March 19, 2013

NRC 2013-0021
10 CFR 50.55a

U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Request for Approval of Risk-Informed/Safety Based
Inservice Inspection Alternative for Class 1 And 2 Piping
In Accordance With 10 CFR 50.55a(a)(3)(i)

In accordance with 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3)(i), NextEra Energy Point Beach, LLC (NextEra) requests that the Nuclear Regulatory Commission (NRC) grant relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI, 2007 Edition with Addenda through 2008 from the requirements of IWB-2200 IWB-2420, IWB-2430, and IWB-2500, which provide the examination requirements for Category B-F and Category B-J welds. Similarly, relief is requested from the requirements IWC-2200, IWC-2420, IWC-2430, and IWC-2500, which provide the examination requirements for Category C-F-1 and C-F-2 welds. Relief is requested on the basis that alternative methods will provide an acceptable level of quality and safety.

Specifically, NextEra proposes to use a risk-informed/safety-based inservice inspection (RIS_B) process as an alternate to the current ISI program for Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI, Division 1.

Code Case N-716 is founded, in large part, on the risk-informed inservice inspection (RI-ISI) process described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, Revised Risk-Informed Inservice Inspection Evaluation Procedure, December 1999 (ML013470102) which was previously reviewed and approved by the NRC.

In general, a risk-informed program replaces the number and locations of nondestructive examination (NDE) inspections based on ASME Code, Section XI requirements with the number and locations of these inspections based on the risk-informed guidelines. These processes result in a program consistent with the concept that, by focusing inspections on the most safety-significant welds, the number of inspections can be reduced while at the same time maintaining protection of public health and safety.

NextEra requests approval of this request prior to January 31, 2014.

Summary of Commitments

This submittal contains no new commitments or revisions to existing commitments.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW
Mr. Mike Verhagan, Department of Commerce, State of Wisconsin

ENCLOSURE

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR APPROVAL OF RISK-INFORMED/SAFETY BASED
INSERVICE INSPECTION ALTERNATIVE FOR CLASS 1 AND 2 PIPING
IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i)

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1. INTRODUCTION

Point Beach Nuclear Plant (PBNP) Units 1 and 2 have entered the Fifth inservice inspection (ISI) Interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code. PBNP plans to implement a risk-informed/safety-based inservice inspection (RIS_B) program in the Fifth ISI Interval. The Fifth ISI Interval began in August 2012.

The ASME Section XI Code of record for the Fifth ISI Interval is the 2007 Edition through the 2008 Addenda for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components.

The RIS_B process used in this submittal is based upon ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI, Division 1, which is founded in large part on the risk-informed inservice inspection (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 for Inservice Inspection of Piping*. Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Risk Assessment (PRA) Quality

NextEra Energy Point Beach, LLC (NextEra) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the probabilistic risk assessment (PRA) models for all operating NextEra Energy nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the PBNP PRA.

<u>PRA Maintenance and Update</u>

The NextEra risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plant. This process is defined in the PBNP PRA Guideline for Model Maintenance and Update. This procedure also delineates the responsibilities and guidelines for updating the full power internal events PRA models at PBNP. This procedure also defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- Existing calculations, as credited in the Model, are reviewed for their impact.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every three to four years.

In addition to these activities, NextEra risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for NextEra Energy nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately four-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. NextEra performed a regularly scheduled update to the Unit 1 and Unit 2 PRA model in December 2011 and March 2013.

<u>PRA Self Assessment and Peer Review</u>

Several assessments of technical capability have been made, and continue to be planned, for the PBNP PRA models. These assessments are as follows:

- In November 2010, a full scope Peer Review was performed by the PWROG against the available versions of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.*

- In August 2011, a focused scope Peer Review of the Internal Flooding Supporting Requirements by independent contractors was reviewed against the available versions of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2.

- In October, 2011, a Peer Review of the findings and suggestions from the November 2010 full scope peer review, except Internal Flooding SRs, by independent contractors, was reviewed against the available versions of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2.

A Full Scope PRA Peer Review for the PBNP PRA model was completed in November 2010. This peer review was performed against the available version of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2 and followed the Follow-On PRA Peer Review process. This peer review included an assessment of the PRA model maintenance and update process. This peer review defined a list of 71 findings for which potential gaps to Capability Category II of the Standard were identified.

A focused peer review of the updated internal flooding study (IF) was conducted in August 2011. This peer review was performed against the available version of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2 and followed the Follow-On PRA Peer Review process. Six of the original 13 findings were not resolved and two new ones were identified during the focused peer review. Attachment A contains a summary of these eight findings, including the status of the resolution for each finding and the potential impact of each finding on this application.

Another peer review of the updated PRA excluding IF was conducted in October 2011. This peer review was performed against the available version of the ASME PRA Standard and Regulatory Guide 1.200, Revision 2 and followed the Follow-On PRA Peer Review process. Twenty-Seven of the original findings were not resolved and 4 new ones were identified during the focused peer review. Attachment A contains a summary of these 31 findings, including the status of the resolution for each finding and the potential impact of each finding on this application.

The PRA model was further updated resulting in the PBNP PRA model (Revision 5.01, March 2013) that was used in this submittal. In updating the PRA, changes were made to the PRA to address most of the remaining findings. Following the update, an assessment concluded that 36 of the findings were fully resolved (i.e., are no longer gaps), and another 3 were not resolved. No additional gaps were identified during the performance of the review relative to the updated requirements in Addendum B of the ASME PRA Standard and criteria in RG 1.200, Revision 2, including the NRC position stating Appendix A and other NRC-issued clarifications after the 2011 gap analysis had been performed. A summary of the current open items including the partially resolved items is provided in Attachment 1.

The remaining gaps will be reviewed for consideration during the future model updates, but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in a database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

General Conclusion Regarding PRA Capability

Based on the full scope Peer Review and the subsequent two peer reviews, the PBNP PRA is considered RG 1.200, Revision 2, compliant for Internal Events. In addition, the PBNP PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection

In the risk-informed inservice inspection (RI-ISI) program at PBNP, the EPRI RI-ISI methodology is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking and delta risk evaluation steps. The importance of PRA consequence results, and therefore the necessary scope of PRA technical capability, is tempered by two processes in the EPRI methodology.

First, PRA consequence results are binned into one of three conditional core damage probability (CCDP) and conditional large early release probability (CLERP) ranges before any welds are chosen for RI-ISI inspection. Table 2 illustrates the binning process.

Table 2 – Consequence Results Binning Groups		
Consequence Category	**CCDP Range**	**CLERP Range**
High	CCDP > 1E-04	CLERP > 1E-05
Medium	1E-06 < CCDP < 1E-04	1E-07 < CLERP < 1E-05
Low	CCDP < 1E-06	CLERP < 1E-07

The risk importance of a weld is therefore not tied directly to a specific PRA result. Instead, it depends only on the range in which the PRA result falls. The wide binning provided in the methodology generally reduces the significance of specific PRA results.

Secondly, the influence of specific PRA consequence results is further reduced by the joint consideration of the weld failure potential via a non-PRA-dependent damage mechanism assessment. The results of the consequence assessment and the damage mechanism assessment are combined to determine the risk ranking of each pipe segment (and ultimately each element) according to the EPRI Risk Matrix. The Risk Matrix, which equally takes both assessments into consideration, is reproduced below.

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	**CONSEQUENCES OF PIPE RUPTURE** IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	[illegible]	[illegible]
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	[illegible]
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

These facets of the methodology reduce the influence of specific PRA results on the final list of candidate welds.

The limited use of specific PRA results in the RI-ISI process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174. Section 2.2.6 of

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

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

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

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

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

The EPRI methodology further provides an alternate means to estimate the pipe rupture consequence, namely lookup tables. By using lookup tables, PRA analysis is not involved, and the impact of the loss of systems or trains is done in a generic (not plant-specific) fashion. This allowable alternative underscores the relatively low dependence of the process on specific PRA capabilities.

In addition to the above, it is noted that welds are not eliminated from the ISI program on the basis of risk information. The risk significance of a weld may fall from Medium Risk Ranking to Low Risk Ranking, resulting in it not being a candidate for inspection. However, it remains in the program, and if, in the future, the assessment of its ranking changes (either by damage mechanism or PRA risk) then it can again become a candidate for inspection. If a weld is determined, outside the PRA evaluation, to be susceptible to either flow-accelerated corrosion (FAC), inter-granular stress corrosion cracking (IGSCC) or microbiological induced cracking (MIC) in the absence of any other damage mechanism, then it moves into an "augmented" program where it is monitored for those special damage mechanisms. That occurs no matter what the Risk Ranking of the weld is determined to be.

Conclusion Regarding PRA Capability for Risk-Informed ISI

The PBNP Unit 1 and Unit 2 PRA models continue to be suitable for use in the RI-ISI application. This conclusion is based on:

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

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

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

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

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

2.2 Augmented Programs

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

- A plant augmented inspection program has been implemented in response to NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*. This program was updated in response to MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines.* The thermal fatigue concern addressed was explicitly considered in the application of the RIS_B process and is subsumed by the RIS_B Program.
- The plant augmented inspection program for flow accelerated corrosion (FAC) per GL 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RIS_B Program.
- Since the issuance of the NRC safety evaluation for EPRI TR 112657, Rev. B-A, several instances of primary water stress corrosion cracking (PWSCC) of unmitigated Alloy 82/182 welds has occurred at pressurized water reactors. For PBNP, the only Alloy 82/182 Category B-F dissimilar metal welds (greater than NPS 1) are the four Unit 2 steam generator hot leg and cold leg primary nozzle to safe-end welds. However, these welds were factory clad with 52/152 material which is considered to be resistant to PWSCC.

PBNP intends to manage these welds per the requirements of Code Case N-770-1 as outlined in 10 CFR 50.55a. The examination frequency for these four welds is currently based on the frequencies established by these requirements. The RIS_B Program will not be used to eliminate any MRP-139 or Regulatory requirements.

3. RISK-INFORMED/SAFETY-BASED ISI PROCESS

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

- Safety Significance Determination (see Section 3.1)
- Failure Potential Assessment (see Section 3.2)
- Element and NDE Selection (see Section 3.3)
- Risk Impact Assessment (see Section 3.4)
- Implementation Program (see Section 3.5)
- Feedback Loop (see Section 3.6)

Each of these six steps is discussed below.

3.1 Safety Significance Determination

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

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

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

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

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

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

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

(5) Any piping segment whose contribution to Core Damage Frequency (CDF) is greater than 1E-6 [and per NRC feedback on the Grand Gulf and D. C. Cook RIS_B applications 1E-07 for Large Early Release Frequency (LERF)] based upon a plant-specific PSA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping. Service water piping in the cable spreading room was identified as HSS due to CDF and LERF exceeding the above criteria. This piping is shared between both PBNP units.

3.2 Failure Potential Assessment

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

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

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

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

   AND

   - $\Delta T > 50^{\circ}F$,

   AND

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

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

- **Turbulent Penetration TASCS**

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

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

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

- **Low flow TASCS**

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

- **Valve leakage TASCS**

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

- **Convection Heating TASCS**

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

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

3.3 Element and NDE Selection

Code Case N-716 and lessons learned from the Grand Gulf and DC Cook RIS_B applications provided criteria for identifying the number and location of required examinations. Ten percent of the high safety significance (HSS) welds shall be selected for examination as follows:

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements (for PBNP, because there are limited inside first isolation valve (IFIV) welds present in the RH and SI systems due to the fact that most branch lines are classified as RC out to the first isolation valve, the overall IFIV two-thirds requirement must be satisfied by selecting RC system welds in lieu of normal system-specific selections.):

   (a) A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.

   (b) If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.

   (c) If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the RCPB welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the IFIV (i.e., isolation valve closest to the RPV) and the RPV (for PBNP, because there are limited IFIV welds present in the RH and SI systems due to the fact that most branch lines are classified as RC out to the first isolation valve, the overall IFIV two-thirds requirement must be satisfied by selecting RC system welds in lieu of normal system-specific selections.).

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

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

In contrast to a number of traditional RI-ISI program applications, where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% of the HSS welds be chosen. A brief summary of the number of welds and the number selected is provided below, and the results of the selections are presented in Table 3.3. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations. Only those RIS_B inspection locations that receive a volumetric examination are included.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		All Piping Welds(3)	
	Total	Selected	Total	Selected	Total	Selected
1	754	77	1143	8	1897(4)	85(4)
2	618	63	1246	10	1864	73

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, 79 are HSS at Unit 1 and 86 are HSS at Unit 2; the remaining are LSS.

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

(4) Two Class 3 service water piping welds in the cable spreading room are defined as HSS and are included in the RIS_B Program. Of these 2 welds, 1 was selected for inspection. This piping is shared between both PBNP units.

3.3.1 Current Examinations

For the fourth interval PBNP was using the NRC approved application using EPRI-TR 112657B-A.

3.3.2 Successive Examinations

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

3.3.3 Scope Expansion

If the nature and type of the flaw is service-induced, then welds subject to the same type of postulated degradation mechanism will be selected and examined per Section 6 of Code Case N-716. The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include HSS elements up to a number equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations need be performed if there are no additional elements identified as being susceptible to the same root cause conditions. The need for extensive root cause analysis beyond that required for the IWB-3600 analytical evaluation will be dependent on practical considerations (i.e., the practicality of performing additional NDE or removing the flaw for further evaluation during the outage).

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

3.3.4 Program Relief Requests

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

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

3.4 Risk Impact Assessment

The RIS_B Program development has been conducted in accordance with Regulatory Guide 1.174 and the requirements of Code Case N-716, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements. This evaluation categorized segments as high safety significant or low safety significant in accordance with Code Case N-716, and then determined what inspection changes were proposed for each system. The changes included changing the number and location of inspections, and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

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

For LSS welds, Conditional Core Damage Probability (CCDP)/Conditional Large Early Release Probability (CLERP) values of 1E-04/1E-05 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. Based on this review there is no Class 2 piping with a CCDP/CLERP that exceeds these values.

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.

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

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

CCDP and CLERP Values Based on Break Location					
Break Location Designation	**Estimated**		**Consequence Rank**	**Upper Bound**	
	CCDP	**CLERP**		**CCDP**	**CLERP**
LOCA	2.5E-02	2.50E-03	**HIGH**	2.5E-02	2.5E-03
RCPB pipe breaks that result in a loss of coolant accident - The highest CCDP for Large LOCA, INIT-A, was used (0.1 margin used for CLERP). Unisolable RCPB piping of all sizes.					
PLOCA[1][2]	9.0E-05	9.0E-06	**MEDIUM**	1.0E-04	1.0E-05
Isolable or Potential LOCA (1 open valve or 1 closed valve) inside containment - RCPB pipe breaks that result in an isolable or potential LOCA - Calculated based on Large LOCA CCDP of 3E-2 and valve fail to close probability of ~3E-3 (0.1 margin used for CLERP). Between 1st and 2nd isolation valve inside containment.					
PPLOCA[1]	<1E-5	<1E-06	**MEDIUM**	1.0E-04	1.0E-05
Potential LOCA (2 closed valves) inside containment – Based on failure of two normally closed valves in series from the ISLOCA analysis. Applies to RHR shutdown cooling suction and discharge paths. Although the CCDP is less than 1E-6, 1E-5 is used as a bounding value in consideration of RHR operation during shutdown.					
FB	1.0E-05	1.0E-06	**MEDIUM**	1.0E-04	1.0E-05
Feedwater breaks based on bounding valve for INIT-FBIC and INIT-FBOC (0.1 margin used for CLERP)					
Class 2 LSS	1.0E-04	1.0E-05	**MEDIUM**	1.0E-04	1.0E-05
Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant. Estimated based on upper bound for Medium Consequence.					

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

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

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

Table 3.4 presents a summary of the RIS_B Program versus the third ISI interval (1986 Edition of ASME Section XI) program requirements on a "per system" basis. The presence of FAC was adjusted for in the quantitative analysis by excluding its impact on the failure potential rank. The exclusion of the impact of FAC on the failure potential rank and therefore in the determination of the change-in-risk, was performed because FAC is a damage mechanism managed by a separate, independent plant augmented inspection program. The RIS_B Program credits and relies upon this plant augmented inspection program to manage this damage mechanism. The plant FAC program will continue to determine where and when examinations shall be performed. Hence, since the number of FAC examination locations remains the same "before" and "after" (the implementation of the RIS_B program) and no delta exists, there is no need to include the impact of FAC in the performance of the risk impact analysis.

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

PBNP Unit 1

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CV - Chemical Volume & Control	-1.12E-10	-1.12E-11	-6.17E-11	-6.17E-12
FW - Feedwater	-2.60E-12	-2.60E-13	1.40E-12	1.40E-13
RC - Reactor Coolant	-6.61E-08	-6.61E-09	-8.13E-09	-8.13E-10
RH - Residual Heat Removal	6.74E-10	6.74E-11	6.74E-10	6.74E-11
SI - Safety Injection	3.19E-10	3.19E-11	3.19E-10	3.19E-11
AF - Auxiliary Feedwater	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS - Main Steam	1.40E-10	1.40E-11	1.40E-10	1.40E-11
Total	**-6.51E-08**	**-6.51E-09**	**-7.05E-09**	**-7.05E-10**

PBNP Unit 2

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
CV - Chemical Volume & Control	-8.10E-11	-8.10E-12	-4.50E-11	-4.50E-12
FW - Feedwater	-7.00E-13	-7.00E-14	2.50E-12	2.50E-13
RC - Reactor Coolant	-4.81E-08	-4.81E-09	1.88E-09	1.88E-10
RH - Residual Heat Removal	7.24E-10	7.24E-11	7.24E-10	7.24E-11
SI - Safety Injection	3.28E-10	3.28E-11	3.28E-10	3.28E-11
AF - Auxiliary Feedwater	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS - Main Steam	1.30E-10	1.30E-11	1.30E-10	1.30E-11
Total	**-4.70E-08**	**-4.70E-09**	**3.01E-09**	**3.01E-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 in Piping*, 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. PBNP identified Class 3 service water piping in the cable spreading room as HSS. This piping is shared between both PBNP units.

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 fifth ISI interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

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

3.6 Feedback (Monitoring)

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

In addition, more frequent adjustment may be required as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

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

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

For preservice examinations, PBNP 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

PBNP is currently in the First Period of the Fifth ISI Interval.

In anticipation of the approval of this RIS_B submittal, selected welds that are being examined during the 1st Period, using the traditional ASME Section XI methodology, also meet the examination requirements of Table 1 of Code Case N-716. After approval of the RIS_B submittal, those welds in the RIS_B scope that were examined during the first period that also met Table 1 requirements may be credited toward the RIS_B requirements for the 1st Period.

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

A comparison between the RIS_B Program and the 1989 Edition of Section XI program requirements for in-scope piping is provided in Table 4. In addition, service water piping in the cable spreading room was identified as high safety significant and is included in the RIS_B Program. Ten percent of the welds will be inspected during the interval. No degradation mechanism was identified for this piping, but a wall thickness type of volumetric exam will be conducted since this is considered most relevant to service water systems.

5. PRECEDENTS

1. NRC letter to Southern Nuclear Operating Company, Inc, dated January 18, 2012, Joseph M. Farley Nuclear Plant, Units 1 and 2 – Risk–Informed Safety-Based Inservice Inspection Alternative for Class 1 and Class 2 Piping Welds (TAC Nos. ME5273 and ME5274), (ML12012A135)
2. NRC letter to Southern Nuclear Operating Company, Inc, dated March 3, 2010, Vogtle Electric Generating Plant, Units 1 and 2 – Risk–Informed Safety-Based Inservice Inspection Alternative for Class 1 and Class 2 Piping Welds (TAC Nos. ME1097 and ME1098), (ML100610470)

3. NRC letter to Indiana Michigan Power Company, dated September 28, 2007, Donald C. Cook Nuclear Plant, Units 1 and 2 – Risk–Informed Safety-Based Inservice Inspection Alternative for Class 1 and Class 2 Piping Welds (TAC Nos. MD3137 and MD3138), (ML072620553)
4. NRC letter to Entergy Operations, Inc, dated September 21, 2007, Grand Gulf Nuclear Station Unit 1 – Request for Alternative GG-ISI-002 – Implement Risk–Informed Inservice Inspection Program Based on American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Code Case N-716 (TAC No. MD3044), (ML072430005)
5. NRC letter to Entergy Operations, Inc, dated April 28, 2008, Waterford Steam Electric, Unit 3 - Request for Alternative W3-ISI-005, Request to Use ASME Code Case N-716 (TAC No. MD7061), (ML080980120)
6. NRC letter to Dominion Nuclear Connecticut, Inc, dated March 27, 2012, Millstone Power Station, Unit No. 2 – Issuance of Relief Request RR-04-11 Regarding Risk-Informed Inservice Inspection Program (TAC No. ME5962), (ML120800433)

6. REFERENCES/DOCUMENTATION

1. EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs.*
2. EPRI TR-112657, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A.
3. ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1.*
4. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.*
5. Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping.*
6. Regulatory Guide 1.200, Rev 2 *An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities.*
7. 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.
8. 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.
9. EPRI Report 1021467 Nondestructive Evaluation: *Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs.*

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

System [1]	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6 [2]	High	Low
CV	215	✓					✓	
FW	50			✓			✓	
RC	353	✓					✓	
	13	✓	✓				✓	
RH	19	✓	✓				✓	
	29		✓				✓	
	271							✓
SI	113	✓					✓	
	41	✓	✓				✓	
	481							✓
AF	226							✓
MS	86							✓
Summary Results for all Systems	681	✓					✓	
	73	✓	✓				✓	
	29		✓				✓	
	50			✓			✓	
	1064							✓
TOTAL	1897							

Notes:

1. System Scope:

   AF = Auxiliary Feedwater
   CV = Chemical Volume and Control System
   FW = Main Feedwater
   MS = Main Steam
   RC = Reactor Coolant
   RH = Residual Heat Removal
   SI = Safety Injection

2. Two service water piping welds in the cable spreading room are included in the HSS scope due to exceeding the CDF > 1E-6 threshold. This piping is shared between both PBNP units.

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

System [(1)]	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6 [(2)]	High	Low
CV	154	✓					✓	
FW	54			✓			✓	
RC	314	✓					✓	
	14	✓	✓				✓	
RH	9	✓	✓				✓	
	32		✓				✓	
	315							✓
SI	89	✓					✓	
	38	✓	✓				✓	
	485							✓
AF	264							✓
MS	96							✓
Summary Results for all Systems	557	✓					✓	
	61	✓	✓				✓	
	32		✓				✓	
	54			✓			✓	
	1160							✓
TOTAL	1864							

Notes:

1. System Scope:

   AF = Auxiliary Feedwater
   CV = Chemical Volume and Control System
   FW = Main Feedwater
   MS = Main Steam
   RC = Reactor Coolant
   RH = Residual Heat Removal
   SI = Safety Injection

2. Two service water piping welds in the cable spreading room are included in the HSS scope due to exceeding the CDF > 1E-6 threshold. This piping is shared between both PBNP units.

Table 3.2
Failure Potential Assessment Summary

	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
System(1)(2)	**TASCS**	**TT**	**IGSCC**	**TGSCC**	**ECSCC**	**PWSCC**	**MIC**	**PIT**	**CC**	**E-C**	**FAC**
CV		✓									
FW	✓										
RC	✓	✓									
RH											
SI		✓	✓								
AF											
MS											

Notes:
1. Systems are described in Table 3.1.
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the AF and MS in its entirety, as well as portions of the RH and SI systems. A degradation mechanism assessment was also performed of the high safety significant service water piping welds in the cable spreading room; no mechanisms were identified.

Table 3.3a: Unit 1 Code Case N716 Selections

System(1)	Weld Count(2)		N716 Selection Considerations					Selections(2)
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CV	27		TT	✓				7
	16		None	✓	✓			7
	172		None	✓				0
FW	11		TASCS					3
	39		None					2
RC	34		TT	✓	✓			14
	10		TT,TASCS	✓	✓			4
	4		TASCS	✓	✓			3
	312		None	✓	✓			37
	6		None	✓				0
RH	19		None	✓				0
	29		None					3
		271	Assumed None					0
SI	1		TT,IGSCC	✓				1
	7		IGSCC	✓				2
	146		None	✓				2
		481	Assumed None					0
AF		226	Assumed None					0
MS		86	Assumed None					0
Summary Results All Systems	27		TT	✓				7
	34		TT	✓	✓			14
	10		TT,TASCS	✓	✓			4
	4		TASCS	✓	✓			3
	11		TASCS					3
	1		TT,IGSCC	✓				1
	7		IGSCC	✓				2
	328		None	✓	✓			44
	343		None	✓				2
	68		None					5
Totals	**833**	**1064**						**85**

Notes:

1. Systems are described in Table 3.1.
2. Two service water piping welds in the cable spreading room are included in the HSS scope due to exceeding the CDF > 1E-6 threshold. One weld will be selected for inspection. This piping is shared between both PBNP units.

Table 3.3b: Unit 2 Code Case N716 Selections

System(1)	Weld Count(2)		N716 Selection Considerations					Selections(2)
	HSS	LSS	DMs	RCPB	RCPB (IFIV)	RCPB (OC)	BER	
CV	20		TT	✓				5
	18		None	✓	✓			4
	116		None	✓				0
FW	8		TASCS					2
	46		None					4
RC	32		TT	✓	✓			12
	10		TT,TASCS	✓	✓			3
	4		TASCS	✓	✓			2
	276		None	✓	✓			33
	6		None	✓				0
RH	9		None	✓				0
	32		None					4
		315	Assumed None					0
SI	1		TT,IGSCC	✓				1
	5		IGSCC	✓				2
	121		None	✓				1
		485	Assumed None					0
AF		264	Assumed None					0
MS		96	Assumed None					0
Summary Results All Systems	20		TT	✓				5
	32		TT	✓	✓			12
	10		TT,TASCS	✓	✓			3
	4		TASCS	✓	✓			2
	8		TASCS					2
	1		TT,IGSCC	✓				1
	5		IGSCC	✓				2
	294		None	✓	✓			37
	252		None	✓				1
	78		None					8
Totals	**704**	**1160**						**73**

Notes:

1. Systems are described in Table 3.1.
2. Two service water piping welds in the cable spreading room are included in the HSS scope due to exceeding the CDF > 1E-6 threshold. One weld will be selected for inspection. This piping is shared between both PBNP units.

Table 3.4a Unit 1 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
CV	High	PLOCA	TT	Medium	0	7	7	-1.26E-10	-7.00E-11	-1.26E-11	-7.00E-12
CV	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CV	High	PLOCA	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
CV Total								**-1.25E-10**	**-6.85E-11**	**-1.25E-11**	**-6.85E-12**
FW	High	FB	TASCS	Medium	4	3	-1	-3.00E-11	1.00E-11	-3.00E-12	1.00E-12
FW	High	FB	None	Low	10	2	-8	4.00E-12	4.00E-12	4.00E-13	4.00E-13
FW Total								**-2.60E-11**	**1.40E-11**	**-2.60E-12**	**1.40E-12**
RC	High	LOCA	TT	Medium	9	14	5	-4.95E-08	-1.25E-08	-4.95E-09	-1.25E-09
RC	High	LOCA	TT,TASCS	Medium	4	4	0	-1.20E-08	0.00E+00	-1.20E-09	0.00E+00
RC	High	LOCA	TASCS	Medium	3	3	0	-9.00E-09	0.00E+00	-9.00E-10	0.00E+00
RC	High	LOCA	None	Low	68	33	-35	4.38E-09	4.38E-09	4.38E-10	4.38E-10
RC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC Total								**-6.61E-08**	**-8.13E-09**	**-6.61E-09**	**-8.13E-10**
RH	High	PLOCA	None	Low	8	0	-8	4.00E-12	4.00E-12	4.00E-13	4.00E-13
RH	High	PPLOCA	None	Low	13	3	-10	5.00E-12	5.00E-12	5.00E-13	5.00E-13
RH	Low	Class 2 LSS		Assume Medium	67	0	-67	6.70E-10	6.70E-10	6.70E-11	6.70E-11
RH Total								**6.79E-10**	**6.79E-10**	**6.79E-11**	**6.79E-11**
SI	High	PLOCA	TT, IGSCC	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	PLOCA	IGSCC	Medium	4	2	-2	2.00E-11	2.00E-11	2.00E-12	2.00E-12
SI	High	PLOCA	None	Low	24	0	-24	1.20E-11	1.20E-11	1.20E-12	1.20E-12
SI	Low	Class 2 LSS		Assume Medium	29	0	-29	2.90E-10	2.90E-10	2.90E-11	2.90E-11
SI Total								**3.22E-10**	**3.22E-10**	**3.22E-11**	**3.22E-11**
AF Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
MS Total	Low	Class 2 LSS		Assume Medium	14	0	-14	**1.40E-10**	**1.40E-10**	**1.40E-11**	**1.40E-11**
Grand Total					**261**	**72**	**-189**	**-6.51E-08**	**-7.04E-09**	**-6.51E-09**	**-7.04E-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 3.4b Unit 2 Risk Impact Analysis Results

System	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
CV	High	PLOCA	TT	Medium	0	5	5	-9.00E-11	-5.00E-11	-9.00E-12	-5.00E-12
CV	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CV	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CV Total								**-9.00E-11**	**-5.00E-11**	**-9.00E-12**	**-5.00E-12**
FW	High	FB	TASCS	Medium	4	2	-2	-1.20E-11	2.00E-11	-1.20E-12	2.00E-12
FW	High	FB	None	Low	14	4	-10	5.00E-12	5.00E-12	5.00E-13	5.00E-13
FW Total								**-7.00E-12**	**2.50E-11**	**-7.00E-13**	**2.50E-12**
RC	High	LOCA	TT	Medium	6	12	6	-4.50E-08	-1.50E-08	-4.50E-09	-1.50E-09
RC	High	LOCA	TT,TASCS	Medium	6	3	-3	-4.50E-09	7.50E-09	-4.50E-10	7.50E-10
RC	High	LOCA	TASCS	Medium	4	2	-2	-3.00E-09	5.00E-09	-3.00E-10	5.00E-10
RC	High	LOCA	None	Low	64	29	-35	4.38E-09	4.38E-09	4.38E-10	4.38E-10
RC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC Total								**-4.81E-08**	**1.88E-09**	**-4.81E-09**	**1.88E-10**
RH	High	PLOCA	None	Low	8	0	-8	4.00E-12	4.00E-12	4.00E-13	4.00E-13
RH	High	PPLOCA	None	Low	13	4	-9	4.50E-12	4.50E-12	4.50E-13	4.50E-13
RH	Low	Class 2 LSS		Assume Medium	72	0	-72	7.20E-10	7.20E-10	7.20E-11	7.20E-11
RH Total								**7.29E-10**	**7.29E-10**	**7.29E-11**	**7.29E-11**
SI	High	PLOCA	TT, IGSCC	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	PLOCA	IGSCC	Medium	2	2	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	PLOCA	None	Low	39	0	-39	1.95E-11	1.95E-11	1.95E-12	1.95E-12
SI	Low	Class 2 LSS		Assume Medium	31	0	-31	3.10E-10	3.10E-10	3.10E-11	3.10E-11
SI Total								**3.30E-10**	**3.30E-10**	**3.30E-11**	**3.30E-11**
AF Total	Low	Class 2 LSS		Assume Medium	0	0	0	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
MS Total	Low	Class 2 LSS		Assume Medium	13	0	-13	**1.30E-10**	**1.30E-10**	**1.30E-11**	**1.30E-11**
Grand Total					**277**	**64**	**-213**	**-4.70E-08**	**3.04E-09**	**-4.70E-09**	**3.04E-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 4a: Unit 1 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
CV	✓		PLOCA	TT	Medium	B-J	27	0	20	7	NA
CV	✓		LOCA	None	Low	B-J	16	0	11	0	7
CV	✓		PLOCA	None	Low	B-J	172	3	117	0	NA
FW	✓		FB	TASCS	Medium	C-F-2	11	4	0	3	NA
FW	✓		FB	None	Low	C-F-2	39	10	0	2	NA
RC	✓		LOCA	TT	Medium	B-F, B-J	34	9	18	14	NA
RC	✓		LOCA	TT,TASCS	Medium	B-J	10	4	0	4	NA
RC	✓		LOCA	TASCS	Medium	B-J	4	3	0	3	NA
RC	✓		LOCA	None	Low	B-F, B-J	312	68	169	33	4
RC	✓		PLOCA	None	Low	B-J	6	0	0	0	NA
RH	✓		PLOCA	None	Low	B-J	19	8	0	0	NA
RH	✓		PPLOCA	None	Low	C-F-1	29	13	0	3	NA
RH		✓	Class 2 LSS		Assume Medium	C-F-1	271	67	0	0	NA
SI	✓		PLOCA	TT, IGSCC	Medium	B-J	1	1	0	1	NA
SI	✓		PLOCA	IGSCC	Medium	B-J	7	4	0	2	NA
SI	✓		PLOCA	None	Low	B-J	146	24	70	0	2
SI		✓	Class 2 LSS		Assume Medium	C-F-1	481	29	23	0	NA
AF		✓	Class 2 LSS		Assume Medium	N/A	226	0	0	0	NA
MS		✓	Class 2 LSS		Assume Medium	C-F-2	86	14	2	0	NA
						Total	**1897**	**261**	**430**	**72**	**13**

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 PBNP 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").

Table 4b: Unit 2 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
CV	✓		PLOCA	TT	Medium	B-J	20	0	13	5	NA
CV	✓		LOCA	None	Low	B-J	18	0	75	0	4
CV	✓		PLOCA	None	Low	B-J	116	0	10	0	NA
FW	✓		FB	TASCS	Medium	C-F-2	8	4	0	2	NA
FW	✓		FB	None	Low	C-F-2	46	14	0	4	NA
RC	✓		LOCA	TT	Medium	B-F, B-J	32	6	6	12	NA
RC	✓		LOCA	TT,TASCS	Medium	B-J	10	6	0	3	NA
RC	✓		LOCA	TASCS	Medium	B-J	4	4	0	2	NA
RC	✓		LOCA	None	Low	B-F, B-J	276	64	114	29	4
RC	✓		PLOCA	None	Low	B-J	6	0	1	0	NA
RH	✓		PLOCA	None	Low	B-J	9	8	0	0	NA
RH	✓		PPLOCA	None	Low	C-F-1	32	13	0	4	NA
RH		✓	Class 2 LSS		Assume Medium	C-F-1	315	72	0	0	NA
SI	✓		PLOCA	TT, IGSCC	Medium	B-J	1	1	0	1	NA
SI	✓		PLOCA	IGSCC	Medium	B-J	5	2	0	2	NA
SI	✓		PLOCA	None	Low	B-J	121	39	43	0	1
SI		✓	Class 2 LSS		Assume Medium	C-F-1	485	31	19	0	NA
AF		✓	Class 2 LSS		Assume Medium	N/A	264	0	0	0	NA
MS		✓	Class 2 LSS		Assume Medium	C-F-2	96	13	2	0	NA
						Total	**1864**	**277**	**283**	**64**	**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 PBNP 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").

ATTACHMENT A
TO ENCLOSURE

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

PROBABILISTIC RISK ASSESSMENT
QUALITY REVIEW

Attachment 1: Point Beach PRA Peer Review Findings				
SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
IE-A1	Not Met Finding IE-A1-01	IE-A5 (CC-I) IE-B2 (Not Met) IE-D2 (Not Met)	**2010 Peer Review Finding:** A systematic process for identifying initiating events was not performed. Table 4 (Plant System Review to Determine Special Initiators) in the initiating events report provides an identification of System IE that impact mitigation equipment but does not fully address the impact of loss any normally operating system that could results in an IE. For example loss of the 4.16 kV AC would lead to an IE due to the loss of component cooling water, loss of instrument air, and loss of CVCS; however, there is no quantitative estimate as a basis for screening out this initiating event. Loss of HVAC in the Electrical Equipment Room HVAC could result in a reactor trip but no documentation or room heat up calculations are provided to support that loss of the system would not generate a trip. Without a systematic review that accounts for plant-specific features an initiating event can be missed. A systematic review should be performed and documented on all normally operating systems. Provide a quantitative basis for screening out the loss of 4.16kV AC bus as an initiator. A recommendation would be that this review would include documentation of possible failure modes and effect on safety system(s) for each system. This is also an ideal location to document possible dual unit impacts.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Table 4, Special initiator column was expanded to include tech spec shutdowns and improved to include explanation of why special initiator is not required and / or how the system is subsumed by another initiating event. The following text was added to Initiating Events Notebook, PRA 2.0, Section 1.3.5, "Special Initiators for PBNP": "A review of Table 2-1 in DBD-27, "ACCIDENT ANALYSIS REACTOR TRIP VARIABLES, LIMITS, and RESPONSE TIMES" was performed if required to determine if the event described in Table 4, column 3, "Description of Event", would cause a direct or indirect reactor trip. The results of this review are captured in "Special Initiating Event?" column of Table 4." New special initiators were identified as part of this review. 4160 VAC Safeguards buses 1A05 and 1A06 on Unit 1, 2A05 and 2A06 on Unit 2. The Unit is required to shut down if one of the safeguards buses cannot be restored within 6 hours. CAFTA runs were performed to determine the impact of these initiators. A flag file was used to set all initiators to false except Initiating Event Transient with PCS which had the probability set to the 1 year failure probability of a 4160 VAC Vital Switchgear Bus. The flag file also set the 4160 VAC Vital Switchgear bus to failed. The results were the CDF due to a failed 4160 VAC Vital Switchgear bus

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2010 Peer Review Finding Response: The systematic process was described in Section 1.2 which has been revised to also provide a list of steps in addition to the descriptive text. The loss of a single 4.16 kV AC bus does not result in a unit trip. This has happened at Point Beach and the unit did not trip. This is not an initiating event. Since this is based on actual plant historical events no quantitative estimate is needed. Loss of HVAC was evaluated in PRA Notebook 05.25. The evaluation for some critical areas was revised and for some areas fault tree models were developed to evaluate the impact of the loss of HVAC. These calculations provide a quantitative basis that these HVAC systems do not contribute and need not be modeled. A systematic review of the plant-specific features was performed in Initiating Events Notebook 2.00, Section 1.3.4, "Review of PBNP Design." A systematic review of all normally operating systems was performed. This is documented in Sections 1.3.4, 1.3.5, and Table 4. Additional documentation is provided in each system notebook in Section 05.xx.4, "Initiating Events Review" and Section 05.xx.8, "Failure Modes and Effects Analysis." A quantitative basis for the loss of a 4.16kV AC bus is not needed. The plant has lost a 4.16 kV AC bus and the unit did not trip. Therefore this is not a potential initiating event. Dual unit impacts are discussed in the Success Criteria Notebook, Section 4.1, "Dual Unit Success Criteria."	initiator was between 1.9E-7 and 1.2E-9. LERF was between 3.9E-10 and 9.4E-12. These are not significant contributions.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2011 Peer Review Finding: Section 1.2, 1.3.4, 1.3.5 and Table 4 provide evidence of a systematic review and addresses loss of 4kv and HVAC. Also, the system notebooks address the potential for initiating event (e.g., 05.25 HVAC). However, based on this review, weaknesses still remain and this finding could not be completed closed. The documentation suggests that an immediate plant trip is required for an equipment failure to be considered an initiating event. However, tech spec shutdowns should also be considered (e.g., <24 hr LCO unlikely equipment failure could be fixed within tech spec). The evaluation must be expanded to include this. For example the plant experienced a 4KV failure that did not cause a trip but resulted in plant shutdown due to tech specs. Was there a tech spec requirement to shutdown. Table 4 should be improved to explain why a special initiator is not required and or how the system is subsumed in another initiating event (Section 1 should have most of the basis along with SY?)	
IE-B2	**Not Met** **Finding IE-B2-01**	**IE-A1-01 (Not Met)**	**2010 Peer Review Finding:** The requirement for this element is to use a structured, systematic process for grouping initiating events. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA). There is no discussion of the use of a structured, systematic process for grouping initiating events.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Column 5, "Special Initiating Event", of Table 4 in the Initiating Event Notebook, 2.0 has been expanded to document the review of the systems and the basis for Special IE exclusion.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			Ensure a structured, systematic process for grouping initiating events was used, and document the process. **2010 Peer Review Finding Response:** Several F&Os have resulted in changes in the IE Notebook. Section 1.2 now explicitly presents the structured, systematic methodology used in the development of the initiating events. Step 3 of this process it the grouping of identified initiating events. Section 2.2 has been revised to better document the systematic process as per the below. Section 2.3 has been added to present the plant operator interview comments. **2011 Peer Review Finding:** Sections 1.2 and 2.2 were revised to better explain the structured approach; however, as described for IE-A1-01 the documentation of the review of all systems in Table 4 and basis for IE exclusion is required.	
IE-D3	**Not Met** **Finding IE-D3-01**		**2010 Peer Review Finding:** No documentation of sources of uncertainty for initiating events could be found in the IE document. The Standard requires this documentation. Add section discussing sources of uncertainty in the Initiating Events calculation.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Section 5.0 of the PRA Notebook 11.0, Quantification Notebook contains a discussion on the sources of uncertainty and their impact to the PRA.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2010 Peer Review Finding Response: Sources of Uncertainty for this and all other PBNP PRA Notebooks are evaluated in PBNP PRA 11.00 Quantification Notebook. A new Section 5 was added to the IE Notebook to state that Sources of Uncertainty are evaluated in PRA 11.00, the Quantification Notebook. **2011 Peer Review Finding:** Section 1.3.6 of IE Notebook identifies assumptions, which are a key source of uncertainty and Section 5 of IE Notebook references QU for uncertainty, but there is no updated QU Notebook	
AS-B1	**Not Met** **Finding AS-B1-01**		**2010 Peer Review Finding:** This element states that for each modeled initiating event, identify mitigating systems impacted by the occurrence of the initiator and the extent of the impact. Include the impact of initiating events on mitigating systems in the accident progression either in the accident sequence models or in the system models. Currently two separate models are being maintained for PBNP - one for Unit 1 and one for Unit 2. By maintaining two separate models, the full impact of dual unit initiating events, and the importance of failures of shared equipment is not adequately addressed. The dual unit impact of shared systems, especially under dual unit initiating events, is very important from a risk perspective, and will become even more important when the PRA is converted to a Fire PRA model. A single top PRA model that reflects both Units would explicitly address the dual unit impacts and the importance of system/equipment failures.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** There are no longer any front line systems shared between units. Auxiliary feedwater is now unit specific. Startup Steam Generator pumps which were the motor driven AFW pumps are shared between units. CSTs are shared but levels are maintained to accommodate an accident on one unit and hot shutdown on the other unit. 13.8 KV - Designed for normal loads on one unit which are greater than accident loads on both units. 4160 VAC - Designed for normal loads on one unit which are greater than accident loads on both units. 480 VAC - Unit Specific EDG - Can supply both units with a single diesel 120 VAC - Unit Specific 125 VDC - The batteries are designed for accident on one unit and hot standby on other unit. Accumulators - Unit Specific SI RHR - Unit Specific AFW - Unit Specific (See note above)

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			Challenge: The two separate top models being maintained for PBNP address the full impact of dual unit initiators. **Response:** From the team review during the week, and our discussions with the PRA people, it did not appear that dual unit initiators were being addressed appropriately. Given the events at Fukashima, this is now of particular concern. In particular, in a dual Unit initiating event – the equipment on the opposite Unit will most likely NOT be available to respond to the initiating event – it will be dedicated to its Unit until its Unit is placed in a safe, stable state, or until it is shown that it is not needed for its Unit. There was no evidence that this was taken into consideration at all in the individual models. Depending upon the Dual Unit Initiating Event, cross-tied/shared systems will most likely have their cross-connect valves closed to isolate the two Units from each other – and will require an operator action to re-open the valves if the system is allowed to be re-cross connected - this consideration was not seen in the individual models. There is also no evidence that the HFEs associated with an event are modified for a dual-unit initiator when Operators will be at a premium, and their availability to respond to outside the control room actions will be impacted. **2010 Peer Review Plant Response:** The two separate top models being maintained for PBNP address the full impact of dual unit initiating events. The PBNP PRA currently uses 2 separate top gates, one for Unit 1 and one for Unit 2. This has been the case historically and the 2 models	Containment Spray - Unit Specific Fan Coolers - Unit Specific Containment Isolation - Unit Specific Service Water System - Success criteria of two pumps can support accident on one unit and hot shutdown on other unit. Component Cooling Water - Unit Specific Actuation Systems - Unit Specific Main Feedwater - Unit Specific Main Steam - Unit Specific Reactor Coolant System - Unit Specific CVCS - Unit Specific Fire Protection - Accident loads on both units are less than fire protection loads. Used to make up CSTs and cool TDAFWP bearings. Instrument/Service Air - Accident loads on both units are less than normal operating loads. Fuel Oil - Designed to support EDG with accident on one unit and hot shutdown on other unit. This should clarify that dependencies between unit shared systems have the capability as required to cope with dual unit shutdowns.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			are maintained in parallel. The standard does not require a single top model for a multiple unit site. While there are some shared systems between the 2 PBNP units (electrical, service water) and some additional systems that have limited cross-connect capabilities (auxiliary feedwater, component cooling water, instrument air, station air), identical fault tree logic is used in both models for these systems and the commonalities and impacts are properly accounted for in the logic models. For example, the AFW system model considers the availability of AFW flow to the opposite unit to determine what pumps can be considered for the unit in question. Some additional gates were added to the model to better reflect dual unit impacts. The following changes were made to the model in response to F&O AS-B1-01, the F&O related to a single model for both units: - Under existing gate GAFM2500, add new AND gate GAFM2501 with two new inputs. One input is new OR gate GAFM2502 and the other input is new OR gate GAFM2503. New OR gate GAFM2503 has as inputs existing initiating events INIT-T1G, INIT-T1GB, INIT-TIP, INIT-T1W, INIT-TD1, INIT-TD2, INIT-TIA, INIT-TSW. - Under gate GAFW1800, add new AND gate GAFW1801. Under new AND gate 1801, add OR gate GAFM2503.	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			- Under gate GAFM2900, add new AND gate GAFM2901. Under new AND gate add OR gate GAFM2503. The Technical Specifications were reviewed to assure that the impact of the status of the opposite unit is correctly modeled and it was determined that there is no impact from unit status. The modeling of the common systems and systems with cross-tie capability described above was reviewed and the modeling correctly captures the dependencies. There is no requirement for a single top event for a multiple unit site. Additionally, none of the duel unit site models that the PRA team is familiar with have a single top model for multiple units. If a single top model were produced it would still be solved at the individual unit level. It is not clear what the meaning would be of solving for simultaneous core damage. **2011 Peer Review Finding:** It does not appear that the Standard requires a single top and the models for each unit appears correct for quantifying risk of each unit with shared equipment. Also, Point Beach can supply both units with a single diesel. What is not clear is whether all dependencies between unit shared systems have this capability and the importance of any that cannot supply both units. Given recent events and the potential importance of dual unit events, this finding will have to remain open.	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
AS-B3	**Not Met** **Finding AS-B3-01**	**SY-B6 (Met)** **SY-B14 (Not Met)**	**2010 Peer Review Finding:** This element is associated identifying and modeling the effects of the phenomenological conditions created by the accident progression. Phenomenological impacts include: generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration. The effects of phenomenological conditions created by the accident progression of Main Steam Line Breaks or Feed Line Breaks outside containment are not adequately addressed. In particular, the analysis states that MSBL's outside containment "result in no adverse Containment atmosphere" or "there are no adverse environmental conditions" from the event. Because of the accident sequence itself - there will be a steam environment in the vicinity of the break, but this adverse environment is not addressed. Because of the potential impact on non-qualified equipment and Operator actions in areas outside of containment that can be subject to the effects of MSLBs outside containment, the potential adverse conditions need to be identified and their impact of equipment and actions in the areas need to be addressed. Additionally, no discussion on debris generated in Containment due to LOCAs or MSLBs inside containment can be found. Evaluate the phenomenological conditions created by the accident progression and include the impacts of any adverse conditions in the fault tree	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Accident Sequence Notebook 3.2, Section 6.5.1 was revised to include discussion of HELB and the impacts on the Aux Bldg and Turbine Bldg.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			model and documentation. Need to evaluate potential steam environments outside containment, Main Feed breaks outside Containment, HELB issues, debris generation inside containment, potential NPSH impacts, etc. **2010 Peer Review Plant Response:** Section 5.4.6 and Table 5.4.5 of the AS Notebook were enhanced with the following: "Large LOCAs may also create an environment (i.e., pressure, temperature, humidity, debris generation) that could impact equipment. This is addressed in the Success Criteria Notebook (Reference 8.1)." Section 5.5.6 and Table 5.5.4 of the AS Notebook were enhanced with the following: "Events inside containment may create an environment (i.e., pressure, temperature, humidity, debris generation) that could impact equipment. This is addressed in the Success Criteria Notebook (Reference 8.1). For events outside containment, collateral damage is explicitly included in the model. For secondary line break events in the turbine building, a loss of all MFW and Instrument Air is assumed. Breaks in the Aux building also impact equipment. For these cases, only qualified equipment or equipment not directly."	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			Section 3.4 of the SC Notebook was enhanced (and Reference 4 was added in Section 9) with the following: "In addition, Section 1.4 and Section 9 (assumption 6) of the SI/RHR System Notebook (Reference 4) address the issue of debris in containment, concluding that debris has no impact on containment sump recirculation." The above enhancements address the concerns of the F&O regarding the environmental impacts of Large LOCAs (Small and Medium LOCAs produce much less force; see responses to GSI-191) and Secondary Line Breaks. **2011 Peer Review Finding:** Sections 5.4.6, 5.5.6 and tables 5.4.5, 5.5.4 of AS Notebook were revised to address this F&O, but there was insufficient information to address HELB outside containment. Plant response refers to SC Notebook Section 3.4 which does not contain anything relevant. The details of HELB (e.g., FW and MS) and the impacts in the Aux Bldg and Turbine Bldg are not described. The RHR system notebook was revised to address issue of containment sump debris and plugging of SI injection path flow orifices.	
AS-B6	**Not Met** **Finding AS-B6-01**	**SY-A5 (Met)** **SY-A21 (Not Met)** **SY-B6 (Met)** **SY-B15 (Met)**	**2010 Peer Review Finding:** This element is associated with ensuring that plant configurations and maintenance practices which create dependencies among various system alignments are defined and modeled in a manner that reflects these dependencies, either in the accident sequence models or in the system models.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Peer Review Plant Response:** The HFE of failure of the operators to properly manage EDG loads was not modeled due to the extremely low probability of the opportunity for this error to result in a loss of the EDG ever occurring. In order for this HFE to be

Attachment 1: Point Beach PRA Peer Review Findings				
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			Because of electrical bus limitations, Point Beach has some unique system alignment restrictions. Currently these system alignment restrictions are not reflected in the PRA model. In particular, there are system alignment restrictions associated with the System Air and CVCS systems such that specific System Air Compressors cannot be in operation if specific CVCS pumps are in operation. A review of the Normal System Operating Procedures should be performed to identify the unique PBNP system alignments and restrictions. Once the unique system alignments and restrictions are identified, the limitations should be reflected in the PRA fault tree models. **2010 Peer Review Plant Response:** The following text sections were added to the AS Notebook, Section 5.6.3, and the EDG and 4160 VAC Notebooks. As OI-35C will not be applicable after the March 2001 Unit 2 outage it is felt that this need not be added to the 480 VAC Notebook. Point Beach Electrical Loads Limitations Per Tim Lensmire, the Point Beach electrical engineer knowledgeable on electrical loads, was interviewed on 2-17-2011 by Stanley Goukas to address load management for normal alignments at Point Beach. The first is OI-35C which Tim says in theory should go away after the upcoming Unit 2 outage once the EPU modifications have been implemented. The second and third are AOP-22 for Unit 1 and Unit 2. These AOP's provide for	viable, there must be a loss of offsite power, a demand for the safety injection pumps (i.e., a LOCA), and a random failure of one of the EDGs. Furthermore, the probability of this HFE is expected to be low due to the clarity of the procedural guidance and the frequent training given to the operators on proper EDG load management. Within AOP-22, a note specifically states that EDG Loading is critical when the site is reduced to a single EDG and the EDG is required to support the equipment required for Safety Injection. A calculation is presented below, which calculates the probability of using this procedure: SI = 1E-2 Includes LOCAs and Steam/Feed line breaks since excessive cooldown will generate an SI LOOP = 3E-2 Sum of all LOOPs Gas Turbine = 1E-1 Out for Maintenance 3 EDGs = 7E-6 Common Cause Failure to Run 1st hour or CCF run 23 hours. Probability of using this procedure with only 1 EDG = 1E-2 * 3E-2 * 1E-1 * 7E-6 = 2.1E-10.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			load management on the diesel generators following a loss of offsite power. Additionally, it was noted that additional loading restrictions may be placed in effect when maintenance is performed on electrical equipment: 13.8KV or 4160 Volt Electrical Load Management When maintenance is performed on some 4160 volt transformers bus load restrictions are placed in effect: • 3.5 When removing 1X-03, Unit 1 High Voltage Station Auxiliary Transformer, or 1X-04, Unit 1 Low Voltage Station Auxiliary Transformer, from service, the following additional measures will ensure operability of offsite power from a potential degraded voltage condition during a unit trip: (Ref 6.6.8 & Attachment F) • For any unit in Mode 1 - defeat one of that units 4160V fast bus transfer, typically the A-03 to A-01 Bus Tie due to the turbine auxiliaries powered from A-02. For any unit in Modes 5, 6 or defueled - maintain that units A01 and A02 4160V motors OFF. • 3.6 When re-energizing the 1X-04 transformer the 13.8KV bus should be aligned to the 1X-03 transformer to reduce possible perturbations to opposite units online equipment. These transformers are the normal supply to the class 1E buses. As such, these transformers	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			would not be taken out of service while the plant is operating. The model assumes that no routine maintenance would be performed on these transformers during unit operation. Therefore, this load restriction has no impact on the model. OI-35C – 480 Volt Electrical Load Conservation The loads considered as discretionary are charging pumps 1P-2A and 1P-2B and instrument air compressor K-2A. To meet the loading requirement, first one of these loads is secured off (no auto-start). If this configuration does not meet the loading requirement, then 2 of these loads are secured off (no auto-start). However, these load management measures are not utilized when an AOP/EOP is in effect. The load management issues addressed in OI-35C will be resolved by modifications being made during the upcoming (March 2011) Unit 2 refueling outage. Therefore, the electrical load considerations contained in OI-35C need not be considered in the model. AOP-22 Unit 1 – EDG Load Management This procedure is applicable when the EDGs are running, loaded, and the bus being supplied is isolated. If the load on a EDG exceeds the 200 hour limit the operators isolate unnecessary plant equipment per Attachment A, Unit 1 Electrical Loads (for EDGs G-01 and G-02) or isolate non-safeguards bus 1B-40 (for EDGs G-03 and G-04) as per Step 1, Response Not Obtained.	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			If the load on a EDG exceeds the 2,000 hour limit the operators isolate additional unnecessary plant equipment per Attachment A. It should be noted that Attachment A simply defines all of the potential loads by bus and does not prioritize these loads or indicate what loads should be shed first. Since Attachment A does not prescribe what loads should be retained or shed, it is not possible to determine directly what impact this restriction has on the model. Since the PRA models the first 24 hours following a Unit trip the applicable loading limits for the PRA are the 2000 hour ratings, or 2,850 kW for G-01 and G-02 and 2,848 kW for G-03 and G-04. An analysis of the loads necessary to safely shutdown both units with a single diesel generator has been performed and it is possible with any single diesel, subject to equipment failure. These facts, and the statement that loads are started and stopped "as directed by the plant procedures" (Step 2), indicate that the necessary equipment to safely shutdown the unit(s) would be identified by the operating procedures in effect at time and operated at direction of the operators. Therefore, no additional modeling is necessary to capture any potential limitations. To ensure that excessive loads are not credited in the PRA, the minimal set of equipment necessary to safely shutdown the unit(s) for the Loss of Offsite Power and Station Blackout accident sequences were reviewed and compared against the analysis of the loads necessary to safely shutdown both units with a single diesel generator	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			mentioned above. This demonstrated that there are no additional impacts of electrical load limitations on the PRA models. AOP-22 Unit 2 – EDG Load Management This procedure is applicable when the EDGs are running, loaded, and the bus being supplied is isolated. If the load on a EDG exceeds the 200 hour limit the operators isolate unnecessary plant equipment per Attachment A (for EDGs G-01 and G-02) or isolate non-safeguards bus 1B-40 (for EDGs G-03 and G-04) as per Step 1, Response Not Obtained. If the load on a EDG exceeds the 2,000 hour limit the operators isolate additional unnecessary plant equipment per Attachment A, Unit 2 Electrical Loads. It should be noted that Attachment A simply defines all of the potential loads by bus and does not prioritize these loads or indicate what loads should be shed first. Since Attachment A does not prescribe what loads should be retained or shed, it is not possible to determine directly what impact this restriction has on the model. Since the PRA models the first 24 hours following a Unit trip the applicable loading limits for the PRA are the 2000 hour ratings, or 2,850 kW for G-01 and G-02 and 2,848 kW for G-03 and G-04. An analysis of the loads necessary to safely shutdown both units with a single diesel generator has been performed and it is possible with any single diesel, subject to equipment failure. These facts, and the statement that loads are started and stopped "as directed by the plant procedures" (Step 2), indicate that the	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			necessary equipment to safely shutdown the unit(s) would be identified by the operating procedures in effect at time and operated at direction of the operators. Therefore, no additional modeling is necessary to capture any potential limitations. To ensure that excessive loads are not credited in the PRA, the minimal set of equipment necessary to safely shutdown the unit(s) for the Loss of Offsite Power and Station Blackout accident sequences were reviewed and compared against the analysis of the loads necessary to safely shutdown both units with a single diesel generator mentioned above. This demonstrated that there are no additional impacts of electrical load limitations on the PRA models. **2011 Peer Review Finding:** Section 5.6.3 of AS Notebook and EDG & 4Kv system notebooks were revised. OI-35c will not be applicable after upcoming Unit 2 outage (480 V). AOP-22 addressed load management on EDGs. Additional restrictions on 13.8 and 4 Kv, but these maintenance alignments are not conducted at power. Appears that load management failure is a possible failure mode for EDGs that is not modeled. AOP-22 indicates load management is critical however could not find basis concluding it could be neglected as an EDG failure. Plant response indicated that HEP "HEP-416-U1-A-3-4" included Load Management within a cross-tie action. This was reviewed in HRA Calculator (HRAC) and no	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			tasks for load management were identified and no mention of load management was visible in the HRA entry. Similar AC cross-tie actions were also reviewed in HRAC and likewise no load management information was in identified. Also, the Load list in AOP-22 was summed to yield over 7000kW. Further, Calculation 2004-002 "Emergency Diesel Loading" appears to credit operator actions for various situations. For example, Page 215 has negative loads (i.e., loads that are secured) for "SW Reduction", "Turn off MDAFP", and "Turn Off SI Pump (P-15)". Thus, in theory it seems possible to overload an EDG (Load limit ~3000kW). No Modeling of Operator to manage EDG Loads found. (See SY-21-01).	
AS-B7	**Not Met** **Finding AS-B7-01**		**2010 Peer Review Finding:** This element is associated with modeling time-phased dependencies (i.e., those that change as the accident progresses, due to such factors as depletion of resources, recovery of resources, and changes in loads) in the accident sequences. Examples are: (a) For SBO/LOOP sequences, key time-phased events, such as: (1) AC power recovery (2) DC battery adequacy (time-dependent discharge) (3) Environmental conditions (e.g., room cooling) for operating equipment and the control room Although time-phased recoveries appear to be considered at PBNP, it is not clear that they are addressed appropriately and completely. For	Yes - The 2011 Peer Review Finding was NOT resolved in the PRA Model. As stated in the finding, only recovering SBO sequences and not considering battery depletion time will conservatively affect the results. **2011 Peer Review Plant Response:** In the current convolution calculation for LOSP, LOOP recovery is applied to only SBO sequences and DC battery life is not considered (i.e. Fails at 0 hours). This is conservative since recoveries which could be applied to reduce CDF and LERF are not applied. The resolution to this F&O will possibly be addressed in the next PRA Model revision (5.02).

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			example, in the HVAC notebook, there are several rooms that are expected to exceed the design limits for the equipment in them, but failure of HVAC to the rooms are not modeled. Additional justification for why HVAC to those rooms is not required needs to be addressed. **2010 Peer Review Plant Response:** The PRA model reasonably accounts for the impacts of time phased dependencies. The model has been improved in 3 areas to better reflect the impact of time phased events. First, the Power Recovery Convolution has been revised. This calculation determines the likelihood of the recovery of offsite power at the specific times that the MAAP and the RCP Seal LOCA analyses identified as being critical to the development of accident sequences. The current Convolution analyses were developed specifically for SBO (no power available from any source) and are therefore not applicable to a partial power situation such as LOOP. Additionally, the modifications to the DC modeling resolve the bulk of the cutsets in LOOP that give the appearance of being long term SBO sequences. Second, the HVAC Notebook analyses have been revised. Additional consideration was given to the available information and additional analyses were performed to quantitatively support the conclusions presented in the notebook.	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			Third, the modeling and tagging of battery depletion and the recovery rules for restoration of power to a DC bus have been revised. The previous model had a single tag to identify a depleted bus and the HEP dependencies are cued off of this tag. This resulted in the failure of a single DC bus effectively failing all DC power (a modeling error). This has been revised such that there is a unique tag for each DC bus and the cutsets in LOOP that looked like they should be in SBO (erroneous cutsets) have been modified to correctly reflect the loss of DC at a specific bus and not the loss of all DC power. **2011 Peer Review Finding:** The model was improved in 3 areas to better reflect the impact of time phased dependencies as described above. HVAC notebook was updated and model includes HVAC as appropriate. However, the Model is still conservative because LOOP recovery for non-SBO scenarios is still neglected and the basis for this is inadequate. Also, DC life is still assumed to be 1 hour when realistic battery life is much greater (DC notebook does not mention true battery life other than full load test takes 2 ½ days). In the convolution analysis, credit is not even taken for the one battery hour. As a minimum greater detail is required to document these assumptions and their impact on the results (QU). Since the 5.00 model is being reviewed the QU results will not address additional model changes being incorporated NEXTERA.	

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SC-A6	**Not Met** **Finding SC-A6-02 Doc Only**	**AS-C3 (Not Met)**	**2010 Peer Review Finding:** This SR requires that success criteria be confirmed to be consistent with features procedures and operating philosophy of the plant. Point Beach set the upper end of their small break LOCA event to 2 inches based on generic information from NUREGs. However, in the accident sequence notebook, Point Beach made a statement to the effect that over much of the range of their small break LOCA spectrum, secondary side heat removal side was not needed (see small LOCA assumptions section). However, in Section 6.2.3 of PRA 3.2, it is stated "The small LOCA event tree (Event Tree Notebook, Figure S2) applies to breaches in the RCS which are large enough that the break flow exceeds the capacity of the normal reactor makeup system. The break size, however, is not large enough to provide core decay heat removal." These two statements are inconsistent. A review of the success criteria calculations did not reveal any calculations to determine the upper end of the small LOCA spectrum based on the need for secondary side heat removal. Run a set of MAAP calculations to determine the break size that is just sufficient to remove decay heat and depressurize the primary side. Use the break size thus determined as the lower bound for the medium LOCA and upper bound for the small LOCA. Use the results to also clarify the small LOCA definition in the SC notebook, the AS notebook and the IE notebook and make them all consistent.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The response to F&O SC-A6-02 was added to PRA 3.2, Success Criteria Notebook, Section 6.2.3.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2010 Peer Review Plant Response: The Westinghouse Design Basis Analysis divides LOCAs into two sizes, large and small. The small LOCA upper end break size is 6 inches diameter. The basis for this division is that large LOCAs exhibit high break exit velocities such that mitigation flows (low pressure injection, accumulators, etc.) bypass the core and exit the break without providing core cooling. This occurs until the end of the "blowdown phase", when the break exit velocities drop such that mitigation flows reach the core to provide core cooling. As such, different design basis codes (SATAN/WREFLOOD for large LOCAs and NOTRUMP for small LOCAs) must be used for the different size LOCAs. The MAAP code cannot be used to analyze the short term timeframes of large LOCAs that produce the conditions that result in core bypass (see Section 3.4 of PRA 3.2). However, MAAP is reasonable for analyzing the longer term time frames of all LOCA sizes. The results of MAAP run PB1ML-25 (4" LOCA without AFW or cooldown and depressurization) indicate that RHR injection only just barely averts core damage. This would indicate that the large LOCA success criteria could be used down to approximately 4 inches. However, the 6" break size is a reasonable point at which to differentiate between PRA-defined large and medium LOCAs based on the core bypass characteristics described above. The differentiation point between PRA-defined medium and small LOCAs is 2 inches diameter.	

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			MAAP run PB1ML-06 shows that sufficient energy is removed via the break that secondary cooling is not required as long as high head SI is successful. However, unlike the 6" and 4" medium LOCAs, a 2" LOCA does require AFW and operator-initiated cooldown and depressurization to use RHR if high head fails (MAAP run PB1SL-17 for failure of high head recirculation, PB1SL-14 for failure of high head injection). As one can see, the requirements for AFW and cooldown and depressurization that exist for medium LOCA successful sequence 3 are set by the lower bound of the medium LOCA (2 inch diameter). One could move more of the break spectrum in the 2" to 4" range into the small LOCA realm such that there was no requirement for AFW and cooldown and depressurization for the medium LOCA. However, there would be more break spectrum in the small LOCA that would not require secondary heat removal if high head injection was available. Conversely, one could move more of the break spectrum in the 2" to 1" range into the medium LOCA realm such that there was always a requirement for secondary heat removal if high head injection was available for the small LOCA. However, there would be more break spectrum in the medium LOCA that would require AFW and cooldown and depressurization, and possibly feed and bleed for failure of AFW. In other words, there are competing conditions such that a "perfect" break point" may not be attainable.	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			What this means is that the 2 inch LOCA is a reasonable break size for the differentiation point between PRA-defined medium and small LOCAs. Therefore, no changes were made to the Point Beach break size spectrum. **2011 Peer Review Finding:** Plant response to peer review seems reasonable but has to be included in the SC (and or IE and AS) notebook to resolve this finding (important documentation based on previous peer review).	
SY-A21	**Not Met** **Finding SY-A21-01**		**2010 Peer Review Finding:** This element states that system conditions that cause a loss of desired system function, (e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.) should be identified. A review of various electrical system notebooks and the EDG system notebook did not identify any consideration of excessive electrical loads on the busses or the EDG. With the electrical margin for some of the busses and the EDGs at Point Beach being minimal to non-existent, a review for potential excessive loading conditions needs to be performed and documented. In particular, a look at Operators starting equipment in response to redundant equipment failures, and failures of equipment to fully load shed should be considered, documented, and explicitly included in the model as appropriate. Currently, with the exception of the EDG start logic, load shed and UV detection is embedded in the bus failure rates, and needs to be explicitly modeled because of the excessive loading concern.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The HFE of failure of the operators to properly manage EDG loads was not modeled due to the extremely low probability of the opportunity for this error to result in a loss of the EDG ever occurring. In order for this HFE to be viable, there must be a loss of offsite power, a demand for the safety injection pumps (i.e., a LOCA), and a random failure of one of the EDGs. Furthermore, the probability of this HFE is expected to be low due to the clarity of the procedural guidance and the frequent training given to the operators on proper EDG load management. Within AOP-22, a note specifically states that EDG Loading is critical when the site is reduced to a single EDG and the EDG is required to support the equipment required for Safety Injection. A calculation is presented below, which calculates the probability of using this procedure:

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			With the electrical margin for some of the busses and the EDGs at Point Beach being minimal to non-existent, a review for potential excessive loading conditions needs to be performed and documented. In particular, a look at Operators starting equipment in response to redundant equipment failures, and failures of equipment to fully load shed should be considered, documented, and explicitly included in the model as appropriate. Currently, with the exception of the EDG start logic, load shed and UV detection is embedded in the bus failure rates, and needs to be explicitly modeled because of the excessive loading concern. **2010 Peer Review Plant Response:** Plant Response: Excessive Electrical Loads are addressed in the response to F&O AS-B6-1. The PRA model requires ALL of the circuit breakers associated with a bus that are required to be shed (be opened) prior to closing a circuit breaker for a new power source to be aligned to the bus. This is explicitly modeled and is not imbedded in the bus failure rates as was thought by the reviewers. No model change is required to address this comment. The limitations associated with the 4.16 KV transformers are related to maintenance that would not be performed during operation, so there is not impact on modeling.	SI = 1E-2 Includes LOCAs and Steam/Feed line breaks since excessive cooldown will generate an SI LOOP = 3E-2 Sum of all LOOPs Gas Turbine = 1E-1 Out for Maintenance 3 EDGs = 7E-6 Common Cause Failure to Run 1st hour or CCF run 23 hours. Probability of using this procedure with only 1 EDG = 1E-2 * 3E-2 * 1E-1 * 7E-6 = 2.1E-10.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2011 Peer Review Finding: AOP-22 for Unit 1 and Unit 2 provide for load management on the diesel generators following a loss of offsite power. Procedure direct operators to isolate/strip unnecessary loads if the EDG load exceeds the 200/2000 hour limit. To ensure that excessive loads are not credited in the PRA, the minimal set of equipment necessary to safely shutdown the unit(s) for the Loss of Offsite Power and Station Blackout accident sequences were reviewed and compared against the analysis of the loads necessary to safely shutdown both units with a single diesel generator mentioned above. This demonstrated that there are no additional impacts of electrical load limitations on the PRA models. See also discussion in AS-B6-01.	
SY-B3	**Not Met** **Finding SY-B3-01**		**2010 Peer Review Finding:** ESTABLISH common cause failure groups by using a logical, systematic process that considers similarity in: a. Service conditions b. Environment c. Design or manufacturer d. Maintenance JUSTIFY the basis for selecting common cause component groups. Candidates for common cause failures include, for example: a. Motor-operated valves b. Pumps c. Safety-relief valves d. Air-operated valves	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Response:** Common Cause Failure of Component Cooling Water Pumps to start and run has been added to the model. The delta CDF due to this change was 4.25E-8 on Unit 1 and 4.28E-8 on Unit 2. There was no delta LERF on either Unit. Sensitivity of common cause failures for component cooling water pumps in mitigation section of model. CC--MDP-FS-1-11A = 1.51E-03/demand Beta = 2.31E-02 NRC Common Cause Database MDP FS

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			e. Solenoid-operated valves f. Check valves g. Diesel generators h. Batteries i. Inverters and battery charger j. Circuit breakers For initiating events, common cause failure groups for the "failure to run" or "failure to operate" modes that involve a normally operating component failing followed by the failure of the standby failure group use an exposure time of 24 hours. Because the exposure in which the normally operating component can fail is one year, and because CCF parameters are dimensionless, the use of 24 hours is incorrect. It should be one year. The times can be changed and then the initiating event requantified. **2010 Peer Review Plant Response:** Plant Response: Because the exposure in which the normally operating component can fail is one year, and because CCF parameters are dimensionless, the use of 24 hours is incorrect." This statement is incorrect. In all of the analyses of common cause data events to date, the definition of the parameter is "failure of identical components due to the same cause within a 24 hour period." Thus, CCF parameters are NOT dimensionless; they are a fraction of failures that occur in a 24 hour period. CCF parameters not dimensionless.	Common cause failure to start = failure rate * Beta Common cause failure to start = 1.51E-03 * 2.31E-02 = 3.49E-05/demand Type Code CC- MDP CM 11S = 3.49E-05/demand CC--MDP-FR-1-11A = 5.86E-06/hour Beta = 5.86E-02 NRC Common Cause Database CC MDP FR Common cause failure to run = failure rate * Beta Common cause failure to run = 5.86E-06 * 5.86E-02 = 3.43E-07/hour Type Code CC- MDP CM 11R = 3.43E-07/hour Results:

U1 CDF No CCW CCF	U1 CDF CCW CCF	U2 CDF No CCW CCF	U2 CDF CCW CCF
5.34E-06	5.39E-06	5.34E-06	5.38E-06
DELTA	4.25E-08	**DELTA**	4.28E-08

U1 LERF No CCW CCF	U1 LERF CCW CCF	U2 LERF No CCW CCF	U2 LERF CCW CCF
7.83E-08	7.83E-08	8.13E-08	8.13E-08
DELTA	0.00E+00	**DELTA**	0.00E+00

The change in CDF for both units was an increase of 4E-8 and there was no change in calculated LERF. Therefore,

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2011 Peer Review Finding: Plant response could be improved as it is common practice in most PRAs to decouple standby pump failures from CCF of running pumps over 8760 hours. Although it is slightly optimistic there is no data for failure of one pump over 8760 and then common cause failure of the second (given start success) to fail before repair of the first pump. This modeling approach needs to be clearly described in the system notebooks. SW modeling was improved using this approach. Note that CCW has one operating pump and one standby pump thus there is no CCF to run for an initiating event using this approach. However, there is no CCF to run or start in the CCW mitigation model, which is required.	the change has an insignificant impact on CDF and LERF for both Unit 1 and Unit 2. The changes to the fault trees are documented in the appropriate System Notebook.
HR-A3	Not Met Finding HR-A3-01		**2011 Peer Review Finding:** Pre-Initiator dependency is based on an incorrect interpretation of SY-B2 "No requirement to model intra-system common-cause" and includes judgments that are not adequately defended. The following excerpts from the HRA Notebook demonstrate this misinterpretation and do not present any compelling justification for the judgments: Per ASME SR SY-B2, there is no requirement to model intersystem common cause failure. As miscalibration of redundant channels is a common cause failure, miscalibration between different systems need not be modeled. By considering diverse input signals to an actuation signal as "different systems", screening of signals	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** RPS system was reviewed. Two groups of sensors, low pressurizer pressure and low low steam generator level were identified as not having diversity. As such, common cause mis-calibration errors were calculated and added to the Unit 1 and Unit 2 models for these groups of sensors. No other signals were identified which did not have diversity. Calculation MSE-EJJ-05-10, "Point Beach Nuclear Power Plant Mis-calibration Human Reliability Analysis", dated December 19, 2005 identified low pressurizer pressure, VCT Level and AFW pressure as common cause mis-calibration failures. The low pressurizer pressure was added above. The VCT Level and AFW pressure were added to the model as a result of this calculation. The

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			can be accomplished based on diversity. A signal channel typically consists of a transducer, transmitter, power supply and an analog-to-digital converter that converts the input from the transmitter to an on-off signal using a bistable. Calibrations are performed on the transducer/transmitter and on the bistable. Miscalibration of either the transmitter or bistable setpoints can defeat the automatic actuation signal. For redundant channels, calibration of the transmitter can be screened out from further consideration, if signals provided by the transmitters are also used for indications in the control room. For example, steam generator level has redundant channels that are monitored closely by control room personnel during normal operation. If one or more redundant channels deviate from the rest, the operators would take notice. However, the calibration of the bistables cannot be screened out as a miscalibration may only become evident when the signal is required. For signals that are simply generated by relay contacts due to loss of voltage across the relay coil, miscalibration is deemed not to be a significant contributor to signal failure. Miscalibration is therefore not to be	values used for the common cause of these sensor groups were obtained from Calculation MSE-EJJ-05-10. The calculation has been included in Appendix A – Mis-calibration.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			modeled for the signals such as loss of offsite power / 4 kV bus undervoltage. RPS should be considered a single system and screening based solely on intra-system common-cause considerations should not apply. The model includes no common-cause miscalibration or misalignment and the basis for this treatment is not adequately defended. Screening of pre-initiators also includes a notion of diversity. The concept of diversity is not adequately developed: The automatic actuation signals are screened on diversity. Two groups of signals which produce automatic actuation were identified. Those related to reactor protection system and those related to ESFAS. Tables A-1 and A-2 in Appendix A show the ESFAS signals as described in the ESFAS system notebook. The reactor protection system signals are outlined in Table 7.2-1 of the FSAR and all signals have been screened from further consideration based on redundancy and diversity. As described in Chapter 14 of the FSAR all events accidents analyzed require at least two diverse parameters.	

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SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			Screening based on diversity such as signals that are actuated based on two separate parameters (e.g., Level OR Pressure) seems reasonable but not all signals have such diversity.	
HR-D1	**Not Met** **Finding HR-D1-01**	**HR-D2 (CC-I)** **HR-D3 (CC-I)**	**2010 Peer Review Suggestion:** This element requires an estimation of the probabilities of human failure events using a systematic process. Acceptable methods include THERP [2-5] and ASEP [2-6]. Point Beach uses screening values for all of their pre-initiator human actions. In the report, the HEPs are givens as the screening value with an error factor. The screening values tend to be medians. Therefore, the values in the model, which are means, are higher than the screening values in the report because of the conversion from medians to means. However, this is not described in the report. Provide a discussion of the conversion from medians to means in the report so that the report values can be traced back to the values used in the model. **2010 Peer Review Plant Response:** This is an incorrect "Suggestion." The Peer Reviewer chose to explain the difference between the documentation and the model provided for the Peer Review as being a problem in the conversion of the value from medians to means. This is an incorrect explanation for the differences. The HRA Calculator ™ does this conversion and exports mean values to the CAFTA .rr database file. The actual cause of the differences is that the HEP	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Screening values were reset from 1E-4 to 5E-4. The model was rerun and a list of BEs showing up in the CDF cutsets at a truncation of 1E-11 was generated. The mis-positioning BEs in the list were then reviewed to see which if any were important. A mis-positioning BE was considered important if the F-V was greater than 0.005 or the RAW was greater than 2. There was one mis-positioning BE on Unit 1 which had a RAW that was slightly less than 2 and one mis-positioning BE on Unit 2 which had a RAW greater than 2. These events had a detailed ASEP analysis performed to generate a specific value for them. The value obtained was then inserted into the model and used for the quantification. Note that the mis-position event was for valve 1AF-109 on Unit 1 and 2AF-109 on Unit 2, the same mis-positioning event.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			data was changed after the QU Notebook had been completed, thus there was a discrepancy between the HR, QU, and CAFTA documentation. To incorporate these data changes into the model would have required a complete revision of the QU Notebook and there was insufficient time to do this prior to the Peer Review. The pre-initiator values listed in the HR documentation will be implemented into the CAFTA .rr database file prior to the final quantification and development of the QU Notebook. Thus, this issue is resolved. **2011 Peer Review Finding:** The use of screening values does not meet HR-D2. The values appear arbitrarily low, for screening values, and are not based on actual procedure-based assessment (i.e., a systematic process). While THERP data may have been used in creating the screening values, neither the THERP or ASEP methods are used. For example, no consideration of independent verification is presented in the analysis approach. SR HR-D2 allows screening (using the ASEP approach) for non-significant HEP. As a test, the BE importance was looked at and the very first pre-initiator randomly selected (HEP-AF--TY-1P29) has a RAW of 2.06.	
HR-G5	**CC-II** **Finding HR-G5-01**	**HR-E3 (CC-I)** **HR-E4 (CC-I)**	**2010 Peer Review Finding:** Talk throughs were performed for E-0, ECA 0.1, E-1.3 and E-1.4 and for risk significant operator actions (Section E.2). Simulator observation was provided for SGTR event that address timing for actions. Appendix E.4 provides simulator	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Screening values were reset from 1E-4 to 5E-4. The model was rerun and a list of BEs showing up in the CDF

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			observations to several procedures but only provides timing information. But after a review of the risk significant operator action in the quantification notebook and Appendix E there was limited or no information on (HEP-SW--START-IE, HEP-AF--CST-FW, HEP-AF--CST—LOW). In general the only insights documented from these interviews were to support timing. There is little or no documentation to support the evaluation of the information that impacts the cognitive, stress levels, and information that to support the THERP. For example the time window information HEP-CCW-STDBY-IE (OPS FAILS TO ALIGN STANDBY HEAT EXCHANGE (PRE REACTOR TRIP)) is based the High CCW temperature alarm the limiting time should be based on high RCP bearing temp. During an operator interview the operators would trip the reactor in a shorter time window than 15 minutes to protect the RCPs. Without this level of detail document it is difficult to reproduce results. Talk throughs and/or simulator run insights should cover information requires to support the evaluation of the information that impacts the cognitive, stress levels, and information that to support the THERP. At a minimal this level of detail should be provided for all risk significant operator actions. An example of information that would be expected to be documented and asked during an interview would, for each scenario, confirmation that the action and procedure steps are correctly performed. Document the number of people required to support the actions (impotent for	cutsets at a truncation of 1E-11 was generated. The mis-positioning BEs in the list were then reviewed to see which if any were important. A mis-positioning BE was considered important if the F-V was greater than 0.005 or the RAW was greater than 2. There was one mis-positioning BE on Unit 1 which had a RAW that was slightly less than 2 and one mis-positioning BE on Unit 2 which had a RAW greater than 2. These events had a detailed ASEP analysis performed to generate a specific value for them. The value obtained was then inserted into the model and used for the quantification. Note that the mis-position event was for valve 1AF-109 on Unit 1 and 2AF-109 on Unit 2, the same mis-positioning event. Section 4.2 of the HRA Notebook, 6.0 has been rewritten to clearly identify simulator observations and operator interviews conducted to confirm the interpretation of the procedures is consistent with observations and response models are correct for scenarios modeled. Appendix E has added Section E.5, “Additional Emails with operations and plant staff to support HRA”. Documentation has been revised to correctly list Appendix E and Appendix F as appropriate. Response to Finding from November 2010 stated that “Appendix F, Section F.2, was inadvertently truncated. This table included additional operator interview insights related to execution on additional risk significant HFEs (As of 8/4/10).” This should read ” Appendix E, Section E.2, was inadvertently truncated. This table included additional operator interview insights related to execution on additional risk significant HFEs (As of 8/4/10).”

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			local actions). Documenting information to support the information necessary to evaluation the cognitive error (clearly of the cue, front panel or back panel, etc). Document the workload during the event (Are you in multipage procedures or one, stress level, etc). Document time window estimates. If time to cue or time to undesirable state is based on T&H ask if these times are consistent with what they have seen on the simulator. **2010 Peer Review Plant Response:** The reviewer and analyst clearly agree that time required to complete actions were based on operator talk-throughs of the procedures or simulator observations. HR-G5 does not require documentation of operator interviews to support the evaluation of the information that impacts the cognitive, stress, levels, and information that to support execution analysis. However, review of Appendix F by the HRA analysts showed that Appendix F, Section F.2, was inadvertently truncated. This table included additional operator interview insights related to execution on additional risk significant HFEs (As of 8/4/10). Not included in Section F.2 were 3 risk significant HFEs related to aligning the battery charger and these HFEs were re-interviewed with operations and the insights are included in the HFE analysis. The overall HEP values were not impacted by additional operator insights.	

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SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			HEP-125-BAT-CHG OPS FAILS TO ALIGN PWR/RELOAD TO BATT CHARGER FROM CONTROL ROOM HEP-125-COG OPS FAILS TO RECOGNIZE NEED TO PWR BATT CHARGER (COMMON COG) HEP-125-COG-REC OPS FAILS TO RECOVERY BATTERY CHARGER AFTER BATTERIES DEPLETE **2011 Peer Review Finding:** Note that this original finding is related to SRs HR-E3, HR-E4, and HR-G5 (HR-E3 and HR-E4 listed as CC: I by original peer review team). HR-E3 requires interviews to confirm procedure interpretation and HR-E4 requires confirmation of response models. Plant response to this F&O does not address HR-E3 and HR-E4. More complete operators interviews appear required. Also, the documentation appears to alternately list the Interview Appendix as Appendix E and Appendix F. This editorial issue should be corrected when additional operator interview information is added.	
DA-C7	**CC-I** **Finding DA-C7-01**		**2010 Peer Review Finding:** For the Level 2 requirements, this SR states "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."	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** MSPI surveillance data was not used. The MSPI Basis Document for Point Beach was used. From **Section 1.1.6** of the MSPI Basis Document "For Point Beach, the numbers of demands for Emergency AC System are based on the actual number of demands and estimated

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			Estimations were made regarding testing frequencies (see Section 2.3.3 in Data Calculation, lastbut-one paragraph). Therefore, it meets CC I. Update the data analysis to include actual plant information instead of estimations. **2010 Peer Review Plant Response:** Plant Response: In Appendix A the component for which the number of tests are stated as "Estimated" were obtained from the system engineer during the system engineer interview (SW pumps). The system engineer described actual practice of pump testing which was erroneously stated as "Estimated." This has been re-worded to properly describe the source in the "Notes" column in Appendix A table. The process of collecting test procedure data is mentioned in Section 2.3.3 Collection of Test Data. Following editorial change in Section 2.3.3 Collection of Test Data has been implemented to address the Cat-II requirement of the standard. **2011 Peer Review Finding:** MSPI surveillance data was used, but this is still an estimation technique (not actual) and at most slightly conservative. In Appendix A it appears that actual demands may have used for non-MSPI equipment, but this is not described in main report.	run hours from surveillance tests performed." **Section 1.3.6** "For Point Beach, the numbers of demands and run hours for Auxiliary Feedwater are based on a combination of estimates based on surveillance frequencies and on actual data from data loggers on the motor driven pumps with PMT demands and run hours removed." **Section 1.4.6** "For Point Beach, the numbers of demands and run hours for Residual Heat Removal are based on a combination of estimates based on surveillance frequencies and on actual data from data loggers on the RH pumps with PMT demands removed." **Section 1.5.6** "For Point Beach, the numbers of demands and run hours for Service Water are based on a combination of estimates based on surveillance frequencies and on actual data from data loggers oil four of the six Service Water pumps with PMT demands removed." **Section 1.6.6** "For Point Beach, the numbers of demands and run hours for Component Cooling Water are based on a combination of estimates based on surveillance frequencies and on actual data obtained from Safety Monitor." **Section 1.2.6** on Safety Injection is estimated based on the number of times surveillance testing is performed and operating instructions for use. This led to 132 demands on the SI pumps used in the data analysis notebook. Subsequently, the number of SI pump demands was determined from the Safety Monitor data for the timeframe used for the data analysis notebook. There were actually 154 start demands on the SI pumps. Therefore, the data analysis notebook is conservative and no change to the model is required. To clarify that actual demand failures were used, the third paragraph of Section 2.3.3 of the Data Analysis Notebook was changed to read as follows:

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SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
				Plant surveillance data of 3 years was obtained from MSPI basis doc, Revision 14, September 30, 2009. This data was annualized for computing the test and maintenance unavailability due to surveillance procedural tests, actual planned maintenance activities and unplanned maintenance acts on actual plant experience at Point Beach. In cases where MSPI basis document demand data was not available, actual demands were determined from data logging devices installed on the equipment or from Safety Monitor. The component demand data is presented in Appendix A.
DA-C8	**CC-II/III** **Finding DA-C8-01 Doc Only**		**2010 Peer Review Finding:** Plant-specific operational records were not used for components such as SW pumps - these were lumped together. Assumed symmetry across the similar components - this meets the CC I in the Standard for DA-C8. Apply component specific data to each individual component for unavailability. **2010 Peer Review Plant Response:** Plant Response: In 6 years of the data period following are the actual observations about service water pumps: 1. There were no failures to run incidents for any of the service water pumps. 2. All the pumps were evenly swapped for running and no preferential treatment was given to any of the pumps. 3. There was only one failure to start out of the system engineer estimated 2592 starts.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The purpose of collecting data is to try to estimate future performance. It is a better estimate of future unavailability by polling the data for identical components in the same system. The data should be long term averages, so the occurrence of unavailability due to failure should average out over the long term. Therefore, long term averages will continue to be used. Component specific unavailability data will not be used.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			4. All the Plant operators logs were studied to look for coherence and absolutely no specific operational preference were given to any pump. The plant specific operational and standby status timing of each SW pump was obtained from the system engineer. Additionally, the time that components in normally operating systems was not imbedded in the PRA. Rather, the PRA was set up to be imported into the Safety Monitor and thus house events had been used to set the status running and standby equipment. In response to F&Os IE-C10-01 and SC-A6-03 specific configurations with the duration for each configuration have been added to the PRA. This incorporates the plant-specific operational records. **2011 Peer Review Finding:** Information could not be found in DA or SW System Notebooks (Doc only).	
DA-C10	**CC-I** **Finding DA-C10-01**		**2010 Peer Review Finding:** This SR states "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." For Level 2 - it also requires "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."	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Per Scientech e-mail from Lincoln Sarmanian dated 10/31/2011 @ 3:29 PM the procedures were evaluated to determine that the appropriate failure modes depending on the type of component were accounted for. The first paragraph of Section 2.3.3 was revised to read as follows (underlined text was added): **Data for the number**

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			This is met at CCI since the first part of requirement appears to be done appropriately, but it does not appear that the component failure modes are decomposed into sub-elements that are fully tested. Review the procedures down to the sub-element level and use the information to credit tests that exercise specific sub-elements. **2010 Peer Review Plant Response:** Plant Response: The Summary Assessment is not true. The review of test procedures was performed. The results of this review are shown in Appendix A of the Data Notebook. As per Capability Category III, Table-7 of the data notebook lists the actual hours of unavailability for the components as per the decomposition of the component failure mode into sub-elements that are fully tested and use tests that exercise specific sub-elements in their evaluation. In the calculation of unavailability hours for the data period it was ensured that double counting of unavailability is avoided. **2011 Peer Review Finding:** Section 2.3,3 describes the process for evaluating test data obtained via surveillance procedures. It does not state that it evaluates procedure to determine if test can be credited for all possible failure modes of the component.	**of demands based on surveillance testing was generated through the demands stated in procedures and requirements (performance per demand) during various plant states is tabulated in Appendix A. The number of times a test procedure was required was recorded for the key components. The procedure was then reviewed to determine the number of demands on the component for each test run. <u>The appropriate failure modes for the type of component were accounted for in the procedure review.</u> In addition, key components which were not being tested but did receive demands were also recorded.**

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
DA-C14	Not Met Finding DA-C14-01		**2010 Peer Review Finding:** The System Notebook Guidance notebook states that a specific review for possible activities which can cause the simultaneous unavailability of redundant equipment is documented in the Data Notebook. No discussion of such a review was found in the Data Notebook. Although the team confirmed that concurrent planned maintenance on redundant equipment is not allowed per plant philosophy, this is not addressed anywhere in the PRA. Because of this, T&M event combinations are showing up in dominant cutsets that are in reality not allowed, and should have been eliminated as mutually exclusive events. Add in a discussion of the plant philosophy that does not allow concurrent planned maintenance on redundant equipment - including redundant equipment in the opposite unit. Once this is complete, a review of ALL T&M events in the PRA should be performed to determine which ones are precluded from being planned concurrently, and these combinations should be added into the system notebooks and the fault tree model as mutually exclusive events. **2010 Peer Review Plant Response:** Plant Response: All T&M events contained in the models were reviewed, along with the plant Technical Specifications. Additional combinations of T&M events were added to the MEX portion of the CAFTA fault tree.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Reviewed Real Time Safety Monitor from December 11, 2009 through December 11, 2010 for Unit 1 and Unit 2. Only concurrent maintenance found on a regular basis was battery and associated battery charger. Whenever battery D05 was OOS, the associated battery charger D07 was also OOS. When battery D06 was OOS, battery charger D08 was also OOS. When battery D105 was OOS, battery charger D107 was also OOS. When battery D106 was OOS, battery charger D108 was also OOS. The converse is not true. When a battery charger was out of service, the associated battery was aligned to the spare battery charger. Since the maintenance for the battery and associated battery charger is concurrent, they are not independent events, but the same event. To account for this the same T&M event was used for the battery and associated battery charger. This change did not affect the CDF or LERF on either unit. Section 3.3 of the Data Analysis Notebook was updated to describe the review that was performed, the findings and the effect on the model.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2011 Peer Review Finding: Finding was that a discussion of concurrent maintenance was not found in the DA Notebook.	
DA-D1	**CC-II** **Finding DA-D1-01**	**DA-D3**	**2010 Peer Review Finding:** CHOOSE prior distributions as either non informative, or representative of variability in industry data. CALCULATE parameter estimates for the remaining events by using generic industry data. The issue is how the posterior distribution is calculated. The data notebook states that the generic priors are taken from NUREG/CR-6928. Those distributions are either beta distributions or gamma distributions depending on whether the failure mode is demands or time related. The parameters of the distributions are real numbers. The means for the posterior distribution are calculated in a manner consistent with the expressions presented on pages 14 and 15 of the PBNB notebook. Several cases were tested (AF-MDP, FR.>=1Hour and FS) and using the expressions on pages 14 and 15. The means calculated were higher than that presented in Table 5. Hence the updated distributions may be optimistic. NUREG/CR-6823 Handbook of Parameter Estimation for Probabilistic Risk Assessment discusses Bayesian updating of beta and gamma functions. Recalculate the posteriors using the information from that NUREG as guidance. Note the posterior parameters are easily calculated as is the mean. The percentile can be calculated from EXCEL or equivalent.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Table 5 and Table 6 were reviewed against NUREG/CR-6928 with changes made as appropriate. Were also checked against .rr files for Unit 1 and Unit 2. .rr files were updated to be consistent with revised Table 5 and Table 6. FW normal operating pump revised in Table 5 and .rr files for both units.

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SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2010 Peer Review Plant Response: Plant Response: This F&O is incorrect and should NOT be a Finding or a Suggestion. As guided by the Cap Cat III realistic parameter estimates based on relevant generic and plant-specific evidence was calculated. The issue is the use of generic data parameters taken from NUREG/CR-6928. The NUREG presents the resulting generic data as both gamma distributions and Mean and ER and does not provide guidance as to how to use these or which set to use. The principal author stated that he had not considered using the results as we did that this is not incorrect. After several lengthy discussions both in-house and with the principal Author of NUREG/CR-6928, the consensus was that using the mean and error factor to generate the parameters for the prior distribution was correct and that using the provided gamma functions would yield incorrect results. This is artifact of the limits imposed in developing the gamma distributions presented in the NUREG. Thus, the approach used at Point Beach is correct. After discussions and review with FPL staff and performing the Bayesian updates in the CAFTA software package and obtaining the same results, it was decided to continue with the first approach, which is correct. That said, the differences between the 2 methods were evaluated. A new Table-5 was developed on	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			the basis of the reviewer's recommendation and new insights observed for certain components like fire water pump. Bayesian update was used for specific characterization of the uncertainty. Prior distribution (characteristic parameters: alpha and beta) was obtained from NUREG/CR-6928 and posterior was calculated. The process has been stated in detail in section 3.1.1 Hardware Failure Rates. The parameter estimates for the remaining events were calculated by using generic industry data from Table-5 of NUREG/CR-6928. The results of the 2 methods were compared. The gamma approach essentially has smaller tails in the prior distribution and as such, the mean value of an update can be influenced to be higher with less uncertainty than the correct approach. **2011 Peer Review Finding:** Based on PBU1.rr dated October 2010, it appears that priors taken from NUREG/CR-6928 (MDP STBY FTS) incorrectly assigned to FW normal operating pump (MDP RNNING FTS). Also, value in Table 5 for FW-MDP FS is inconsistent with PBU1.rr. Agree that using NUREG/CR data distribution (e.g., beta) mean and EF as input to a lognormal for Bayesian updating has a minor effect. Suggestion, ensure that the 5.01 model, including Table 5 are reviewed for consistency and the correct prior.	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
DA-D4	CC-1 Finding DA-D4-01		**2010 Peer Review Finding:** When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. The team did not find evidence that the posterior distribution was checked to determine if it is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. For a discussion of what is intended in the standard refer to NUREG/CR-6823 Handbook of Parameter Estimation for Probabilistic Risk Assessment. From the ASME Standard: Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application includethe following: (a) Confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) Examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) Examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) Confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Additional text added to Appendix C, Section 1.0. New text states: "Table 5 provides inputs where the prior was updated with plant specific data for the posterior. Upon completion of the update process a reasonableness check is performed. Each posterior distribution is reviewed against the prior distribution and the weight of the plant specific evidence to ensure that the result of the update is reasonable. The generic value, plant specific value and updated value were considered on a case by case basis. For those cases where the generic and plant specific data were close, the posterior was reviewed to ensure it was close. Where the generic and plant specific were different, the posterior was reviewed to ensure this was reflected. The balance of the data applied the generic prior as the posterior. Since the generic was applied, the posterior is reasonable and appropriate relative to the generic prior."

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			(e) Confirmation of the reasonableness of the posterior distribution mean value **2010 Peer Review Plant Response:** Plant Response: Additional text added to Appendix C, Section 1.0. New text states: "Upon completion of the update process a reasonableness check is performed. Each posterior distribution is reviewed against the prior distribution and the weight of the plant specific evidence to ensure that the result of the update is reasonable." **2011 Peer Review Finding:** This information is not contained in Appendix C or Section 1.0. Also there has to be a discussion about the comparison not just a statement that you did one.	
QU-D1	**Not Met** **Finding QU-D1-01**	**QU-D2 (Not Met)** **QU-D3 (Not Met)** **QU-D5 (Not Met)**	**2010 Peer Review Finding:** This SR requires a review of a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct. The cutset review presented in the Quantification notebook is not adequate. Reviews describe the sequence that the cutset represents or list the failed equipment, but do not describe how the specific component failures in the cutset lead to the end state defined by the sequence. Since the equipment failures were not analyzed, significant cutsets exist that do not appear to make physical sense and do not reflect the as built, as operated plant.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** At the time of the Peer Review for the Internal Events PRA, the Quantification Notebook was in Draft. Based on the final version of the Quantification Notebook, the tables, cutset descriptions, and cutsets were updated.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			A specific example of a suspect cutset is cutset #4. The cutset is either invalid or represents a design deficiency of the plant. The cutset indicates failure of a single air handling unit while the non-safety related gas turbine generator is in maintenance leads directly to core damage. Review indicates that the cutset may be invalid or overly conservative due to a convolution of items: • HRA recovery rules are assuming a HRA failure since the power supplies for the cues to the event are not explicitly modeled. This is likely over conservative and skewing the results of the quantification. • The cutset may be due to the assumed alignment of the service water pumps, which are assumed to be in the most restrictive alignment for this type of event. This alignment assumption is most likely over conservative and skewing the model quantification results. A further example of a suspect cutset is cutset #12501, which includes simultaneous planned maintenance on both turbine driven AFW pumps. This condition would not be entered during plant operation, making the cutset invalid. Both of these cutsets were reviewed and determined valid in the quantification notebook. Cutsets need to be reviewed to ensure the results make sense and reflect the as built, as operated plant. Cutsets need to be adequately described to facilitate understanding of the PRA.	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2010 Peer Review Plant Response: No response provided. **2011 Peer Review Finding:** No Plant Response. The QU notebook is set up properly to address SR-D1, D2, D3 and D5, however, this notebook including critical tables are not update and competed yet. Based on review of new cutsets, old cutest #4 now requires failure of 2 AHUs versus 1, however, description of this cutset in Table 3.3-1 in previous QU Notebooks appears erroneous and inconsistent with actual cutset. Old cutset 12501 (TIA-005_SEQ) is lower in frequency, but procedurally the plant is currently allowed to have both TDPs unavailable for maintenance. Thus, this modeling is appropriate and the frequency of this cutset is less than 1E-7. Table 3.2-1 provides as discussion of significant event tree sequences and a discussion of the underlying logic (QU-D2 and D3). Table 3.3-5 provides a review of non significant cutsets (QU-D5 and D3)	
QU-D4	**CC-I** **Finding QU-D4-01**		**2010 Peer Review Finding:** This SR requires the PRA to compare its results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another? While the CDF results and initiating event contributions from several plants are compared to the results from the Point Beach PRA, there is no discussion of the causes for significant differences in those results. A discussion of the reasons for	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** On January 10, 2012 PRA analysts from Point Beach, Prairie Island, Kewaunee and Ginna participated in a conference call/meeting to discuss the differences in the PRA results. The insights provided by this discussion have been added to Section 5.4 of the Quantification Notebook, 11.0. Differences now included are batteries, service water header arrangement, safety injection pumps

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			the differences is necessary to meet Category II/III. Provide a discussion of the reasons for significant differences in plant results. **2010 Peer Review Plant Response:** No response provided. **2011 Peer Review Finding:** No Plant response. Section 5.4 and Tables 5.4-1 and 2 provides a high level comparison, however the description of differences in results should be enhanced (the only difference cited is the 1 hour battery life assumed for Point Beach). This is a limited description that requires more detail. For example, an explanation of why loss of 4Kv is 0.0 at Point Beach and not so at other plants.	and power uprate.
QU-D7	**Not Met** **Finding QU-D7-01**		**2010 Peer Review Finding:** This SR requires a review of the importance of components and basic events to determine that they make logical sense. No review of components or basic event importances was documented in the PRA documentation. Perform and document a review of the significant components/basic events. **2010 Peer Review Plant Response:** No response provided.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** At the time of the Peer Review for the Internal Events PRA, the Quantification Notebook was in Draft. Based on the final version of the Quantification Notebook, the importance measures were updated and reviewed. See resolution below. Reviews of risk significant basic events are discussed in section 4.2 of PRA 11.0, Quantification Notebook.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2011 Peer Review Finding: No Plant Response. Based on draft QU presently available it does not appear this has been addressed yet. Importance are listed, but review of values is not discussed	
QU-E4	**Not Met** **Finding QU-E4-01**	**QU-F4** **LE-F3**	**2010 Peer Review Finding:** Many of the items identified in Table C-1 are listed as not impacting because the assumption is considered conservative. The use of realistic assumptions in place of conservative assumptions would be expected to reduce the core damage frequency for many of the items identified as conservative. In addition, some assumptions represent the modeling approach used (e.g., length of time that secondary side cooling is available using AFW). These assumptions are very conservative based on a consideration of plant design and operation. SRs QU-E1 and QU-E2 require modeling uncertainties and assumptions to be identified. SR QUE4 requires these be properly characterized. Table C-1 should be reviewed and conservative assumptions should be realistically characterized. In addition, plant configuration assumptions can have a significant impact on model results. Plant configuration assumptions should be included in Table C-1 and their impact on the overall model results assessed. **2011 Peer Review Finding:** No Plant Response. Alignment assumptions (e.g., SW and CCW	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Section 5.0 of the PRA Notebook 11.0, Quantification Notebook contains a discussion on the sources of uncertainty and their impact to the PRA. In particular, battery life is discussed in Section 5.2.2.11, in which no impact is assessed because the current depletion time of 1 hour matches the existing documentation.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			configurations) were updated and therefore do not need to be added, however, characterization of assumptions are all minimal or no impact is questionable and still needs improvement. For example, battery life of 1 hour is characterized as "no impact" yet in comparison with other plants (Section 5.4) this is characterized as a significant difference with other plants.	
QU-F5	**Not Met** **Finding QU-F5-01**	**LE-G5**	**2010 Peer Review Finding:** This SR requires documentation of limitations in the quantification process that would impact applications. The discussion of limitations in the quantification process does not appear to be adequate. There is a discussion in the quantification notebook that is limited to quantifying sequences with failure probabilities greater than 0.1. There is no discussion of quantification process items which could impact applications. For example, the model assumes certain equipment alignments in the master flag file. The assumed alignments could impact applications, and the assumed alignments should be noted in the limitations discussion so that the impact on applications can be addressed when the model is used in support of applications. Review the model and quantification process and identify process and modeling items that are unique to the Point Beach PRA quantification that an analyst needs to be aware of when supporting applications.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The following text was added to Section 2.4.4, "Logic Flags" in the PRA 11.0 Quantification Notebook. Flags were used in the Point Beach model to indicate a particular condition (ie. "A" Steam Generator Intact) and to establish normal alignments (ie. D-49 supplying D-53). In most cases, the flags were either set to 0.0 or 1.0. When a flag setting is set to a value between 0.0 and 1.0, the flag is being used to establish a split fraction. For example, 1 of 2 component cooling water pumps is typically running on each unit. To account for this in the model, the flag for CCW pump running is set to 0.5 and the flag for CCW pump in standby is set to 0.5. This is the way that flags are used to provide a model which represents the as-built, as-operated plant. The potential impact on the results if the flags are not set properly is to create a model which does not accurately reflect the plant. Depending on the flag settings this can have a large or small impact on the results. For example, if the flags were set to have both component cooling water pumps running, the core damage frequency would increase.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2010 Peer Review Plant Response: The model was changed to address all possible configurations of the normally operating systems. The change added alignments so all pumps have T&M, are standby or running, etc. Changes were made to the service water system, the component cooling water system, and the instrument air system. As a result of these changes there is no need to discuss the impact of the assumed alignments as these are no longer used in the model. **2011 Peer Review Finding:** Finding LE-G5-01 is not addressed above. Also, a discussion of flag file setting and their potential impact on results, importance and applications has not been presented as requested by the original peer review.	The importance of the flags is that they enable the model to be changed to reflect changes in operating philosophy. If instead of normally operating 3 of 6 service water pumps, the plant went to normally operating 2 of 6 service water pumps the value of the flags for the service water pumps running and standby would change. No changes to the model would be required. The other importance of flag settings is that risk can easily be evaluated when equipment is set up to an alternate alignment by changing the value of the flag, rather than changing the model. The internal events PRA can be used for applications by resetting basic events values to the desired values through the use of a flag file. To find out the impact of flooding out a room, all the equipment and operator actions failed by the flooding in the room would be reset using a flag file to failed (TRUE). The impact of the applications on the model is application specific.
LE-B1	**CC-II** **Finding LE-B1-01 Doc Only**		**2010 Peer Review Finding:** No credit was taken for manual actions to vent the reactor pressure vessel (post core damage) to reduce vessel pressure. Venting is ultimately credited for venting through a stuck open pressurizer PORV or safety valve when addressing an induced SGTR. However, if the severe accident management guidelines call for depressurization, additionally fidelity and use of the low RCS pressure branch in the event tree. Include the action (with hardware properly accounted for) or justify not including it.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The technical basis explaining how the operator action was subsumed was added to Section 4.0 of the Large Early Release Frequency Notebook, 12.0.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2010 Peer Review Plant Response: Credit for manual action to vent the reactor vessel was taken. Although not applied as a functional heading in the Containment Event Tree (Figure 4-1 of PRA 12.0), a 50% probability of early RCS depressurization, which represents manual opening of a PORV or a stuck-open PORV, was used when calculating the probabilities of PI-SGTRs and TI-SGTRs in Appendix C of PRA 12.0 (see Appendix C item # 6, listed as "APET Heading F"). This treatment is very similar to the treatment provided for early RCS depressurization in WCAP-16341-P (for which, unfortunately, Point Beach was not a participant). Generally, this WCAP is considered to represent a Category II model. However, there are three differences between the WCAP modeling and the NUREG modeling: The WCAP used a success probability of 0.9 for early RCS depressuization, based upon Engineering Judgment. A value of 0.5 is used based upon NUREG-1570 (which also used Engineering Judgment). The WCAP assumes successful early RCS depressurization prevents TI-SGTR. The NUREG-1570 model assumes that if the S/Gs are depressurized, there is still a probability of TI-SGTR even with early RCS depressurization. In the WCAP, successful early RCS depressurization potentially yields a different early containment failure probability (although both	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			failure probabilities are approximately equal). The PB CET containment failure probabilities (i.e., CF_LOW and CF_HIGH) are assumed equal. Thus, an early RCS depressurization would not yield different containment failure results. One final comment. The NUREG-1570 Induced SGTR model contains three main branches or pathways: 1) the RCS is initially intact and may become depressurized via a number of ways, including manual depressurization (path A); 2) the RCS has a seal LOCA (path B); and 3) the RCS has a very early stuck open PORV (path C). Given the PB tube integrity, the only possibility of a TI-SGTR is for the path that contains a seal LOCA (path B). Because the RCS has a depressurization pathway, intentional depressurization is not questioned for this pathway in the NUREG-1570 model. Thus, only a PI-SGTR may be impacted by intentional depressurization (thus addressing the second bulleted modeling difference from above). A sensitivity was performed in which the successful depressurization probability was changed from 0.5 to 0.9. The PI-SGTR probability had an insignificant change (thus addressing the first bulleted modeling difference from above). Given the above discussion, the following conclusions are provided: Credit for manual action to vent the reactor vessel was taken. A sensitivity on the probability of successful early RCS depressurization was performed and the PI-SGTR probability had an insignificant change. Successful early RCS depressurization would have	

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			no impact to the TI-SGTR probability. Successful early RCS depressurization would have no impact on the early containment failure probability. Therefore, no changes were made to the Point Beach LERF Notebook. **2011 Peer Review Finding:** The technical basis described above explaining how the operator action is subsumed needs to be added to the LE Notebook to close this Finding.	
LE-C2	**CC-I** **Finding LE-C2-02**	**LE-C4 (CC-I)**	**2010 Peer Review Finding:** One action, HEP-CI--EOP-0-A2, which is to close the containment isolation valves given a LOCA, was included in the model. As stated in Section 5.2.2 of PRA 11.0, " MAAP analyses for break sizes larger than 6" had insufficient time available to perform the action. Therefore, this HEP is "OR"ed in the model with the Large LOCA initiating event. However, a review of the model revealed that this operator action was "AND"ed with the large LOCA initiator. Change the Gate from "AND" to "OR" and verify that the rest of the logic remains valid. **2010 Peer Review Plant Response:** This action was reviewed in the CAFTA fault tree logic. Agree with the suggested gate change. Change gate GCI1444 from an "AND" gate to an "OR" gate. The remainder of the logic is valid. **2011 Peer Review Finding:** Gate GCI1444 has not been changed.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Changed gate GCI1444 from an "AND" gate to an "OR" gate. The change does not affect CDF since it is in the LERF fault tree, containment isolation. The delta Unit 1 LERF due to this change was 2E-12. There was no delta Unit 2 LERF due to this change.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
LE-C3	**CC-I** **Finding LE-C3-01 New**		**2011 Peer Review Finding:** CC-II requires review of significant accident sequences and justification for any repairs credited. There is no evidence that this has been considered.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** A review of significant accident sequences was performed. No repairs were identified which could be credited. This was documented in PRA notebook 12.0, LE notebook, Section 1.6.2 by adding the following paragraph. The Point Beach PRA models limited credit for systems or functional recovery, this applies to both the Level 1 PRA and the Level 2 PRA. NRC and industry PRA expectations regarding modeling of recovery of failed equipment or functions during accident progression is that any such credited recovery in the PRA is scrutinized. As such, typical of most industry PRAs, the Point Beach PRA does not model significant credit in the Level 2 PRA for recovering failed equipment functions (other than offsite AC power).
LE-C9	**CC-I** **Finding LE-C9-01**	**LE-C10 (CC-I) LE-C11 (CC-I) LE-C12 (CC-I) LE-D3 (CC-I)**	**2010 Peer Review Suggestion:** Point Beach uses a unit-specific NUREG/CR-6595 CAFTA one-top model covering both CDF and LERF. The model should be expanded to address the dual unit impacts but no additional level of detail would be required. The NRC has indicated that for most applications, they are only interested in LERF. Furthermore, they have indicated that the use of a simplified NUREG/CR-6595 LERF model was acceptable as long as it addressed plant-specific differences from the model in NUREG/CR-6595. However, for any application that directly addresses containment performance or the actual source terms and timing, a more detailed analysis approximating a Level 2 PRA would likely be required.	No - The 2011 Peer Review Finding was NOT resolved in the PRA Model. See resolution below. The documentation should be updated at the next PRA Model revision to document the plant specific differences from the model in NUREG/CR-6595 **2011 Peer Review Plant Response:** The LERF Analysis does meet CC-I requirements. Use of the PRA which meets CC-1 requirements is conservative.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2010 Peer Review Plant Response: First, this is a suggestion only and there is no requirement to address this issue. Second, the Peer Review was of the Level 1 CDF and LERF Model. The issue raised suggests that A Level 2 PRA would likely be required to evaluate some issues of containment performance. This is outside the bounds of the areas to be evaluated in a Level 1 Peer Review. Additionally, it should be noted that Point Beach has a complete Level 2 PRA model, which is in the process of being updated, that is available to be used for any evaluation of containment performance for which the Level 1 LERF model is deemed insufficient. **2011 Peer Review Finding:** Original suggestion is limited in scope and since CC-II is not obtained for SR-C9 through C12, this should have been a finding. Above response does not address CC-II requirements (e.g., need to explain that containment analysis went beyond NUREG and is CC-II and add it to the notebook if this is true). Also, the present LERF Notebook is not a complete Level 2 model. If it exists it is not available.	
LE-D5	**CC-II** **Finding LE-D5-01 Doc Only in AS**		**2010 Peer Review Finding:** The secondary side isolation analysis was performed in a conservative manner without a detailed analysis of isolation capability. For example, all steam generator tube rupture core damage sequences are conservatively assumed to lead to LERF. The analysis meets Capability	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** The technical explanation was added to the end of Section 5.8.4 of the Accident Sequence Notebook.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			Category I, but a more detailed, realistic analysis is necessary to meet Capability Category II or higher. Perform a more detailed realistic secondary side isolation analysis to meet Capability Category II or higher. **2010 Peer Review Plant Response:** The end states of the SGTR event are GLH (SGTR, Late core damage, High pressure) and GEH (SGTR, Early core damage, High pressure), and only GEH is considered LERF (PRA 12.0, Section 3.2.3). The following SGTR GEH sequences either do not question isolation or isolation was successful, yet are considered LERF for PB: R-022: Isolation not questioned, AFW to ruptured S/G used, thus ruptured S/G not isolated. Therefore isolation is not successful. A more detailed isolation analysis not required for this sequence. R-023: Isolation not questioned, all AFW fails. A Level 2 MAAP analysis shows that this sequence results in RCS and S/G pressures rising to the point of S/G ASD and/or MSSV actuation. Therefore isolation is not successful. A more detailed isolation analysis not required for this sequence. R-026: Isolation successful, cooldown and depressurization fails. Because of the failure of cooldown and depressurization, the ruptured S/G continues to fill due to primary to secondary flow resulting in opening of the S/G ASD and/or MSSV. Therefore isolation is not successful.	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			A more detailed isolation analysis not required for this sequence. R-031: Isolation successful, cooldown and depressurization fails. Because of the failure of cooldown and depressurization, the ruptured S/G continues to fill due to primary to secondary flow resulting in opening of the S/G ASD and/or MSSV. Therefore isolation is not successful. A more detailed isolation analysis not required for this sequence. Note that the PB SGTR emergency procedure (EOP-3) cannot prevent the MSSVs from opening, and the ruptured S/G ASD controller is set to 1050 psig. This setting does not preclude the ASD from opening under the sequence conditions described above. Thus, successful isolation does not preclude the possibility of a release from the ruptured S/G. Therefore, no changes were made to the Point Beach LERF Notebook. **2011 Peer Review Finding:** Agree with technical explanation, but this explanation must be added to the AS Notebook for SGTR to close this Finding.	
LE-F1	**Not Met** **Finding LE-F1-01**		**2010 Peer Review Finding:** There were some discrepancies between Unit 1 and Unit 2 LERF results for loss of offsite power. Point Beach did identify the error but has not corrected as yet. Point Beach did provide the LERF results for both units in tables and pie charts. The pie chart for Unit 2 did not agree with the tables for ISLOCA results or SGTR results.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Peer Review Plant Response:** Model was revised such that GINIT-S2 is "OR"ed with GSC1120 when input to gate CET-013F.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			Specifically, the table shows a LERF contribution of 9.3 % for ISLOCA but ISLOCA does not show up in the pie chart for unit 2. Also, the table shows a LERF contribution of 16.4% for Unit 2 but the pie chart for Unit 2 shows 1%. Correct the known error with respect to the Loss of Offsite Power Contributions. Review the LERF results and adjust the tables and pie charts so that they are consistent with each other. **2010 Peer Review Plant Response:** Some of the issues with LERF were related to logic errors that have been identified and corrected in other systems, most notably 125 VDC and AFW, and one pair event tree errors on Unit 2 (TD1 and TD2). All of the Unit 1 and Unit 2 event trees were reviewed against the LERF notebook to verify that the event tree sequences are binned to the correct category. Following this, the LERF Unit 1 and Unit 2 logics were reviewed against the verified event trees to ensure that the sequences are being properly classified. Additional differences were due to discrepancies in data between the two Units (calc type, values). A set of changes to the CAFTA fault tree structures were made. The most notable was adding gate GSC1120 to AND gate CET-013F. This imposed the missing condition of RCP Seal LOCA.	

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			2011 Peer Review Finding: Errors corrected. However, it appears that S2 gate GINIT-S2 should be ORed with GSC1120 when input to Gate CET-013F.	
LE-F3	**Not Met** **Finding LE-F3-01**		**2010 Peer Review Finding:** Appendix A of PRA 12.0 contains a list of 25 assumptions specific to the LERF analysis but did not include a characterization of the potential of the assumptions. Section 5.2 for PRA 11.0 discussed sensitivities and other sources of uncertainties. This section did contain one sensitivity analysis related to LERF but the selected case was not related to any of the assumptions in PSA 12.0. Appendix C of PRA 11.0 also contains a list of the assumptions with a "characterization". These did match the assumptions listed in PRA 12.0. The characterization was at best terse. The assumptions were characterized as "Realistic" or "Conservative". However, for the "conservative" assumptions, there was no discussion of the potential impact on the model or the results. Also, the one issue for which a sensitivity analysis was performed was not on the list. This is a good indication that the list is incomplete. a) Provide an assessment of potential impact on the model for the assumptions characterized as "conservative". b) Tie the LERF sensitivity case to a LERF assumption by adding a new assumption and review the LERF analyses to determine if there are any other assumptions that might impact the analysis.	No - The 2011 Peer Review Finding was NOT resolved in the PRA Model. See resolution below. The documentation should be updated at the next PRA Model revision to document the potential impact on the model for the identified assumptions. Include sensitivities in this assessment. **2011 Peer Review Plant Response:** No response provided.

Attachment 1: Point Beach PRA Peer Review Findings				
SR	**Category and Finding**	**Other Affected SRs**	**Issue and Proposed Resolution**	**Impact to Applications and Peer Review Resolution**
			2010 Peer Review Plant Response: No response provided. **2011 Peer Review Finding:** No plant response yet.	
IFSO-A1	**Met** **Finding IFSO-A1-01**		**2011 Flooding Focused Peer Review Finding:** This SR states: For each flood area, IDENTIFY the potential sources of flooding including (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, feedwater system, condensate and steam systems); (b) plant internal sources of flooding (e.g., tanks or pools) located in the flood area; (c) plant external sources of flooding (e.g., reservoirs or rivers) that are connected to the area through some system or structure; (d) in-leakage from other flood areas (e.g., back flow through drains, doorways, etc.) Listing of potential flood sources in Section 7.3 appears to be reasonably complete, but a couple of systems/tanks do not appear to be addressed including Reheat Steam, Extraction Steam, Fuel oil tanks, Lube oil reservoirs, Spent Fuel pools, etc. Since the Extraction Steam and Reheat Steam systems are specifically identified as potential HELB concerns in Section 7.1.2 of the report, they need to be included in the Section 7.3 analysis. These other fluid sources are not expected to result in new significant floods but are required to be identified for completeness.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** A review of all systems at Point Beach was performed to identify which systems were liquid systems. The Table in Section 7.3 was then updated to show the review of all liquid systems including justification for why they are not required to be addressed further or to indicate they were included.

SR	Category and Finding	Other Affected SRs	Issue and Proposed Resolution	Impact to Applications and Peer Review Resolution
			To meet the intent of the supporting requirement a review of all fluid systems should be performed and their presence needs to be mentioned, including a justification for why they are not required to be addressed further (e.g. insufficient volume, location of suction/discharge piping, with respect to pool level, presence of pool leak detection systems, etc.).	
IFSO-A5	**Not Met** **Finding IFSO-A5-01**		**2011 Focused Flooding Peer Review Finding:** For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE: (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) flow rate (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source Section 7.3.2 of the Internal Flooding Analysis provides the details of the postulated internal flooding events, although temperature and pressure is not discussed in this section, and is not documented in the WEFLOOD.XLS spreadsheet. Temperature and pressure information needs to be provided to meet the requirements of the SR.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** The Temperature and Pressure information for each system modeled in the Flooding PRA was added to section 7.1.2 of the Flooding Notebook.
IFQU-A1	**Met** **Finding IFQU-A1-01**	**IFEV-B2 (Met)**	**2011 Focused Flooding Peer Review Finding:** As noted in F&O IFQU-A1-01 of the 2010 Peer Review, the process used to identify flood-induced initiating events is in the documentation, but the initiating events identified in the documentation does not always propagate properly into the flag files used for the quantification. In particular, the HEP events listed in Appendix 7.1J, Section 3.0 do	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** SR IFEV-B2 was not resolved from the 2010 Full Scope PRA Peer Review (F&O 2010 IFQU-A1-01). This was

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			not appear to have been included in the appropriate flag files. To address this finding, the flag files should be reviewed for completeness and reflect the basis information in the documentation so that all results are complete and reproducible.	identified in the Draft Internal Flooding Focused Peer Review. This F&O stated that there is still an issue with the Flag Files matching the documentation. In this instance, Flag Files U1-AFPN.flg, U1-AFPS.flg, U2-AFPN.flg, U2-AFPS.flg and U2-FOPH.flg did not have the associated HEP listed in Appendix 7.1J, Section 3.0 set to TRUE. The associated flag files were revised and affected the results as follows: <table><tr><th>Flood Area</th><th>Flag File</th><th>Peer Review CDF</th><th>Revised CDF</th><th>Change in CDF</th></tr><tr><td>U1 AFPN</td><td>U1-AFPN.flg</td><td>5.3977E-11</td><td>5.5843E-11</td><td>1.866E-12</td></tr><tr><td>U1 AFPS</td><td>U1-AFPS.flg</td><td>5.1662E-08</td><td>5.1662E-08</td><td>0.000E+00</td></tr><tr><td>U2 AFPN</td><td>U2-AFPN.flg</td><td>4.1403E-10</td><td>4.1403E-10</td><td>0.000E+00</td></tr><tr><td>U2 AFPS</td><td>U2-AFPS.flg</td><td>6.1560E-08</td><td>6.1560E-08</td><td>0.000E+00</td></tr><tr><td>U2 FOPH</td><td>U2-FOPH.flg</td><td>1.2410E-09</td><td>1.2410E-09</td><td>0.000E+00</td></tr></table> As a result of this change, Appendix 7.1K was revised accordingly.
IFQU-A6	**Not Met** **Finding IFQU-A6-01**		**2011 Focused Flooding Peer Review Finding:** F&O IFQU-A6-01 from the 2010 Peer Review identified that several internal events related HFEs were modified to support the quantification. However, the basis behind the "adjustments" to the HFEs did not appear to be justified - for example – a straight multiplier has been applied to the Internal Events HFEs instead of re-evaluating the HFE in detail. This is inappropriate since the conditions associated with the original equipment failures that	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** ASME-ANS RA-Sa-2009 IFQU-A6 has the following requirements: For all human failure events in the internal flood scenarios, INCLUDE the following scenario specific impacts on PSFs for control room and ex-control room actions as appropriate to the

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			resulted in the need for the HFE have changed, and the potential to respond to the failure is completely different. For the HFEs credited in the Internal Flooding analysis, a completely "new evaluation" of the HFEs need to be performed including an evaluation of the time available, the stress levels, and the potential that the equipment is even recoverable post flooding. Although additional information has been provided to attempt to justify the generic factors applied, this approach does not meet the requirements of this SR since it still does not include consideration of the scenario-specific impacts on the performance shaping factors which is required in order to meet this SR.	HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises) The Internal Flooding Analysis Notebook addresses each as follows: (a) Section 3.3 of Appendix 7.1H. The impacts of stress on operator reliability are addressed in NUREG/CR-1278 (Reference H-2) and take the form of multipliers that are applied to the nominal HEPs that are used to evaluate the subtasks of an operator action. These stress multipliers are considered to be applicable to flood-related stress and are used as the basis for quantifying the effects of flood-induced equipment failures and confusion on operator reliability. Table 20-16 of NUREG/CR-1278 (Reference H-2) provides a list of stress multipliers for step-by-step and dynamic actions over a range of different stress levels for both experienced and novice crews. These are used as the basis for the flood multipliers. Multiplier values for the Point Beach internal flooding analysis are determined by the following flow charts. The first flowchart is applicable to In-Control Room actions and the second flowchart is for Ex-Control Room actions. All Operator Actions in the internal events PRA model were reviewed and evaluated for the internal flooding analysis. For each PRA model operator action, the HRA Calculator data

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				associated with the action was reviewed and the appropriate multiplier applied to the HEP per the flow charts above (See Tables 1 and 2). The HRA Calculator provided the internal event HEP value for each event, the location of the action, and the time available to complete the action. Additional workload and stress have been evaluated for workload and stress per NUREG/CR-1278 which is appropriate to the HRA methodology used in the Point Beach PRA. (b) Cue availability has been evaluated and is documented in Table 1 and Table 2 of Section 3.3 in Appendix 7.1H. (c) The effect of flooding on mitigation was documented in Section 3.0 of Appendix 7.1J. The table in this section lists the HFEs which were set to guaranteed failure for selected flood initiating events because of the flooding effects in the areas where these actions are performed. (d) Flooding specific job aids are provided in Attachment 2 to Appendix 7.1H. Attachment 2 to Appendix 7.1H are the detailed HEP calculations for the operator actions credited in mitigating the flood. No operator training on mitigating flooding is provided at Point Beach.
IFQU-A10	**Not Met** **Finding IFQU-A10-01**		**2011 Focused Flooding Peer Review Finding:** F&O IFQU-A10-01 of the 2010 Peer Review identified that although the Internal Flooding Analysis documentation contained a couple of tables that had LERF values provided in them, no discussion could be found that any review of the LERF analysis was performed to confirm the applicability of the LERF sequences. The current Internal Flooding analysis does include a review of cut sets to ensure that the cut sets make sense	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** A discussion has been added to Appendix 7.1K, "Quantification", Section 3.2 which describes the review performed to determine if any new LERF sequences needed to be considered due to the unique impacts of a flood. No additional sequences were identified.

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			from a LERF perspective; however, it still does not contain a discussion of the review that was performed to determine if any NEW LERF sequences needed to be considered due to the unique impacts of a flood. Potential NEW LERF impacts could be required if the flood could cause bypass scenarios that were not previously evaluated due to multiple spurious operations, inadvertent openings of Containment purge valves due to flood impacts on control panels, etc. These types of potential NEW LERF impacts need to be included in the analysis, either by confirming that there are no NEW LERF sequences, or by modifying the LERF analysis as appropriate to address any NEW LERF sequences that are identified.	
IFQU-B2	**Met** **Finding IFQU-B2-01**	**IFEV-A5 (Not Met)**	**2011 Focused Flooding Peer Review Finding:** Although significant work has been done, and detailed spreadsheets are provided, there is still an inconsistency between the write-up in the front of the documentation (Section 7.4.4) and the information contained in the WEFLOOD spreadsheet printout in Appendix 7.1B and the WEFLOOD.XLS file provided to the Peer Review Team. For example, Section 7.4.4.8 states a frequency of 3.34E-05/yr for the DG1 scenario while the WEFLOOD spreadsheet shows a frequency of 3.5E-05/yr, and section 7.4.4.9 states a frequency for the DG2 scenario of 5.97E-05/yr while the WEFLOOD spreadsheet shows a frequency of 5.01E-05/yr. Note that the flag files appear to match the WEFLOOD.XLS spreadsheet.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** SR IFEV-A5 was not resolved from the 2010 Full Scope PRA Peer Review (F&O 2010 IFQU-B2-01). This was identified in the Draft Internal Flooding Focused Peer Review. This F&O stated that there is still and issue with the IF Notebook matching the electronic files, and in this case it was the WEFLOOD.xls matching Section 7.4.4.1 through 7.4.4.35. To resolve this issue in Rev. 2 of the 2011 IF Notebook, the values referenced from WEFLOOD.xls was removed and added a statement that the flooding frequency can be found in the spreadsheet WEFLOOD.xls found in Appendix 7.1C, WEFLOOD.xls.

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			This inconsistency should be eliminated and the analysis results consistent across the applications and documentation to avoid confusion and to support future updates.	
IFQU-B3	**Not Met** **Finding IFQU-B3-01**	**IFPP-B3 (Not Met)** **IFQU-B3 (Not Met)** **IFSN-B3 (Not Met)** **IFSO-B3 (Not Met)**	**2011 Focused Flooding Peer Review Finding:** In the 2010 Peer Review, F&O IFQU-B3-01 was written to identify that the Internal Flooding analysis did not identify some of the major conservatisms and assumptions in the analysis. This appears to remain an open item for multiple reasons. Some specific examples include: • Although the methodology selected for estimating initiating event frequencies is valid, it does not appear that uncertainty bounds associated with the IE frequencies were determined or documented. • All flood related initiating events are currently mapped to the general transient initiating event (INIT-T3), but some of the floods, by definition, impact entire systems that would result in another type of initiating event - e.g. CCW pump room flood would really be a "loss of CCW" initiating event, but it is not mapped to INIT-TCC. Mapping the flood-induced initiating events in this manner is considered an assumption and needs to be identified as such • The analysis specifically assumes the number of pumps running and the flow rate at which they are operating. These types of assumptions can result in major impacts on the resulting analysis and need to be addressed in the Uncertainty analysis.	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below. **2011 Focused Flooding Peer Review Plant Response:** 1. Uncertainty has been added to all flooding initiating events. The following text was added to Appendix 7.1I, "Uncertainty Analysis", Section 2.3, "Flood-Induced Initiating Events". "Initiating event frequencies are assumed to have a lognormal distribution. The error factors associated with the initiating event frequencies are determined by the order of magnitude associated with the initiating event frequency. If the initiating event frequency is E-3, an error factor of 10 is used. For an initiating event frequency of E-4, an error factor of 15 is assumed. If the initiating event frequency is E-5 or lower, an error factor of 20 is applied. The application of the lognormal distributions for the data range is consistent with how the other initiating events are treated in the PRA." 2. The following text was added to Appendix 7.1I, "Uncertainty Analysis", Section 2.3, "Flood-Induced Initiating Events". "It is assumed that all flood related initiating events can be mapped to the general transient initiating event, INIT-T3. This assumption is valid even though some floods impact entire systems. This is because in those cases where the entire system is impacted, the entire system is failed by the flag file. For example, if the

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			• The number of pumps and therefore flow rates associated with various flood scenarios are assumed, critical heights are assumed, etc. Because these assumptions have a direct impact on timing for HRA and on determining the potential consequences of a flood, they need to be captured and sensitivities performed to determine their significance. • There is an inherent assumption that all flood events will result in a reactor trip with PCS available. In reality, this is not true since some floods would result in a loss of CCW, others would result in a loss of Instrument Air, etc. These types of assumptions need to be identified, and their potential impact on quantification and results need to be evaluated, especially for flood scenarios that have a very large contribution to either CDF or LERF. • The qualitative and quantitative screening processes involve several subjective criteria and interpretations of the Standard that are by definition, assumptions and sources of uncertainties. Since these subjective criteria impact the entire analysis, they should be considered as significant sources of uncertainties and significant assumptions.	flooding in the PAB reaches the height of the CCW pumps, the CCW system would be lost. The loss of the CCW system is accounted for by setting the flag files associated with the scenario to fail the CCW pumps. The reactor will trip and the CCW pumps will be lost." 3. Section 3.6 was added to 7.1l which looks at the number of pumps running and the impact on the analysis. The conclusion is that this assumption does not have a major impact on the resulting analysis. 3.6 Number of Running Pumps There are three systems which are involved with submergence flooding, the only type of flooding which would be impacted by the number of pumps running. HELB, spray and jet impingement are independent of the number of pumps in service. The three systems are service water, circulating water and fire protection. Of the three systems, only the service water system has an operator action which would be affected by the number of pumps going to a runout condition. One fire protection pump at runout is larger than any of the fire protection scenario flowrates. So, whether one pump is running or two the time to submergence is unchanged. The circulating water system floods do not credit operator actions and will not be considered further. The time to reach critical flood height can be affected by the number of pumps running if the pipe break size is large enough to accommodate all running pumps in a runout condition. For example, the maximum flow out of a service water pipe break in the cable spreading room is about 7,000 gpm which

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				is less than the runout flow from one operating service water pump. This means the cable spreading room time to reach critical flood height is the same for one service water pump running or six. It does not create any new flooding hazards. So, submergence floods which Four of the submergence flooding areas credit operator action to stop the flood before the critical flood height is reached for service water flooding. These are AFPN, U2F, DG2 and PAB. U2F does not have any service water piping large enough to cause pump runout flow and will not be considered further. AFPN does not credit operator actions for floods which would cause runout of three service water pumps and will not be considered further. The remaining two areas had the operator actions which prevent reaching critical flood height for major flooding. It was assumed that if 4, 5, or 6 pumps were running, the operator would not be able to respond before the critical flood height was reached. The operator action failure rate was changed from the value with three service water pumps running to 1.0, always failed. The initiating event frequencies calculated from this were then substituted for the existing values in the internal flooding cutset. This changed the core damage frequency on Unit 1 from 1.98190E-6 to 1.98197E-6. The difference is 7E-11 which is not significant and the number of pumps running does not significantly affect the core damage frequency.
IFSO-A4	**Finding IFSO-A4-01**	**IFEV-A7 (Met) IFSO-A6**	**2011 Focused Flooding Peer Review:** F&O IFSO-A4-01 from the 2010 Peer review identified that human induced flooding needed to	No - The 2011 Peer Review Finding was resolved in the PRA Model. See resolution below.

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		(Met) **IFSO-B2** **(Met)**	be added to the Internal Flooding analysis. The current analysis has a very detailed evaluation of industry-related human induced flood events, and the potential for Point Beach specific human induced flood events. In the detailed evaluation, it was identified that over the last 8 years, approximately 20% of the major flood events in industry that were determined to be applicable to Point Beach were human-induced flood events. However, the conclusion was that the probability of human-induced significant flooding in power modes was judged to be insignificant. Since human-induced significant floods had a 20% contribution to significant flood frequency, this conclusion appears to be unjustified. Additionally, the IFSO-A4-01 F&O stated that the walkdown sheets should consider and document the potential for human-induced flooding for each flood area. Note, there is some discussion in Section 7.3.2 associated with the potential for human-induced floods in each flood area, but it is not always easy to find or understand, and it does not provide a systematic evaluation for human-induced flooding potential.	**2011 Focused Flooding Peer Review Plant Response:** Section 7.5.3 of PRA 7.1, Internal Flooding Notebook has been revised to include human induced inadvertent fire protection system actuation and maintenance induced flooding. The spreadsheet used to calculate submergence initiating event frequencies (WEFLOOD.xls) has been updated to include human induced inadvertent fire protection system actuations and maintenance induced flooding. The table was updated to show the results of the major flood events which were applicable to Point Beach. All of the major industry flooding events which were human induced and applicable to Point Beach are in the inadvertent fire protection actuation category. Section 7.5.4 was added to PRA 7.1 which discusses maintenance-induced flooding at Point Beach. The conclusion was that maintenance induced flooding will have a negligible impact on flooding because of the low probability of mechanical failure and the negligible probability of human failure to incorrectly position maintenance isolation boundary valves. However, maintenance induced flooding was included for 3 submergence flooding zones. Section 7.5.6 (formerly 7.5.4) provides a detailed review of action requests at Point Beach over the last 10 years looking for cases of maintenance induced flooding. None were identified which confirms the conclusion of Section 7.5.4. Regarding the second part of the finding, that walkdown sheets should consider and document the potential for human-induced flooding for each flood area, no

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				requirement to that effect could be identified. IFSO-A4 is related to identification of flooding mechanisms and does not discuss walkdowns. IFEV-A7 says to include consideration of human-induced floods during maintenance through application of generic data. It does not discuss walkdowns. IFSO-A6 says to conduct plant walkdowns to verify the accuracy of information obtained from plant information sources and determine or verify the location of flood sources and in-leakage pathways. This was performed as part of the flooding walkdowns. However, a separate walkdown was performed to confirm the maintenance induced flooding evaluation and is documented in added Section 7.5.5.

[2] The gap analysis is conducted independently of RI-ISI and is based on comparing the PRA model against the supporting requirements of ASME PRA standard at capability category II. Many of the identified gaps are not applicable to RI-ISI since in general capability category I is sufficient. For completeness, all current gaps are identified in Attachment 1.