Entergy

Entergy Nuclear Operations, Inc.
600 Rocky Hill Road
Plymouth, MA 02360

Pilgrim Nuclear Power Station

LETTER NUMBER 2.13.030

April 10, 2013

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

SUBJECT: Request for Alternative - Implementation of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping
Request to Use ASME Code Case N-716

Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Nuclear Operations, Inc. (Entergy) requests NRC authorization to implement a risk-informed Inservice Inspection (RI-ISI) program based on the American Society of Mechanical Engineers (ASME) Code Case N-716 as documented in the attached Pilgrim Relief Request (PRR) –22, "Implementation of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping" (Attachment 1). This template format is similar to the submittals the Nuclear Regulatory Commission (NRC) Staff has approved for Waterford 3 and Grand Gulf. This request includes information to address NRC requests for additional information available at the time of development of this submittal.

In accordance with 10 CFR 50.55a(a)(3)(i), the proposed alternative to the referenced requirements may be approved by the NRC provided an acceptable level of quality and safety are maintained. Entergy believes the proposed alternative meets this requirement.

Entergy requests approval of the proposed alternative on or before April 1, 2014. Entergy plans to implement this alternative during the third period of the fourth ISI interval. Although this review is neither exigent nor emergency, your prompt review is requested.

This request for alternative to use ASME Code Case N-716 includes two new commitments that are summarized in Attachment 2.

A047
NRR

If you have any questions or require additional information, please contact me at (508) 830-8403.

Sincerely,

Joseph R. Lynch

Joseph R. Lynch, Licensing Manager

JRL/mew

Attachments:

1. Pilgrim Relief Request (PRR) –22, "Implementation of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping"
2. Licensee-Identified Commitments

cc: Mr. William M. Dean
Regional Administrator, Region 1
U. S. Nuclear Regulatory Commission
2100 Renaissance Boulevard, Suite 100
King of Prussia, PA 19406-1415

Mr. Richard V. Guzman
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
MS O-8C2A
Washington, DC 20555

NRC Senior Resident Inspector
Pilgrim Nuclear Power Station

"

Attachment 1 to Letter 2.13.030

Pilgrim Relief Request (PRR) –22, "Implementation of Risk-Informed/Safety Based Inservice Inspection Alternative for Class 1 and 2 Piping"

(46 Pages)

Technical Acronyms/Definitions

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

Technical Acronyms/Definitions (Continued)

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

Table of Contents

1. INTRODUCTION

Pilgrim Nuclear Power Station (PNPS) is currently in the fourth Inservice Inspection (ISI) interval, as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. PNPS plans to implement a risk-informed/safety-based inservice inspection (RIS_B) program in the third period of the fourth interval, July 1, 2012 through June 30, 2015.

The ASME Section XI Code of record for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping welds for the fourth interval is the 1998 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code with the 2000 Addenda.

The objective of this submittal is to request the use of the RIS_B process for the ISI of 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*, which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure.*

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*, and Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping*. Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Safety Assessment (PSA) Quality

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

The Pilgrim Nuclear Power Station (PNPS) Probabilistic Risk Assessment (PRA) is based on a detailed model of the plant that was originally developed for the Individual Plant Examination (IPE) and Individual Plant Examination for External Events (IPEEE) projects. The PNPS internal events PRA model has been upgraded since the original IPE to meet the guidance of RG 1.200 Rev 1 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," as well as the American Society of Mechanical Engineers and American National Standard (ASME/ANS) PRA Standard RA-Sb-2005.

A formal, BWROG-sponsored industry peer review of the upgraded internal events model was completed in June 2008. The peer review utilized the process described in Nuclear Energy Institute document NEI 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard," January 2005, and the ASME/ANS PRA Standard. This

review confirmed that the PRA model met the requirements of RG 1.200, Revision 1, and ASME/ANS RA-Sb–2005. There were 34 findings identified by the peer review team. Appendix A contains a summary of these findings, including the status of the resolution for each finding and the potential impact of each finding on this application. The PNPS PRA technical capability evaluations and the maintenance and update processes described above and Appendix A provide a robust basis for concluding that the PNPS PRA model is suitable for use in the risk-informed process used for this application.

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

2. PROPOSED ALTERNATIVE TO CURRENT INSERVICE INSPECTION 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 (i.e., Class 1, 2 and 3 piping).

- The plant augmented BWRVIP-75-A program for Class 1 stainless steel and nickel-based alloy piping welds is relied upon to manage Intergranular Stress Corrosion Cracking (IGSCC) at PNPS.

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

- The plant augmented inspection program for localized corrosion per Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, is relied upon to manage this damage mechanism.

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 (see Section 3.5)
- Feedback Loop (see Section 3.6)

Each of these six steps is discussed below:

3.1 Safety Significance Determination

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

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii)

(2) Applicable portions of the shutdown cooling pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling flow path either:

(a) As part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or

(b) Other systems or portions of systems from the RPV to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds

(3) That portion of the Class 2 feedwater system [> 4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the steam generator to the outer containment isolation valve,

(4) Piping within the break exclusion region (BER) greater than 4" NPS for high-energy piping systems as defined by the Owner. Per Code Case N-716, this may include Class 3 or Non-Class piping.

(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.

3.2 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in NRC approved EPRI TR-112657 (i.e., the EPRI RI-ISI methodology), with the exception of the deviation discussed below.

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

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

1. The potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids; or
2. The potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids; or
3. The potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid; or
4. The potential exists for two phase (steam/water) flow; or
5. The potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow;

   AND

   $\Delta T > 50°F$,

   AND

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

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

- **Turbulent Penetration TASCS**

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

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

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

- **Low flow TASCS**

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

- **Valve leakage TASCS**

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

- **Convection Heating TASCS**

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

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. Consideration of cycle severity was used in previous NRC approved RIS_B program submittals for D. C. Cook, Grand Gulf Nuclear Station, Waterford-3, and the Vogtle Electric Generating Plant. The methodology used in the PNPS RIS_B application for assessing TASCS potential conforms to these updated criteria.

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 shall be selected.

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

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

selections are presented in Table 3.3. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations. Only those RIS_B inspection locations that receive a volumetric examination are included.

Class 1 Welds(1)		Class 2 Welds(2)		All Piping Welds(3)	
Total	**Selected**	**Total**	**Selected**	**Total**	**Selected**
641	72	1017	0	1658	72

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, none are HSS at PNPS.

(3) Regardless of safety significance, Class 1, 2, and 3 ASME Section XI in-scope piping components will continue to be pressure tested as required by the ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the pressure test program that remains unaffected by the RIS_B Program.

3.3.1 Current Examinations

PNPS is currently using the NRC previously approved application using Code Case N578 for ISI examination of Category B-F and B-J piping welds and using the traditional ASME Section XI inspection methodology for ISI examination of Category C-F-1, and C-F-2 piping welds per the 1998 Edition of ASME Section XI through the 2000 Addenda.

3.3.2 Successive Examinations

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

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, PNPS will calculate coverage and use additional examinations or techniques in the same manner it has for traditional Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until the examinations are performed. Relief requests for those cases where greater than 90% coverage is not obtained will be submitted per the requirements of 10 CFR 50.55a(g)(5)(iv).

Other than the previous RI-ISI N578 relief request (PRR-10), no other PNPS 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 Revision 1 and the requirements of Code Case N-716, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

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

3.4.1 Quantitative Analysis

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

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

The updated internal flooding PRA was also reviewed to ensure that there is no LSS Class 2 piping with a CCDP/CLERP greater than 1E-4/1E-5. Because some of the RHR piping has a CCDP slightly larger than 1E-4, the whole system was assigned this higher value in the risk impact assessment.

With respect to assigning failure potentials for LSS piping, the criteria are defined in Table 3 of Code Case N-716. That is, those locations identified as susceptible to FAC are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion, or stress corrosion cracking are assigned a medium failure potential, unless they have an identified potential for water hammer loads. In such cases, they will be assigned a high failure potential. Finally, those locations that are identified as not susceptible to degradation are assigned a low failure potential.

In order to streamline the risk impact assessment, a review was conducted that verified that the LSS piping was not susceptible to water hammer. LSS piping may be susceptible to FAC; however, the examination for FAC is performed per the FAC program. This review was conducted similar to that done for a traditional RI-ISI application. Thus, the high failure potential category is not applicable to LSS piping. In lieu of conducting a formal degradation mechanism evaluation for all LSS piping (e.g. to determine if thermal fatigue is applicable), these locations were conservatively assigned to the Medium failure potential ("Assume Medium" in Table 3.4) for use in the change-in-risk assessment. Experience with previous industry RIS_B applications shows this to be conservative.

PNPS has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716 that is consistent with the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657. The analysis estimates the net change-in-risk due to the positive and negative influences of adding and removing locations from the inspection program.

The CCDP and CLERP values used to assess risk impact were estimated based on pipe break location. Based on these estimated values, a corresponding consequence rank was assigned per the requirements of EPRI TR-112657 and upper bound threshold values were used as provided in the table below. Consistent with the EPRI methodology, the upper bound for all break locations that fall within the high consequence rank range was based on the highest CCDP value obtained (e.g., Intermediate LOCA CCDP bounds the large and small LOCA CCDPs).

Break Location	Estimated		Consequence Rank	Upper / Lower Bound		Description of Affected Piping
	CCDP	CLERP		CCDP	CLERP	
LOCA	2E-03	3E-04	HIGH	2E-03 1E-04	3E-04 1E-05	Unisolable RCPB piping of all sizes
Intermediate LOCA (IE-S1) bounds for CCDP and large LOCA bounds for CLERP						
ILOCA[1]	6E-06	9E-07	MEDIUM	1E-04 1E-06	1E-05 1E-07	Piping between 1st and 2nd normally open isolation valve inside containment (RWCU, RCIC, MS, FW, HPCI)
Calculated based on LOCA CCDP and CLERP times valve fail to close probability of ~3E-3						
PLOCA[1]	4E-06	6E-07	MEDIUM	1E-04 1E-06	1E-05 1E-07	Piping between 1st and 2nd normally closed isolation valve inside containment (RHR, CS, RECIRC, RWCU, SBLC)
Calculated based on LOCA CCDP and CLERP times valve rupture probability of ~2E-3						
ILOCA-OC	3E-03	3E-03	HIGH	3E-03 1E-04	3E-03 1E-05	Piping between penetration and outside containment isolation valve with normally open isolation valve inside containment (RWCU, RCIC, MS, FW, HPCI)
Isolable LOCA outside containment CCDP based on valve fail to close probability ~3E-3 (CCDP = CLERP)						
PLOCA-OC	2E-03	2E-03	HIGH	2E-03 1E-04	2E-03 1E-05	Piping between penetration and outside containment isolation valve with normally closed isolation valve inside containment (RHR, CS, SBLC)
Potential LOCA outside containment CCDP based on valve rupture probability ~2E-3 (CCDP = CLERP)						
IILOCA-OC	<1E-05	<1E-05	MEDIUM	1E-05 1E-06	1E-05 1E-07	Class 1 piping upstream of two check valves where the second valve is normally open (RWCU) or closed (RCIC and HPCI) that connect to FW
Isolable LOCA outside containment CCDP based on two valve failures to close probability <1E-5 (CCDP = CLERP)						
Class 2 LSS	1E-04	1E-05	MEDIUM	(U) 1E-04 (L) 1E-06	(U) 1E-05 (L) 1E-07	All other Class 2 system piping designated as low safety significant except for RHR
Estimated based on upper bound for Medium Consequence						
Class 2 RHR	6E-04	6E-05	HIGH	(U) 6E-04 (L) 1E-04	(U) 6E-05 (L) 1E-05	Class 2 RHR lines designated as low safety significant.
Because some of RHR (IE-FL-CRD-RHR) piping has a high CCDP from the internal flooding analysis, the whole system is assigned the above CCDP (0.1 is used for CLERP)						

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

The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RIS_B approach.

Table 3.4 presents a summary of the RIS_B Program versus the 1989 Edition of ASME Code for the selection of Category B-F and B-J piping welds for the 3rd Interval (N578 Application) and 1998 Edition, 2000 Addenda for C-F-1 and C-F-2 piping welds for the 4th Interval ISI Program. The presence of FAC was adjusted for in the quantitative analysis by excluding its impact on the failure potential rank. The exclusion of the impact of FAC on the failure potential rank and therefore in the determination of the change-in-risk, was performed because FAC is a damage mechanism managed by a separate, independent plant augmented inspection program. The RIS_B Program credits and relies upon this plant augmented inspection program to manage this damage mechanism. The plant FAC program will continue to determine where and when examinations shall be performed. Hence, since the number of FAC examination locations remains the same "before" and "after" (the implementation of the RIS_B program) and no delta exists, there is no need to include the impact of FAC in the performance of the risk impact analysis.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and that the acceptance criteria of Regulatory Guide 1.174 Revision 1 and Code Case N-716 are satisfied.

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CRD - Control Rod Drive	4.00E-11	4.00E-12	4.00E-11	4.00E-12
CS - Core Spray	2.90E-10	6.00E-11	2.90E-10	6.00E-11
FW - Feedwater	-2.38E-09	-1.11E-09	-4.24E-10	-3.06E-10
HPCI - High Pressure Coolant Injection	-4.15E-10	-5.15E-10	-1.75E-10	-2.75E-10
MS - Main Steam	-8.10E-10	-5.55E-10	-4.10E-10	-2.91E-10
RCIC - Reactor Core Isolation Cooler	2.51E-11	-1.25E-11	2.51E-11	-1.25E-11
RECIRC - Reactor Recirculation	1.00E-10	1.50E-11	1.00E-10	1.50E-11
RHR - Residual Heat Removal	2.89E-09	3.01E-10	4.49E-09	5.41E-10
RPV - RPV Nozzles	7.90E-10	1.19E-10	7.90E-10	1.19E-10
RWCU - Reactor Water Cleanup	-1.95E-11	2.29E-11	-1.95E-11	2.29E-11
SBLC - Standby Liquid Control	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	**5.06E-10**	**-1.67E-09**	**4.71E-09**	**-1.24E-10**

As shown in Table 3.4, new RIS_B locations were selected such that the RIS_B selections exceed the Section XI selections for certain categories (Delta column has a positive number). To show that the use of a conservative upper bound CCDP/CLERP does not result in an optimistic calculation with regard to meeting the acceptance criteria, a conservative sensitivity was conducted where the RIS_B selections were set equal to the Section XI selections (Delta changed from positive number to zero). The acceptance criteria are met when the number of RIS_B selections is not allowed to exceed Section XI.

3.4.2 Defense-in-Depth

The intent of the inspections mandated by 10 CFR 50.55a for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for selecting inspection locations is based upon terminal end locations, structural discontinuities, and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this methodology has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, a generic assessment of high-consequence sites has been determined by Code Case N-716, supplemented by plant-specific evaluations, thereby requiring a minimum threshold of inspection for important piping whose failure would result in a LOCA or BER break. Finally, Code Case N-716 requires that any piping on a plant-specific basis that has a contribution to CDF of greater than 1E-06 (or 1E-07 for LERF) be included in the scope of the application.

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 period of the fourth interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RIS_B process, as appropriate.

3.6 Feedback (Monitoring)

The RIS_B Program is a living program that is required to be monitored continuously for changes that could impact the basis for which welds are selected for examination. Monitoring encompasses numerous facets, including the review of changes to the plant configuration, changes to operations that could affect the degradation assessment, a review of NDE results, a review of site failure information from the corrective action program, and a review of industry failure information from industry operating experience (OE). Also included is a review of PRA changes for their impact on the RIS_B program. These reviews provide a feedback loop such that new relevant information is obtained that will ensure that the appropriate identification of HSS piping locations selected for examination is maintained. As a minimum, this review will be conducted on an ASME period basis. In addition, more frequent adjustment may be required as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

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

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

For preservice examinations, PNPS 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

PNPS is currently in the third period of the fourth ISI interval and plans to implement this RIS_B submittal for the third period of the fourth ISI interval.

A comparison between the RIS_B Program and the 1989 Edition of ASME Code for the selection of Category B-F and B-J piping welds for the 3rd Interval (N578 Application) and 1998 Edition, 2000 Addenda for C-F-1 and C-F-2 piping welds for the 4th Interval ISI Program is provided in Table 4.

5. REFERENCES/DOCUMENTATION

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

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

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

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

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

USNRC Safety Evaluation for Grand Gulf Nuclear Station Unit 1, Request for Alternative GG-ISI-002-Implement Risk-Informed ISI based on ASME Code Case N-716, dated September 21, 2007. ADAMS Accession No. ML072430005

USNRC Safety Evaluation for DC Cook Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based ISI program for Class 1 and 2 Piping Welds, dated September 28, 2007. See ADAMS Accession No. ML072620553.

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

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

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

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

Supporting Onsite Documentation

EPRI Report IR-2012-527 "ASME Code Case N716 Evaluation, Pilgrim Nuclear Plant"

Table 3.1
Code Case N-716 Safety Significance Determination

System	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6	High	Low
CRD	48							✓
CS	44	✓					✓	
	158							✓
FW	76	✓					✓	
	4							✓
HPCI	41	✓					✓	
	160							✓
MS	92	✓					✓	
	4							✓
RCIC	36	✓					✓	
	91							✓
RECIRC	70	✓					✓	
RHR	58	✓	✓				✓	
	552							✓
RPV	36	✓					✓	
RWCU	116	✓					✓	
SBLC	72	✓					✓	
Summary Results all Systems	583	✓					✓	
	58	✓	✓				✓	
	1017							✓
Totals	**1658**							

(1) System Scope:
CRD – Control Rod Drive
CS – Core Spray
FW – Feedwater
HPCI – High Pressure Coolant Injection
MS – Main Steam
RCIC- Reactor Core Isolation Cooling
RECIRC – Reactor Recirculation System
RHR – Residual Heat Removal
RPV – Reactor Pressure Vessel Nozzles
RWCU – Reactor Water Cleanup
SBLC – Standby Liquid Control

Table 3.2
Failure Potential Assessment Summary

System[1]	*Thermal Fatigue*		*Stress Corrosion Cracking*				Localized			Flow	
		TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CRD - Control Rod Drive											
CS - Core Spray											
FW - Feedwater	✓	✓									
HPCI - High Pressure Coolant Injection	✓										
MS - Main Steam		✓									
RCIC - Reactor Core Isolation Cooler											
RECIRC - Reactor Recirculation											
RHR - Residual Heat Removal	✓	✓									
RPV - RPV Nozzles	✓	✓							✓		
RWCU - Reactor Water Cleanup											
SBLC - Standby Liquid Control											

Notes:

1. Systems are described in Table 3.1
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes CRD in its entirety, as well as portions of the CS, FW, HPCI, MS, RCIC and RHR systems.

Table 3.3: Code Case N716 Selections

System	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CRD		48	None					0
CS	27		None	✓	✓			4
CS	6		None	✓				0
CS	11		None	✓		✓		1
CS		158	None					0
FW	24		TT	✓	✓			4
FW	12		TASCS,TT	✓	✓			2
FW	1		TASCS,TT	✓				0
FW	6		TASCS,TT	✓		✓		2
FW	25		None	✓	✓			0
FW	1		None	✓				0
FW	7		None	✓		✓		0
FW		4	None					0
HPCI	2		TASCS	✓		✓		1
HPCI	10		None	✓	✓			4
HPCI	7		None	✓				0
HPCI	22		None	✓		✓		0
HPCI		160	None					0
MS	2		TT	✓	✓			1
MS	1		TT	✓				0
MS	3		TT	✓		✓		1
MS	66		None	✓	✓			6
MS	4		None	✓				0
MS	16		None	✓		✓		2
MS		4	None					0
RCIC	5		None	✓	✓			3
RCIC	12		None	✓				0
RCIC	19		None	✓		✓		1
RCIC		91	None					0
RECIRC	66		None	✓	✓			8
RECIRC	4		None	✓				0
RHR	7		TT	✓	✓			1
RHR	8		TASCS	✓	✓			3
RHR	12		None	✓	✓			0
RHR	14		None	✓				0
RHR	17		None	✓		✓		2
RHR		552	None					0
RPV	4		TASCS,TT,CC	✓	✓			1
RPV	32		None	✓	✓			5
RWCU	50		None	✓	✓			7

System	Weld Count		N716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
RWCU	17		None	✓				0
RWCU	49		None	✓		✓		4
SBLC	11		None	✓	✓			6
SBLC	31		None	✓				0
SBLC	30		None	✓		✓		3
Total All Systems	12		TASCS,TT	✓	✓			2
	4		TASCS,TT,CC	✓	✓			1
	1		TASCS,TT	✓				0
	6		TASCS,TT	✓		✓		2
	33		TT	✓	✓			6
	1		TT	✓				0
	3		TT	✓		✓		1
	8		TASCS	✓				3
	2		TASCS	✓		✓		1
	304		None	✓	✓			43
	96		None	✓				0
	171		None	✓		✓		13
Totals	641	**1017**						72

Notes:

(1) Systems are described in Table 3.1

Table 3.4 Risk Impact Analysis Results

System	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
CRD Total	**Low**	Class 2 LSS		Assume Medium	**4**	**0**	-4	**4.00E-11**	**4.00E-11**	**4.00E-12**	**4.00E-12**
CS	High	LOCA	None	Low	12	4	-8	8.00E-11	8.00E-11	1.20E-11	1.20E-11
CS	High	PLOCA	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
CS	High	PLOCA-OC	None	Low	4	1	-3	3.00E-11	3.00E-11	3.00E-11	3.00E-11
CS	Low	Class 2 LSS		Assume Medium	18	0	-18	1.80E-10	1.80E-10	1.80E-11	1.80E-11
CS Total					36	5	-31	**2.91E-10**	**2.91E-10**	**6.01E-11**	**6.01E-11**
FW	High	LOCA	TT	Medium	5	4	-1	-8.40E-10	2.00E-10	-1.26E-10	3.00E-11
FW	High	LOCA	TASCS,TT	Medium	0	2	2	-7.20E-10	-4.00E-10	-1.08E-10	-6.00E-11
FW	High	ILOCA	TASCS,TT	Medium	1	0	-1	6.00E-12	1.00E-11	6.00E-13	1.00E-12
FW	High	ILOCA-OC	TASCS,TT	Medium	1	2	1	-9.00E-10	-3.00E-10	-9.00E-10	-3.00E-10
FW	High	LOCA	None	Low	5	0	-5	5.00E-11	5.00E-11	7.50E-12	7.50E-12
FW	High	ILOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
FW	High	ILOCA-OC	None	Low	1	0	-1	1.50E-11	1.50E-11	1.50E-11	1.50E-11
FW	Low	Class 2 LSS		Assume Medium	1	0	-1	1.00E-11	1.00E-11	1.00E-12	1.00E-12
FW Total					15	8	-7	**-2.38E-09**	**-4.15E-10**	**-1.11E-09**	**-3.05E-10**
HPCI	High	ILOCA-OC	TASCS	Medium	0	1	1	-5.40E-10	-3.00E-10	-5.40E-10	-3.00E-10
HPCI	High	LOCA	None	Low	1	4	3	-3.00E-11	-3.00E-11	-4.50E-12	-4.50E-12
HPCI	High	ILOCA	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
HPCI	High	ILOCA-OC	None	Low	1	0	-1	1.50E-11	1.50E-11	1.50E-11	1.50E-11
HPCI	High	IILOCA-OC	None	Low	2	0	-2	1.00E-13	1.00E-13	1.00E-13	1.00E-13
HPCI	Low	Class 2 LSS		Assume Medium	14	0	-14	1.40E-10	1.40E-10	1.40E-11	1.40E-11
HPCI Total					21	5	-16	**-4.13E-10**	**-1.73E-10**	**-5.15E-10**	**-2.75E-10**
MS	High	LOCA	TT	Medium	0	1	1	-3.60E-10	-2.00E-10	-5.40E-11	-3.00E-11
MS	High	ILOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS	High	ILOCA-OC	TT	Medium	0	1	1	-5.40E-10	-3.00E-10	-5.40E-10	-3.00E-10
MS	High	LOCA	None	Low	11	6	-5	5.00E-11	5.00E-11	7.50E-12	7.50E-12
MS	High	ILOCA	None	Low	4	0	-4	2.00E-12	2.00E-12	2.00E-13	2.00E-13
MS	High	ILOCA-OC	None	Low	4	2	-2	3.00E-11	3.00E-11	3.00E-11	3.00E-11

System	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
MS	Low	Class 2 LSS		Assume Medium	1	0	-1	1.00E-11	1.00E-11	1.00E-12	1.00E-12
MS Total					20	10	-10	**-8.08E-10**	**-4.08E-10**	**-5.55E-10**	**-2.91E-10**
RCIC	High	LOCA	None	Low	0	3	3	-3.00E-11	-3.00E-11	-4.50E-12	-4.50E-12
RCIC	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RCIC	High	ILOCA-OC	None	Low	0	1	1	-1.50E-11	-1.50E-11	-1.50E-11	-1.50E-11
RCIC	High	IILOCA-OC	None	Low	1	0	-1	5.00E-14	5.00E-14	5.00E-14	5.00E-14
RCIC	Low	Class 2 LSS		Assume Medium	7	0	-7	7.00E-11	7.00E-11	7.00E-12	7.00E-12
RCIC Total					8	4	-4	**2.51E-11**	**2.51E-11**	**-1.25E-11**	**-1.25E-11**
RECIRC	High	LOCA	None	Low	18	8	-10	1.00E-10	1.00E-10	1.50E-11	1.50E-11
RECIRC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RECIRC Total					18	8	-10	**1.00E-10**	**1.00E-10**	**1.50E-11**	**1.50E-11**
RHR	High	LOCA	TT	Medium	6	1	-5	3.60E-10	1.00E-09	5.40E-11	1.50E-10
RHR	High	LOCA	TASCS	Medium	6	3	-3	-3.60E-10	6.00E-10	-5.40E-11	9.00E-11
RHR	High	LOCA	None	Low	6	0	-6	6.00E-11	6.00E-11	9.00E-12	9.00E-12
RHR	High	PLOCA	None	Low	7	0	-7	3.50E-12	3.50E-12	3.50E-13	3.50E-13
RHR	High	PLOCA-OC	None	Low	3	2	-1	1.00E-11	1.00E-11	1.00E-11	1.00E-11
RHR	Low	Class 2 RHR		Assume Medium	47	0	-47	2.82E-09	2.82E-09	2.82E-10	2.82E-10
RHR Total					75	6	-69	**2.89E-09**	**4.49E-09**	**3.01E-10**	**5.41E-10**
RPV	High	LOCA	TASCS,TT,CC	Medium	4	1	-3	6.00E-10	6.00E-10	9.00E-11	9.00E-11
RPV	High	LOCA	None	Low	24	5	-19	1.90E-10	1.90E-10	2.85E-11	2.85E-11
RPV Total					28	6	-22	**7.90E-10**	**7.90E-10**	**1.19E-10**	**1.19E-10**
RWCU	High	LOCA	None	Low	2	7	5	-5.00E-11	-5.00E-11	-7.50E-12	-7.50E-12
RWCU	High	ILOCA	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
RWCU	High	ILOCA-OC	None	Low	6	4	-2	3.00E-11	3.00E-11	3.00E-11	3.00E-11
RWCU	High	IILOCA-OC	None	Low	8	0	-8	4.00E-13	4.00E-13	4.00E-13	4.00E-13
RWCU	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RWCU Total					19	11	-8	**-1.81E-11**	**-1.81E-11**	**2.31E-11**	**2.31E-11**
SBLC	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SBLC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00

System	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
SBLC	High	PLOCA-OC	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SBLC Total					0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Grand Total					**244**	**63**	**-181**	**5.22E-10**	**4.73E-09**	**-1.67E-09**	**-1.22-10**

Notes

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

Table 4: Inspection Location Selections Comparison

System	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
CRD		✓	Class 2 LSS		Assume Medium	C-F-2	48	4		0	NA
CS	✓		LOCA	None	Low	B-F,B-J	27	12		4	NA
CS	✓		PLOCA	None	Low	B-J	6	2		0	NA
CS	✓		PLOCA-OC	None	Low	B-J	11	4		1	NA
CS		✓	Class 2 LSS		Assume Medium	C-F-1,C-F-2	158	18	2	0	NA
FW	✓		LOCA	TT	Medium	B-J	24	5		4	NA
FW	✓		LOCA	TASCS,TT	Medium	B-J	12	0		2	NA
FW	✓		ILOCA	TASCS,TT	Medium	B-J	1	1		0	NA
FW	✓		ILOCA-OC	TASCS,TT	Low	B-J	6	1		2	NA
FW	✓		LOCA	None	Low	B-J	25	5		0	NA
FW	✓		ILOCA	None	Low	B-J	1	1		0	NA
FW	✓		ILOCA-OC	None	Low	B-J	7	1		0	NA
FW		✓	Class 2 LSS		Assume Medium	C-F-2	4	1		0	NA
HPCI	✓		ILOCA-OC	TASCS	Medium	B-J	2	0		1	NA
HPCI	✓		LOCA	None	Low	B-J	10	1		4	NA
HPCI	✓		ILOCA	None	Low	B-J	7	3		0	NA
HPCI	✓		ILOCA-OC	None	Low	B-J	8	1		0	NA
HPCI	✓		IILOCA-OC	None	Low	B-J	14	2		0	NA
HPCI		✓	Class 2 LSS		Assume Medium	C-F-2	160	14		0	NA
MS	✓		LOCA	TT	Medium	B-J	2	0		1	NA
MS	✓		ILOCA	TT	Medium	B-J	1	0	1	0	NA
MS	✓		ILOCA-OC	TT	Medium	B-J	3	0	1	1	NA
MS	✓		LOCA	None	Low	B-J	66	11		6	NA
MS	✓		ILOCA	None	Low	B-J	4	4		0	NA
MS	✓		ILOCA-OC	None	Low	B-J	16	4		2	NA
MS		✓	Class 2 LSS		Assume Medium	C-F-2	4	1		0	NA
RCIC	✓		LOCA	None	Low	B-J	5	0	2	3	NA
RCIC	✓		ILOCA	None	Low	B-J	12	0	1	0	NA

System	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
RCIC	✓		ILOCA-OC	None	Low	B-J	4	0	2	1	NA
RCIC	✓		IILOCA-OC	None	Low	B-J	15	1		0	NA
RCIC		✓	Class 2 LSS		Assume Medium	C-F-2	91	7		0	NA
RECIRC	✓		LOCA	None	Low	B-J	66	18	3	8	NA
RECIRC	✓		PLOCA	None	Low	B-J	4	0		0	NA
RHR	✓		LOCA	TT	Medium	B-J	7	6		1	NA
RHR	✓		LOCA	TASCS	Medium	B-J	8	6		3	NA
RHR	✓		LOCA	None	Low	B-J	12	6		0	NA
RHR	✓		PLOCA	None	Low	B-F,B-J	14	7		0	NA
RHR	✓		PLOCA-OC	None	Low	B-J	17	3		2	NA
RHR		✓	Class 2 LSS		Assume Medium	C-F-1,C-F-2	552	47		0	NA
RPV	✓		LOCA	TASCS,TT,CC	Medium	B-F	4	4		1	NA
RPV	✓		LOCA	None	Low	B-F	32	24	5	5	NA
RWCU	✓		LOCA	None	Low	B-F,B-J	50	2	12	7	NA
RWCU	✓		ILOCA	None	Low	B-J	15	3		0	NA
RWCU	✓		ILOCA-OC	None	Low	B-J	25	6		4	NA
RWCU	✓		IILOCA-OC	None	Low	B-F,B-J	24	8		0	NA
RWCU	✓		PLOCA	None	Low	B-J	2	0		0	NA
SBLC	✓		LOCA	None	Low	B-J	11	0		0	6
SBLC	✓		PLOCA	None	Low	B-J	31	0	11	0	NA
SBLC	✓		PLOCA-OC	None	Low	B-J	30	0	6	0	3
						Totals	**1658**	**244**	**46**	**63**	**9**

Notes

1. Systems are described in Table 3.1
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 4 of Code Case N-716. Code Case N-716 allows the existing plant augmented inspection program for IGSCC (Categories B through G) in a BWR to be credited toward the 10% requirement. This option is not applicable for the PNPS RIS_B application. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals and to indicate when RIS_B selections will receive a VT-2 examination (these are not credited in risk impact assessment).
3. The failure potential rank for high safety significant (HSS) locations is assigned as "High", "Medium", or "Low" depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").

Appendix A

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

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

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

PRA Maintenance and Update

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

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

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

Regulatory Guide 1.200 BWROG Peer Review of the PNPS Internal Events PRA Model

The PNPS PRA internal events model went through a Regulatory Guide 1.200 BWR Owners Group peer review in June 2008. The NEI 05-04 process [Reference 3], the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard [Reference 4], and Regulatory Guide 1.200, Rev. 1 [Reference 5] were used for the peer review.

The 2008 PNPS PRA Peer Review was a full-scope review of all the Technical Elements of the internal events, at-power PRA:

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

During the PNPS PRA model Peer Review, the technical elements identified above were assessed with respect to Capability Category II criteria to better focus the Supporting Requirement assessments. The ASME/ANS PRA Standard has 331 individual Supporting Requirements. Of the 303 ASME/ANS PRA Standard Supporting Requirements that are applicable to the PNPS PRA model, approximately 86% were satisfied at Capability Category II criteria or greater. The Facts and Observations (F&Os) for the PNPS PRA peer review are provided in the report, entitled, "Pilgrim Station PRA Peer Review Report Using ASME PRA Standard Requirements" [Reference 6]. Of the 93 Facts and Observations (F&Os) generated by the Peer Review Team, 34 were considered Findings, 58 were Suggestions, and one was a Best Practice.

As a result of the Regulatory Guide 1.200 BWROG peer review, all the above mentioned F&Os (other than best practices) have been identified as potential improvements to the PNPS PRA model and are tracked in the Entergy Model Change Request (MCR) database. The 34 findings resulting from the peer review were reviewed against the table of technical adequacy requirements in the EPRI Topical Report 1021467 [Reference 7]. The PNPS peer review identified 14 of 34 findings meeting at least Capability Category I (CCI) supporting requirements. Since CCI was determined to be sufficient for these supporting requirements for the purpose of

this application, no further resolution of these findings is necessary. In addition, two findings were associated with supporting requirements that were not applicable to the RI-ISI program since they are limited to Maintenance and Update tasks, and two findings were associated with supporting requirements that did not need to be 'Met'. Therefore, based on the EPRI Streamlined RI-ISI capability category requirements, resolution of these 18 peer review findings, as shown in Table 1, is not required for this PNPS risk informed ISI application. The status of the resolution of the other 16 peer review findings and the potential impact of each finding on this application are discussed in Table 2. Most of the findings in Table 2 are related to documentation and have no material impact. Resolution of the remaining findings is expected to have a minor impact on the input for this application and will have a negligible, if any, impact on the conclusions of this application.

Since the revised documents will be formally issued with the final update, addressing all the peer review findings, those findings are considered resolved but will not be considered closed until the final revised model and report are formally issued.

External Events

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

Summary

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

References

[1] Engineering Report, PNPS-RPT- PNPS-NE-07-00006, Rev.0, "Pilgrim Probabilistic Safety Assessment (PSA), Rev. 2", April 2008.

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

[3] NEI 05-04, Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard, Nuclear Energy Institute, Rev. 1, November 2007.

[4] American Society of Mechanical Engineers/American Nuclear Society, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASE RA-Sb-2005), December 2005.

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

[6] Pilgrim Station PRA Peer Review Report Using ASME PRA Standard Requirements, October 2008.

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

[8] NRC Final Safety Evaluation for the EPRI Topical Report 1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs", (TAC NO. ME1057), January 2012.

Table 1 - PNPS Peer Review Findings With No Further Resolution Required for This Application		
Assoc. SR	**Finding Description**	**Reason**
IE-A6	There is no evidence provided for independent review by, or interview of, plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
IE-B1	Initiating Event TMS - Controlled Manual Shutdown is listed in the following places in the Initiating Event Notebook: Table 1.5.1.1.1 Table 1 - PNPS Initiating Event Data Sources Appendix A1, Initiating Events Notebook, Section 6.7 Attachment A5, PNPS Initiator Frequency Results This event is also included in the PNPS_07_maint.rr data file. However, it is not included in the PNPSCombo.caf. In Appendix A1, section 6.7, this event is described as being included in the T3A initiating event.	Capability Categories I, II and III were met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
IE-C1a	The initiating event updating includes review and screening of LERs but the screening is not documented. The justification for excluding data is not provided as required by the SR	This SR does not need to be met in accordance with the EPRI Streamlined RI-ISI requirements.
SY-A4	This SR requires discussions with system engineers to confirm the adequacy of the system modeling (Cat 1) or the performance of plant walkdowns and interviews with system engineers and Operations (Cat 2/3). There was no objective evidence that the plant walkdowns or interviews were conducted in the system notebooks. There is evidence of reviews by system engineers, however.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.

Table 1 - PNPS Peer Review Findings With No Further Resolution Required for This Application		
Assoc. SR	**Finding Description**	**Reason**
HR-D3	Using the NRC clarification for this SR, additional documentation is needed to meet Capability Category 2. The clarification requires additional evaluation of procedure quality and human-machine interface to meet Capability Category 2 as described below: For written procedures, quality includes the evaluation of format, logical structure, ease of use, clarity, and comprehensiveness For Administrative Controls for Independent Review, quality includes the configuration control process, technical review process, training process, and management emphasis on adherence to procedures. For Human-Machine Interface, quality includes adherence to human factors guidelines [NUREG-0700, Rev. 2, Human-System Interface Design Review Guidelines, O'hara, et al, May 2002] and results of any quantitative evaluations of performance per functional requirements.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
HR-E3	As an enhancement to the HFE summaries, a section related to Operator Interviews should be added to document the discussion and any changes made based on those observations. This should include a description of the process used for selection of HFEs to review and for the interviews / talk-throughs. The standard identifies the difference between capability category 1 (i.e. "REVIEW the interpretation ...") and capability category 2/3 (I.e. "TALK THROUGH (i.e., review in detail).	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
HR-E4	As an enhancement to the HFE summaries, a section related to simulator observation could be added to document the observation and any changes made based on those observations. Similarly to the recommendation in HR-E3-01, the process used to identify which HFEs would be observed and for evaluation of simulator observations should be included with the interview documentation.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
HR-G5	The time required documented in the notebook is not consistently referenced to a JPM or to simulator exercises. There are many instances where no reference or basis of the time required is documented. Note that the timing used in these cases does appear to be reasonable. There was documentation of review of the HRA notebook by Operations Training but this is deemed to not meet this requirement.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.

Table 1 - PNPS Peer Review Findings With No Further Resolution Required for This Application		
Assoc. SR	**Finding Description**	**Reason**
DA-A1	IDENTIFY from the systems analysis the basic events for which probabilities are required. Examples of basic events include: (a) independent or common cause failure of a component or system to start or change state on demand (b) independent or common cause failure of a component or system to continue operating or provide a required function for a defined time period (c) equipment unavailable to perform its required function due to being out of service for maintenance (d) equipment unavailable to perform its required function due to being in test mode (e) failure to recover a function or system (e.g., failure to recover offsite-power) (f) failure to repair a component, system, or function in a defined time period The components are identified in the System Analysis, also included are maintenance information and surveillance intervals, however the failure modes, common cause grouping, and repair times are not called out. The fault trees developed contain this information so the process was performed but not described.	Capability Categories I, II and III were met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
DA-E3	The PRA performs a parametric uncertainty analysis, but generally does not provide an analysis of sources of uncertainty and related assumptions for Data Analysis (and in other PRA elements) consistent with the intent of these related SRs in the ASME Standard and the NRC expectations (as evidenced by NRC Memorandum, "Notice of Clarification to Rev. 1 of Regulatory Guide 1.200", July 27, 2007 (NRC ADAMS Accession number ML071170054). As such, the intent of SR DA-E3 is judged not met by the current PRA documentation.	This SR does not need to be met in accordance with the EPRI Streamlined RI-ISI requirements.
IF-C3b (IFSN-A8)	The analysis does not clearly indicate that structural failures of doors/walls were considered in the propagation analysis. This SR and the NRC clarification of this SR require explicit consideration of this potential.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.

Table 1 - PNPS Peer Review Findings With No Further Resolution Required for This Application		
Assoc. SR	**Finding Description**	**Reason**
QU-A2b	ESTIMATE the mean CDF from internal events, accounting for the "state-of- knowledge" correlation between event probabilities when significant. Meeting CAT-II requirement would require revising the type code for which generic data or plant-specific data being used is the same to be generic code that could be used in any system before running Monte Carlo simulation using UNCERT. However, point estimate CDF was estimated from internal events.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
QU-D5a	Significant contributors to CDF are identified in Appendix I of the PNPS PSA; however, SSCs and operator actions that contribute to initiating event frequencies are not identified.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
LE-C2b	It is expected that the accident sequence progression is examined to determine whether repair can be credited in the analysis. This examination is not documented.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
LE-C8b	There is no evidence of review of LERF sequences for the purpose of identifying equipment or actions that could reduce LERF frequency.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
LE-E2	This SR requires that realistic parameter estimates be used in the characterization of significant accident progression sequences resulting in a LERF. "Significance" is not defined and conservative parameter estimates are used in the analysis that are based on the plants and generic sources.	Capability Category I was met for this SR, therefore the EPRI Streamlined RI-ISI requirements are satisfied.
MU-E1	The PRA configuration control process shall include a process for maintaining control of computer codes used to support PRA quantification. The PRA software is classified and controlled by the corporate IT process. No formal process used for the control of the spreadsheets data/input/output files.	Not applicable to the RI-ISI program

Table 1 - PNPS Peer Review Findings With No Further Resolution Required for This Application		
Assoc. SR	**Finding Description**	**Reason**
MU-F1	The PRA configuration control process shall be documented. *Documentation typically includes:* (a) Description of the process used to monitor PRA inputs and collect new information, *(b) Evidence that the aforementioned process is active,* (c) Descriptions of proposed changes, (d) Descriptions of changes in PRA due to each Update or Upgrade, (e) Record of the performance and result of the appropriate PRA reviews, (f) Record of the process and results used to address the cumulative impact of pending changes, (g) Record of the process and results used to evaluate changes on previously implemented risk-informed decisions (pursuant to MU-D1), (h) Description of the process used to maintain software configuration control. Reviewers found no evidence of the following items being fully implemented and documented, (e) Record of the performance and result of the appropriate PRA reviews, (f) Record of the process and results used to address the cumulative impact of pending changes, (g) Record of the process and results used to evaluate changes on previously implemented risk-informed decisions (pursuant to MU-D1), (h) Description of the process used to maintain software configuration *control.*	Not applicable to the RI-ISI program

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
IE-A5	The SR directs that in identification of IEs, events that occurred at other than at-power must be considered, included events that resulted in a controlled shutdown, unless the event is NOT applicable at-power. However, the data included review of ONLY events that occurred at power.	This is a potentially non-conservative omission because initiating events could have been missed.	Perform a review of events at other than at-power for potential initiating events.	**Resolved – No Impact** All PNPS Event Reports that occurred during the update period were reviewed and no additional shutdown events which could have been initiating events during power operation were identified. The list of modeled initiating events was formally presented to, and reviewed with plant personnel who raised no issues that pointed to the need to consider additional initiating events.
SC-B5	No evidence was found that results of the thermal/hydraulic analyses (and other success criteria supporting calculations) were checked for reasonableness by comparison with analyses performed for similar plants, or alternate calculation. The review team found that the justification given in the self-assessment (i.e., "checked by examining the FSAR & Design Basis Documents") is not sufficient. For example, there are calculations to determine the time available for certain operator actions, time that is not found in the FSAR or design basis documents.	Required by SR. Low impact on PSA results. Although there is potential for measurable impact on CDF due to inappropriate success criteria, the peer review team did not find any such instances.	Check the reasonableness of PNPS calculation results by comparing with results of similar calculations performed for a different plant where applicable, perform alternate calculations when needed. Document process for reasonableness checking.	**Resolved – No Impact** This is a documentation enhancement issue. Inserted a new section into Appendix B1, "Success Criteria Notebook" comparing the Pilgrim success criteria with those selected for the Fitzpatrick and Vermont Yankee nuclear plants. This information is provided in Section B1.3.2.8, "Success Criteria Comparison with other PSAs" of Appendix B1.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application

Assoc. SR	Finding Description	Basis for Significance	Peer Review Team Suggested Resolution	Reason
SY-B3	The system analysis does not provide justification for the selection of the system specific common cause groups.	It appears that the approach used is reasonable but the documentation in system notebooks is not provided to identify the groups selected for application of common cause, other than the fault tree model itself. The justification for the CCCGs modeled, required by the SR, is not provided.	Provide a common cause section in each system notebook identifying the components that are grouped and why they are grouped (based on the grouping process).	**Resolved – No Impact** This is a documentation enhancement issue. The system notebooks were revised to include a justification for selection of CCF groups and each notebook now has a Table 2.1 "Common Cause Failure Basic Events" which lists the CCF basic events in each system fault tree and identifies the CCF event size and the group size. No additional CCF groups requiring modeling were identified.
SY-B8	There is no evidence of a systematic approach to identification of spatial/environmental hazards that could impact multiple systems or redundant components.	The system notebooks do not provide systematic evidence of the search for spatial/environmental hazards. This is a potentially non-conservative omission.	Include a section in the notebooks documenting the results of the search for spatial/environmental hazards (negative or positive)	**Resolved – No Impact** This is a documentation enhancement issue. The system notebooks were revised to more adequately address spatial and environmental hazards. Additionally, the dependency table in each system note book provides spatial and environmental information from the system walkdown results. No additional hazards were identified.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
HR-A3	The work practices (T&M activities, procedures/practices) do not identify mechanisms that simultaneously affect equipment in different trains/redundant systems. This is a potentially non-conservative omission.	The guidance talks about the consideration of simultaneous equipment impact but Table H2-1 does not address the issue. The SR directs that the work practices be identified.	It would be straight forward to add a column to Table H2-1 that included other impacted equipment, if applicable for each procedure/practice.	**Resolved – No Impact** This is a documentation enhancement issue. The HRA documentation was reviewed following the peer review. Based on this review it was concluded that the current documentation provides the information identified in the finding. The work practices which can simultaneously impact redundant components have in fact been identified. The applicable procedure is cited for each pre-initiator human failure event (HFE), along with a list of components which could be misaligned or miscalibrated as a result of restoration errors committed during performance of the procedure. In addition, the evaluation performed for each HFE states which common-mode errors are relevant. For example, improper setup of the test equipment is a common-mode failure considered for instrument calibration procedures.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
DA-C10	DETERMINE the number of plant-specific demands on standby components on the basis of the number of plant records. BASE number of surveillance tests on plant surveillance requirements and actual practice. BASE number of planned maintenance activities on plant maintenance plans and actual practice. BASE number of unplanned maintenance acts on actual plant experience. When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into sub-elements (or causes) that are fully tested, then USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element sometimes has many more successes than another. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid tests would be significantly decreased.] There was no cross-referenced check provided for review of test procedures to determine whether a test should be credited for each possible failure mode. Self-assessment for this SR indicated that some systems might not meet this requirement based on boundary definition.	From the PRA documents it is unclear as to what information was extracted from plant records, or if/how it was screened. The PRA could be missing or have incorrect information.	Provide more detail of information extracted from plant records, how it was treated, screened, or modified, and where it was used in the PRA.	**Resolved – No Impact** This is a documentation enhancement issue. The Data Analysis documentation was reviewed and it was confirmed that the Bayesian update of generic data was limited to: 1) System component failure modes for which plant failures had occurred during the update period, and 2) Important safety system components, where reliable plant data existed, provided that the demand failure generic mean was approximately 1.0E-06 (or lower), or the hourly failure generic mean was approximately 5.0E-06 (or lower). This was done to avoid potentially non-conservative Bayesian updates when no plant failures had occurred. For all other components, the generic data was used with no Bayesian update. Failures were assigned to the applicable failure mode. Where Bayesian updates were done, plant records from the MSPI program were used to identify the appropriate demand and/or hourly data to be used for the update. When no plant data was readily available, conservative estimates were made based on the applicable current plant surveillance test procedures and programs, operational practices and unplanned operational demands. These estimates are documented in the EXCEL spreadsheet that is included in the PRA update documents.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application

Assoc. SR	Finding Description	Basis for Significance	Peer Review Team Suggested Resolution	Reason
DA-C13	EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) based on actual plant experience. CALCULATE coincident maintenance unavailabilities that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares," i.e., plant systems that have more redundancy than is addressed by tech specs. For example, the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with tech specs. There is no evidence provided for examination and/or calculation of coincident unavailability due to maintenance for redundant equipment (both in transystem and intersystem) based on actual plant experience.	There is no evidence provided for examination and/or calculation of coincident unavailability due to maintenance for redundant equipment (both in transystem and intersystem) based on actual plant experience.	Perform and document examination of coincident unavailability due to maintenance for redundant equipment	**Resolved – No Significant Impact** Plant maintenance activities and unavailability data were reviewed and the LPCI Loop Selection Logic tests were identified to result in coincident unavailability of both LPCI trains. An average coincident UA term of about 2.67E-04 was added to the model and the net impact on the CDF/LERF was determined to be an increase of less than 1E-6/1E-7 per year respectively. No additional coincident UA terms were determined to be required for the system models. This is based on the following; 1. Coincident UA is administratively restricted by Technical Specifications and/or administrative controls in accordance with EN-WM-101 and EN-WM-104. 2. Multiple components routinely taken out of service on a "train schedule" are strictly controlled by the maintenance schedule to minimize the risk impact and maintenance and/or surveillances are routinely conducted to have minimal coincident UA. 3. The potential impact on CDF for non-routine maintenance is accounted for probabilistically in the model quantification with the UA terms in the system models. 4. Plant configuration risk management procedures and corporate procedures would not permit concurrent UA for extended time periods unless the evaluated risk was of low significance, or would be for a limited time if the evaluated risk was above the low threshold.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
IF-D5a (IFEV-A6)	Flood-initiating event frequencies were determined using predominantly generic data sources. Capability Category II requires gathering plant-specific information and utilization of that information in the IE frequency determination	Insufficient use of plant specific information to satisfy the SR.	Expand collection and utilization of plant-specific information in determining the flood-initiating event frequencies, as required by the SR for Capability Category II.	**Resolved – No Impact** A review of the PNPS Condition Reports relevant to flood, rupture and break was performed following the peer review. The results do not bring into question the validity of using the generic pipe rupture data from EPRI 1013141 Rev 1 dated March 2006.
IF-E5 (IFQU-A5)	There is no justification for the operator response assumption (10 minutes to isolate a flood).	The operator response assumption is used consistently throughout the flooding scenarios. In some cases this may be conservative and others it may be non-conservative. The application of HRA methodologies would provide a more realistic impact from this action. A sensitivity was performed using 30 minutes to determine the impact of this HEP.	Perform an HRA evaluation for this action.	**Resolved – No Significant Impact** The internal flooding analysis has been revised to eliminate any dependence on prompt operator action to mitigate flooding effects. All credited operator actions have been subjected to a HRA evaluation using input from interviews with Operations personnel.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application

Assoc. SR	Finding Description	Basis for Significance	Peer Review Team Suggested Resolution	Reason
IF-E5a (IFQU-A6)	Re-evaluate HFEs that are used in internal flooding scenarios in accordance with the requirements of SR IF-E5a.	SR IF-E5a provides direction of changes in existing HFE analysis to account for the impact of the internal flooding. Consideration of the PSFs associated with flooding events could have significant impact on the likelihood of failure to perform an action.	Development of additional HRA evaluations for operator actions credited in internal flooding scenarios.	**Resolved – No Impact** The process for quantifying operator actions in response to internal events is provided in Appendix H, "Human Reliability Analysis". A review of the internal flooding analysis was performed. Local operator actions that could be impeded by the flood event were not credited Actions performed from within the control room that are credited for mitigation of internal flood initiators are not unique to internal flood initiators). The quantification of those actions is considered to be similar to that for non-internal flood initiators. In other words, the performance shaping factors (e.g., stress, cues, timing) for control room actions are not considered to be materially different for internal flood initiators.
QU-F6	DOCUMENT the quantitative definition used for significant basic event, significant cutset, significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative. The model quantification was documented in PNPS PSA Section 3. Table 3-1 PNPS Top 95 Percent Internal Core Damage Accident Sequences lists significant accident sequence. Section 3.3 DOMINANT SEQUENCES provides the top cutset for each of the top 10 accident sequences. No definition provided for significant basic event or significant cutset.	The ASME Standard calls for the definition of significant basic event, significant cutset, and significant accident sequence.	Include definitions and table of each in analysis.	**Resolved – No Impact** This is a documentation enhancement issue. The quantitative definition used for significant basic event, significant cutset, and significant accident sequence is now specified in Entergy procedure EN-DC-151 (PSA Maintenance and Update).

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
LE-C6	The Level 2 HFEs are not analyzed consistent with the Level 1 HFEs. The approach to Level 2 quantification ignores dependency.	The lack of a dependency analysis that includes the Level 2 actions is non-conservative. The HFEs are to be treated in a manner similar to the treatment of the Level 1 HFEs.	Evaluate the Level 2 HFEs similar to the Level 1 HFEs and link the fault trees.	**Resolved – No Impact** Re-confirmation that the PNPS Level 2 analysis does not need to model HRA dependencies has been performed. This conclusion was based on the following: 1) The PNPS LERF analysis includes only a few operator actions and there are no combinations of post core damage operator actions in the Level 2 cutsets. 2) There is no significant dependency between Level 2 operator actions and Level 1 operator actions, since: • The TSC is manned prior to the onset of substantial core damage • The TSC personnel are expected to make recommendations on severe accident management strategies to control room personnel based on Severe Accident Management Guidelines. • The time lag between the Level 2 operator actions (as compared to similar Level 1 operator actions) and the additional opportunities for human intervention and recovery would reduce any dependencies to such a degree that their influence is deemed to be inconsequential.

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
LE-C7	System dependencies are not treated consistently with the applicable requirements of 4.5.2 in that containment systems are quantified offline and then treated as a probability in the LERF model rather than linking the fault trees and carrying the dependencies forward.	Cutsets that include support system failures are expected to be underestimated using this approach because the probability for failure of the support system portion of the LERF contribution does not recognize that the support system has failed.	Develop "extended event trees" that include the containment systems and link those systems using CAFTA such that the dependencies are considered across the containment systems as well as the Level 1 systems.	**Resolved – No Impact** The LERF model was revised to include a direct linking of the Level 1 model to LERF model via the development of extended event trees and fault trees. The linking of the Level 1 and LERF models more adequately address dependencies across the containment systems as well as the Level 1 systems.
LE-E1	LERF analysis must be performed consistent with the requirements of the Level 1 analysis for the corresponding Capability Category. The LERF analysis was not performed in consistent with Level 1 in several areas described in the SR assessments. This F&O is written generally to be applicable to several SRs related to this topic.	The Level 2 analysis was not performed in a manner that is consistent with the level of detail and level of plant specificity and level of realism corresponding with that required for Cat II for corresponding elements of the Level 1 analysis,	Modify the Level 2 analysis to be consistent with the modeling detail, specificity and realism warranting Cat II.	**Resolved – No Impact** As noted above, the current LERF analysis includes a direct linking of the Level 1 and LERF models. Therefore, the LERF analysis bins directly connects the Level 1 accident sequence and system logic to the LERF and is considered to be adequate for the this application.
LE-F1b	REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.). A review was performed as documented in the results Section J1.10.3.5 and Section 4.9.9.4, but no discussion about the reasonableness of acceptability of the results could be found.	Evidence is that review was performed, however no conclusions about reasonableness were recorded as required by the SR.	Present the conclusions of the review and provide a statement about the reasonableness of the results.	**Resolved – No Impact** This is a documentation enhancement issue. As noted in the finding, the LERF contributors for reasonableness are identified and summarized in Appendix J1; (specifically throughout Appendix J1 and in Appendix K0 were review comments and resolutions are documented).

Table 2 - PNPS Peer Review Findings Requiring Resolution for This Application				
Assoc. SR	**Finding Description**	**Basis for Significance**	**Peer Review Team Suggested Resolution**	**Reason**
LE-G6	Definition of "significance" for LERF is not provided	The definition of significance is used in the results analysis and presentation; it is a documentation issue and does not impact the results of the model. However, it is require to be documented to satisfy this SR.	Adopt the RG 1.200 definition of "significance" or justify another definition in the documentation.	**Resolved – No Impact** This is a documentation enhancement issue. The quantitative definition used for significant basic event, significant cutset, and significant accident sequence is now specified in Entergy procedure EN-DC-151 (PSA Maintenance and Update).

Attachment 2 to Letter 2.13.030

Licensee-Identified Commitments

LICENSEE-IDENTIFIED COMMITMENTS

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The request for alternative pertaining to the use of Code Case N-578 (PRR-10) will be withdrawn for use at PNP upon NRC approval of this request for alternative.	X		90 Days after NRC approval of this request for alternative
Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program.	X		90 Days after NRC approval of this request for alternative