
RC	✓		LOCA	PWSCC	Medium	B-F	4	4	0	4	NA
RC	✓		LOCA	TASCS	Medium	B-J	8	0	2	8	NA
RC	✓		LOCA	TASCS,TT	Medium	B-J	12	10	0	6	NA
RC	✓		LOCA	TT	Medium	B-J	23	3	6	6	NA
RC	✓		LOCA	None	Low	B-F, B-J	207	55	27	5	NA
RC	✓		PLOCASD	None	Low	B-J	35	0	1	2	NA
RC	✓		MVLOCA	None	Low	B-J	12	0	0	0	NA
RHR	✓		PLOCASD2	None	Low	C-F-1	6	0	0	2	NA
RHR		✓	LSS	N/A	Assume Medium	C-F-1	401	37	0	0	NA
SI	✓		PLOCA	IGSCC	Medium	B-F	10	6	0	3	NA
SI	✓		LOCA	TASCS,TT	Medium	B-F	12	0	12	8	4 VT2
SI	✓		LOCA	TT	Medium	B-F	8	0	8	4	NA

SI	✓		PLOCA	TT, IGSCC	Medium	B-F	4	0	0	0	1 VT2
SI	✓		LOCA	None	Low	B-F	42	22	0	26	NA

Table 4a
VEGP-1 Inspection Location Selection Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank (3)			Vol	Surface	RIS_B	Other (2)
SI	✓		PLOCA	None	Low	B-F	410	18	44	8	NA
SI	✓		PPLOCA	None	Low	C-F-1	126	5	0	8	NA
SI		✓	LSS	N/A	Assume Medium	C-F-1	462	37	1	0	NA
CS		✓	LSS	N/A	Assume Medium	C-F-1	216	19	0	0	NA

Notes

1. Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).
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 VEGP RIS_B application. The “Other” column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals and to indicate when RIS_B selections will receive a VT-2 examination (these are not credited in risk impact assessment).
3. The failure potential rank for high safety significant (HSS) locations is then assigned as “High”, “Medium”, or “Low” depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”).

Table 4b
VEGP-2 Inspection Location Selection Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank (3)			Vol	Surface	RIS_B	Other (2)
AFW	✓		SSBI	TT	Medium	C-F-2	141	13	0	19	NA
AFW	✓		SSBI	None	Low	C-F-2	41	1	0	0	NA
CVCS	✓		LOCA	TT	Medium	B-J	9	0	6	2	NA
CVCS	✓		IPLOCA	TT	Medium	B-J	4	0	4	2	NA
CVCS	✓		AOVLOCA	TT	Medium	B-J	2	0	0	0	NA
CVCS	✓		LOCA	None	Low	B-J	75	0	27	6	NA
CVCS	✓		PLOCA	None	Low	B-J	2	0	2	0	NA
CVCS	✓		ILOCA	None	Low	B-J	8	0	0	0	NA
CVCS		✓	LSS	N/A	Assume Medium	B-J	329	41	2	0	NA
FW	✓		SSBI	TT	Medium	C-F-2	12	2	0	3	NA
FW	✓		SSBI	None	Low	C-F-2	57	4	0	5	NA
FW	✓		SSBO	None	Low	C-F-2	11	0	0	0	NA
FW	□	✓	LSS	N/A	Assume Medium	C-F-2	28	5	0	0	NA
MS	✓		SSBI	None	Low	C-F-2	45	3	0	6	NA
MS	✓		SSBO	None	Low	C-F-2	8	2	0	0	NA
MS		✓	LSS	N/A	Assume Medium	C-F-2	106	10	0	0	NA
RC	✓		LOCA	PWSCC	Medium	B-F	4	4	0	4	NA
RC	✓		LOCA	TASCS	Medium	B-J	8	0	0	4	NA
RC	✓		LOCA	TASCS,TT	Medium	B-J	13	12	0	6	NA
RC	✓		LOCA	TT	Medium	B-J	26	2	8	6	NA
RC	✓		LOCA	None	Low	B-F, B-J	235	73	24	10	NA
RC	✓		PLOCASD	None	Low	B-J	36	1	0	2	NA
RC	✓		MVLOCA	None	Low	B-J	12	0	1	2	NA
RHR	✓		PLOCASD2	None	Low	C-F-1	6	0	0	2	NA
RHR		✓	LSS	N/A	Assume Medium	C-F-1	399	28	0	0	NA
SI	✓		PLOCA	IGSCC	Medium	B-F	10	5	0	3	NA
SI	✓		LOCA	TASCS,TT	Medium	B-F	12	0	12	8	4 VT2
SI	✓		LOCA	TT	Medium	B-F	8	0	4	4	NA

Table 4b
VEGP-2 Inspection Location Selection Comparison

System (1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank (3)			Vol	Surface	RIS_B	Other (2)
SI	✓		PLOCA	TT, IGSCC	Medium	B-F	4	0	0	0	1 VT2
SI	✓		LOCA	None	Low	B-F	42	20	0	27	NA
SI	✓		PLOCA	None	Low	B-F	408	16	49	7	NA
SI	✓		PPLOCA	None	Low	C-F-1	128	4	0	8	NA
SI		✓	LSS	N/A	Assume Medium	C-F-1	432	35	1	0	NA
CS		✓	LSS	N/A	Assume Medium	C-F-1	204	17	0	0	NA

Notes

1. Systems are described in Table 3.1a (Unit 1) and Table 3.1b (Unit 2).
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 VEGP RIS_B application. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals and to indicate when RIS_B selections will receive a VT-2 examination (these are not credited in risk impact assessment).
3. The failure potential rank for high safety significant (HSS) locations is then assigned as "High", "Medium", or "Low" depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").

Attachment A to VEGP N716 Template

Consideration of the Adequacy of Probabilistic Risk Assessment Model for Application of Code Case N716

Summary Statement of VEGP PRA Model Capability for Use in Risk-Informed Inservice Inspection Program Licensing Actions

Introduction

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

PRA Maintenance and Update

The SNC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated units. This process is defined in the SNC risk management program which is described in SNC procedure NL-PRA-001[1], "Generation of PRA models and Associated Updates". SNC Procedure NL-PRA-001 delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating SNC nuclear generation sites. The overall SNC risk management program, including NL-PRA-001, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the VEGP PRA model has been updated according to the requirements in the following sections of VEGP procedure NL-PRA-001:

- Pertinent modifications to the physical plant (i.e. those potentially affecting the Base Line PRA (BL-PRA) models, calculated core damage frequencies, or large early release frequencies to a significant degree) shall be reviewed to determine the scope and necessity of a revision to the baseline model within six months following the Unit 2 refueling outage or a specific major plant modification occurring outside a refueling outage. The BL-PRAs should be updated as necessary in accordance with a schedule approved by the PRA Services Supervisor following the scoping review. Upon completion of the lead unit's BL-PRA, the other unit's BL-PRA will be regenerated by modification of the updated BL-PRAs to account for unit differences which significantly impact the results.

- Pertinent modifications to plant procedures and technical specifications shall be reviewed annually for changes which are of statistical significance to the results of the BL-PRA and those changes documented. Reliability data, failure data, initiating events frequency data, human reliability data, and other such PRA INPUTs shall be reviewed approximately every three years for statistical significance to the results of the BL-PRAs. Following the tri-annual review, the BL-PRAs shall be updated to account for the significant changes to these two categories of PRA INPUTS in accordance with an approved schedule.

- BL-PRAs shall be updated to reflect germane changes in methodology, phenomenology, and regulation as judged to be prudent or as required by regulation.

In addition to these activities, SNC risk management procedures [2,3,4,5,6] provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for SNC nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximate 3-year cycle; however, longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. Table A-1 shows the brief history of the major VEGP PRA model updates.

The PRA model for internal events (except internal flooding) used for the RIS_B evaluation was Vogtle PRA L2UP model [7]. The Vogtle PRA L2UP model was previously used for the Vogtle Severe Accident Management Alternatives (SAMA) Analysis, which had been submitted in 2007 as a part of Vogtle License renewal submittal. The PRA adequacy was addressed in the SAMA analysis report [8] and the responses to the Request for Additional Information in 2007 [9].

The Vogtle PRA L2UP model includes an upgraded level 1 internal event PRA model and a level 2 PRA model. The upgraded level 1 PRA model included in the VEGP L2UP model was based on VEGP Level 1 PRA model Rev 3 [10], in which all PWROG PRA peer review B Findings and Observations (F&Os) were addressed (there were no A findings). The upgraded level 2 PRA model included in the L2UP model was based on a PWROG methodology (WCAP-16341-P [11]) which was intended to reflect ASME PRA standard Capability Category II.

In addition, during 2008, the VEGP internal flooding PRA was re-performed in order to meet ANS PRA standard Capability Category II. The revised internal flooding PRA model [12] was used for the VEGP RIS_B evaluation. Self assessment findings (by an independent external contractor) and the associated resolutions were also documented as a part of the re-performed internal flooding analysis to ensure that the internal flooding evaluation met all requirements for Capability Category II.

In the following section, details of PRA self assessment, peer review, and resolution of findings and gaps were documented. Also, the impact of non-compliance of some gaps on the VEGP RIS_B program is described.

Table A-1: History of the Major VEGP PRA Model Updates

Model	Document No.	Scope	Updated Items	CDF and LERF
IPE	WCAP-13553 (WH report) by WH and SNC, 11/1992	At-power, internal and external, CDF and Level 2	The original	CDF: 4.9E-5 LERF: 1.78E-6
Rev. 0	SAIC prepared reports, 3/1998.	At-power, internal, CDF and LERF	Converted from a large Event Tree/small Fault Tree approach to a small Event Tree/large Fault Tree approach (linked fault tree model method). The PRA software changed from WESQT/GRAFTER (Westinghouse Event Tree and Fault tree software) to CAFTA	CDF: 3.62E-5 LERF: 1.72E-6 The CDF reduction was mainly due to changes, such as, removal of unrealistic SBO scenarios, addition of more realistic assumptions regarding the effect of loss of room cooling, and removal of a ‘guaranteed failure’ assumption made during IPE for event CON (operator action to depressurize one SG to cause feed flow from the condensate pumps if AFW failed).
Rev. 1	PSA-V-99-002 by SNC, 9/1999	At-power, internal, CDF and LERF	Enhanced the treatment of operator action dependency, removed circular logic, and made minor corrections/improvements.	

Table A-1: History of the Major VEGP PRA Model Updates

Model	Document No.	Scope	Updated Items	CDF and LERF
Rev. 2	PSA-V-99-012 by SNC, 1/2000	At-power, internal, CDF and LERF	Update of plant specific failure data. Update for initiating event frequencies, component failure data, and maintenance unavailablities using plant specific data collected though the end of 1998. Incorporated plant changes.	CDF: 1.48E-5 LERF:1.15E-6 There was a considerable reduction in CDF mainly due to reduction in the transient event frequency. The sum of frequencies of eight transient subcategories was reduced from 4.04/yr to 2.64/yr after the data update. Also, items updated during revision 0a, 0b, and 0c, especially the crediting of the plant Wilson switchyard for alternate AC power source, contributed to the reduction in CDF. The reduction in LERF was mainly due to reduced failure probabilities of some of the components, especially NSCW pumps, which have a significant contribution to the LERF after the Bayesian update of failure data using VEGP specific failure data.
Rev. 2c	PSA-V-00-030 by SNC, 11/2001	At-power, internal, CDF and LERF	Peer reviewed model by the WOG PRA peer review team. Revised the LERF model based on the new WOG LERF modeling guidelines. Updated the initiating event frequencies using the more recent generic data source (NUREG/CR-5750). Some SGTR scenarios were removed from the LERF scenarios and minor changes were made to facilitate RIS_B analysis. Removed circular logic in normal charging pump fault trees.	CDF: 1.602E-5, LERF:7.802E-8 The CDF decrease (rev.2a-> rev.2c) was mainly due to a decrease in LOCA frequencies after an update of initiating frequencies using NUREG/CR-5750 data. The decrease in LERF was due to the removal of some SGTR scenarios from the LERF model.

Table A-1: History of the Major VEGP PRA Model Updates

Model	Document No.	Scope	Updated Items	CDF and LERF
Rev. 3	PRA-BC-V-06-001, by SNC, 2/2006	At-power, internal, CDF and LERF	This is the most extensive upgrade of the VEGP PRA model since the IPE. • All level 1 PRA tasks, from the selection and grouping of initiating events to the final quantification were practically re-done. • Resolved all WOG PRA peer review B F&Os (there were no A F&O for VEGP).	CDF: 1.28E-5 LERF: 1.10E-7 The CDF changes were due to combined effects of many changes during revision 3. The main cause of the LERF increase (from Rev 2c -> Rev. 3) was the regrouping of all of the SGTR sequences back into the containment bypass scenarios, and the removal of the credit for mitigating systems for some ISLOCA scenarios (as resolutions of peer review findings).
VEGPL2UP model	P0293060001-2707 (ERIN report) by SNC and ERIN, 11/2006	At-power, internal, CDF and full level 2	Based on the Rev.3 level 1 PRA logic. This model was used for the Severe Accident Management Alternative Analysis for the VEGP license renewal which was submitted in 2007. Upgraded the full Level 2 PRA model, based on WCAP-16341-P guidelines which aims for producing an ASME PRA capability category II LERF model. Incorporated success terms in level 1 and level 2 logic. Corrected an error in the level 1 PRA failure data.	CDF: 1.552E-5 1.529E-5 (after treating success terms) LERF: 1.819E-7 The increase in CDF (before treating success terms) from revision 3 to VEGPL2UP model was due to a correction of RCP seal LOCA probability from WCAP-16141. The above LERF value is the sum of four LERF release categories: LERF-BYPASS, LERF-ISO, LERF-CFE, and LERF-SGTR.
Rev. 4	Under development	At power, internal, CDF and full level 2	The following items are complete: • Site review of initiating events list for gap closure. • Site review of event trees for gap closure. • Re-performed pre-initiator HFE screening for gap closure. • Re-performed internal flooding PRA	Under development

PRA Self Assessment and Peer Review

In addition to independent internal and external review during each VEGP PRA model development and update, several assessments of the technical capability have been made, and continue to be planned, for the VEGP PRA models. These assessments are as follows:

- An independent PRA peer review was conducted under the auspices of the Westinghouse Owners Group (WOG) in December 2001, following the Industry PRA Peer Review process [13]. This peer review included an assessment of the PRA model maintenance and update process.

- During 2005, the VEGP PRA model results were evaluated in the WOG PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process. Results of this cross-comparison are presented in WCAP-16464 [14]. The PRA Cross comparison Candidate Outlier Status was described in section 3.4 of VEGP MSPI base document [15]. Noted in this document was the fact that, after allowing for plant-specific features, there are no MSPI cross-comparison outliers for VEGP PRA.

- In 2006, a gap analysis was performed against the available versions of the ASME PRA Standard [16] and Regulatory Guide 1.200, Revision 0 (2003 trial version) [17].

All B facts and observations (F&Os) from the 2001 Industry PRA Peer Review for VEGP PRA [18] were addressed in VEGP PRA model revision 3 [10]. There were no A F&Os. Table A-2 shows the summary of disposition of B F&Os from the 2001 WOG peer review for VEGP PRA (details were documented as part of a VEGP PRA model revision 3 report).

Table A-2: Resolutions of VEGP PRA WOG Peer Review Level B Findings in VEGP PRA R3

F&O	Issues (All Significance Level B, no "A" F&O)	Resolutions in VEGP PRA Revision 3
IE-06	CCF NSCW pumps among pumps with different operating cycle &histories in special initiating events should be based on plant specific CCF analysis.	CCF of NSCW pumps with different operating cycles & histories were reevaluated through a detailed VEGP plant specific CCF analysis using NRC CCF Data base and by considering VEGP specific design features.
AS-04	The success state of ISLOCA and SGTR after 24 hours should be no core damage and "a stable" state.	Basically, for revision 3, the MAAP analyses for determination of the success criteria ran for 30 hours for most of the accident sequences. The 30 hour duration included 24 hours mission time, plus 6 additional hours. Generally, if core damage did not occur within 30 hours, it was assumed that core damage had been avoided. This approach would prevent sequences which would result in core damage just after the PRA mission time (24 hours) from being categorized as non-core damage sequences. Furthermore, the following modifications were made in ISLOCA and SGTR modeling: • Each ISLOCA potential path was re-examined using an event tree approach and identified ISLOCA paths were modeled as fault trees. The success state of ISLOCA was isolation of the ISLOCA path by closing (auto or manual) isolation valves before RWST depletion. Inventory makeup until the ISLOCA path is isolated is also required for the success. • If the ISLOCA break size was smaller than or equal to 1.0" in diameter, an additional success state was considered: the plant would be in stable condition if the RCS was cooled down and depressurized to minimize the leak with AFW and high pressure injection available. Once depressurized, the ECCS injection flow requirement would be minimal. For an ISLOCA path which could not be isolated by isolation valves and the break size was greater than 1" in diameter, core damage was assumed.

Table A-2: Resolutions of VEGP PRA WOG Peer Review Level B Findings in VEGP PRA R3

F&O	Issues (All Significance Level B, no "A" F&O)	Resolutions in VEGP PRA Revision 3
AS-04 (continued)	The success state of ISLOCA and SGTR after 24 hours should be no core damage and "a stable" state.	In revision3, the SGTR event tree was revised to more accurately reflect VEGP procedures and actual scenarios. For SGTR, obtaining a long term stable state was an issue only when the SG Valves stuck open after the SG was overfilled due to the failure of SG isolation because, if no recovery actions are taken, there would be a continuous primary-to-secondary-to-atmosphere leakage. The MAAP analysis for VEGP, for such a case, showed that core damage would not occur within 30 hours even when SG ARV or SVs stuck open (multiple valves stuck open) and all CCPs, SIPs, and 200% AFW flow are running. This was because VEGP has a relatively large RWST inventory (~700,000 gal). Thus, even without additional RWST water (refilling RWST), operators would have more than enough time to cool down and depressurize the RCS to stop or minimize the SG tube leak and stabilize the plant. MAAP analyses also showed that in the case of stuck open SG valves due to overfilling, continuous high pressure injection was not a critical mitigating function to prevent core damage. Core damage would not occur even after depletion of the RWST, as long as AFW was supplied. MAAP analyses showed that one CST (VEGP has two CSTs) will be enough to prevent core damage for about 35.5 hours. In revision 3, however, it was conservatively assumed that an additional AFW water source either from the secondary CST, or makeup from demineralized water tank (automatic or manual) would be required to prevent core damage, for such cases. With the additional AFW supply, the plant would be in a stable state well beyond 70 hours.

Table A-2: Resolutions of VEGP PRA WOG Peer Review Level B Findings in VEGP PRA R3

F&O	Issues (All Significance Level B, no "A" F&O)	Resolutions in VEGP PRA Revision 3
AS-05	For some ISLOCA paths, ECCS can not be credited. An ISLOCA through the RHR suction or injection lines may result in a leak rate much greater than 120 gpm (the leak rate was based on the assumption that the break occurs at the RHR pump seal) used in the VEGP IPE, if the RHR HX ruptures due to over-pressurization.	ISLOCA paths were re-identified using an event tree method and modeled as fully developed fault trees. Impacts of an ISLOCA to the mitigating systems were modeled in the ISLOCA core damage fault trees. For ISLOCA paths through RHR, it was assumed that the break location would be at the RHR HX and the size of the break was defined by the size of the piping in the path ways, a 6" diameter break for an ISLOCA though the RHR injection paths and a 12" diameter break for an ISLOCA through a hot leg suction line. For an ISLOCA through a RHR hot leg suction line, it was assumed that core damage would directly occur because it would cause a 12" diameter break and the path could not be isolated (there is no isolation valve between hot leg suction and RHR HX). ECCS operation would not affect the consequences. An ISLOCA in a RHR injection line would cause a 6" diameter LOCA. A 6" break (highest end of medium LOCA category) can be handled by 2 of 4 CCPs/SIPs until RWST depletion. In order to prevent core damage, however, operators must isolate the ISLOCA path by closing the RHR injection isolation motor operated valves. For the isolation to be successful, operators must close the required valves before the RWST is depleted. Core damage was assumed if operator failure or high pressure injection failure occurs. High pressure injection by the charging pumps or safety injection by the safety injection pumps was not credited in the ISLOCA scenarios, if any of the flow paths in the system were involved in the scenarios. For example, the safety injection system was not credited for inventory makeup for the ISLOCA through the cold leg injection lines of the safety injection system. Also, see the resolution to AS-04.
AS-08	Some SGTR sequences that were modeled as non-LERF scenarios may actually be LERF sequences.	All SGTR core damage sequences were included in LERF sequences with exceptions. The exceptions were SGTR-1, SGTR-2, and SGTR-3 sequences which were not considered as LERF sequence because MAAP analyses showed that without refilling RWST, and without having additional AFW water source, core damage would not occur within 30 hrs into the event (late core damage sequence)

Table A-2: Resolutions of VEGP PRA WOG Peer Review Level B Findings in VEGP PRA R3

F&O	Issues (All Significance Level B, no "A" F&O)	Resolutions in VEGP PRA Revision 3
DA-02	MGL factors used for evaluating VEGP IPE CCF probabilities seem to be too low as compared to generic industry data.	The VEGP Plant specific CCF analysis was redone using the NRC CCF Data Base, in order to estimate the VEGP specific CCF factors, while considering VEGP specific defenses against CCF events. The Alpha factor model, which is more statistically correct than the MGL method, was used for the update. VEGP specific environments, procedures, designs, operations, and measures implemented to prevent CCF were considered in the analysis.
DA-03	The same MGL factors were used for pump failure to start and failure to run CCFs.	The VEGP plant specific CCF analysis for the pumps, as well as other major components, was updated. CCFs for a pump failure to run were evaluated using only CCFs of pump failure to run events. CCFs for a pump failure to start were separately evaluated using only failure to start events. Pumps in different systems were evaluated separately.
DA-04	The probability of a safety valve to reclose after passing two phase flow should be higher than that after passing only steam in ATWT and SGTR overfill.	For ATWT, a higher number was used for PZR Safety Valves to fail to reseat because the PZR safety valves are not designed for passing two-phase flow. However, the PZR PORVs are designed for passing either steam or water (Table 5.4.13-1 of VEGP FSAR), thus their failure probability was not changed to a higher value. For SGTR overfill, it was conservatively assumed that SG overfill would cause the secondary side relief or safety valves to stick open.
HR-02	No reference analysis is available for operator action timing.	HRA was updated using the EPRI HRA-Calculator. Review of the training materials, interviews with operators and instructors, and timing information from VEGP specific MAAP analyses were used as inputs to the HRA update.

A gap analysis for VEGP PRA model revision 3 was completed in 2006. This gap analysis was performed against the available version of the ASME PRA Standard [16] and Regulatory Guide 1.200, revision 0 (2003 trial version) [17]. The summary of gap analyses and the impact of gap non-compliance on the VEGP RIS_B program are presented in Table A-3. Most of the gaps, except for uncertainty correlation, were related to documentation. It should be noted that since the gap analysis, the internal flooding PRA for VEGP was re-performed in 2008 in order to meet all capability category II requirements for internal flooding analyses. In addition, a self assessment by a third part was also performed and documented as part of the internal flooding PRA report [12] in order to ensure that all capability category II requirements for internal flooding analyses are being met. The VEGP RIS_B evaluation used the revised VEGP internal flooding PRA.

Following the VEGP PRA model revision 3, a major update of the level 2 PRA model was performed and the VEGP PRA L2UP model was issued in 2006. This update integrated the upgraded level1 PRA model from the VEGP RPA model revision 3 and the updated level 2 PRA model. The level 2 PRA model in the VEGP L2UP model was developed using new WOG level 2 PRA modeling guidelines, WCAP-16341-P “WOG Simplified Level 2 Modeling Guidelines”. WCAP 16341-P aimed for developing an ASME PRA standard Capability Category II large early release frequency (LERF) PRA model. The VEGP PRA L2UP model was used for Severe Accident Mitigation Alternatives (SAMA) analysis for the VEGP license renewal submitted in 2007. The technical adequacy of the VEGP PRA L2UP model was discussed in the SAMA evaluation reports [8] and in the Responses to the Request for Additional Information (RAI) [9]. No additional PRA quality questions were asked by the NRC after the SNC sent the response to the RAI. Therefore, the VEGP PRA L2UP model which was used in the VEGP RIS_B evaluation is considered to be of sufficient quality for SAMA evaluation for license renewal.

Since the gap analysis for VEGP PRA model in 2006 was based on the 2003 trial version of Regulatory Guide 1.200, an additional analysis was performed to identify the differences in requirements and their impacts between the old version of RG 1.200, RG 1.200, revision 1 [19] and ASME PRA Standard RA-SB-2005 [20]. For internal flooding and LERF, no additional gap analyses were performed because the models had been developed to meet the ASME PRA standard capability category II and Regulatory Guide 1.200, Revision 1. Table A-4 summarizes the additional gap analysis results. No additional gaps were found; however, it was determined that the impact of non-compliance related to the treatment uncertainty correlation, especially in the interfacing system LOCA, needed to be investigated. A discussion of the uncertainty correlation is provided below after the tables.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
1	Perform interviews with plant staff for potentially overlooked events and document results.	IE-A6	RG1.200	This gap has been closed.
2	Either use precursor data or document rationale for exclusion.	IE-A7	RG1.200	VEGP operating experiences were already used in identifying initiating events. The only item needed for completion is to enhance the documentation. Since there is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
3	Revise ISLOCA IE Calculation to account for correlated failure probabilities.	IE-C12	IE-02	Uncertainty correlation will be treated when a parametric uncertainty analysis is performed. The parametric uncertainty analysis has not been performed. This was investigated further for this application and found not to impact the HSS determination, and the risk acceptance criteria have been shown to be met even when conservative upper bound CCDP and CLERP values are used in the risk impact assessment.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
4	Perform a systematic review of the model and its assumptions with knowledgeable plant personnel to ensure the model reflects the current operating experience, maintenance, and design.	AS-A4, AS-A5, SY-A2, SC-A8, SY-A20, SY-B6, SY-C2	SY-03	This is only a documentation issue because technically this gap has been closed by the following: • Event trees have been reviewed by A. Chan (former SRO) and the comments have been resolved. • Interviewed site personnel for HRA and event tree development. • Communicated with site personnel via e-mails to identify the current operations and practices. • Current drawing, procedures, documentation from SyncPowr (electronic data base for SNC) were used. • System models were reviewed by a review group which included VEGP personnel, PRA analysts, out side contractors. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
5	Check the screening assumptions used in the flooding analysis and ensure that the flooding events do not hamper an operator's ability to mitigate the event. Use realistic HEPs to model the probability of not isolating floods within 30 minutes. Further analysis needs to be made of floods that impact SSCs but do not trip the plant, as well as, as flood propagation into adjacent rooms.	AS-B3, SC-C1, SY-A4, SY-A19, SY-B9,	DE-01	(See note 1)

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
6	Ensure the new MAAP analyses and the HEP analyses are documented.	AS-C3, AS-C4	RG 1.200	This item has been closed. MAAP analyses have been documented as several separate calculations. HRA also has been documented as a separate report.
7	Develop documentation discussing shared systems between units.	SC-A4	RG 1.200	The only shared system credited is “cross tying an opposite unit DG”. It was documented in an SBO event tree analysis. Thus, this item has been closed.
8	Although some searches have been performed to refine success criteria, guidance should be developed to broaden and formally document sensitivity analyses.	SC-B8,QU-D2, QU-D5, QU-F3	QU-01	This is only a documentation issue because extensive MAAP analyses were used in determining success criteria.
9	Fault tree modeling assumptions need to be readily available to support and document modeling decisions. For example, the discussion of AFW room cooling dependencies and operator response to its failure is not readily found.	SC-C1, SY-A4, SY-A17, SY-A18, SY-A20, SY-B8, SY-B9, QU-D2	SY-02	FT modeling assumptions are available in system note books. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
10	In the current PRA update ensure there is a reviewer signoff, indication of review performed, comments shown and incorporated, evidence of sensitivity analysis of important contributors, and detailed background of the source of each model change. In addition, the calc document should have more detail than the summary document.	SC-C1, SC-C4, SY-C1, SY-C3, QU-D3, QU-D5, QU-F1, QU-F2, LE-F1	MU-01	Most of the documentation is currently available. Some enhancement of documentation may be needed. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
11	Ensure that system notebooks or other supporting documentation defines system boundaries.	SY-A8	RG 1.200	System boundaries are defined and documented in system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
12	Provide explicit documentation of the rationale for exclusions from modeling in accordance with the modeling.	SY-A12	RG 1.200	This information is in the system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
13	System model enhancements should be considered such as adjacent pump discharge check valve failures due to close or gross back-leakage, strainer common cause, and traveling screen clogging.	SY-A13	SY-05	VEGP NSCW does not have traveling screens nor pump suction strainers because the NSCW pumps use the NSCW cooling tower basin for the suction source and makeup water to the cooling tower basin comes from clean well water. Therefore, this item is not applicable to VEGP. Potential for gross back leakage may be need to be investigated but their contributions to the major mitigating system failures would be small because a pump running failure should be combined with all check valves failures in the redundant trains.
14	Ensure the documentation of systems includes assumptions regarding which components have and have not been included in the model.	SY-A14	RG 1.200	Such information is in the system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
15	Screen the system maintenance procedures in order to establish conditions where a pre-initiator could be present.	SY-A15, HR-A1, HR-A2, HR-A3, HR-B1, HR-B2, HR-C3, LE-E2	HR-01	Screening of Pre-initiator HFEs was documented in each system notebook. Documentation may need to be enhanced to integrally document pre-initiator screening. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
16	Ensure that system documentation includes details on what could cause a system to isolate or trip.	SY-A17	RG 1.200	Such information is in the system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
17	Develop detailed documentation of mutually exclusive portion of the plant fault tree. If possible tie the structure to Tech Spec and other plant operating guidance	SY-A18, DA-A3, DA-C1, DA-C2, DA-C3, DA-C6, DA-C7, DA-C9, QU-B7	DA-01	Mutually exclusive event sets were developed based on Technical Specifications. Documentation needs to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
18	Ensure that system documentation includes specific conditions or requirements for room cooling because of room heatup concerns.	SY-A19	RG 1.200	This item has been closed
19	Ensure that system documentation does not take credit beyond the design basis without justification.	SY-A20	RG 1.200	This item is not applicable to VEGP PRA because no such credit was used in VEGP PRA. So failing to close this item has no impact on this specific application.
20	Ensure that system documentation addresses success criteria variability as a function of accident scenario.	SY-B6	RG 1.200	This item has been closed.
21	Confirm that system documentation does not eliminate support systems if the sole basis is the existence of recovery procedures for them.	SY-B13	RG 1.200	This item is not applicable to VEGP PRA because there is no such case in VEGP PRA. So failing to close this item has no impact on this specific application.
22	Provide documentation of procedure quality to support crew response within the times assigned in the models	HR-D3	RG 1.200	This item has been closed. Such information was provided as part of HRA update (one of the PSFs in the HRA-calculator).

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
23	Assign maximum credit for multiple recovery actions or provide justification for existing credit.	HR-D4, HR-G8	RG 1.200	This item is considered to be technically closed because: • Modeling recovery actions were based on Emergency Operating Procedures. • If MAAP results show that a recovery action is not feasible because of limited time, it was not credited. • Cutset level recovery allowed only one recovery. This item is now just a documentation issue. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
24	Provide documentation of “reasonableness” of HEPs.	HR-D7, HR-G6	RG 1.200	This item has been closed.
25	As part of next HRA Update, document the process used to identify post-initiator operator actions that are subjected to detailed evaluation.	HR-E2	HR-04	OAs were identified and described as part of the event tree analysis. Thus this item has been closed.
26	Add opposite unit hardware and outage unavailabilities to the model for the cross-tie, and perform a more detailed quantification of the operator action HEP. Also, add common cause across all 4 diesel generators.	HR-E2	HR-05	This item has been closed (cross tying an opposite unit EDG model is only the related case and it included operator error, EDG failure, CCF with other EDGs).
27	Document “talkthroughs” with plant staff to confirm that interpretations of procedures are consistent with plant observations and training procedures.	HR-E3	RG 1.200	This item has been closed.
28	Document simulator observations or “talkthroughs” to confirm response models	HR-E4, HR-G5	RG 1.200	This item has been closed.
29	Documentation should include the availability of cues and other indications for detection and evaluation of errors.	HR-F2	RG 1.200	This item has been closed.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
30	Add a reference or basis for the time available to each operator action summary for actions included in the PRA model.	HR-F2, HR-G4	HR-02	MAAP analyses performed for determination of success criteria and operator action timing have been documented as separate calculations. Thus, this item has been closed.
31	Review components with generic failure rates to ensure that outliers (rarely tested or unlikely to be operated) do not use the same generic failure probabilities as components with more common testing and usage experience. Ensure that obvious outliers were not included in component grouping while collecting and processing data.	DA-B2	RG 1.200	Component data collections were done by systems. Thus, the obvious outliers were not included.
32	Ensure that in the latest revision that the component notebook provides the number of failures, demands, and operating hours used in the calculations, and provide assumptions or rules that form a "basis for identification of events as failures" as required by the standard.	DA-C4, DA-C6	RG 1.200	This item has been closed.
33	Ensure that in the latest version of the data notebook that any repeat failures are addressed.	DA-C5	RG 1.200	Repeated failures of similar components were examined during plant specific common cause failure analysis. Such information is available from the NRC CCF Data base analysis system. Thus, this item is considered to be closed. Documentation may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
34	Ensure that the current data notebook describes how completed and logged surveillance test data is used in the analysis. Also address tests that only exercise sub-elements of a component.	DA-C10	RG 1.200	Such information is in system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
35	Ensure that the current data notebook verifies the review of component unavailability against its ability to mitigate an accident.	DA-C11	RG 1.200	Such information is in system notebooks. System notebooks may need to be enhanced. Since it is only a documentation issue, failing to close this gap would not affect the conclusion made for this specific application.
36	Ensure that the current data notebook addresses coincident outages based on plant experience.	DA-C13	RG 1.200	Coincident outage of NSCW fans (allowed by Tech Spec.) was included in the model. Thus this item has been closed.
37	Ensure that in the latest data notebook shows the sources of generic data and that plant components are identified when the generic data is applied.	DA-D2	RG 1.200	This item has been closed.
38	Develop a parametric uncertainty analysis of CDF and LERF.	DA-D3, QU-E3, QU-E4	QU-04	A parametric uncertainty analysis has not been performed. This has no impact on this application because the EPRI approach uses an order of magnitude approach to risk ranking and grouping, and the risk acceptance criteria have been shown to be met even when conservative upper bound CCDP and CLERP values are used in the risk impact assessment.
39	Ensure that in the current data notebook that tests are discussed for reasonableness of results.	DA-D4	RG 1.200	Failure data was collected by system engineers under the direction of PRA analysts.
40	Ensure that in the current data notebook that there is discussion of whether a change in maintenance practices has invalidated any historical data.	DA-D7	RG 1.200	For major Maintenance Rule (MR) scope components (pumps and EDGS), only the data after MR implementation was used. Thus, this item has been partially closed.
41	Consider expanding flood sources to include human induced failures such as maintenance errors, operator overfilling or draining.	IF-B2	RG 1.200	(See note 1)

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
42	For breaks considered in VEGP Design Manual ensure that the nature of the break is characterized, (leak, rupture, spray) and its form.	IF-B3	RG 1.200	(See note 1)
43	Ensure supporting documentation considers flood build up and back flow, including flow into HVAC ducting or adjacent rooms.	IF-C1	RG 1.200	(See note 1)
44	Consider estimating flood frequencies and developing scenarios from them, e.g. loss of service water flood.	IF-D1	RG 1.200	(See note 1)
45	Provide documentation of an analysis of potential flooding precursors including the alignment of support systems.	IF-D2	RG 1.200	(See note 1)
46	If flooding initiating events are developed, care should be taken in grouping those with similar characteristics such as timing, plant response, and available mitigative equipment.	IF-D3	RG 1.200	(See note 1)
47	Describe the process for identifying or excluding potential multi-unit flood initiators.	IF-D4	RG 1.200	(See note 1)
48	When developing plant specific flooding initiators consider plant characteristics, design, expert judgment, and historical experience.	IF-D5	RG 1.200	(See note 1)
49	Modify documentation to list the assumptions used and the model changes made in order to model flood scenarios in Appendix B of the flooding report.	IF-E1, IF-E6, IF-F1	RG 1.200	(See note 1)
50	Ensure the VEGP Design Manual is part of the flooding analysis documentation package.	IF-E2, IF-F1	RG 1.200	(See note 1)
51	Develop scenario specific HEPs based on procedures, stress levels, plant conditions and uncertainty in scenario progression.	IF-E5	RG 1.200	(See note 1)

Table A-3: Gap Analysis Summary and Current VEGP Compliance Status

#	Description	Applicable ASME SRs	Applicable F&Os	Current VEGP Compliance Status
52	For quantified flood scenarios determine the contribution to LERF.	IF-E7	RG 1.200	(See note 1)
53	Perform or document LERF analysis, sensitivity analyses, and importance measures.	IF-F2	RG 1.200	(See note 1)
54	Perform a HFE dependency analysis when the current revision is in the final stages of completion.	QU-C2	RG 1.200	This item has been resolved.
55	A formally documented review and checking of results against other plants should be performed.	QU-D3, QU-D5, QU-F1, QU-F2, LE-F1	QU-05	The VEGP PRA model has been reviewed many times by site personnel; inter-PRA analysts, external contractors, PWROG peer review team, and MSPI peer teams. Thus failing to close this item will not affect this specific application.
56	The model documentation should address model limitations that may impact application.	QU-F6	RG 1.200	VEGP L2UP PRA model is for internal events at power level 1 and level 2 PRA model. Modeling limitations and uncertainties will not have an impact on this application because the EPRI approach uses an order of magnitude approach to risk ranking and grouping, and the risk acceptance criteria have been shown to be met even when conservative upper bound CCDP and CLERP values are used in the risk impact assessment.
57	Document rationale for UET treatment and AMSAC modeling changes.	LE-B3	AS-09	This item has been resolved.
58	Update the Level 2 analysis to include pre-core damage and post- core damage actions.	LE-C5, LE-C7, LE-C8, LE-C9	L2-01	This item has been resolved.
59	Revise ISLOCA IE Calculation to account for correlated failure probabilities	LE-D3	IE-02	Item #59 is the same as item#3

Note 1: There were no A or B F&Os for the internal flooding analysis from the previous VEGP PRA peer review. Even so, the internal flooding analysis has been re-performed in 2008 in order to meet all Capability Category II requirements for IF in the ASME PRA standard. A self assessment by a third party was also performed and all issues have been resolved and documented as a part of the revised internal flooding report [12]. None of the internal flooding scenarios were found to be risk significant.

Table A-4 Additional Gap Analysis Using RG 1.200 Rev 1[1), 2)]

ASME PRA Standard SR Index No..	Requirement	Vogtle PRA L2UP model status	Impacts of non-compliance on RIS_B application
IE-C13	Characterize the uncertainties in the initiating event (IE) frequencies and provide mean values in the quantification of the PRA results	Partially met: Mean values were used for IEs modeled as single basic events. For IEs modeled as a fault tree, parametric uncertainty analysis needs to be performed.	A detailed parametric uncertainty analysis is not necessary for EPRI RIS_B methodology because it uses bounding PRA values. Uncertainty correlation needs to be investigated in interfacing system LOCA scenarios.
SY-A12a	Do not include beneficial failures	Met	NA
SY-A12b	Include those failures that can cause flow diversion pathways	Partially met.	Addressed as item 13 in the original gap analysis table.
SY-A18a	Include simultaneous unavailability of redundant equipments when this is a results of planned activity	Met	NA
HR-I2	Document details of human reliability analysis	Met:	NA
HR-I3	Document key assumptions and key sources of uncertainty	Partially met. Documentation of Pre-initiator human failure events screening needs to be enhanced	Negligible impacts.
DA-C11a	When an unavailability of a front line system component is caused by an unavailability of a support system, count it as support system unavailability	Met	NA
DA-D6a	In CCF analysis, screening both CCF events and independent events	Met	NA
DA-E2	Document Data Analysis details	Met	NA
DA-E3	Document key assumptions and key sources of uncertainty associated with the data analysis	Documentation needs to be enhanced	Negligible impacts
QU-A2a	Provide estimates of the individual sequences in a manner with the estimation of total CDF	Met. The fault tree linking modeling structure enables one to estimate any core damage sequence in the same manner as	NA

Table A-4 Additional Gap Analysis Using RG 1.200 Rev 1[1), 2)]

ASME PRA Standard SR Index No..	Requirement	Vogtle PRA L2UP model status	Impacts of non-compliance on RIS_B application
		the total CDF is evaluated	
QU-A2b	Capability category II: Estimate the mean CDF from internal events accounting the uncertainty correlation	Parametric uncertainty analysis considering an uncertainty correlation is needed	Detailed parametric uncertainty analysis is not necessary for the EPRI RIS_B methodology because it uses bounding PRA values. The effect of the uncertainty correlation needs to be investigated.
QU-B7a	Identify cutsets containing mutually exclusive events in the results	Met. Mutually exclusive events cutsets were removed from mutually exclusive events logic during cutset generation	NA
QU-B7b	Correct castes containing mutually exclusive events	Met. Mutually exclusive events cutsets were removed from mutually exclusive events logic during cutset generation	NA
QU-D1a	Review a sample of significant accident sequences/cutsets sufficient to determine the logic of the cutset or sequence is correct	Met	NA
QU-D1b	Review of the results of the PRA for modeling consistency and operational consistency	Met	NA
QU-D1c	Review results to determine that the flag event settings, mutually event rules and recovery rules yield logical results	Met	NA
QU-D5a	For Capability Category II: Identify significant contributors to the CDF	Met	NA
QU-D5b	Review importance of components and basic events to determine that they make logical sense	Met	NA

1) SC-B6,SC-C4, SY-A23, and HR-G8 were removed from the ASME PRA standards and any gaps identified related to these requirement during the gap analysis based on RG1.200 2003 trial version need not to be closed
2) HR-D7 is no longer required for Capability Category II. Thus any gaps related to HR-D7 needs not to be closed for Capability Category II.

The gap analyses for VEGP PRA model (as summarized in Tables A-3 and A-4) identified that one gap related to the uncertainty correlation needs to be investigated. Considering the state of knowledge, an uncertainty correlation is especially important in estimating the Interfacing System LOCA. The point estimate for the VEGP interfacing system LOCA core damage frequency, which is also the large early release frequency for interfacing system LOCA case, was 3.03E-8/yr. In order to evaluate the impacts of not including an uncertainty correlation, a parametric uncertainty analysis was performed for the interfacing system LOCA core damage frequency (CDF) using EPRI's UNCERT code. The uncertainty correlation was evaluated by using the same sampled value for the same type of valve in the same system during Monte Carlo sampling in UNCERT. The following show the results for interfacing systems LOCA CDF:

Mean:	1.97E-07
5%:	3.76E-10
50%:	8.64E-09
95%:	3.81E-07
Std. Dev.:	3.32E-06

The use of an uncertainty correlation resulted in a significant increase in the mean value. However, the failure data for the rupture of a motor operated valve and that of check valve used in the VEGP L2UP PRA model were based on old generic failure data bases. The rupture failure rates for check valve and motor operator valves in the most recent failure data base, NUREG CR 6928[21], are almost an order of magnitude lower than those used in VEGP L2UP model. NUREG CR-6928 which was published in 2007 was based on more extensive collected data and more recent experiences. If the most recent data from NUREG CR 6928 is used, the results of uncertainty analysis for interfacing LOCA CDF are:

Mean:	3.46E-09
5%:	4.72E-13
50%:	3.47E-10
95%:	1.63E-08
Std. Dev.:	1.09E-08

Furthermore, even the use of the data from NUREG CR 6928 introduced a conservatism, because the VEGP PRA model assumed that the leakage rate would be the equivalent to the case when a valve disk is completely blown away, while the NUREG CR 6928 failure rate for check valves and motor operated valves are for those for leakage rates of 50 gpm or greater. For example, the VEGP PRA model assumed that if an interfacing system LOCA occurs through a RHR hot leg suction line , the leakage rate would be equivalent to that of 12" diameter line break. In such cases, use of the NUREG CR 6928 failure rate is conservative.

Therefore, even after considering the state of knowledge uncertainty correlation, the interfacing system LOCA CDF, which is the same as LERF for interfacing LOCA case, would be less than 1E-8/yr if the most recent failure data from NUREG CR 6928 is used.

General Conclusion Regarding PRA Capability

The VEGP PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection

In the risk-informed inservice inspection program at VEGP, the EPRI RIS_B methodology [Code Case N-716] is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RIS_B plan development to support the safety significance determination and delta risk evaluation steps.

The limited use of specific PRA results in the RIS_B process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174 [23].

Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*
>
> *An example is risk-informed inservice inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.*

Further, Table 1.3-1 of the ASME PRA Standard[1] [20] identifies the bases for PRA capability categories. The bases for Capability Category I for scope and level of detail attributes of the PRA states:

> *Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level including associated human actions.*

Based on the above, in general, Capability Category I should be sufficient for PRA quality for a RIS_B application.

In addition to the above, it is noted that welds are not eliminated from the ISI program on the basis of risk information. The risk significance of a weld may become low. However, it remains in the program, and if, in the future, the assessment of its ranking changes (either by damage mechanism or PRA risk) then it can again become a candidate for inspection. If a weld is determined, outside the PRA evaluation, to be susceptible to either flow-accelerated corrosion

[1] Table A-1 of Regulatory Guide 1.200 identifies the NRC staff position as “No objection” to Section 1.3 of the ASME PRA Standard, which contains Table 1.3-1.

(FAC), primary water stress corrosion cracking (PWSCC), or microbiological induced cracking (MIC) in the absence of any other damage mechanism, then it moves into an “augmented” program where it is monitored for those special damage mechanisms. That occurs no matter what the Risk Ranking of the weld is determined to be.

<u>Conclusion Regarding PRA Capability for Risk-Informed ISI</u>

The VEGP PRA models are suitable for use in the RIS_B application. This conclusion is based on:

- the PRA maintenance and update processes in place,
- the PRA technical capability evaluations that have been performed and are being planned, and
- the RIS_B process considerations, as noted above, that demonstrate the relatively limited reliance of the process on PRA capability.

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