Entergy

River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775
Tel 225-381-4149

Jerry C. Roberts
Director, Nuclear Safety Assurance

RBG-46922

June 16, 2009

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

SUBJECT: Request for Alternative – Implementation of a Risk-Informed Inservice Inspection Program Based on ASME Code Case N-716
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) hereby requests authorization to implement a risk-informed Inservice Inspection (RI-ISI) program based on the American Society of Mechanical Engineers (ASME) Code Case N-716, as documented in the attached Request for Alternative RBS-ISI-013. RBS-ISI-013 is being submitted in a template format as Attachment 1. This template format is similar to the submittals the NRC Staff has approved for Waterford 3 and Grand Gulf. This format is also similar to the recently submitted request for alternative by Calvert Cliffs and Arkansas Nuclear One for the same subject. This request includes information to address NRC requests for additional information available at the time of development of this submittal.

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

Entergy requests to implement this alternative beginning with the third period of the second ISI interval to cover the remaining weld examinations that were not performed during the second interval. The second ISI interval was previously extended by RBS-ISI-005 as approved by NRC on May 17, 2007 (TAC No. MD3442). A separate relief request (RBS-ISI-012) has been submitted requesting further extension of the second interval to allow for completion of NRC's review of RBS-ISI-013.

Entergy requests approval of the proposed alternative by December 1, 2010. RBS will withdraw the Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 for use at RBS upon NRC approval of this RI ISI program submittal. Although this request is neither exigent nor emergency, your prompt review is requested.

The request for alternative includes several new commitments that are summarized in Attachment 2.

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NRR

If you have any questions or require additional information, please contact David Lorfing, Manager, Licensing at (225) 381-4157.

Sincerely,

Director, Nuclear Safety Assurance
River Bend Station - Unit 1

JCR/DNL/bmb

Attachments:

1. Request for Alternative RBS-ISI-013
2. Licensee Identified Commitments

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
612 E. Lamar Blvd., Suite 400
Arlington, TX 76011-4125

NRC Senior Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

U. S. Nuclear Regulatory Commission
Attn: Mr. Alan B. Wang
MS O-7 D1
Washington, DC 20555-0001

Mr. Jeffrey P. Meyers
Louisiana Department of Environmental Quality
Office of Environmental Compliance
Attn. OEC - ERSD
P. O. Box 4312
Baton Rouge, LA 70821-4312

ATTACHMENT 1 TO

RBG-46922

REQUEST FOR ALTERNATIVE

RBS-ISI-013

REQUEST FOR ALTERNATIVE
ENTERGY OPERATIONS, INC.
RIVER BEND STATION – UNIT 1

REQUEST FOR ALTERNATIVE
RBS-ISI-013

APPLICATION OF ASME CODE CASE N-716

RISK-INFORMED / SAFETY-BASED INSERVICE INSPECTION PROGRAM PLAN

Table of Contents

ENTERGY OPERATIONS, INC.
RIVER BEND STATION – UNIT 1

REQUEST FOR ALTERNATIVE
RBS-ISI-013

1. INTRODUCTION

River Bend Station – Unit 1 (RBS) is currently in the second inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. RBS plans to complete the current (second) ISI interval by implementing a risk-informed / safety-based inservice inspection (RIS_B) program during the third inspection period of the interval. Entergy will also implement 100% of the RIS_B program in the third ISI interval.

The ASME Section XI code of record for the second ISI interval at RBS is the 1992 Edition for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components. The ASME Section XI code of record for the third ISI interval at RBS is the 2001 Edition through 2003 Addenda for items in these Examination Categories.

The objective of this submittal is to request the use of the RIS_B process for the ISI of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*, which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure.*

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

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

1.2 Probabilistic Safety Assessment (PSA) Quality

The River Bend Station PRA has been demonstrated to be adequate for this application. The PRA and its supporting processes are described in further detail in Appendix 1. As described in the Appendix, the RBS PRA internal events model has been reviewed as part of the Boiling Water Reactor Owners Group (BWROG) Peer Review process in 1998. A self-assessment of the current PRA model against Regulatory Guide 1.200 was conducted in Fall 2008. Results of the self-assessment are discussed in the Appendix. The Internal Flooding PRA was updated to fully meet RG 1.200 requirements in 2009. Future PRA changes under the PRA maintenance and update process will address identified gaps against Regulatory Guide 1.200. As discussed in the Appendix, most of the gaps are considered documentation issues and all gaps have been reviewed to support the conclusion that the RBS PRA is fully capable of supporting the request to use the RIS_B process, based upon ASME Code Case N-716, for a Risk-Informed In-Service Inspection program at River Bend.

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, except as amended by the application of ASME Code Case N-663 (Request for Alternative CEP-ISI-007) that was approved for use at RBS by the NRC on August 26, 2003.

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

2.2 Augmented Programs

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

- The original plant augmented inspection program for high-energy line breaks, implemented in accordance with RBS Final Safety Analysis Report (FSAR) Sections 5.2.4.8 "Augmented Inservice Inspection to Protect Against Postulated Class 1 Piping Failures" and 6.6.8, "Augmented Inservice Inspection to Protect Against Postulated Class 2 Piping Failures," were revised in accordance with the risk-informed break exclusion region methodology (RI-BER) described in EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs*. EPRI Report 1006937 was approved by the NRC in 2002. The results of the RI-BER application demonstrated that the volumetric examination requirement for this scope of piping could be reduced from 100% to approximately 15%. As a result, 15% of the BER population will be examined during the course of each ten-year interval which exceeds the 10% requirement imposed by Code Case N-716.

- The RBS augmented inspection program for intergranular stress corrosion cracking (IGSCC) per Generic Letter (GL) 88-01, *NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping*, is relied upon to manage this damage mechanism. GL 88-01 specifies the examination extent and frequency requirements for austenitic stainless steel welds classified as Categories A through G, depending on their susceptibility to IGSCC. In accordance with EPRI TR-112657, piping welds identified as Category A are considered resistant to IGSCC and are assigned a low failure potential provided no other damage mechanisms are present. Consequently, the examination of welds identified as Category A inspection locations is subsumed by the RIS_B Program. The existing RBS augmented inspection program for the other piping welds susceptible to IGSCC (Categories "B" and "C") remains unaffected by the RIS_B Program submittal.

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

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
- Failure Potential Assessment
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

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 used to determine the treatment requirements. HSS welds are determined in accordance with the requirements below. 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 (> NPS 4) for high-energy piping systems as defined by the Owner. This may include Class 3 or Non-Class piping; and

(5) Any piping segment whose contribution to CDF is greater than 1E-06 (and per NRC feedback on the Grand Gulf and DC Cook RIS_B pilot applications 1E-07 for LERF) based upon a plant-specific PSA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping.

3.2 Failure Potential Assessment

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

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

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for RBS. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling, and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 include:

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

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

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

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

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

   AND

   $\Delta T > 50°F$,

   AND

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

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

- **Turbulent Penetration TASCS**

  Turbulent penetration 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 keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, 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 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. The above criteria have previously been submitted by EPRI to the NRC for generic approval [letters dated February 28, 2001, and March 28, 2001, from P.J. O’Regan (EPRI) to Dr. B. Sheron (USNRC), *Extension of Risk-Informed Inservice Inspection Methodology*]. The methodology used in the RBS RIS_B application for assessing TASCS potential conforms to these updated criteria. Final Materials Reliability Program (MRP) guidance on the subject of TASCS will be incorporated into the RBS RIS_B application, if warranted. It should be noted that the NRC has granted approval for RI-ISI relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (NRC letter dated September 28, 2001) and South Texas Project (NRC letter dated March 5, 2002).

3.3 Element and NDE Selection

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

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:
  (a) A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.
  (b) If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.
  (c) If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the Reactor Coolant Pressure Boundary (RCPB) welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the inside first isolation valve (IFIV) (i.e., isolation valve closest to the Reactor Pressure Vessel (RPV)) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (OC) (e.g., portions of the main feedwater system in BWRs) shall be selected.

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

In contrast to a number of 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% be chosen. A brief summary 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.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		Class 3 Welds(3)		All Piping Welds(4)	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected
1	763	83	1338	0	4	0	2105	83

Notes
(1) Includes all Category B-F and B-J locations. All 763 Class 1 piping weld locations are HSS.
(2) Includes all Category C-F-1 and C-F-2 locations. Of the 1338 Class 2 piping weld locations, 13 are HSS and the remaining 1325 are LSS.
(3) All four of these Class 3 piping weld locations are HSS.
(4) Regardless of safety significance, Class 1, 2 and 3 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 station's pressure test program that remains unaffected by the RIS_B Program.

Prior to developing the RIS_B Program, RBS had planned to inspect locations scheduled for examination under a traditional ASME Section XI inspection program. Examination activities during refueling outages are planned well in advance. In general, only designated plant areas and components are accessible for examination during a given refueling outage due to other ongoing plant maintenance and modification activities. Hence, any location previously scheduled for examination in the third period via the traditional program will remain scheduled for examination in the third period, for locations selected for RIS_B Program purposes. Additional samples will be selected, if necessary, to achieve equal representation of the degradation mechanisms. Other factors, such as accessibility and scaffolding requirements, will also factor into the selection process.

3.3.1 Additional Examinations

If the flaw is original construction or otherwise is acceptable, Code rules do not require any additional inspections. Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500 and/or IWB-3600. As part of performing evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. The process for ordinary flaws is to perform the evaluation using ASME Section XI. If the flaw meets the criteria, then it is noted and appropriate successive examinations scheduled. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000 and/or applicable ASME Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation is dependent on practical considerations (i.e., the practicality of performing additional NDE or removing the flaw for further evaluation during the outage). The NRC is involved in the process at several points. For preemptive weld overlays, a relief request in accordance with 10 CFR 50.55a(a)(3) is usually required for design and installation. Should a flaw be discovered during an examination, a notification in accordance with 10 CFR 50.72 or 10 CFR 50.73 may be required. IWB-3600 requires the evaluation to be submitted to the NRC. Finally, the evaluation will be documented in the corrective action program and the Owner submittals required by Section XI.

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.

3.3.2 Program Relief Requests

An attempt has been made to select RIS_B locations for examination such that a minimum of >90% coverage (i.e., Code Case N-460 criteria) is attainable. However, some limitations will not be known until the examination is performed since some locations may be examined for the first time by the specified techniques. In instances where locations at the time of the examination fail to meet the >90% coverage requirement, the process outlined in 10 CFR 50.55a will be followed.

Per footnote 3 of Table 1 of Code Case N-716, when the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the change-in-risk evaluation (paragraph 5 of Code Case N-716). The change in risk evaluation of Code Case N-716 is consistent with previous RI ISI applications and meets RG 1.174 change-in-risk acceptance criteria. Areas with acceptable limited examinations, and their bases, shall be documented.

Consistent with previously approved RI-ISI submittals, RBS 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 that time. Relief requests will be submitted per the guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 will be withdrawn for use at RBS upon NRC approval of the RIS_B Program submittal.

3.4 Risk Impact Assessment

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

This evaluation categorized segments as HSS or LSS in accordance with Code Case N-716, and then determined what inspection changes are proposed for each system. The changes include 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 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 RG 1.174 and 1.178. The EPRI criterion 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, CCDP and CLERP values of 1E-4 and 1E-5 are generally conservatively used, unless pipe segments in the plant internal flooding study are found with higher values. For the RBS RIS_B application, CCDP and CLERP values of 3.4E-4 and 1.4E-5 have been used for LSS welds to bound plant internal flooding study results. The 3.4E-4 and 1.4E-5 values used for CCDP and CLERP is based on results from the plant internal flooding study for a postulated rupture of Class 2 Feedwater system piping outside containment and have been conservatively applied as an upper bound for all LSS welds.

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of the Code Case. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion or stress corrosion cracking are assigned to a medium failure potential, and 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 to verify that the LSS piping was not susceptible to FAC or water hammer. 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 3.4-1) for use in the change-in-risk assessment. Experience with previous industry RI-ISI applications shows this to be conservative.

RBS 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 below. Consistent with the EPRI risk-informed methodology, the upper bound for all HSS break locations that fall within the high consequence rank range was based on the highest CCDP value obtained dependent upon whether the piping break occurs inside or outside of containment. For piping breaks inside containment, the upper bound is based on RBS plant specific Initiator T3B1 (i.e., loss of Feedwater, condenser, reactor core isolation cooling and shutdown cooling). For piping breaks outside containment, the upper bound is based on RBS plant-specific flood scenarios F-4-2-1-21a and F-4-2-1-21b that assess a rupture of Class 1 HSS and Class 2 LSS Feedwater system piping outside containment.

CCDP and CLERP Values Based on Break Location

Break Location Designation	Estimated		Consequence Rank	Upper Bound	
	CCDP	CLERP		CCDP	CLERP
LOCA	1.0E-4	2.3E-6	**HIGH**	1.7E-4	3.5E-5
Pipe breaks that result in a LOCA – Estimated based on highest CCDP for LOCA (Intermediate Break) from PSA model					
ILOCA	<1.0E-6	<1.0E-7	**MEDIUM**(1)	1.0E-4	1.0E-5
Pipe breaks that result in an isolable LOCA inside containment – Estimated based on Intermediate LOCA CCDP of 1.0E-4 and valve fail to close probability of 3.4E-3					
ILOCA – FW	1.7E-4	3.5E-5	**HIGH**	1.7E-4	3.5E-5
Pipe breaks that result in an isolable LOCA inside containment – Estimated based on loss of Feedwater, condenser, reactor core isolation cooling and shutdown cooling					
ILOCA – OC	3.4E-4	1.4E-5	**HIGH**	3.4E-4	1.4E-5
Pipe breaks that result in an isolable LOCA outside containment – Estimated based on flood scenarios F-4-2-1-21a and F-4-2-1-21b that assess a rupture of Class 1 HSS Feedwater system piping outside containment; conservatively applied to all ILOCA – OC designated break locations					
PLOCA	<1.0E-6	<1.0E-7	**MEDIUM**(1)	1.0E-4	1.0E-5
Pipe breaks that result in a potential LOCA – Estimated based on Intermediate LOCA CCDP of 1.0E-4 and valve rupture probability of 1.0E-3					
ISLOCA	1.0E-3	1.0E-3	**HIGH**	1.0E-3	1.0E-3
Pipe breaks that result in an interfacing system LOCA outside containment – Estimated based on valve rupture probability of 1.0E-3					
2ISLOCA	8.3E-6	8.3E-6	**MEDIUM**	1.0E-4	1.0E-5
Pipe breaks that result in an interfacing system LOCA outside containment – Estimated based on two valve rupture probability of 8.3E-6					
MSD – 3	3.4E-6	3.4E-6	**MEDIUM**	1.0E-4	1.0E-5
Pipe breaks that occur in main steam drain system piping outside containment – Estimated based on an assumed steam LOCA CCDP outside containment of 1.0E-3 and valve fail to close probability of 3.4E-3					
DTM – 1/MSI – 1	2.0E-6	2.0E-6	**MEDIUM**	1.0E-4	1.0E-5
Pipe breaks that occur in main steam drain and main steam isolation valve leakage control system piping outside containment – Estimated based on an assumed steam LOCA CCDP outside containment of 1.0E-3 and valve fail to close probability of 2.0E-3					
Class 2 LSS	3.4E-4	1.4E-5	**HIGH**	3.4E-4	1.4E-5
Pipe breaks that occur in Class 2 system piping designated as LSS – Estimated based on flood scenarios F-4-2-1-21a and F-4-2-1-21b that assess a rupture of Class 2 LSS Feedwater system piping outside containment; conservatively applied to all Class LSS system piping					

Note

(1) Although the estimated CCDP and CLERP values for ILOCA and PLOCA break locations fall in the "Low" consequence rank range, a "Medium" consequence rank is conservatively used for risk impact.

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 3.4-1 presents a summary of the RIS_B Program versus 1992 ASME Section XI Code Edition program requirements on a "per system" basis. The presence of FAC and IGSCC was adjusted for in the quantitative analysis by excluding their impact on the failure potential rank. The exclusion of the impact of FAC and IGSCC on the failure potential rank and therefore in the determination of the change in risk is appropriate, because FAC and IGSCC are damage mechanisms managed by separate, independent plant augmented inspection programs. The RIS_B Program credits and relies upon these plant augmented inspection programs to manage these damage mechanisms. The plant FAC and IGSCC Programs will continue to determine where and when examinations are performed. Hence, since the number of FAC and IGSCC examination locations remains the same "before" and "after" and no delta exist, there is no need to include the impact of FAC and IGSCC in the performance of the risk impact analysis.

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

RBS Risk Impact Results

System(1)	ΔR_{CDF} Results		ΔR_{LERF} Results	
	w/ POD	w/o POD	w/ POD	w/o POD
RPV	2.04E-11	8.84E-11	4.20E-12	1.82E-11
RDS	1.36E-10	1.36E-10	5.60E-12	5.60E-12
RCS	-1.02E-11	-1.02E-11	-2.10E-12	-2.10E-12
FWS	1.97E-10	5.44E-10	3.50E-11	9.52E-11
MSS	7.94E-11	7.94E-11	2.96E-12	2.96E-12
SLS	-4.90E-12	-4.90E-12	-8.50E-13	-8.50E-13
CSH	4.43E-10	4.77E-10	1.84E-11	2.54E-11
RHS	2.46E-09	2.46E-09	1.16E-10	1.16E-10
CSL	3.09E-10	3.09E-10	1.31E-11	1.31E-11
MSI	-2.50E-12	-2.50E-12	-2.50E-13	-2.50E-13
ICS	3.39E-10	3.39E-10	1.41E-11	1.41E-11
WCS	1.00E-12	1.00E-12	1.00E-13	1.00E-13
DTM	-5.05E-12	-5.05E-12	-7.75E-13	-7.75E-13
TOTAL	3.97E-09	4.42E-09	2.05E-10	2.86E-10

Note
(1) Systems are described in Table 3.1.

3.4.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI 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 structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients which are 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 plant-specific piping with a contribution to CDF of greater than 1E-06 (or 1E-07 for LERF) be included in the scope of the application. No such piping was identified at RBS.

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.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program. The new program will be implemented into the second 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.

The monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of the condition identified
   (2) Evaluate, develop a corrective action plan or plans
D. Decide
E. Implement
F. Monitor
G. Trend

The RIS_B Program is a living program requiring feedback of new relevant information to ensure the appropriate identification of HSS piping locations. As a minimum, this review will be conducted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or GL requirements, or by industry and plant-specific feedback.

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

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RIS_B Program and ASME Section XI 1992 Code Edition program requirements for in-scope piping is provided in Table 5.

Currently, RBS is in its extended second ISI interval. By letter dated November 1, 2006, the licensee stated that it planned to implement a risk-informed/safety based ISI (RIS_B) program during the third inspection period of the current (second) ISI interval. In the subject extension request, RBS committed to perform the same percentage of examinations which remained incomplete from the second ISI interval. These welds would be selected from the welds included under the new risk informed program and would have been completed by the end of RF-15 currently scheduled for the Fall 2009.

Due to delays in completing the updated flooding study in support of the RIS_B submittal, an additional extension of the second ISI interval is being requested under Request for Alternative RBS-ISI-012. In this request, RBS commits to complete approximately 60% of the remaining examinations selected under the conventional ISI program by the end of RF-15. All of the required first period examinations of the third interval would be performed during RF-16 currently scheduled for 2011.

The third ISI interval will implement 100% of the inspection locations selected for examination per the RIS_B Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

6. REFERENCES/DOCUMENTATION

USNRC Safety Evaluation pertaining to the use of ASME Code Case N-663, dated August 26, 2003 (Letter CNRI-2003-00010)

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

EPRI TR-1018427, *Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs*

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

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

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

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

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

Supporting Onsite Documentation

Structural Integrity (SI) Calculations

- RBS-12Q-301, Rev. 0, *Degradation Mechanism Evaluation*
- RBS-12Q-302, Rev. 0, *Risk Informed Break Exclusion Region Evaluation for River Bend Station*
- RBS-12Q-303, Rev. 0, *Service History Review*
- ENTP-19Q-310, Rev. 2, *Degradation Mechanism Evaluation for River Bend*
- ENTP-19Q-311, Rev. 0, *N-716 Evaluation of River Bend Station*

Table 3.1
N-716 Safety Significance Determination

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	>1E-6 CDF >1E-7 LERF	High	Low
RPV – Reactor Pressure Vessel (050)	40	✓					✓	
RDS – Control Rod Drive (052)	48							✓
RCS – Reactor Coolant (053)	1	✓	✓				✓	
	113	✓					✓	
FWS – Feedwater (107)	10	✓	✓		✓		✓	
	50	✓	✓				✓	
	4	✓			✓		✓	
	2	✓					✓	
	23							✓
MSS – Main Steam (109)	20	✓			✓		✓	
	122	✓					✓	
	16							✓
SLS – Standby Liquid Control (201)	63	✓					✓	
CSH – High Pressure Core Spray (203)	19	✓					✓	
	129							✓
RHS – Residual Heat Removal (204)	13	✓	✓				✓	
	53	✓					✓	
	866							✓
CSL – Low Pressure Core Spray (205)	18	✓					✓	
	78							✓
MSI – Main Steam Leakage Control (208)	43	✓					✓	
ICS – Reactor Core Isolation Cooling (209)	9	✓			✓		✓	
	10	✓					✓	
	165							✓
WCS – Reactor Water Cleanup (601)	8	✓			✓		✓	
	83	✓					✓	
	17				✓		✓	
DTM – Steam Drains (609)	82	✓					✓	
SUMMARY RESULTS FOR ALL SYSTEMS	10	✓	✓		✓		✓	
	64	✓	✓				✓	
	41	✓			✓		✓	
	648	✓					✓	
	17				✓		✓	
	1325							✓
TOTALS	**2105**						**780**	**1325**

Table 3.2
Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
RPV	✓	✓	✓								
RDS[2]											
RCS											
FWS[2]	✓	✓									✓
MSS[2]											
SLS											
CSH[2]		✓									
RHS[2]											
CSL[2]											
MSI											
ICS[2]											
WCS											✓
DTM											

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 RDS system in its entirety, as well as portions of the FWS, MSS, CSH, RHS, CSL and ICS systems.

Table 3.3
N-716 Element Selections

System[1]	Selections	HSS	DMs[2]		RCPB	RCPB$^{IFIV(3)}$	RCPB^{OC}	BER
RPV	Required	4 of 40	TASCS, TT, (IGSCC) TASCS, TT TT, (IGSCC) None (IGSCC)	4(7) of 28	4 of 40	3	n/a	n/a
	Made	4	TASCS, TT, (IGSCC) TASCS, TT TT, (IGSCC) None (IGSCC)	1 1 0 2	4	4	n/a	n/a
RDS	Required	n/a	n/a		n/a	n/a	n/a	n/a
	Made	n/a	n/a		n/a	n/a	n/a	n/a
RCS	Required	12 of 114	n/a		12 of 114	8	n/a	n/a
	Made	12	n/a		12	12	n/a	n/a
FWS	Required	7 of 66	TASCS, TT, (FAC) TASCS, TT TT	7(17) of 66	7 of 66	5	1 of 10	2 of 14
	Made	7	TASCS, TT, (FAC) TASCS, TT TT	5 2 0	7	5	2	2
MSS	Required	15 of 142	n/a		15 of 142	10	3 of 25	2 of 20
	Made	15	n/a		15	10	3	8
SLS	Required	7 of 63	n/a		7 of 63	5	3 of 28	n/a
	Made	7	n/a		7	4	3	n/a
CSH	Required	2 of 19	TT	1 of 4	2 of 19	2	1 of 4	n/a
	Made	2	TT	1	2	1	1	n/a
RHS	Required	7 of 66	n/a		7 of 66	5	2 of 16	n/a
	Made	7	n/a		7	5	2	n/a
CSL	Required	2 of 18	n/a		2 of 18	2	1 of 5	n/a
	Made	2	n/a		2	1	1	n/a
MSI	Required	5 of 43	n/a		5 of 43	n/a	5 of 43	n/a
	Made	5	n/a		5	n/a	5	n/a
ICS	Required	2 of 19	n/a		2 of 19	2	1 of 5	2 of 9
	Made	2	n/a		2	1	1	2
WCS	Required	11 of 108	None (FAC)	1 of 4	10 of 91	7	1 of 2	3 of 25
	Made	11	None (FAC)	1	11	9	1	3
DTM	Required	9 of 82	n/a		9 of 82	6	6 of 52	n/a
	Made	9	n/a		9	3	6	n/a
TOTAL	**Made**	**83**	**13**		**83**	**55**	**25**	**15**

Notes

(1) Systems are described in Table 3.1.

(2) For RPV and FWS systems, no more than 10% of HSS piping welds are required to be selected for examination.

(3) For SLS, CSH, CSL, ICS and DTM systems, it was not possible to meet the requirement that 2/3 of the RCPB piping welds selected for examination be located between the first isolation valve and the reactor pressure vessel, while also ensuring that a minimum of 10% of the RCPB piping welds that lie outside containment were selected for examination. This lesson learned from the RBS RIS_B application is being addressed in Revision 1 to Code Case N-716.

Table 3.4-1
Risk Impact Analysis Results

System[(1)]	Safety *Significance*	Break Location[(2)]	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank[(3)]	SXI[(4)]	RIS_B[(5)]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RPV	High	LOCA	TASCS, TT, (IGSCC)	Medium (Medium)	3	1	-2	0.00E+00	3.40E-11	0.00E+00	7.00E-12
RPV	High	LOCA	TASCS, TT	Medium	1	1	0	-2.04E-11	0.00E+00	-4.20E-12	0.00E+00
RPV	High	LOCA	TT, (IGSCC)	Medium (Medium)	2	0	-2	2.04E-11	3.40E-11	4.20E-12	7.00E-12
RPV	High	LOCA	None (IGSCC)	Low (Medium)	22	2	-20	1.70E-11	1.70E-11	3.50E-12	3.50E-12
RPV	High	LOCA	None	Low	4	0	-4	3.40E-12	3.40E-12	7.00E-13	7.00E-13
TOTAL								**2.04E-11**	**8.84E-11**	**4.20E-12**	**1.82E-11**
RDS	Low	Class 2 LSS	N/A	Assume Medium	4	0	-4	1.36E-10	1.36E-10	5.60E-12	5.60E-12
TOTAL								**1.36E-10**	**1.36E-10**	**5.60E-12**	**5.60E-12**
RCS	High	LOCA	None	Low	0	12	12	-1.02E-11	-1.02E-11	-2.10E-12	-2.10E-12
TOTAL								**-1.02E-11**	**-1.02E-11**	**-2.10E-12**	**-2.10E-12**
FWS	High	LOCA	TASCS, TT, (FAC)	Medium (High)	15	5	-10	0.00E+00	1.70E-10	0.00E+00	3.50E-11
FWS	High	LOCA	TASCS, TT	Medium	16	0	-16	1.63E-10	2.72E-10	3.36E-11	5.60E-11
FWS	High	LOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FWS	High	ILOCA – FW	TASCS, TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FWS	High	ILOCA – OC	TASCS, TT	Medium	1	2	1	-1.02E-10	-3.40E-11	-4.20E-12	-1.40E-12
FWS	Low	Class 2 LSS	N/A	Assume Medium	4	0	-4	1.36E-10	1.36E-10	5.60E-12	5.60E-12
TOTAL								**1.97E-10**	**5.44E-10**	**3.50E-11**	**9.52E-11**
MSS	High	LOCA	None	Low	8	10	2	-1.70E-12	-1.70E-12	-3.50E-13	-3.50E-13
MSS	High	ILOCA	None	Low	1	2	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
MSS	High	ILOCA – OC	None	Low	11	3	-8	1.36E-11	1.36E-11	5.60E-13	5.60E-13
MSS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MSS	High	DTM – 1 / MSI – 1	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MSS	Low	Class 2 LSS	N/A	Assume Medium	2	0	-2	6.80E-11	6.80E-11	2.80E-12	2.80E-12
TOTAL								**7.94E-11**	**7.94E-11**	**2.96E-12**	**2.96E-12**
SLS	High	LOCA	None	Low	0	4	4	-3.40E-12	-3.40E-12	-7.00E-13	-7.00E-13
SLS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SLS	High	ISLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SLS	High	2ISLOCA	None	Low	0	3	3	-1.50E-12	-1.50E-12	-1.50E-13	-1.50E-13
TOTAL								**-4.90E-12**	**-4.90E-12**	**-8.50E-13**	**-8.50E-13**

Table 3.4-1 (Cont'd)
Risk Impact Analysis Results

System[1]	Safety Significance	Break Location[2]	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank[3]	SXI[4]	RIS_B[5]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
CSH	High	LOCA	TT	Medium	3	1	-2	0.00E+00	3.40E-11	0.00E+00	7.00E-12
CSH	High	LOCA	None	Low	1	0	-1	8.50E-13	8.50E-13	1.75E-13	1.75E-13
CSH	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSH	High	ISLOCA	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSH	Low	Class 2 LSS	N/A	Assume Medium	13	0	-13	4.42E-10	4.42E-10	1.82E-11	1.82E-11
TOTAL								**4.43E-10**	**4.77E-10**	**1.84E-11**	**2.54E-11**
RHS	High	LOCA	None	Low	3	5	2	-1.70E-12	-1.70E-12	-3.50E-13	-3.50E-13
RHS	High	PLOCA	None	Low	7	0	-7	3.50E-12	3.50E-12	3.50E-13	3.50E-13
RHS	High	ISLOCA	None	Low	4	1	-3	1.50E-11	1.50E-11	1.50E-11	1.50E-11
RHS	High	2ISLOCA	None	Low	0	1	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
RHS	Low	Class 2 LSS	N/A	Assume Medium	72	0	-72	2.45E-09	2.45E-09	1.01E-10	1.01E-10
TOTAL								**2.46E-09**	**2.46E-09**	**1.16E-10**	**1.16E-10**
CSL	High	LOCA	None	Low	4	1	-3	2.55E-12	2.55E-12	5.25E-13	5.25E-13
CSL	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSL	High	ISLOCA	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CSL	Low	Class 2 LSS	N/A	Assume Medium	9	0	-9	3.06E-10	3.06E-10	1.26E-11	1.26E-11
TOTAL								**3.09E-10**	**3.09E-10**	**1.31E-11**	**1.31E-11**
MSI	High	DTM – 1 / MSI – 1	None	Low	0	5	5	-2.50E-12	-2.50E-12	-2.50E-13	-2.50E-13
TOTAL								**-2.50E-12**	**-2.50E-12**	**-2.50E-13**	**-2.50E-13**
ICS	High	LOCA	None	Low	2	1	-1	8.50E-13	8.50E-13	1.75E-13	1.75E-13
ICS	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ICS	High	ILOCA – OC	None	Low	0	1	1	-1.70E-12	-1.70E-12	-7.00E-14	-7.00E-14
ICS	Low	Class 2 LSS	N/A	Assume Medium	10	0	-10	3.40E-10	3.40E-10	1.40E-11	1.40E-11
TOTAL								**3.39E-10**	**3.39E-10**	**1.41E-11**	**1.41E-11**
WCS	High	LOCA	None (FAC)	Low (High)	0	1	1	-8.50E-13	-8.50E-13	-1.75E-13	-1.75E-13
WCS	High	LOCA	None	Low	9	8	-1	8.50E-13	8.50E-13	1.75E-13	1.75E-13
WCS	High	ILOCA	None	Low	3	1	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
WCS	High	ILOCA - OC	None	Low	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL								**1.00E-12**	**1.00E-12**	**1.00E-13**	**1.00E-13**

Table 3.4-1 (Cont'd)
Risk Impact Analysis Results

System(1)	Safety Significance	Break Location(2)	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank(3)	SXI(4)	RIS_B(5)	Delta	w/ POD	w/o POD	w/ POD	w/o POD
DTM	High	LOCA	None	Low	0	3	3	-2.55E-12	-2.55E-12	-5.25E-13	-5.25E-13
DTM	High	ILOCA	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
DTM	High	MSD – 3	None	Low	0	3	3	-1.50E-12	-1.50E-12	-1.50E-13	-1.50E-13
DTM	High	DTM – 1 / MSI – 1	None	Low	0	3	3	-1.50E-12	-1.50E-12	-1.50E-13	-1.50E-13
TOTAL								**-5.05E-12**	**-5.05E-12**	**-7.75E-13**	**-7.75E-13**
GRAND TOTAL								**3.97E-09**	**4.42E-09**	**2.05E-10**	**2.86E-10**

Notes

(1) Systems are described in Table 3.1.

(2) The "Class 2 LSS" break location designation in Table 3.4-1 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).

(3) The failure potential rank for high safety significant (HSS) locations is assigned as "High", "Medium" or "Low" dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").]

(4) Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.

(5) Inspection locations selected for RIS_B purposes that are in the plant's augmented inspection program for IGSCC are subject to the requirements provided below dependent upon other damage mechanisms identified. These requirements dictate how these inspection locations are accounted for in the risk impact analysis.

(a) TACSC, TT, (IGSCC) and TT, (IGSCC) Damage Mechanism Combinations – these inspection locations are susceptible to thermal fatigue damage mechanisms in addition to IGSCC. In these cases, inspection locations selected for examination by both the IGSCC and RIS_B Programs should be included in both counts, but only those locations that were previously being credited in the Section XI Program and are now being credited in the RIS_B Program. The examination performed for IGSCC is judged adequate to have detected the other damage mechanisms subsequently identified by the RIS_B Program. For the RBS RIS_B application, one of these inspection locations was selected for examination per the plant's augmented inspection program for IGSCC and for RIS_B purposes due to the presence of other damage mechanisms. This inspection location was previously credited in the Section XI Program.

(a) None (IGSCC) Damage Mechanism – these inspection locations are susceptible to IGSCC only. In these cases, inspection locations selected for examination by both the IGSCC and RIS_B Programs should be included in both counts, but only those locations that were previously credited in the Section XI Program and are now being credited in the RIS_B Program. For the RBS RIS_B application, two of these inspection locations were selected for examination per the plant's augmented inspection program for IGSCC and are being credited for RIS_B purposes. These two inspection locations were previously credited in the Section XI Program.

Table 5
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank[2]			Vol/Sur	Sur Only	RIS_B	Other[3]
RPV	✓		LOCA	TASCS, TT, (IGSCC)	Medium (Medium)	B-F	3	3	0	1[4]	–
RPV	✓		LOCA	TASCS, TT	Medium	B-J	1	1	0	1	–
RPV	✓		LOCA	TT, (IGSCC)	Medium (Medium)	B-F	1	1	0	0	–
						B-J	1	1	0	0	–
RPV	✓		LOCA	None (IGSCC)	Low (Medium)	B-F	18	18	0	1[5]	–
						B-J	4	4	0	1[5]	–
RPV	✓		LOCA	None	Low	B-J	12	4	2	0	–
RDS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	48	4	0	0	–
RCS	✓		LOCA	None	Low	B-J	114	0	0	12	–
FWS	✓		LOCA	TASCS, TT, (FAC)	Medium (High)	B-J	25	15	0	5	–
FWS	✓		LOCA	TASCS, TT	Medium	B-J	28	16	0	0	–
FWS	✓		LOCA	TT	Medium	B-J	1	0	0	0	–
FWS	✓		ILOCA – FW	TASCS, TT	Medium	B-J	2	0	0	0	–
FWS	✓		ILOCA – OC	TASCS, TT	Medium	B-J	10	1	2	2	–
FWS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	23	4	0	0	–
MSS	✓		LOCA	None	Low	B-J	111	8	12	10	–
MSS	✓		ILOCA	None	Low	B-J	4	1	0	2	–
MSS	✓		ILOCA – OC	None	Low	B-J	17	11	0	3	–
MSS	✓		PLOCA	None	Low	B-J	2	0	1	0	–
MSS	✓		DTM – 1/MSI – 1	None	Low	B-J	8	0	8	0	–
MSS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	16	2	0	0	–

Table 5 (Cont'd)
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank[2]			Vol/Sur	Sur Only	RIS_B	Other[3]
SLS	✓		LOCA	None	Low	B-J	17	0	13	4	–
SLS	✓		PLOCA	None	Low	B-J	18	0	12	0	–
SLS	✓		ISLOCA	None	Low	B-J	5	0	2	0	–
SLS	✓		2ISLOCA	None	Low	B-J	23	0	3	3	–
CSH	✓		LOCA	TT	Medium	B-J	4	3	0	1	–
CSH	✓		LOCA	None	Low	B-J	1	1	0	0	–
CSH	✓		PLOCA	None	Low	B-J	10	0	0	0	–
CSH	✓		ISLOCA	None	Low	B-J	4	1	0	1	–
CSH		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	7	3	0	0	–
						C-F-2	122	10	0	0	–
RHS	✓		LOCA	None	Low	B-J	13	3	0	5	–
RHS	✓		PLOCA	None	Low	B-J	37	7	0	0	–
RHS	✓		ISLOCA	None	Low	B-J	14	4	0	1	–
RHS	✓		2ISLOCA	None	Low	B-J	2	0	0	1	–
RHS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	81	15	0	0	–
						C-F-2	785	57	0	0	–
CSL	✓		LOCA	None	Low	B-J	5	4	0	1	–
CSL	✓		PLOCA	None	Low	B-J	8	0	0	0	–
CSL	✓		ISLOCA	None	Low	B-J	5	1	0	1	–
CSL		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	5	4	0	0	–
						C-F-2	73	5	0	0	–
MSI	✓		DTM – 1/MSI – 1	None	Low	B-J	43	0	26	5	–

Table 5 (Cont'd)
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank[2]			Vol/Sur	Sur Only	RIS_B	Other[3]
ICS	✓		LOCA	None	Low	B-J	12	2	0	1	–
ICS	✓		ILOCA	None	Low	B-J	2	0	0	0	–
ICS	✓		ILOCA – OC	None	Low	B-J	5	0	0	1	–
ICS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	6	0	0	0	–
						C-F-2	159	10	0	0	–
WCS	✓		LOCA	None (FAC)	Low (High)	B-J	4	0	1	1	–
WCS	✓		LOCA	None	Low	B-J	81	9	15	8	–
WCS	✓		ILOCA	None	Low	B-J	4	3	0	1	–
						C-F-2	9	0	0	0	–
						Class 3	1	0	0	0	–
WCS	✓		ILOCA – OC	None	Low	B-J	2	0	0	1	–
						C-F-2	4	1	0	0	–
						Class 3	3	0	0	0	–
DTM	✓		LOCA	None	Low	B-J	27	0	15	3	–
DTM	✓		ILOCA	None	Low	B-J	3	1	2	0	–
DTM	✓		MSD – 3	None	Low	B-J	9	0	6	3	–
DTM	✓		DTM – 1/MSI – 1	None	Low	B-J	43	0	6	3	–

Notes

(1) Systems are described in Table 3.1.

(2) The failure potential rank for high safety significant (HSS) locations is assigned as "High", "Medium" or "Low" dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").]

(3) 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) to be credited toward the 10% requirement. RBS selected a 10% sampling without relying on IGSCC Program locations beyond those selected for RIS_B purposes either due to the presence of other damage mechanisms, or where no other damage mechanism is present. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals.

(4) This piping weld selected for examination per plant augmented IGSCC inspection program (Category C) and RIS_B purposes due to presence of other damage mechanisms.

(5) Two piping welds (Code Category B-F and B-J) selected for examination per the plant augmented IGSCC inspection program (Category C) and RIS_B purposes.

APPENDIX 1

River Bend Station PRA Model Capability for Use in Risk-Informed Inservice Inspection Applications

Introduction

The River Bend Station Probabilistic Risk Assessment (PRA) was initially developed in response to NRC Generic Letter (GL) 88-20, Individual Plant Examinations (IPE's). The IPE was submitted to the NRC via letter RBG-38077 dated 15 January 1993. The RBS IPE consisted of the Level 1 PSA, including addressing internal flooding, and the back-end analysis (Level 2) consistent with the requirements of GL 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities.* The NRC provided a Staff Evaluation in a letter dated October 17, 1996, which approved the RBS IPE. The Staff Evaluation concluded that the RBS IPE met the intent of GL88-20, that is, the RBS IPE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities for RBS.

Several PRA model updates have been completed on the RBS PRA since the IPE was submitted. These were done to maintain the PRA model reasonably consistent with the as-built as-operated plant. The scope of the updates was based on review of results, plant input to the model, updated plant and component failure and initiating event data, modifications to plant design, changes to plant procedures, as well as model enhancements. As part of major updates, an internal review of PRA model results is performed utilizing an expert panel composed of experienced personnel from various plant organizations, including Operations and Engineering.

The RBS PRA has been used as a basis for risk-informed submittals to the NRC, including Amendment 125 to the RBS license, approved by a Sept. 25, 2002, NRC letter. While the NRC did not review the RBS PRA, the staff had asked River Bend to perform various calculations, the results of which caused the NRC staff to agree with the overall assessment of the previous 1998 BWROG peer review that the RBS PRA was suitable for supporting risk-informed applications. The NRC found that the RBS PRA was adequate to support a RG 1.177 risk assessment.

The RBS PRA is currently at Revision 4.a. This minor revision was approved in March 2008 and implemented a model for cooling of the Control Building switchgear rooms. Core Damage Frequency (CDF) is predicted to be 3.55E-06 per year with a truncation limit of 1E-11/year.

The previous full revision of the RBS PRA was Revision 4. This revision was approved in September 2005. The Rev.4 CDF was calculated to be 1.94E-06/year with a truncation limit of 1E-10/year. The Large Early Release Frequency (LERF) for this model was calculated to be 2.53E-08/year. Model changes incorporated in the March 2008 update result in minimal impact (approximately 10%) upon LERF results.

As discussed below, the RBS PRA is more than adequate for this RI-ISI application. The PRA model used for this application has been evaluated against RG 1.200 Revision 1 and all gaps with respect to RG 1.200 have been evaluated. Most of the gaps are documentation issues. The few remaining gaps which could have been potentially applicable to use of the model for RI-ISI have been successfully addressed, as documented herein. It is concluded that the RBS PRA model fully supports the needs of this RI-ISI submittal, as the internal flooding calculation CDF and LERF results for each scenario are well below the risk thresholds for ASME Code Case N-716.

PRA Maintenance and Update

Entergy employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Entergy nuclear plants. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews.

The Entergy 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 Entergy Risk Management program. The overall Entergy Risk Management program, through procedure EN-DC-151, "PSA Maintenance and Update", 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 operating 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 plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is updated as part of the model revision process.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four to five years.

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

- Documentation of the PRA model, PRA products, and bases documents. (Procedures EN-DC-126 and EN-DC-151.)
- Guidelines for updating the full power, internal events PRA models for Entergy nuclear generation sites. (Procedures EN-DC-126 and EN-DC-151.)
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)). (River Bend Station Procedure ADM-0096)

Issues requiring action are entered into the Model Change Request (MCR) database which is controlled under EN-DC-151. These issues are prioritized in accordance with their significance for implementation into future PRA updates. Significant issues that are a result of errors are entered into the Entergy corrective action program under EN-LI-102.

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately four to five 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. Entergy performs regularly scheduled updates to the RBS PRA model. The next RBS PRA model update is scheduled for 2009-2010 and is expected to be approved in 2010.

PRA Self Assessment and Peer Review

RG 1.178 [10] specifies that a description of industry reviews performed on the PRA be included. Therefore, the independent PRA Peer Review and the ASME PRA Standard self-assessment review are included here along with the resolution of the review comments.

Several assessments of technical capability have been made, and continue to be planned for RBS PRA model. These assessments are as follows and further discussed in the paragraphs below.

- An independent PRA peer review was conducted under the auspices of the BWR Owners' Group in November 1998, following the industry PRA Peer Review process [1] which was the basis for the industry PRA Peer Review process NEI 00-02. The predecessor to the ASME PRA Standard Peer Review process was NEI-00-02 which identified the critical PRA elements and their attributes necessary for a quality PRA. This peer review included an assessment of the PRA model maintenance and update process.
- During 2005 and 2006, the RBS PRA model results were evaluated in the BWR Owners' Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process.
- In 2008, a self-assessment analysis [14] was performed against the available version of the ASME PRA Standard [2] and Regulatory Guide 1.200, Rev. 1 [9].
- As part of the PRA model update scheduled for 2009-2010 and expected to be approved in 2010, the self-assessment analysis will be updated to reflect pertinent changes to both the PRA Standard and Regulatory Guide 1.200.

<u>Internal Flooding Model</u>

The RBS Internal Flooding Analysis (IFA) was significantly upgraded to meet the requirements of RG 1.200 in 2009. This analysis was used in the subject RI-ISI evaluation to determine the High Safety-Significant (HSS) scope and as an input to Low Safety-Significant (LSS) scope Conditional Core Damage Probability (CCDP) values used in the risk impact assessment. This analysis is a substantial improvement over the previous IPE version of the Internal Flooding PRA (IFPRA). As an example, the IPE IFA conservatively used a 1E-7 screening value, and no scenarios resulted in CDF higher than the screening value. The current IFA has approximately 500 quantified scenarios; with CDF's ranging from about 2E-7 to less than 1E-12 range. Many of the scenarios have a CDF lower than the quantification truncation value used (1E-12). Further, due to the conservative simplifications required to analyze the large number of scenarios for an Internal Flooding PRA, these results are considered to be more conservative in inherent nature compared to the base Internal Events PRA. Other improvements including accounting for all liquid systems (e.g., Fire Protection) as flood sources, use of improved pipe rupture frequency data, and accounting for the frequency of breaks of up to a full guillotine break in the affected piping.

PRA Peer Review

An independent assessment of the RBS PRA, using the Self-Assessment Process developed as part of the Boiling Water Reactor Owners' Group PRA Peer Review Certification Program, was completed in 1998. Certification was performed by a team of independent PRA experts from U.S. nuclear utility PRA groups and PRA consulting organizations. This intensive peer review involved about two man-months of engineering effort by the review team and provided a comprehensive assessment of the strengths and limitations of each element of the PRA. The peer review concluded that the PRA was suitable for supporting risk-informed applications, such as Technical Specification changes, provided some enhancements were made, including four of High significance and numerous of less significance. Items evaluated through a peer review which were considered to not meet technical element requirements are documented with F&O's. Each F&O is provided with a level of significance (A: Extremely Important; B: Important; C: Desirable; D: Minor). The high significance items and all but a few of the lower significance items had been adequately addressed by the time the NRC approved an extension to the Allowed Outage Time (AOT) for RBS Emergency Diesel Generators via Amendment 125 to the RBS License (NRC letter dated Sept. 25, 2002).

In 2006, a summary of the disposition of Industry PRA Peer Review facts and observations (F&Os) arising from the BWROG PRA peer review for the RBS PRA model was documented as part of the statement of PRA capability for MSPI in the RBS MSPI Basis Document [4]. As noted in the RBS MSPI Basis Document, all of the "A" level F&Os identified in the PRA Peer Review were addressed and resolved in the RBS update model (Revision 4) approved for use in 2005. In addition, all of the "B" level F&Os were resolved within the model except for two. Entergy subsequently demonstrated that these two F&Os were insignificant for inclusion in the MSPI evaluation. Also noted in the MSPI Basis Document was the fact that, after allowing for plant-specific features, there are no MSPI cross-comparison outliers for RBS. Since these F&O's are insignificant with respect to MPSI, they also do not impact the ability to use the RBS PRA model in support of RIISI applications. Table 3 addresses the two remaining "B" level F&O items.

Self-Assessment

A 2008 Self-Assessment analysis for the RBS PRA model approved in 2005, including a March 2008 minor revision, was completed in December 2008. [14] This Self-Assessment analysis was performed against ASME PRA Standard [13] as endorsed by Regulatory Guide 1.200 Revision 1 [9]. This self-assessment analysis identified a list of 72 supporting requirements from the Standard which did not meet the Standard.

Table 1 provides the results of the River Bend Self-Assessment and identifies those ASME PRA Supporting Requirements that could require a sensitivity study or other disposition to more fully support the RI-ISI analysis. Table 2 provides a disposition of these identified gaps, including discussion of applicable sensitivity calculations.

The River Bend Station Internal Flooding PRA has recently been updated and meets ASME PRA Standard requirements for Internal Flooding (IF), including documenting compliance with all ASME standard supporting requirements. Entergy has also conducted an internal review of its processes to ensure they meet the ASME Standard supporting requirements for Maintenance and Update (MU).

Plant modifications which have not yet been incorporated into the RBS PRA and other potential issues identified since the model was revised have been reviewed. It is concluded that none of these items would significantly impact the ability of the RBS PRA to support a RI-ISI application.

Self-Assessment Interpretation

PRAs can be used in applications despite not meeting all of the Supporting Requirements (SRs) of the ASME PRA Standard. This is well recognized by the NRC and is explicitly stated in the ASME PRA Standard and RG 1.174. RG 1.174 states the following in Section 2.2.6:

> *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 in-service 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.*

Therefore, a RI-ISI PRA application requires no more than Capability Category I.

It is also acknowledged that for PRAs with SRs ranked as "Not Met", the PRA may be used for PRA applications but may require additional justification and support to allow their use.

Finally, it is judged that no PRA has Capability Category III for all of its SRs, nor is this currently expected as part of the NRC PRA Quality Program.

A review is performed of these Supporting Requirements (SRs) in Table 1 assessed as "Not Met" based on the self-assessment. The evaluation, disposition, and justification for the 72 "Not Met" supporting requirements is included in Table 1. The vast majority of the "Not Met" supporting requirements are documentation deficiencies rather than technical issues with the model itself. The importance of the Supporting Requirements to the RI-ISI application considers a spectrum of possible outcomes. For these "Not Met" SRs, they are dispositioned as follows:

- PA: Potentially applicable. Sensitivity Case may be required.
- NA: Not applicable. Areas of model that are not used in the RI-ISI evaluation
- NS: Not significant for the RI-ISI PRA application. Associated with areas that have no effect on the RI-ISI process or the risk significance determination, e.g.:

Areas that are clearly not significant to the quantification of the risk for welds are:

- Strictly documentation issues
- Parametric uncertainty analyses
- Modeling uncertainties not part of the LOCA scenarios

- NO: None. Meets at least Capability Category I
- SI: Specific Issue. A specific technical issue could influence the risk assessment as it affects one or more systems.

For those SRs that are "Not Met" and have a potential impact on the RI-ISI evaluation, a sensitivity case is defined that would demonstrate the capability of the PRA to appropriately characterize the PRA for use in RI-ISI to meet the ASME PRA Standard and RG 1.200.

Table 1 summarizes the disposition of the "Not Met" Supporting Requirements (SRs). As can be seen, approximately 66% of the "Not Met" SRs are related to documentation or modeling uncertainty assessments. These are expected to have no significant impact on the RI-ISI evaluation. Six SRs are identified for additional sensitivity cases. These six SRs subsume an additional two SRs. Table 2 provides a disposition for these gaps which demonstrates that the River Bend PRA provides a basis of sufficient quality to support the RI-ISI application. This includes documentation of sensitivity cases performed in response to the RG1.200 PRA self-assessment.

General Conclusions Regarding PRA Quality for RI-ISI:

The RBS 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 RI-ISI risk-informed licensing actions. In the risk-informed inservice inspection program at RBS, the EPRI Risk Informed ISI methodology (Reference 7) 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, element selection, and risk differential evaluation steps.

The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three fundamental components of 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 as illustrated below. Broad ranges are used to define these bins so that the impact of uncertainty is minimized and only substantial PRA changes would be expected to have an impact on the consequence ranking results. Further, the LSS classifications were conservatively binned as High Risk. None of the Medium Risk break locations challenged the High classification (Highest was 3.4E-04 for CCDP and 1.4E-05 for CLERP).

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. As a consequence, any PRA modeling uncertainties would be mitigated by the wide binning provided in the methodology. Additionally, conservatism in the binning process (e.g., as would typically be introduced through PRA attributes meeting ASME PRA Standard Capability Category I versus II) will tend to result in a larger inspection population. Secondly, the impacts of particular PRA consequence results are further dampened 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.

Thirdly, the EPRI RI-ISI methodology uses an absolute risk ranking approach. As such, conservatism in either the consequence assessment or the failure potential assessment will result in a larger inspection population rather than masking other important components. That is, providing more realism into the PRA model (e.g., by meeting higher capability categories) most likely would result in a smaller inspection population. These three facets of the methodology reduce the importance and influence of PRA on the final list of candidate welds.

Conclusion Regarding PRA Capability for RI-ISI:

The RBS PRA models are suitable for use in the RI-ISI application. This conclusion is based on:

- The PRA maintenance and update process 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 sensitivity of the EPRI RI-ISI process to PRA attribute capability beyond ASME PRA Standard Capability Category I.

As the PRA analysis continues to be improved during the 10-year interval, these results will be reviewed to determine which, if any, would merit RI-ISI specific sensitivity studies.

References

1. Boiling Water Reactors Owners' Group, *BWROG PSA Peer Review Certification Implementation Guidelines*, Revision 3, January 1997.
2. American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME RA-S-2002, New York, New York, April 2002.
3. U.S. Nuclear Regulatory Commission, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Draft Regulatory Guide DG-1122, November 2002.
4. River Bend MSPI Basis Document, RBS-SA-06-00001. Rev. 2, dated April 5, 2007.
5. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-Sb-2005, New York, New York, December 2005.
6. U.S. Nuclear Regulatory Commission Memorandum to Michael T. Lesar from Farouk Eltawila, "Notice of Clarification to Revision 1 of Regulatory Guide 1.200," for publication as a Federal Register Notice, July 27, 2007.
7. *Revised Risk-Informed Inservice Inspection Evaluation Procedure,* EPRI TR-112657, Revision B-A, December 1999.
8. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 1, November 2002.
9. NRC Regulatory Guide, RG 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Rev. 1, January 2007.
10. An Approach for Plant-Specific Risk-Informed Decision making for In service Inspection of Piping, Regulatory Guide 1.178, Rev. 1, US NRC, September 2003.
11. Alternative Piping Classification and Examination Requirements (Section XI, Division 1), Case N-716, Cases of ASME Boiler and Pressure Vessel Code, Approval Date: April 19, 2006.

12. Standard Review Plan for the Review of Risk-Informed Inservice Inspection of Piping, Standard Review Plan Office of Nuclear Reactor Regulation, NUREG-0800, September 2003.

13. ASME RA-Sb-2005, Addenda to ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, December 2005.

14. River Bend PRA Self-Assessment, Report C247080005-8620, Rev. 2, dated February 2, 2009.

15. Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs, EPRI, 1018427, December 2008.

16. PRA Quality for RI-ISI PRA Application, Report C247080010-8506 dated February 4, 2009.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #1	Initiators appropriately categorized and plant specific features accounted for. • The implementation of the ISLOCA evaluations in PRA-RB-01-002S08 has assumed that any 4 valves in a line qualify to allow screening a line from consideration. This is judged to be inconsistent with NSAC-154 and typical PRA practice. The isolation valves that are to be counted must: (1) be able to close against the differential pressure or (2) must be closed as their normal position The lines screened are the 3 LPCI and 1 LPCS injection lines. Low pressure rated pipe in the LPCI, SDC, and LPCS systems has been hydrostatically tested at rated RPV pressure. Discussion with Entergy [16] indicated that the low pressure pipe for LPCI and LPCI has been hydrostatically tested at normal operating pressures of the RPV (>1000 psig). Therefore, there is high confidence that the pipe is capable of withstanding any potential high pressure condition that could result from a failure of the high pressure to low pressure interface valves. Based on this plant unique resolution to the ISLOCA question, no additional sensitivity cases are needed. However, this information should be documented in the PRA to support the ISLOCA evaluation.. • Breaks outside containment in high energy lines beyond the 2nd isolation valve (Main Steam, FW, HPCS, RWCU, RCIC) are also in need of evaluation to ensure these are properly accounted for. (See IE-C4)	IE-A2	• **NS: Not Significant** Category I pipes are already placed in the High Safety Significant Category. Therefore, this SR is judged not to result in changing the HSS categorization of the interfacing lines inside the 2nd isolation valve. • **PA: Potentially Applicable** For lines outside the 2nd isolation valve in high pressure lines (e.g., main steam lines), use bounding estimates of the valve interface rupture, isolation capability, pipe rupture frequency, and the CCDP[(1)] to assess whether the pipe segments need to be added (See IE-C4 for breaks outside containment in high energy lines).

(1) CCDP = Conditional Core Damage Probability

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #1 (cont'd)			Code Case N-716 allows the use of bounding estimates for both the pipe rupture frequency and the CCDP. See Section 5 of Code Case N-716. With this allowance, the pipe segments that lie outside the 2nd isolation valve for ECCS, RWCU, MS, and FW can be conservatively evaluated to determine if these pipe segments have a sufficiently low risk contribution to be placed in the low safety significant category.
	• Reference Leg leak-down is an initiating event that can compromise multiple systems. This initiator should be identified for disposition. A loss of Ref. Leg is not assessed. (See SLI-8211 [1], SLI8218 [2], SLI8221 [3] for typical approaches used in BWR PRAs).		**NS: Not Significant** Reference leg leakdown should be explicitly evaluated as part of a PRA update but this is not required for RI-ISI application because it has a negligible influence on pipe break frequency and the overall risk profile.
Gap #2	Challenges to safe shutdown from power conditions occurring at power levels other than 100% power are included in the Initiating Events Assessment. Examples of non-applicable events which are not included: refueling events, external events. See plant specific initiating event data assessment. However, the exclusion of the loss of transformer event at 10% power from the plant specific assessment is contrary to IE-A5 requirement to incorporate such events. Exclusion of LERs should be reconsidered in light of the ASME SR IE-A5 to consider events if they could have occurred at power specifically April 11, 2003 outage #FO 03-02. The methodology section does not discuss how LP/SD events are addressed in the analysis.	IE-A5	• **NS: Not Significant** None of the events eliminated from RBS consideration would significantly influence the assessment of pipe failure frequencies. As such, the RI-ISI can be performed effectively without the resolution of this SR to a higher Capability Category.
Gap #3	Refer to SR IE-C4 regarding screening of excessive LOCA and assessment of Breaks Outside Containment (BOC) initiators.	IE-B4	See IE-C4.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #4	The resulting IE frequencies are in units of critical years; the posterior critical year values were not converted to reactor calendar year using the predicted plant availability as required by SR IE-C3. In the next PRA Update, update the Initiating Event Notebook (River Bend PSA-001) so that it addresses the expected future plant availability and how that relates to the availability used in the predictive IE frequency calculations.	IE-C3	• <u>NO: None</u> This change would decrease the initiating event frequency and resulting CDF and LERF. As such, the RI-ISI can be performed effectively without the resolution of this SR to a higher Capability Category. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #5	Initiating events within the screening criteria are retained. Initiating events are in general appropriately subsumed into the resulting initiating event categories. Therefore, the screening criteria in the ASME Standard are not explicitly used to eliminate initiators from consideration. The implementation of the ISLOCA evaluations in PRA-RB-01-002S08 has assumed that any 4 valves in a line qualify to allow screening a line from consideration. This is judged to be inconsistent with NSAC-154 and typical PRA practice. The isolation valves that are to be counted must: (1) be able to close against the differential pressure <u>or</u> (2) must be closed as their normal position The lines eliminated incorrectly are the 3 LPCI and 1 LPCS injection lines.	IE-C4	See IE-A2. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #5 (cont'd)	• For SDC Suction Line The probability of line failure given the interface fails should also be assessed probabilistically instead of setting to 1.0. • Vessel rupture has been excluded from consideration based solely on a frequency estimate. This is not consistent with SR IE-B4 and IE-C4 requirements. • Breaks Outside Containment (BOC) in high energy lines are not evaluated for their impact and frequency.		• **NS: Not Significant** – This would result in decreasing CDF. As such, the RI-ISI can be performed effectively without the resolution of this SR to a higher Capability Category. – Vessel rupture has negligible influence on the CDF – Category I pipes are already placed in the High Safety Significant Category. Therefore, this SR is judged not to result in changing the HSS categorization of the interfacing lines inside the 2nd isolation valve except as noted in IE-A2.
Gap #6	List of generic priors is consistent with "NRC Issue" expectation. However, the "NRC Resolution" for this SR also requires that differences be explained.	IE-C10	• **NS: Not Significant** Documentation related item to explain differences.
Gap #7	"Loss of intake" initiators and associated plant-specific issues not discussed in the IE analysis. RBS will want to document CR-RBS-2007-4447 loss of makeup event, for which there was a significant and quick down power, but for which the plant did not scram or go off-line.	IE-C11	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #8	All of the ISLOCA features are not explicitly addressed in the derivation of the ISLOCA frequency. Items (b) – (e) are not included in the ISLOCA analysis.	IE-C12	See IE-A2.
Gap #9	• Modeling uncertainties are not discussed or identified. • Peer review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	IE-D3	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. To be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #9 (cont'd)			EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #10	Insufficient information included to ascertain this, e.g., effect of RHR in SPC during power operation.	AS-B5a	• **<u>NS: Not Significant</u>** Subsequent Entergy input [16] indicated that: Many different alignments are considered in the PRA. The particular alignment for SPC initially running when LPCI is actuated is not normally an issue for BWR-6 plants. At best, this is a Specific Issue rather than a Potentially Significant issue. Informal documentation from 1998 indicates the water hammer issue was not considered an issue at RBS due to the difference in elevations between the high point in the line and the suppression pool lower limit was not sufficient to result in voiding. Thus, no significant water hammer would occur. This becomes a documentation issue that needs to be addressed under separate SRs.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #11	PRA-RB-01-002S01 provides some information. Event Trees that appear to be missing include (p. 58 of 78): • ISLOCA • ATWS • Internal Flood • The event trees are not presented in the Accident Sequence Calc PRA-RB-01-002S01 • The functional fault trees that relate the Event Tree Nodes to the systems supporting the functions are not presented. • The individual sequence descriptions are presented but not the event tree. There appears to be significant gaps in the documentation regarding how the model is assembled. The gaps are primarily in the following areas: • The connection between systems (which have relatively limited documentation) and the function used in the event tree nodes. Examples include the following: – LOSP in the transient event tree – X1: RPV depressurization for different sequences (e.g., initiators) ➢ Transients, Small LOCA, Med. LOCA – V2, V3 for transients or Large LOCA • The event trees do not specify the branch fault trees used in the quantification; e.g., – RCIC for 24 hours – RCIC for 4 hours – Other	AS-C1	• <u>**NS: Not Significant**</u> The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. Event Trees reviewed as part of Internal Flooding analysis. Internal Events PRA event tree confirmed to be appropriate for Internal Flooding.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #11 (cont'd)	• The treatment of sequence transfers is not clear. Are the successes carried forward with the transfers; if so, how? • Transfer 4 for transient p. 4 does not appear to be connected. • The small LOCA Seq. 42 appears to be truncated at 3E-8/yr. This would appear to be inappropriate. • For small LOCA, Seq. 9 & 10 appear to allow RCIC success for the 24 hr. mission time. This would require crew alignment of RHR within a short time (2-4 hrs) to ensure HCTL is not exceeded. • In addition, there could be an SORV. RCIC is not a success for a 24 hr. mission with an SORV.		• **NS: Not Significant** A sensitivity case to extend the truncation level to that used for other sequences is desirable but not required for RI-ISI. • **NA: Not Applicable** – Pipe inside containment (Class I) is already classified as HSS. This difference has no material effect on the pipe classification. – Restructure the SLOCA event tree to require depressurization and LP injection for SLOCA with RCIC and SORV with RCIC.
Gap #12	See Uncertainty Calculation (PRA-RB-01-002S13). The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	AS-C3	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #13	Not defined. The parameter basis for determining success or failure of a sequence is not defined • Core Damage – Core temperature • RPV – Pressure Capacity • Containment – Pressure Capacity – Hydrodynamic loading under ATWS conditions	SC-A2	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #14	Mission times are discussed in Accident Sequence Calculation PRA-RB-01-002S01. The mission times for failure to run calculations are assessed at 24 hours or less if specifically justified. Extending the FTR mission time beyond 24 hours for loss of DHR sequences is considered to be an unnecessary complication and does not affect PRA insights nor does it significantly affect its quantitative evaluation.	SC-A5	See SC-A1.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #14 (cont'd)	The evaluation of safe stable states in a PSA has generally involved the assessment of equipment operation and operator actions over an extended period of time. This extended period of time is nominally taken to be sufficiently long such that offsite resources can be brought to bear to mitigate or further prevent accident progression. The considerations that have dominated the choice of the mission time are as follows: • Equipment failure rates (failures/hour) are judged to be too conservative for times greater than a few hours of operation. • For times greater than a few hours, the ability to repair and recover equipment can compete with the failure rate such that there can be considered to be a steady state equilibrium condition reached. • For times greater than 24 hours, the TSC and EOF would be manned, and additional expertise could be available by phone or transported to these facilities. • For times greater than 24 hours, it is considered highly likely that offsite resources (e.g., equipment, power, vehicles) would be available as back-ups to primary methods of prevention and mitigation. • From a risk perspective, actual data from natural and man-caused disasters have indicated that public evacuations can be effectively carried out in time frames of less than 24 hours. Therefore, prevention of accidents through 24 hours of mission time have the largest potential for early health effects risk reduction.		

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #14 (cont'd)	• Finally, beyond time frames of 24 hours, "ad hoc" procedures can be written and reviewed to perform alignments and equipment usage that are not part of current plant practices or training. Such ad hoc procedures and equipment usage can cover such a wide spectrum of possibilities that it is judged not useful to develop all possible contingencies at this time. Based on the above considerations, it has been considered in past PSAs that it is to appropriate to use an equipment mission time of 24 hours. This consideration dictates the use of equipment "run" failure rates (per hour) coupled with a 24 hour mission time to calculate the "run" failure probability of equipment. This calculated "run" failure probability is then treated conservatively by applying this "run" failure probability as a failure that is postulated at time zero.		

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #15	The modeling has been performed consistent with the as-built, as-operated plant as of the PRA modeling freeze date with the possible exceptions: • There is no basis presented for SSW as a successful injection source except the assertion that the EOP-0001 specifies its use. This is not an adequate basis. Based on discussions with Entergy [16], the SW system has a discharge head of approximately 80 psig. This head should be more than adequate to provide significant flows to the RPV via the cross tie connection when the RPV is fully depressurized. Therefore, the primary gap is associated with providing these facts in the PRA documentation (e.g., success criteria notebook or SW system notebook). • The RCIC back pressure trip is taken at 25 psig and is then related to the pressure in containment at 25 psig. There is generally a lower containment pressure associated with the trip. This is usually approximately 6 psig less or 19 psig as measured in containment. Based on discussions with Entergy [16] the RCIC operation for a 24 hour mission time requires RHR success in suppression pool cooling. For SBO, the operation of RCIC is limited by a number of factors: – CST inventory – Suppression pool temperature – Battery capacity Because of the potential limitations, RCIC operation for 6 hours is credited. RCIC is not credited beyond 6 hours for the SBO conditions. The RCIC high turbine back pressure trip (even for cases of recirc seal LOCA) is expected to occur beyond 6 hours and therefore, is not limiting.	SC-A6	• **NS: Not Significant** – Credit for use of Service Water as a vessel injection source is part of BWR EOP's. Grand Gulf justification is that the Service Water flow is comfortably greater than LPCI injection flow. This is also the case for River Bend, where Standby Service Water pumps are rated at 7690 gpm per SAR Table 9.2-15 whereas LPCI injection flow credited in accident analyses is 4470 gpm per SAR Appendix 15.b. Per Process Flow Diagrams (0221.434-000-015, -016, -017), the SWP system is capable of supplying at least 5800 GPM to the input of the RHR HX, including flow through the tube side of both heat exchangers and return flow to the Standby Cooling Towers, which is considered a higher resistance flow path than injection to the reactor vessel. Per TSG-001 (Severe Accident Procedure Technical Support Guidelines), each Standby Service Water pump is capable of providing 9600 gpm of flow (best estimate calculations; on same basis, LPCI can provide 5650 gpm). In addition, based on discussions with Entergy [16], the SW system has a discharge head of approximately 80 psig. This head should be more than adequate to provide significant flows to the RPV via the cross tie connection when the RPV is fully depressurized. Therefore, the primary gap is associated with providing these facts in the PRA documentation (e.g., success criteria notebook or SW system notebook). As such, this "gap" is judged to be not significant for the assessment of the RI-ISI applications. – RCIC success for extended SBO events has properly considered the availability of RHR and the potential for a shorter RCIC mission time.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #15 (cont'd)	Success Criteria Tables in Appendix C of the Accident Sequence Calculation (PRA-RB-01-002S01) appear to have the following additional needs: • Specific deterministic calculations (e.g., MAAP) to demonstrate success of systems • Additional clarifications to identify the mission time for success of systems such as RCIC (e.g., see p. 6 of 78 of PRA-RB-01-002S01) as an example of a clarification on RCIC mission time. • The assumption that venting and 1 fan cooler is adequate for containment pressure control is not based on reaching a safe stable state. It is said to be "more stable" than the unstable case. However, the ability of containment coolers or venting to mitigate containment pressure rise is documented in Section 3.1 of PRA-RB-001-002S01. Given the significant effect each has in lengthening the time to containment failure, it would appear to be a reasonable engineering judgment that the two combined would result in preventing containment failure. • The failure of HPCS in large and medium LOCAs if SPC is unavailable does not appear to be explained in a clearly unambiguous fashion. It is not reflected in the success criteria table. • RCIC failure during loss of DHR may occur due to: (1) High turbine exhaust back pressure trip (~19 psig in containment) (2) Required depressurization to meet PSP or HCTL requirements in the EOPs. These failure modes of RCIC are not discussed in Section 3.12 Item D nor in the T1 and SBO Success Criteria Tables in Appendix C.		• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item. • **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item. • **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item. • **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item. • **NS: Not Significant** Based on discussions with Entergy [16], the RCIC operation for a 24 hour mission time requires RHR success in suppression pool cooling. For SBO, the operation of RCIC is limited by a number of factors: – CST inventory – Suppression pool temperature – Battery capacity

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #15 (cont'd)	• HPCS dependence on SPC for success under Large LOCA conditions needs to be identified in the Success Criteria Table.		Because of the potential limitations, RCIC operation for 6 hours is credited. RCIC is not credited beyond 6 hours for the SBO conditions. The RCIC high turbine back pressure trip (even for cases of recirc seal LOCA) is expected to occur beyond 6 hours and therefore, is not limiting. • **NS: Not Significant**. This is a documentation issue. The model is not being changed to address this item.
Gap #16	Limitations of codes are not discussed.	SC-B4	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #17	Comparative analysis with other plants is not performed for support.	SC-B5	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #18	Documented. Basis is not clearly documented to allow a Peer Review Team to independently assess the adequacy.	SC-C1	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #19	The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	SC-C3	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this
Gap #19 (cont'd)			Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This

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			evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #20	All alternate system alignments are not considered in the model development and documented in the System Notebooks.	SY-A5	• **NS: Not Significant** See AS-B5a.
Gap #21	Cannot ascertain whether the system boundary is appropriately drawn for each system.	SY-A6	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #22	Component model boundaries are not explained in the component data notebook and are not shown to be consistent with the data used.	SY-A8	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #23	System and functional success criteria are sequence dependent and time dependent but these are not explained in the PRA-RB-01-002S11.	SY-A11	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #24	Screening criteria from Supporting Requirement SY-A14 are not referenced or discussed as they may be used to limit consideration of low probability failure modes. See PRA-RB-01-002S11.	SY-A13	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #25	No indication that this criteria is met regarding incorporation of failure modes. See PRA-RB-01-002S11.	SY-A14	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #26	The following is developed as supporting documentation for River Bend: • A list of the PRA systems to consider for test and maintenance actions • Procedures reviewed, the potential test and maintenance actions associated with the procedures, and the disposition of the action (screened or evaluated). Not addressed are the following: • Identify T&M activities that require realignment of the system outside its normal operational or stand by status. • Rules for identifying and screening test and maintenance actions from the PRA. See PRA-RB-01-002S11.	SY-A15	N/A • **NS: Not Significant** This task is found not to have significant impact on the PRA model and results or the RI-ISI application. The pre-initiator HEP analysis is supportive of Capability Category II applications. (Pre-initiator HEPs included as required.)
Gap #27	The River Bend model does not include system dependencies on accident progression including isolations and trips under severe accident conditions; e.g., RCIC back pressure trip; L8 trip on ref. leg leakdown; MSIV closure interlock on low level and the bypass interface. See PRA-RB-01-002S11.	SY-A17	• **PA: Potentially Applicable** Provide sensitivity by setting RCIC failure to 1.0 for LOCAs, loss of DHR, and long term SBO sequences and MSIV closure for turbine trip ATWS to 1.0. Alternatively, update the model to address these two items.
Gap #28	Realistic functional requirements are not discussed to characterize system operation. See PRA-RB-01-002S11.	SY-A19	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #29	Repair appears to be used even if data is unavailable to support the repair probability. Estimated "repair/recovery" values from NUREG/CR-4550 no longer meet the latest expectation for PRA established in the ASME PRA Standard. Therefore, these repair events are considered potentially significant in establishing the risk baseline and not adequately supported by available data for: – EDC – PCS – DHR See PRA-RB-01-002S11 and Ref [16].	SY-A22	• **PA: Potentially Applicable** Perform a sensitivity evaluation to set all repair to 1.0 failure unless appropriate generic or plant specific data are available to support the quantification. (Excludes offsite AC and on-site AC power recoveries.) The ones of specific interest are: – ORA-EDC4HRS – ORA-PCS1HRS – ORA-PCS4HRS – ORA-DHRLT This was confirmed in Ref. [16].
Gap #30	See SY-B5. (Address room cooling with appropriate analysis.	SY-B6	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #31	Inventories of air, water, and cooling are not treated explicitly in the model documentation. Not discussed in a Dependency Notebook, individual System Notebooks or Event Sequence Notebook.	SY-B12	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #32	System Notebooks do not provide this information. The System Notebooks do not contain all the information requested by the ASME PRA Standard.	SY-C1	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #33	The System Notebooks or Component Data Notebook do not contain a significant fraction of the information requested by the ASME PRA Standard.	SY-C2	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #34	The system assumptions are clearly documented in PRA-RB-01-002S11. No uncertainty evaluation is performed. The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	SY-C3	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #35	The following are not available as supporting documentation for River Bend: • A list of the PRA systems to consider for test and maintenance actions • Rules for identifying and screening test, inspection, and maintenance actions from the PRA • A list of procedures reviewed, the potential test and maintenance actions associated with the procedures, and the disposition of the action (screened or evaluated). • T&M activities that require realignment of the system outside its normal operational or stand by status.	HR-A1	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #36	System Notebooks and System Manager Interviews offer the opportunity to include the identification of the work practices that could influence pre-initiators.	HR-A3	• **NS: Not Significant** In general, pre-initiators are not significant contributors to the BWR risk profile. In addition, the system manager interviews have marginal value in establishing the pre-initiators for the PRA. No additional action for RI-ISI is required.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS
OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #37	The HEP screening process in the River Bend pre-initiator evaluation is not identified consistent with ASME PRA Standard.	HR-B1	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #38	Dependent pre-initiator HEPs are addressed where multiple trains or functions are affected. Miscalibration dependencies using Figure 2 of the PRA-RB-01-002S03 appears to be too optimistic regarding the assignment of multiple miscalibration errors because it does not reflect: • Common measuring standards • Common crews • Common procedures The 1E-8 used for miscalibration is judged to be non-conservative and not supported by the THERP or ASEP methods. Miscalibration probabilities of 9E-16 in Table 2 are judged to be unsupportable and detract from the high quality of the RBS PRA. Miscalibration of Low Rx Pressure Signals (LPCI/LPCS interlock) is listed as negligible. This is contrary to HR SR-B2 and is judged to be unsupportable and detract from the high quality of the RBS PRA.	HR-B2	• **PA: Potentially Applicable** Perform a sensitivity calculation to set all miscalibrations of multiple channels to 3E-5.
Gap #39	Miscalibration is included. Miscalibration dependencies using Figure 2 of the PRA-RB-01-002S03 appears to be too optimistic regarding the assignment of multiple miscalibration errors because it does not reflect: • Common measuring standards • Common crews • Common procedures The 1E-8 used for miscalibration is judged to be non-conservative and not supported by the THERP or ASEP methods. Miscalibration probabilities of 9E-16 in Table 2 are judged to be unsupportable and detract from the high quality of the RBS PRA. Miscalibration of Low Rx Pressure Signals (LPCI/LPCS interlock) is listed as negligible. This is contrary to HR SR-B2 and SR-C3 and is judged to be unsupportable and detract from the high quality of the RBS PRA.	HR-C3	• **PA: Potentially Applicable** See HR-B2.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #40	Appropriate operator actions are included in HRA. For actions other than "skill of the craft", the incorporation of CR (Recovery HEPs) for non-proceduralized actions is generally not allowed without significant documentation support. This is not met.	HR-E2	• **PA: Potentially Applicable** Perform a sensitivity calculation to remove credit for all non-proceduralized recoveries and repairs unless there is explicit documentation provided to justify the assigned "recovery". (See SY-A22 for related SR.)
Gap #41	Plant specific MAAP calculations are not used to provide allowed times for crew response. Thermal hydraulic analyses appropriate for River Bend are not used to set time available. No generic analysis is presented or referenced to support allowable action times. Interface of success criteria and plant specific calculations: HEP – BA-SSWINJ (Section 5.2.5 in PRA-RB-01-002S03) allows 20 min. for SW alignment to prevent core damage. This appears to be in need of a clear definition of core damage (based on a measurable parameter) as required by SR SC-B2 and a method to calculate the parameter (e.g., RBS MAAP calculation). Neither of these two could be found. The 20 min. time allowed for a DBA LOCA is judged to be significantly longer than any other BWR reviewed by the BWROG during the BWROG certification process.	HR-G4	No effect on the pipe rupture effects on CDF or LERF are identified. As such, the RI-ISI can be performed effectively without the resolution of this SR to a higher Capability Category. • **SI: Specific Issue** There is some possibility that Service Water Importance can be influenced by this SR deficiency. Therefore, SW pipe segments need to be evaluated for this potential change in importance if this SR is resolved to be Capability Category I. This can be resolved by using bounding CCDP and initiating rupture frequencies if they could be determined. See SC-A6 for sensitivity case to subsume this item.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #42	Evidence that HEPs are reviewed for reasonableness is not presented. The calculated HEP derivations are not provided. Section 5.2.6 (PRA-RB-01-002S03) ADS recovery. The HEP of 1.2E-5 is lower than considered possible using the cause based and THERP approaches. RPT: For ATWS sequences subsumed into the ATWS model, manual RPT cannot be credited; in fact only RPT on high dome pressure can be credited. SSW cross tie for ATWS and LOCA response would appear to be optimistic especially considering that no deterministic calculation is available to support its flow rate and timing as adequate as the sole RPV injection source.	HR-G6	• **<u>NS: Not Significant</u>** This is a documentation issue. The model is not being changed to address this item.
Gap #43	The apparent separate treatment of HEPs, in-model recoveries, and ex-model recoveries needs to be better described and integrated. There needs to be an explanation of the combined HEP values that result from this treatment.	HR-I1	• **<u>NS: Not Significant</u>** The treatment of the combination of HEPs within a given cutset is assessed by Entergy. See Table 10 of the HRA Calculational Notebook. This evaluation was submitted by Entergy [16] and leads to the conclusion that the HEPs are adequately modeled for dependencies and may only require additional documentation to describe the process and display the results. This is a documentation issue. The model is not being changed to address this item.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #44	The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	HR-I3	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #45	PRA-RB-01-002S05 Rev. 0 Component boundaries are not explicitly provided in the Data Notebook.	DA-A1a	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #46	Data Notebook provides no discussion or indication that coincident maintenance was considered or evaluated. To be consistent with SR DA-C13, the PRA should include an examination of coincident outage times for redundant equipment (both intra- and inter-system) and incorporate the results into the modeling and documentation. However, it is judged that it is not practical to model all potential combinations of coincident maintenance unavailabilities, and that a review of maintenance experience would not be sufficient to allow the prediction of the dominant risk contributor combinations. As such, an approach to identify dominant risk contributor combinations based on knowledge of the accident sequences modeling, and model such combinations of coincident maintenance outages in the fault tree logic is judged prudent. A review of recent maintenance experience can be performed to identify events of coincident maintenance outages for these combinations to support probability estimation for the events.	DA-C13	• **NS: Not Significant** Since work practices call for a Protected Division philosophy, as documented in ADM-0096 on-line maintenance procedure, cross-divisional maintenance unavailabilities would be limited to emergent situations and thus coincident maintenance unavailabilities would be expected to be negligible.
Gap #47	Data Notebook provides no discussion or indication repair was considered or included in PRA.	DA-C14	• **PA: Potentially Applicable** See SY-A22.
Gap #48	Repair is not discussed in Data Notebook. Data is not sufficient to support the ASME requirement.	DA-C15	• **PA: Potentially Applicable** See SY-A22.
Gap #49	No indication from Data Notebook that past data is examined for applicability of the data. Data used ranges from: 1988 – 2003 1993 – 2003 1998 – 2003 Provide confirmation that the data used is applicable given that plant modifications and procedures have significantly changed the as-built, as-operated plant.	DA-D7	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #50	No discussion or indication of repair actions in Data Notebook.	DA-D8 (New NRC SR)	See SY-A22.

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OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #51	Assumptions are documented in Section 3.2 of PRA-RB-01-002S05, Rev. 0. The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	DA-E3	• <u>**NS: Not Significant**</u> The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #52	Modules are used, but the requirements of QU-B9 are not described in the RBS documentation. PRA-RB-01-005 and PRA-RB-01-002S02.	QU-B9	• <u>**NS: Not Significant**</u> This is a documentation issue. The model is not being changed to address this item.
Gap #53	No evidence of a sample of the significant accident sequences/cutsets have been appropriately reviewed is presented. PRA-RB-01-005 and PRA-RB-01-002S02.	QU-D1a	• <u>**NS: Not Significant**</u> This is a documentation issue. The model is not being changed to address this item.

TABLE 1
STATUS OF "NOT MET" SUPPORTING REQUIREMENTS OF THE ASME PRA STANDARD TO SUPPORT CODE CASE N-716 FOR RI-ISI

<table>
<tr><th>Gap</th><th>Description of Gap to Capability Category II</th><th>Applicable SRs</th><th>Importance to RI-ISI</th></tr>
<tr><td>Gap #54</td><td>The results of the PRA model reflect the sequence models, system models, success criteria, and the as-built, as-operated plant.

Appendix F of PRA-RB-01-002S02 give the PRA Quant results at 1E-11/yr truncation to be 1.197E-5/yr. This appears to be significantly different than the reported point estimate of 3.62E-6/yr. No explanation provided.
It also differs from Appendix D at 2.34E-6/yr.
Table 7 of PRA-RB-01-002S02 gives:
<table><tr><th>CDF(/yr)</th><th>Truncation (yr)</th></tr><tr><td>3.62E-6</td><td>1E-11</td></tr><tr><td>3.20E-6</td><td>1E-10 (Cutset Truncation)</td></tr></table>
PRA-RB-01-005 and PRA-RB-01-002S02.</td><td>QU-D1b</td><td><u>Action</u>

Provide the CDF and LERF along with its truncation.

Ensure that the results are properly vetted.

• <u>NS: Not Significant</u>
This is a documentation issue. The model is not being changed to address this item.</td></tr>
<tr><td>Gap #55</td><td>No evidence is found that dominant sequences are reviewed and found consistent with model, plant, procedures, and mutually exclusive file.
PRA-RB-01-005 and PRA-RB-01-002S02.</td><td>QU-D1c</td><td>• <u>NS: Not Significant</u>
This is a documentation issue. The model is not being changed to address this item.</td></tr>
<tr><td>Gap #56</td><td>No evidence is found that non-dominant sequences are reviewed and found appropriate.
PRA-RB-01-005 and PRA-RB-01-002S02.</td><td>QU-D4</td><td>• <u>NS: Not Significant</u>
This is a documentation issue. The model is not being changed to address this item.</td></tr>
</table>

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #57	The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	QU-E1	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #58	See PRA-RB-01-005 and PRA-RB-01-002S02. The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	QU-F4	• **NS: Not Significant** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #59	No evidence is found that the process for using CAFTA is part of CAFTA documentation.	QU-F5	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #60	No evidence is found that the limitations of model are documented.	QU-F6	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #61	Plant damage states are not defined. The end states of the Level 1 do not appear to distinguish among possible plant damage states. They are all listed as CD (core damage) or OK. This may create difficulty in the Level 2 assessment because of the significant differences in RPV, containment, and plant conditions given the various accident sequence paths to core damage in Level 1. PRA-RB-01-002S12 Rev. 0.	LE-A5	• **NS: Not Significant** This is a documentation issue. The model is not being changed to address this item.
Gap #62	Scrubbing of fission products is not explicitly modeled with MAAP	LE-C10	• **PA: Potentially Applicable**

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
	or equivalent deterministic code. The use of NUREG/CR-6595 satisfies Capability Category I for pool scrubbing assessments. However, scrubbing by Aux. Bldg is credited by assumption in reducing the radionuclide releases. A Decontamination Factor (DF) is developed based strictly on engineering judgment.		Perform a sensitivity calculation to eliminate all DF in the Auxiliary Building unless a calculation is available to support the assessment.
Gap #63	ISLOCA frequency development is plant specific and considers plant details. Discussion with Entergy [16] indicated that the low pressure pipe for LPCS and LPCI has been hydrostatically tested at normal operating pressures of the RPV (>1000 psig). Therefore, there is high confidence that the pipe is capable of withstanding any potential high pressure condition that could result from a failure of the high pressure to low pressure interface valves. Based on this plant unique resolution to the ISLOCA question, no additional sensitivity cases are needed. However, the ISLOCA evaluation does not include critical lines that need to be addressed.	LE-D3	• **<u>NS: Not Significant</u>** See IE-A2.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #64	Containment isolation failure probability is judged not to be adequate. Containment bypass is modeled in the LERF model, as documented in PRA-RB-01-002S12 Section 3.1.3. This also documents that Containment Isolation is modeled as part of the RBS System Notebook. A more detailed containment isolation notebook (CIS-22) had previously existed in the 1990s that contained much more detail than is currently in PRA-RB-01-002S11 / S20. RBS LERF model does include gates/models for failure to isolate containment vent and purge (KJB-Z31, KJB-Z33), reactor floor drains (KJB-Z35), and reactor equipment drains (KJB-Z38). Discussion in PRA-RB-01-002S12 implies a NUREG/CR-6595 LERF model, such as RBS, does model whether or not the Suppression Pool provides scrubbing of releases. (See LE-C10.) In addition, Entergy stated in Ref. [16] that the LERF model for RBS uses the assumptions from NUREG/CR-6595. The containment isolation notebook needs to be reissued and updated (CIS-22). The containment isolation analysis appears to be missing treatment of: • Pre-existing DW and containment failures PRA-RB-01-002S12 Rev. 0.	LE-D6	• **<u>NS: Not Significant</u>** No impact on CDF and minimal impact on LERF. Note that radionuclide scrubbing treatment in the Auxiliary Building is assessed in SR LE-C10.
Gap #65	Dominant contributors to LERF are not provided. Attachment 6, PRA-RB-01-002S12 Rev. 0.	LE-F1a	• **<u>NS: Not Significant</u>** This is a documentation issue. The <u>model</u> does not need to be changed to address this item.
Gap #66	Dominant contributors to LERF are not discussed for reasonableness. Attachment 6, PRA-RB-01-002S12 Rev. 0.	LE-F1b	• **<u>NS: Not Significant</u>** This is a documentation issue. The <u>model</u> does not need to be changed to address this item.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #67	Compliance with SR LE-F2 includes the LERF analysis and associated documentation which incorporates: • Quantitative sensitivity studies of the LERF analysis to reflect variations in the significant LERF contributors The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	LE-F2	• **<u>NS: Not Significant</u>** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #68	The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	LE-F3	• **<u>NS: Not Significant</u>** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The RBS approach is to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #69	Accident sequences, binning, and plant damage status are marginally documented. Additional documentation is judged to be necessary. This includes the nodal fault trees and their assumptions.	LE-G1	• **<u>NS: Not Significant</u>** This is a documentation issue. The model is not being changed to address this item.
Gap #70	Additional documentation is judged to be necessary. This includes the nodal fault trees and their assumptions.	LE-G3	• **<u>NS: Not Significant</u>** This is a documentation issue. The model is not being changed to address this item.

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Gap	Description of Gap to Capability Category II	Applicable SRs	Importance to RI-ISI
Gap #71	The Level 2 Notebook and the PSA Summary Notebook provide the following: • Quantitative sensitivity studies of the LERF analysis to reflect variations in the significant LERF contributors • Assumptions that could impact LERF results The RG 1.200 endorsement of the ASME PRA Standard has included a requirement to document the assumptions and sources of uncertainty associated with each PRA element. The NRC and the industry are working together to clarify what this means and to develop a structure that will satisfy these SRs. As of the performance of this self-assessment, this cooperative effort has not been completed. Peer Review expectations and an edict from the BWROG to declare these SRs as Not Met will in general lead to "Not Met" categorization by Peer Review Teams.	LE-G4	• **<u>NS: Not Significant</u>** The RBS PRA includes a substantial number of sensitivity calculations that demonstrate the range of uncertainties associated with specific assumptions and modeling uncertainties. These sensitivity studies are consistent with the expected NUREG-1855 approach to sensitivity analyses for evaluation of modeling uncertainties. The sensitivity cases are judged to be determined once the new NRC/EPRI guidance is available (e.g., NUREG-1855). However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results. EPRI [15] has reviewed the ASME PRA Standard Supporting Requirements for their applicability in ensuring the technical quality of RI-ISI risk-informed decisions. Based on the EPRI evaluation [15], this Supporting Requirement does not need to be met in order to adequately support the EPRI RI-ISI application methodology. This evaluation is also judged to apply to the methodology used in the Code Case N-716 RI-ISI.
Gap #72	The quantitative definition used for significant accident progression sequence is not included.	LE-G6	• **<u>NS: Not Significant</u>** This is a documentation issue. The model is not being changed to address this item.

TABLE 2
DISCUSSION OF POTENTIALLY APPLICABLE GAPS
TO REGULATORY GUIDE 1.200

Gap	Description of Gap to Capability Category II	Applicable SRs	Disposition
Gap #1	Initiators appropriately categorized and plant specific features accounted for. Internal flooding initiators to be completed by Entergy. The implementation of the ISLOCA evaluations in PRA-RB-01-002S08 has assumed that any 4 valves in a line qualify to allow screening a line from consideration. This is judged to be inconsistent with NSAC-154 and typical PRA practice. The isolation valves that are to be counted must: (1) be able to close against the differential pressure or (2) must be closed as their normal position The lines screened are the 3 LPCI and 1 LPCS injection lines. Low pressure rated pipe in the LPCI, SDC, and LPCS systems has been hydrostatically tested at rated RPV pressure. Discussion with Entergy [16] indicated that the low pressure pipe for LPCI and LPCI has been hydrostatically tested at normal operating pressures of the RPV (>1000 psig). Therefore, there is high confidence that the pipe is capable of withstanding any ISLOCA condition. Based on this plant unique resolution to the ISLOCA question, no additional sensitivity cases are needed for the LPCI, SDC, or LPCS lines. Breaks outside containment in high energy lines beyond the 2nd isolation valve (Main Steam, FW, HPCS, RWCU, RCIC) are also in need of evaluation to ensure these are properly accounted for. (See IE-C4) Reference Leg leak-down is an initiating event that can compromise multiple systems. This initiator should be identified for disposition. A loss of Ref. Leg is not assessed. (See SLI-8211 [1], SLI8218 [2], SLI8221 [3] for typical approaches used in BWR PRAs).	IE-A2	This gap is inherently addressed through the Internal Flooding PRA, which will determine the risk significance of the individual ECCS/etc. line segments. Use of the "Sensitivity" cases which account for the contribution of component rupture frequencies in addition to the EPRI pipe rupture frequencies will ensure that this ECCS / RCIC / FWS / RWCU piping will be appropriately characterized as part of the RBS Code Case N-716 RIISI program.

TABLE 2
DISCUSSION OF POTENTIALLY APPLICABLE GAPS TO REGULATORY GUIDE 1.200

Gap	Description of Gap to Capability Category II	Applicable SRs	Disposition
Gap #27	The River Bend model does not include system dependencies on accident progression including isolations and trips under severe accident conditions; e.g., RCIC back pressure trip; L8 trip on ref. leg leakdown; MSIV closure interlock on low level and the bypass interface. See PRA-RB-01-002S11.	SY-A17	River Bend includes failure of RCIC on loss of NPSH and other Level 8 trip. River Bend has incorporated a reference leg backfill system which would mitigate against a reference leg leak. The MSIV closure interlock is of lower importance to the River Bend PRA model since RBS has motor-driven, vice steam-driven, Main Feedwater Pumps. The RBS PRA fault tree models long-term RCIC failure on a loss of containment heat removal to account for the RCIC turbine trip on high exhaust back pressure. We believe that many of the issues have been addressed in the model and the gap may be primarily documentation in nature. Sensitivity cases on RCIC will be considered during the Sensitivity and Uncertainty Calculation base model update. Thus, upon review, this gap is considered to not impact the use of the River Bend PRA for support of RI-ISI applications and is considered to be a documentation or very small significance issue that will be reviewed and addressed during the next PRA model update.
Gap #29	Repair appears to be used even if data is unavailable to support the repair probability. Estimated "repair/recovery" values from NUREG/CR-4550 no longer meet the latest expectation for PRA established in the ASME PRA Standard. Therefore, these repair events are considered potentially significant in establishing the risk baseline and not adequately supported by available data for: • EDC • PCS • DHR See PRA-RB-01-002S11.	SY-A22	A sensitivity was performed with these recoveries set to 1.0. This run showed that CDF only increased by 26% (primarily due to the Power Conversion System (PCS) recovery), but this is for the baseline internal events CDF. Internal Flooding Analyses are insensitive to offsite power recovery actions since internal flooding scenarios do not lead to Loss of Offsite Power related initiators. We believe that the impact of this issue on the internal flooding results is not significant enough to change any conclusions made in this submittal
Gap #38	Dependent pre-initiator HEPs are addressed where multiple trains or functions are affected. Miscalibration dependencies using Figure 2 of the PRA-RB-01-002S03 appears to be too optimistic regarding the assignment of multiple miscalibration errors because it does not reflect: • Common measuring standards • Common crews • Common procedures The 1E-8 used for miscalibration is judged to be non-conservative and not supported by the THERP or ASEP methods.	HR-B2	Appendix B of the River Bend HRA calculation addresses the use of these low miscalibrations for Reactor Level and Pressure. The HEPs for Reactor Level and Reactor Pressure miscalibrations are considered negligible because there are multiple sets of these pressure and level signals (narrow range and wide range) for multiple system actuations. If all of the Reactor Level sensors for RCIC actuation are miscalibrated, the operators would notice the miscalibration based on a comparison to level sensors for HPCS, RPS, etc. None of these transmitters are calibrated at the same time and use different procedures. In addition, the level and pressure sensors would have to be grossly miscalibrated to fail to actuate ECCS components in a manner that would impact the success criteria.

TABLE 2
DISCUSSION OF POTENTIALLY APPLICABLE GAPS
TO REGULATORY GUIDE 1.200

Gap	Description of Gap to Capability Category II	Applicable SRs	Disposition
	Miscalibration probabilities of 9E-16 in Table 2 are judged to be unsupportable and detract from the high quality of the RBS PRA. Miscalibration of Low Rx Pressure Signals (LPCI/LPCS interlock) is listed as negligible. This is contrary to HR SR-B2 and is judged to be unsupportable and detract from the high quality of the RBS PRA.		Therefore, River Bend has justified the values used for miscalibration of these instruments. Further, Sensitivity analyses were conducted using the recommended 3.0E-05 value for miscalibration. These analyses demonstrated that there was no change in the ocntribution of flooding events to CDF or LERF.
Gap #40	Appropriate operator actions are included in HRA. For actions other than "skill of the craft", the incorporation of CR (Recovery HEPs) for non-proceduralized actions is generally not allowed without significant documentation support. This is not met.	HR-E2	The River Bend PRA has only one Control Room human action event for manually opening SWP-AOV599 for non-SBO events. The operators are trained on this valve for SBO events. This action is a "skill of the craft" based on the simplicity of the task and the training for the SBO sequence.
Gap #62	Scrubbing of fission products is not explicitly modeled with MAAP or equivalent deterministic code. Containment bypass accident sequences are not explicitly modeled. Scrubbing by Aux. Bldg is credited by assumption in reducing the radionuclide releases. A Decontamination Factor (DF) is developed based strictly on engineering judgment.	LE-C10	To determine the potential impact of containment bypass, a sensitivity case was run eliminating all credit for Auxiliary Building scrubbing in the LERF model. Event L2-ABSCRUB is set to "TRUE" vice assuming a 0.25 value. This resulted in a 205% increase in the LERF calculated due to flooding scenarios, to 6.31E-08/year. Even with this conservative modeling, the maximum LERF contributor (Feedwater breaks in the Main Steam Tunnel) had a calculated LERF of 3.62E-08, continuing to meet the 10^{-7} N-716 criteria. Thus, LERF results were demonstrated to meet the 10^{-7} criteria even with conservatively crediting no scrubbing due to the Auxiliary Building. In the determination of CLERP values greater than 10^{-5} for ASME Class II piping, no other scenarios with Class 2 pipe had CLERP values greater than 3E-06, thus revisions to the auxiliary building scrubbing model would be expected to have no impact on the identification of high CLERP Class 2 lines. Also, the 1E-08/year EPRI per system LERF criteria and the total ΔR_{LERF} would not be challenged by this conservative modeling (see Section 3.4.1). There is no impact on CDF associated with this modeling issue. Thus, this issue is concluded to have no impact on the use of the RBS PRA for RI-ISI applications.

TABLE 3
DISCUSSION OF OPEN "A" AND "B" PEER REVIEW FACTS AND OBSERVATIONS (F&O'S)

PEER REVIEW				IMPACT
ELEMENT ID / LEVEL OF SIGNIFICANCE / OBSERVATION			POSSIBLE RESOLUTION	
S-29	B	AOP-0050 (Attachment 7) Issues related to SBO and Suppression Pool Temperature Determination are the following: 1. Asserts Suppression Pool Temperature (SP/T) Indication is unavailable in Control Room for an SBO. 2. States Design Calculations Show SP/T does not exceed HCTL for 4 hours. 3. AOP-0050 directs using local temperature measurements. Related operator actions are not included in the HRA for control of RPV pressure below HCTL for SBO sequences using RCIC. The assertion that HCTL is not exceeded for 4 hours in an SBO is contrary to the design calculation G13.18.12.4 * 4 for the no operator action to depressurize case, which would be the situation if no other guidance is available. It is believed normal training is in place to allow initial RPV pressure reduction to 500 psig and this would increase margin to HCTL. This action is not included in PSA model.	Make PRA model consistent with AOP-0050. Consider whether feedback to AOP from PSA is desirable to make guidance on SBO response optimized to avoid HCTL and maintain RCIC operability.	An HRA was performed that evaluated the probability that the crew would not perform properly Attachment 7 of the Abnormal Procedure for Station Blackout (AOP-050). The HEE was included in the fault tree and a sensitivity analysis was run at 1E-13 truncation. The Birnbaums were compared to the model that does not include the HEE. None of the Birnbaums increased by a factor of 3. Typically they increased by less than 1%. Therefore it was concluded that the consequences of improper operator action during local temperature measurements directed by AOP-0050, Attachment 7, have an insignificant effect on MSPI calculation basis (or RIISI).
IE-6	B	Special initiating events were discussed in the initiating event notebooks. However the following initiating events are believed to be incorrectly screened from the quantification process: Breaks outside containment(BOC) 1)Main steam line, 2)Feedwater lines, 3)RCIC & RWCU Lines The analysis is completely adequate and appropriate for an IPE study or a study comparable	Include consideration of Level 2 and consider the lower truncation that may be used in the application of PSA in deciding on the retention of special initiators. This could be performed qualitatively if information is available to support diverse and	Break outside containment is not included in the current model. However, BOC, MSL breaks, and FWL breaks are expected to be a minimal risk impact to the CDF. The frequency of an unisolated large or medium LOCA would be about 1.0E-8/yr with no mitigation efforts: This is calculated with (INI-A+INI-S1)*MSIV CCF to close. The Large LOCA frequency is 3.2E-5/yr. The Medium LOCA frequency is 3.32E-5/yr. The common cause failure of MSIVs to close is 1.54E-4. Therefore, the BOC frequency is 1.0E-

		to NUREG-1150. Because of the potential for Level 2 impacts (e.g., LERF), there is not a good reason presented to eliminate these from quantification for a PSA that may be used for detailed applications. These sequences emphasize the need for the isolation function and the need to address the consequences of isolation failures. Applications involving the isolation function and potential large offsite releases could be affected. NUREG-1602 (DRAFT) has indicated similar concerns by stating that a steam generator event may have a relatively low contribution to the total CDF but may constitute a significant fraction of total large early releases.	redundant equipment available to mitigate the initiator.	8/yr. This is calculated without credit for recovery actions. Smaller breaks and leaks are significantly less important because smaller breaks and leaks damage less equipment. An unisolated break outside containment does not lead directly to core damage. The impact of this event is tied to an operator action for keeping the water level below the MSIV lines to prevent inventory loss other than from steam. HPCS and low pressure injection sources would be available to maintain level. This could become an inventory issue since steam would go to the turbine building instead of back to the suppression pool. When failure of the injection systems and failure of MSIVs is considered, the conditional core damage probability is 1E-3 or less. When these failures are in sequences with a BOC initiator, the CDF is 1E-11/yr or less. The impact of failure to model the BOC initiator is negligible and would have very limited impact on the importance of systems/components part of MSPI. This impact is not expected to exceed the LOCA CDF contribution. The ISLOCA evaluation addresses RCIC and RWCU line breaks. For Internal Flooding, these breaks outside containment have been explicitly modeled, thus this F&O has no impact upon use of the RBS PRA for the RIISI application.

Attachment 2 to

RBG-46922

Licensee-Identified Commitments

LICENSEE-IDENTIFIED COMMITMENTS

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

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Request for Alternative CEP-ISI-007 pertaining to the application of ASME Code Case N-663 will be withdrawn for use at RBS upon NRC approval of the RIS_B Program submittal.	X		Upon NRC approval of this request for alternative
Consistent with previously approved RI-ISI submittals, RBS 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 that time. Relief requests will be submitted per the guidance of 10 CFR 50.55a(g)(5)(iv).	X		Within one (1) year after the end of the interval
Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program.	X		Upon NRC approval of this request for alternative