Entergy

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RBG-47620

October 29, 2015

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

SUBJECT: License Amendment Request for change to Technical Specification; 5.5.13, to be extended to 15 years, Drywell Bypass Test Frequency to 15 Years and Type C Test Frequency to 75 Months
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the Technical Specifications (TS) for River Bend Station (RBS), Unit 1. The proposed changes would:

1. Allow for the extension to the ten-year frequency of the RBS Type A or Integrated Leak Rate Test (ILRT) that is required by TS 5.5.13 to be extended to 15 years on a permanent basis, Attachment 1.
2. Allow for the extension to the ten-year frequency of the RBS Drywell Bypass Test that is required by TS SR 3.6.5.1.3 to be extended to a 15 years on a permanent basis, Attachment 1.
3. Allow for the extension to the 60 month frequency of the RBS Type C or Local Leak Rate Test (LLRT) that is required by TS 5.5.13 to be extended to 75 months on a permanent basis, Attachment 2.

As required by the guidance in Nuclear Energy Institute (NEI) 94-01, Revision 3-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, appendix J", a risk assessment of this change is included in Attachment 3.

Attachment 4 contains markups of the proposed Technical Specifications and Attachment 5 contains markups of the proposed BASES for information.

The proposed change includes a new commitment. This commitment is summarized in Attachment 6.

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The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards consideration. The bases for these determinations are included in the attached submittal.

Entergy requests approval of the proposed amendment within one year of submittal. Once approved, the amendment shall be implemented within 60 days.

In addition to the requested changes to the ILRT and LLRT schedule, Entergy has identified an editorial correction to the Technical Specification Table of Contents. Section 3.6.5.2, Drywell Air Lock, is on page 3.6-63 versus 3.6-62. The proposed change (mark-up) is also included in Attachment 4.

Although this request is neither exigent nor emergency, your prompt review is requested. If you have any questions or require additional information, please contact Mr. J. A. Clark at (225) 381-4177

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 29, 2015.

Sincerely,

EO/JAC/bmb

Attachments:
1. Analysis of Proposed Technical Specification Change (ILRT/Drywell Bypass)
2. Analysis of Proposed Technical Specification Change (LLRT)
3. Risk Analysis
4. Proposed Technical Specification Changes (mark-up)
5. Proposed Technical Specification Bases Changes (mark-up) – For Information Only
6. List of Regulatory Commitments

cc: Regional Administrator
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NRC Senior Resident Inspector
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U. S. Nuclear Regulatory Commission
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RBF1-14-0159
LAR 2014-04

Attachment 1

RBG-47620

Analysis of Proposed Technical Specification Change

1 SUMMARY DESCRIPTION

Entergy Operations, Inc. (Entergy) is requesting an amendment to its Operating License for River Bend Station (RBS). This amendment would incorporate changes related to 10 CFR 50, Appendix J, Option B testing frequency based on Nuclear Energy Institute (NEI) 94-01, Revision 3-A.

2 DETAILED DESCRIPTION

Entergy is requesting an amendment to Operating License NPF-47, Docket No. 50-458 for RBS. The proposed change contained herein would revise Appendix A, Technical Specifications (TS), to allow extension of the Type A or Integrated Leak Rate Test (ILRT) frequency, required by TS 5.5.13, from ten (10) to fifteen (15) years on a permanent basis. This proposed TS change would also allow extension frequency of the Drywell Bypass leak rate test (DWBT) that is required by TS 3.6.5.1 from 10 to 15 years.

The specific proposed changes are listed in the following section.

Proposed Changes

The proposed amendment would change the wording of TS 5.5.13, "Primary Containment Leakage Rate Testing Program", to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", instead of Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program". This proposed change will also change the DWBT frequency requirement in SR 3.6.5.1.3 from 120 months (10 years) to 180 months (15 years) to align with the performance of the ILRT.

Current TS 5.5.13:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the next Type A test performed after the August 15, 1992, Type A test shall be performed no later than April 14, 2008.

Proposed TS 5.5.13:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 3-A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012.

Current Technical Specification Surveillance Requirement SR 3.6.5.1.3:

------Note-----
SR 3.0.2 is not applicable for extensions > 12 months
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24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit

AND

48 months following a test with bypass leakage greater than the bypass leakage limit

AND

120 months except that the next drywell leakage rate test performed after the June 24, 1994 test shall be performed no later than June 23, 2009.

Proposed Technical Specification Surveillance Requirement SR 3.6.5.1.3:

------Note-----
SR 3.0.2 is not applicable for extensions > 9 months
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24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit

AND

48 months following a test with bypass leakage greater than the bypass leakage limit

AND

180 months.

Attachment 4 provides a marked up copy of the proposed changes to the Technical Specifications described above.

Attachment 5 provides a marked up copy of the proposed changes to the Technical Specification Bases, for information.

3 Background

General

The testing requirements of 10 CFR 50, Appendix J provide assurance that leakage from the containment, including systems and components that penetrate the containment, does

not exceed the allowable leakage values specified in the TS. Furthermore, the requirements ensure that periodic surveillance testing of the containment, containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, the systems and penetrations. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests:

> Type A tests, intended to measure the containment overall integrated leakage rate; Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for containment penetrations; and Type C tests, intended to measure containment isolation valve leakage. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall integrated containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based' in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

NEI 94-01 History

RBS TS currently require that the Primary Containment Leakage Rate Program conform to RG 1.163 which endorses the use of NEI 94-01, Revision 0 for guidance in implementing extended surveillance frequencies allowed by 10 CFR 50, Appendix J, Option B.

NEI 94-01, Revision 2-A provides NRC approved guidance for the extension of ILRT surveillance frequencies up to 15 years. The NRC approval is subject to the limits and conditions provided in the Safety Evaluation Report (SER) for NEI 94-01, Revision 2. The NRC provides two sets of limits and conditions in the SER. Section 4.1 provides limits and conditions for NEI 94-01, Revision 2; Section 4.2 provides limits and conditions for Electric Power Research Institute (EPRI) Report number 1009325, revision 2.

NEI 94-01, Revision 3-A provides NRC approved guidance for the extension of LLRT surveillance frequencies up to 75 months. The NRC approval is subject to the limits and conditions provided in the SER for NEI 94-01, Revision 3 Section 4.0.

This request will change the RBS TS to require that the Primary Containment Leakage Rate Program conforms to NEI-94-01, Revision 3-A instead of RG 1.163. The change to NEI 94-01, Revision 3-A also assures that the limits and conditions of NEI 94-01, Revision 2-A for ILRT surveillance frequency extension are met.

NEI 94-01 does not directly address the RBS Drywell Bypass surveillance frequency extension to 15 years, it is used for guidance to support the requested Drywell Bypass surveillance frequency change.

Previous Submittals

RBS submitted a TS change on October 24, 1995 requesting the implementation of 10 CFR 50, Appendix J, Option B. This request was supplemented by letter dated November 22, 1995. The NRC approved this request as Amendment 84 issued by NRC letter on December 19, 1995. The NRC noted the proposed TS changes were in compliance with the requirements of Option B, and is consistent with the guidance in RG 1.163. With the approval of the amendment, RBS transitioned to a performance-based 10 year frequency for the Type A tests.

RBS submitted an Amendment request on May 30, 1995 to increase the DWBT surveillance interval from 18 months to 10 years with an increased testing frequency required if performance degrades. This request was supplemented by letters dated November 20, 1995 and December 12, 1995. The NRC approved this request as Amendment 87 on January 29, 1996.

RBS submitted an Amendment request to extend the ILRT interval one time from 10 to 15 years in a letter dated May 14, 2002 that was supplemented by letter dated December 20, 2002. This one-time extension was approved by the NRC as license Amendment 131 on March 5, 2003. RBS then submitted an amendment request on March 8, 2005 which was supplemented by letter dated January 17, 2006 to extend the test interval an additional 4 months beyond the 5 year extension already granted by the staff to the nominal 10 year interval. This extension was approved by the NRC as license Amendment 150 on February 9, 2006. RBS later submitted an amendment request on August 17, 2007 to extend the test interval to 15 years and 8 months which was approved by the NRC as Amendment 155 on December 3, 2007.

By application dated February 16, 2004 as supplemented by letters dated June 8, 2004 and August 26, 2004, Entergy requested a one-time change in the DWBT interval from 10 years to 15 years. The NRC granted this TS change under Amendment 144, which was approved on October 15, 2004.

4.0 Technical Evaluation

Containment System Description

RBS is a General Electric Boiling Water Reactor (BWR 6) reactor with a Mark III containment. The primary containment consists of a reinforced concrete cylindrical drywell structure, which houses the reactor system, and an annular shaped suppression pool, both enclosed within the freestanding steel primary containment structure. The primary containment vessel is enveloped by an open annulus area and a concrete shield building which serves as a secondary containment. The containment serves both as a suppression chamber and as a leak tight vapor barrier to protect against release of fission products in the event of a Loss of Coolant Accident (LOCA). The maximum allowable leak

rate (L_a) of the RBS freestanding steel containment is 0.325 percent of the total contained free volume per day at the calculated peak containment pressure resulting from the postulated Design Basis Accident (DBA).

The drywell serves to contain the steam released during a Loss of Coolant Accident (LOCA) and direct the steam to the suppression pool. The drywell serves as a buffer to the containment to ensure that the containment is not over-pressurized or directly pressurized following a LOCA. The drywell is a cylindrical reinforced concrete structure surrounding the reactor, the two recirculation loops, and other branch connections of the reactor coolant system. The Drywell is a Category I Seismic structure, with an internal design pressure of 25 psid, an external design pressure of 20 psid, and an internal design temperature of 330°F. A cylindrical weir wall, concentric with the lower portion of the drywell wall, forms the inner boundary of the suppression pool. Water in the annulus between the weir wall and the drywell wall is connected to the main body of the suppression pool by horizontal vent pipes directed radially through the lower portion of the drywell wall. The vents are spaced uniformly around the drywell circumference at three elevations. There are 43 vents in each of 3 rows for a total of 129 vents.

The secondary containment consists of the shield building and auxiliary building, and completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be operable, or that take place outside primary containment. Sixteen penetrations bypass the secondary containment. Leakage through these penetrations could leak to the environment without processing by the SGT or the fuel building filtration system.

Regulatory Requirements

As required by 10 CFR 50.54(o), the RBS containment is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J allows test intervals for Type A, Type B, and Type C testing to be determined by using a performance-based approach. Currently, the 10 CFR 50, Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0.

This proposed amendment would revise TS 5.5.13, "Containment Leakage Rate Testing Program," by replacing the reference to RG 1.163, "Performance-Based Containment Leak Test Program," with NEI topical report NEI 94-01, "Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012. This implementation document will be used by Entergy to develop the RBS performance-based leakage testing program in accordance with 10 CFR 50, Appendix J, Option B. The proposed amendment would also revise TS SR 3.6.5.1.3 frequency for the DWBT from 120 months to 180 months, to align it with the performance of the ILRT. The extension allowed by SR 3.0.2 will also be revised from 12 to 9 months

based upon the guidance in NEI 94-01.

This proposed interval extension for the primary containment ILRT and DWBT from 10 to 15 years would revise the next scheduled ILRT and DWBT from February 2018 to February 2023. This is approximately 15 years since the last ILRT and DWBT performance completed in February 2008. The previous performance of the ILRT and DWBT prior to 2008 was performed in August 1992. The extended interval of 15 years 8 months was approved in Amendment number 155, issued December 2007. The currently proposed change would allow successive ILRT and DWBTs to be performed at 15-year intervals, assuming acceptable performance. The performance of fewer ILRT and DWBT would result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

Limitations and Conditions

In the June 25, 2008 SER, the NRC concluded that NEI 94-01, Revision 2 describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2 is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SER.

In the June 8, 2012 SER, the NRC concluded that NEI 94-01, Revision 3 describes an acceptable approach for implementing the optional performance-based requirements of Option B (Type B and C tests were addressed), and found that NEI 94-01, Revision 3 is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SER.

The 2008 NRC SER Limitations and Conditions address the extension of the ILRT to 15 years and the 2012 SER Limitations and Conditions address increasing the test frequency of Type C LLRTs to 75 months. The following Table 4.0-1 lists the 2008 SER Section 4.1 Limitations and Conditions as well as compliance. Discussions on the 2012 SER Limitations and Conditions are contained in Attachment 2 of this submittal.

Table 4.0-1

Limitations and Conditions (Section 4.1 of Safety Evaluation Report of NEI 94-01 Rev 2)	**River Bend Station Compliance**
For calculating the Type A leakage rate, the licensee should use the definition in the NEI 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002.	Implementation of NEI 94-01, Rev 3-A will require use of the definition of "performance leakage rate" defined in Section 5.0 for calculating the Type A leakage rate when performing Type A tests.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	NEI-94-01, 3-A, Section 9.2.3.2 requires a general visual examination prior to each Type A test and at least 3 other outages before the ILRT. This will be scheduled in conjunction with or coordinated with examinations required by ASME Code, Section XI, Subsection IWE. A schedule of containment inspections is provided in Table 4.0-7.

Table 4.0-1	
Limitations and Conditions (Section 4.1 of Safety Evaluation Report of NEI 94-01 Rev 2)	**River Bend Station Compliance**
The licensee addresses the areas of the containment structure potentially subjected to degradation.	A general visual examination of accessible interior and exterior surfaces is conducted per the Containment Inservice Inspection Plan which implements the requirements of ASME, Section XI, Subsection IWE. No areas of potential degradation have been identified in the RBS containment structure.
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable.	The Entergy design change process will address any testing and inspection requirements following future major modifications to the containment structure. This process provides a disciplined approach for determining the program and system interfaces associated with design change. This process evaluates requirements pertaining to the ASME Containment In-Service Inspection, 10 CFR 50 Appendix J (Primary Containment Leak Rate Testing), and ASME Section XI programs.
The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.	RBS acknowledges and accepts this NRC staff position.
For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable to RBS.

Previous Test results

Integrated Leak Rate Testing (ILRT)

The last three ILRT tests for RBS had results of less than .75 L_a (0.24375 %/day by weight). Therefore, a test frequency of 15 years in accordance with NEI 94-01, Revision 3-A would be acceptable. The results of the last three ILRTs (as-found) are shown in Table 4.0-2 below.

The tests in 1989 and in 1992 were performed prior to implementing the Performance-Based Containment Leak Test Program and were not required to document an as-found

integrated leakage rate value, which includes leakage savings for any penetrations that were adjusted or reworked during the outage. The as-found ILRT value during 2008 was 0.2106 (% Containment weight per day) against a regulatory limit of 1.0 L_a (0.325 % Containment weight per day).

Table 4.0-2

Date	Leakage (% Containment weight per day)	Regulatory Limit - .75 L_a (% Containment weight per day)	Test Pressure (psig)
February 27, 2008	0.1939	0.24375	8.67
August 15, 1992	0.169	0.195	8.7
May 29, 1989	0.090171	0.195	8.22

Drywell Bypass Leakage Testing (DWBT)

Previous DWBT at RBS confirm that the drywell structure's leakage is acceptable with respect to the TS limit of 0.81 ft^2. Since the last three results were significantly less than the regulatory limit, a test frequency of 15 years would be acceptable. The results of the last three DWBT are as follows:

Table 4.0-3

Date	Drywell Bypass Summation (ft^2)	Current Regulatory Limit (ft^2)
February 27, 2008	0.0208	0.81
August 24, 1992	0.0188	0.81
June 4, 1989	0.00025	0.81

By letter dated December 12, 1995, RBS agreed to perform a qualitative assessment of drywell leak tightness once per operating cycle to support the change to a 10 year DWBT. The drywell leak tightness assessment involves trending drywell pressure versus containment pressure changes and observing the time it takes for the pressure to recover. This assessment provides a gross evaluation of the drywell integrity and reasonable assurance that the drywell can perform its safety function. This method for determining drywell leak tightness was approved by the NRC under Amendment 87, which increased the DWBT frequency from 18 months to 10 years, on January 29, 1996. The drywell leak tightness qualitative assessments since the last DWBT on February 27, 2008 had the following results:

Table 4.0-4

Date	**Pressure Drop (psid)**	**Elapsed Time (min)**
March 25, 2015	0.20	40
February 6, 2013	0.12	45
December 27, 2010	0.08	40
September 8, 2009	0.10	40
December 19, 2005	0.13	40

August 2, 2004	0.08	45

Note: The acceptance criteria is a pressure drop of ≤ 0.25 psid after approximately 32 minutes.

RBS will continue to perform this qualitative assessment of drywell leak tightness once per operating cycle to support a change to performing the DWBT every 15 years.

This change will be incorporated into specification SR 3.6.5.1.3. The extension allowed by SR 3.0.2 will also be revised from 12 to 9 months based upon the guidance in NEI 94-01.

Type B and C Testing Summation Margin

The as-found minimum pathway summation results and as-left maximum pathway summation results for Type B and Type C testing from the November 2004 outage and each successive outage are provided below. The Percentage of Regulatory Limit illustrates the margin between the Type B and Type C leakage rate summation and its acceptance criterion.

Table 4.0-5

Date	As-Found Min Path Leak Rate (sccm)	Percentage of Regulatory Limit	As-Left Max Path Leak Rate (sccm)	Percentage of Regulatory Limit	Regulatory Limit - .6 La (sccm)
March 2015	2,909.5	3.5%	12,144.5	14.6%	83,061
March 2013	6,701	8.1%	23,784	28.6%	83,061
February 2011	4,078	4.9%	16,889.9	20.3%	83,061
October 2009	5,034	6.1%	12,653	15.2%	83,061
February 2008	7,346	8.8%	15,983	19.2%	83,061
May 2006	7,363	8.9%	18,981	22.9%	83,061
November 2004	7,574	9.1%	17,842	21.5%	83,061

Secondary Containment Bypass Testing Summation Margin

The minimum pathway summation results and as-left maximum pathway summation results for Secondary Containment Bypass testing from the November 2004 outage and each successive outage are provided below. The Percentage of Regulatory Limit illustrates the margin between the Secondary Containment Bypass leakage rate summation and its acceptance criterion.

Table 4.0-6					
Date	**As-Found Min Path Leak Rate (sccm)**	**Percentage of Regulatory Limit**	**As-Left Max Path Leak Rate (sccm)**	**Percentage of Regulatory Limit**	**Regulatory Limit - TS (sccm)**
March 2015	974.8	6.7%	2,853.2	19.5%	14,600
March 2013	1,113	7.7%	2,777	19.0%	14,600
February 2011	1,536	10.5%	4,055.6	27.8%	14,600
October 2009	1,427	9.8%	8,653	59.3%	14,600
February 2008	1,803	12.3%	4,783	32.8%	14,600
May 2006	2,180	14.9%	9,857	67.5%	14,600
November 2004	1,373	9.4%	9,400	64.4%	14,600

CISI Code Inspections

The Containment Inservice Inspection (CISI) Plan implements the requirements of ASME Section XI, Subsection IWE 2001 Edition through the 2003 Addendum. The 10 year CISI Interval is divided into three periods. A visual inspection of accessible interior and exterior surfaces of the containment for structural deficiencies that may affect the containment leak tight integrity is performed once each period.

The CISI examinations performed in accordance with Subsection IWE satisfy the general visual examination requirements specified in 10 CFR 50 Appendix J, Option B. The identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix).

Therefore, the frequency of the examinations performed in accordance with the CISI program will satisfy the requirements of NEI 94-01, Revision 3-A, Section 9.2.3.2, to perform a general visual examination before the Type A test and during at least three other outages before the next Type A test if the interval is extended to 15 years.

The following Table illustrates the CISI to be performed prior to and between Type A tests. The scope of the visual inspections includes all interior and exterior accessible surfaces of the containment liner.

Table 4.0-7					
Inspection Interval	**Inspection Period**	**Period Start Date**	**Period End Date**	**Refuel Outage**	**Refuel Month/Year**
2	3		May 30, 2008	RF-14*1 *2	2-2008*1
3	1	May 31, 2008	June 1, 2011	RF-15*2 RF-16	Fall 2009 Spring 2011

3	2	June 1, 2011	June 1, 2015	RF-17*2 RF-18	Spring 2013 Spring 2015
3	3	June 1, 2015	November 30, 2017	RF-19*2	Spring 2017
4*4	1	December 01, 2017	November 30, 2020	RF-20*2	Spring 2019
4*4	2	December 01, 2020	November 30, 2024	RF-21 RF-22*2*3	Spring 2021 Spring 2023
4*4	3	December 01, 2024	November 30, 2027	RF-23 *2RF-24	Spring 2025 Spring 2027

*1Last ILRT Performed
*2CISI examination scheduled/performed
*3 15 Year ILRT Scheduled
*4The fourth Inspection Interval schedule has not been finalized

Conclusion

NEI 94-01, Revision 3-A describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. RBS is proposing to adopt the guidance of NEI 94-01, Revision 3-A for the 10 CFR 50, Appendix J testing program plan.

This request includes an associated extension to the DWBT to 15 years and the extension allowed by SR 3.0.2 will also be revised from 12 to 9 months based upon the guidance in NEI 94-01.

Based on the previous LLRT/ILRT tests conducted at RBS, supplemented by risk analysis studies, including the RBS risk analysis provided in Attachment 3, it may be concluded that extension of the containment ILRT interval from 10 to 15 years represents minimal risk when performed in accordance with 10 CFR 50, Appendix J, Option B and the guidance of NEI-94-01, Revision 3-A.

5.0 Confirmatory Analysis

Methodology

An evaluation has been performed to assess the risk impact of extending the RBS ILRT interval from the current 10 years to 15 years. This plant-specific risk assessment followed the guidance in NEI 94-01, Revision 3-A, the methodology outlined in EPRI TR-104285, August 1994, TR-1009325 Revision 2-A, and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implication of corrosion-induced leakage of steel containment liners going undetected during the extended ILRT interval was also used for sensitivity analysis.

In the June 25, 2008 SER, the NRC concluded that a 15 year extension to the Type A ILRT interval was acceptable and that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing in a proposal to amend TS to extend the ILRT surveillance interval to 15 years. This approval was subject to the limitations and conditions noted in Section 4.0 of the SER. The following Table 5.0-1 lists the SER Section 4.2 Limitations and Conditions and a description of how the RBS analysis complies with those four limitations and conditions

Table 5.0–1

Limitations and Conditions of Risk Assessment	**RBS Compliance**
The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.	The technical adequacy of the RBS PRA and consistency with the RG 1.200 requirements relevant to the ILRT extension are summarized below and detailed in Appendix A of Attachment 3.
The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SER. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point. While acceptable for this application, the NRC staff is not endorsing these threshold values for other applications. Consistent with this limitation and condition, EPRI Report No. 1009325 will be revised in the "-A" version of the report, to change the population dose acceptance guidelines and the CCFP guidelines.	The RBS risk evaluation is summarized below and described in detail in Attachment 3. The results of that evaluation demonstrates that the estimated risk increase is small and consistent with the criteria discussed in the SER.
The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 L_a.	The RBS analysis used a pre-existing containment leak rate of $100L_a$ to calculate the increase in population dose for the large leak rate accident case (EPRI Class 3b) (Attachment 3, Section 3.0).
A LAR is required in instances where containment over-pressure is relied upon for ECCS performance.	Containment overpressure is not relied upon for ECCS performance (Attachment 3, Section 6.3).

PRA Quality

The risk assessment performed for the RBS ILRT extension request is based on the current Level 1 and Level 2 PRA model of record, which was released in April 2011. A discussion of the Entergy model update process, the peer review performed on the RBS model, the results of that peer review and the potential impact of peer review findings on the ILRT extension risk assessment are provided in Attachment 3, Appendix A, Section A.2.

Summary of Plant-Specific Risk Assessment Results

The findings of the RBS risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval to one in 15 years is small. The RBS plant-specific results for extending the ILRT interval to 15 years, taken from Attachment 3, Section 7.0, Conclusions, are summarized below.

1. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small" changes in risk as resulting in increases of Core Damage Frequency (CDF) below 1.0E-06/yr and increases in Large Early Release Frequency (LERF) below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension has no impact on CDF for RBS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included is 2.51E-08/yr (see Attachment 3, Table 6.1-1), which is within the small change region of the acceptance guidelines in RG 1.174. In using the EPRI Expert Elicitation methodology, the change is estimated as 3.95E-09/yr (see Attachment 3, Table 6.2-2), which is within the very small change region of the acceptance guidelines in RG 1.174.

2. The change in dose risk for changing the Type A test frequency and the DWBT interval from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences, is 7.22E-03 person-rem/yr using the EPRI guidance with the base case corrosion case (Attachment 3, Table 5.6-1). The change in dose risk drops to 3.39E-03 person-rem/yr when using the EPRI Expert Elicitation methodology (Attachment 3, Table 6.2-2).

3. The increase in the conditional containment failure frequency from the three-per-ten years interval to once-per-fifteen years including corrosion effects using the EPRI guidance (see Table 5.6-1) is 1.15%. This value drops to about 0.34% using the EPRI Expert Elicitation methodology (see Attachment 3, Table 6.2-2). This is below the acceptance criteria of less than 1.5% defined Attachment 3 in Section 1.3.

4. To determine the potential impact from external events, a bounding assessment from the risk associated with external events utilizing the latest information from various sources was performed. As shown in Attachment 3 Table 5.7-3, the total increase in LERF for RBS due to internal events and the bounding external events assessment is 3.16E-07/yr. This value is in Region II of the RG 1.174 acceptance

guidelines. As also shown in Table 5.7-3, the other acceptance criteria for change in population dose and change in Conditional Containment Failure Probability (CCFP) are also still met when the other hazard groups are considered in the analysis.

5. As shown in Attachment 3, Table 5.7-4, the same bounding analysis indicates that the total LERF from both internal events and the other hazard groups is 7.0E-07/yr for RBS, which is less than the RG 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).

6. Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for RBS.

Therefore, increasing the ILRT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be risk significant. Details of the RBS risk assessment are contained in Attachment 3.

6.0 REGULATORY EVALUATION

Entergy has evaluated the safety significance of the proposed change to the RBS TS which revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program" and SR 3.6.5.1.3 to allow a permanent extension to the frequency of the Type A and Drywell Bypass tests to 15 years. The proposed changes have been evaluated according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject change does not involve a Significant Hazards Consideration, as discussed below

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The accidents previously evaluated in the Updated Safety Analysis Report (USAR) that could be potentially impacted by the proposed amendment change are the LOCA Inside Containment and the Fuel Handling Accident.

   In order to affect the frequency of occurrence of an accident, the change has to affect an accident initiator. There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause an accident LOCA; however, it is evaluated without the cause being identified because it provides an upper limit estimate to the effect of pipe breaks. A fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto stored fuel bundles. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment from postulated accidents. Changing the frequency of leakage rate testing has no impact upon the likelihood of a LOCA or of a Fuel Handling Accident. The testing requirements to periodically

demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment adopts the NRC accepted guidelines of NEI 94-01, Revision 3-A for development of the RBS performance-based testing program for the Type A and Drywell Bypass testing. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components would limit leakage rates to less than the values assumed in the plant safety analyses.

The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. Entergy has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The use of the NRC accepted NEI 94-01, Revision 3-A does not alter any condition such that any other accidents previously not considered credible should become credible. No new structures, systems, or components are being added or altered by the proposed amendment, therefore no new failure modes can be postulated that would create the possibility of a different type of accident. There are no accident initiators created or altered by the proposed amendment. The proposed changes would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses.

Based on the above discussion, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the development of the RBS performance-based leakage rate testing program, and establishes a 15-year interval for the performance of the Primary Containment ILRT and DWBT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A.

Containment Inservice Inspections (CISI) performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current RBS PSA model concluded that extending the ILRT test interval from 10 years to 15 years results in a very small change to the RBS risk profile.

Based on the above discussion, the proposed amendment does not involve a significant reduction in a margin of safety.

Entergy concludes that the proposed amendment to the RBS TS presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of 'no significant hazards consideration' is justified.

7.0 Environmental Considerations

The proposed changes to the RBS TS do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

8.0 PRECEDENTS

This request is similar in nature to the following license amendments, which have been approved by the NRC:

1. Nine Mile Point Nuclear Station, Unit 2 – Issuance of Amendment Re: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650, Accession Number ML100730032) approved March 30, 2010.

2. Arkansas Nuclear One, Unit 2 – Issuance of Amendment Re: Technical Specification Change to Extend the Type A Test Frequency to 15 years (TAC No. ME4090, Accession Number ML110800034) approved April 7, 2011.

3. Palisades Nuclear Plant – Issuance of Amendment to Extend the Containment Type A Leak Rate Test Frequency to 15 Years (TAC No. ME5997, Accession Number ML120740081) approved April 23, 2012.

4. Surry Power Station, Units 1 and 2 - Issuance of Amendment Regarding the Containment Type A and Type C Leak Rate Tests (TAC Nos. MF2612 and MF2613, Accession Number ML14148A235) approved July 3, 2014.

5. Beaver Valley Power Station, Units 1 and 2 - Issuance of Amendment Regarding License Amendment Request to Extend Containment Leakage Rate Test Frequency (TAC Nos. MF3985 and MF3986, Accession Number MIL15078A058) approved April 8, 2015.

Attachment 2

RBG-47620

Analysis of Proposed Technical Specification Change

75 Month Local Leak Rate Test Frequency

1.0 SUMMARY DESCRIPTION

Entergy Operations, Inc. (Entergy) is requesting an amendment to its Operating License for RBS (RBS). This amendment would incorporate changes related to 10 CFR 50, Appendix B testing frequency based on Nuclear Energy Institute (NEI) 94-01, Revision 3-A.

2.0 DETAILED DESCRIPTION

Entergy is requesting an amendment to Operating License NPF-47, Docket No. 50-458 for RBS. The proposed change contained herein would revise Appendix A, Technical Specifications (TS), to increase the maximum interval extension for eligible Type C tested valves from 60 months to 75 months.

The specific proposed changes are listed in the following section.

Proposed Changes

The proposed amendment would change the wording of TS 5.5.13, "Primary Containment Leakage Rate Testing Program", to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," instead of Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program".

Current TS 5.5.13:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the next Type A test performed after the August 15, 1992, Type A test shall be performed no later than April 14, 2008.

Proposed TS 5.5.13:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 3-A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012.

3.0 BACKGROUND

General

The testing requirements of 10 CFR 50, Appendix J provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. Furthermore, the requirements ensure that periodic surveillance testing of the containment, containment

penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, the systems and penetrations. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

Appendix J identifies three types of required tests: Type A tests, intended to measure the containment overall integrated leakage rate; Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for containment penetrations; and Type C tests, intended to measure containment isolation valve leakage. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall integrated containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based, Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based' in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

NEI 94-01 History

RBS TS currently require that the Primary Containment Leakage Rate Program conform to RG 1.163, which endorses the use of NEI 94-01, Revision 0 for guidance in implementing extended surveillance frequencies allowed by 10 CFR 50, Appendix J, Option B.

NEI 94-01, Revision 2-A provides NRC approved guidance for the extension of ILRT surveillance frequencies up to 15 years. The NRC approval is subject to the limits and conditions provided in the Safety Evaluation Report (SER) for NEI 94-01, Revision 2. The NRC provides two sets of limits and conditions in the SER. Section 4.1 provides limits and conditions for NEI 94-01, revision 2; Section 4.2 provides limits and conditions for Electric Power Research Institute (EPRI) Report number 1009325, revision 2.

NEI 94-01, Revision 3-A provides NRC approved guidance for the extension of LLRT surveillance frequencies up to 75 months. The NRC approval is subject to the limits and conditions provided in the SER for NEI 94-01, Revision 3, Section 4.0.

This LAR will change the RBS TS to require that the Primary Containment Leakage Rate Program conformance to NEI-94-01, Revision 3-A instead of RG 1.163. The change to NEI 94-01, Revision 3-A also assures that the limits and conditions of NEI 94-01, Revision 2-A for ILRT surveillance frequency extension are met.

Previous Submittals

By letter dated October 24, 1995, RBS submitted a TS change that was supplemented by letter dated November 22, 1995 requesting the implementation of 10 CFR 50, Appendix J, Option B. The NRC approved this request as Amendment 84 issued in NRC letter on December 19, 1995. The NRC noted the proposed TS changes were in compliance with the requirements of Option B and are consistent with the guidance in RG 1.163. With approval of the amendment, RBS transitioned to a performance-based 120 month maximum frequency for the Type B tests and 60 month maximum frequency for Type C tests.

4.0 Technical Evaluation

Containment System Description

RBS is a General Electric Boiling Water Reactor (BWR 6) reactor with a Mark III containment. The primary containment consists of a reinforced concrete cylindrical drywell structure, which houses the reactor system, and an annular shaped suppression pool, both enclosed within the freestanding steel primary containment structure. The primary containment vessel is enveloped by an open annulus area and a concrete shield building which serves as a secondary containment. The containment serves both as a suppression chamber and as a leak tight vapor barrier to protect against release of fission products in the event of a Loss of Coolant Accident (LOCA). The maximum allowable leak rate (L_a) of the RBS freestanding steel containment is 0.325 percent of the total contained free volume per day at the calculated peak containment pressure resulting from the postulated DBA.

The drywell serves to contain the steam released during a LOCA and direct the steam to the suppression pool. The drywell does not serve as a barrier to fission product release, but more as a buffer to the Containment to ensure that the Containment is not over-pressurized or directly pressurized following a LOCA. The drywell is a cylindrical reinforced concrete structure surrounding the reactor, the two recirculation loops, and other branch connections of the reactor coolant system. The drywell is a Category I Seismic structure, with an internal design pressure of 25 psid, an external design pressure of 20 psid, and an internal design temperature of 330°F. A cylindrical weir wall, concentric with the lower portion of the drywell wall, forms the inner boundary of the suppression pool. Water in the annulus between the weir wall and the drywell wall is connected to the main body of the suppression pool by horizontal vent pipes directed radially through the lower portion of the drywell wall. The vents are spaced uniformly around the drywell circumference at three elevations. There are 43 vents in each of 3 rows for a total of 129 vents.

The secondary containment consists of the shield building and auxiliary building, and completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines

penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be operable, or that take place outside primary containment.

Sixteen penetrations bypass the secondary containment. Leakage through these penetrations would be released to the environment, bypassing the Standby Gas Treatment system. Thirteen of these penetrations are eligible for interval extension under 10 CFR 50, Appendix J, Option B. The leakage rates for these penetrations are summed separately and have an acceptance criterion, independent of L_a, in the TS (SR 3.6.1.3.9). Frequency extension beyond 60 months for these valves will meet the requirements of NEI 94-01, Revision 3-A as applicable (i.e. TS limit in lieu of $.6L_a$).

Forty two electrical penetrations and 18 penetration bellows are Type B tested. All Type B electrical penetrations and penetration bellows are performance-based components and are qualified for 120 month testing frequency. These Type B testing frequencies are unaffected by this LAR.

RBS has 104 valves that are Type C tested and eligible for performance-based extended testing intervals. All are currently qualified for a 60-month testing frequency with the following exceptions:

E51-MOVF063 The frequency was reset to 30 months in RF-13, April 2006, because the as found LLRT was not performed prior to maintenance work to lower stem thrust. The as found LLRT failed during RF-14, January 2008. The valve seat was reworked in RF-15, September 2009, with an as-left LLRT result of 338 sccm. In RF-16, January 2011, the as-found leak rate jumped to 11,020 sccm. In RF-17, February 2013, as-found leak rate was 9,980 sccm. Corrective action was taken during RF-17 by increasing the actuator torque switch setting to try to improve performance. The as-left leak rate deteriorated to 12,010 sccm. In RF-18, February 2015,, as-found leak rate was 15,943 sccm. Corrective action was taken during RF-18 by replacing the disc and machining the seat to improve performance. The as-left leak rate improved to 2,280 sccm. The test frequency remains at 30 months.

E51-MOVF064 The frequency was reset to 30 months during RF-16 because the valve exceeded its administrative limit of 1600 sccm. In RF-17 the valve again exceeded its administrative limit of 1600 sccm. To improve performance, the valve disc and stem were replaced and the as-left result improved to 240 sccm. The as-found test result in RF-18 was 200 sccm, which is below its administrative limit. If the valve is within its Administrative limit in RF-19, scheduled for January 2017, the frequency can be reset to 60 months.

SVV-V31 The frequency was reset to 30 months during RF-16 because the valve exceeded its administrative limit of 300 sccm. The valve was reworked resulting in an as-left of 105 scccm. The as-found test result in RF-17 was 20 sccm, which is below its administrative limit. In RF-18, the as-found leak rate was 4,600 sccm. To improve performance, the valve was

replaced and the as-left result improved to 21.2 sccm. The test frequency remains at 30 months.

WCS-RV144 This relief valve is replaced for the IST program on a frequency that precludes test interval extension. The test frequency remains at 30 months.

WCS-RV154 This relief valve is replaced for the IST program on a frequency that precludes test interval extension. The test frequency remains at 30 months.

The following components were limited to 30-month test frequency in the previous cycle but were extended to a 60-month frequency after RF-18:

HVN-MOV128 The frequency was reset to 30 months during RF-16 because the valve was repacked prior to the performance of an as-found leak rate test. The as-found test result in RF-17 was 2.1 sccm, which is below its administrative limit. In RF-18 the as-found test result was 20.2 sccm. Since the as-found leak rate was again under its administrative limit, the frequency is reset to the 60 month extended frequency.

SAS-V486 The frequency was reset to 30 months during RF-16 because the valve exceeded its administrative limit of 400 sccm. The valve was reworked resulting in an as-left test result of 261 sccm. The as-found test result in RF-17 was 175 sccm, which is below its administrative limit. In RF-18 the as-found test result was 108.4 sccm. Since the as-found leak rate was again under its administrative limit, the frequency is reset to the 60 month extended frequency.

SSR-SOV131 The frequency was reset to 30 months during cycle (Cy)-16 because the valve position indicator was reworked without an as-found test. In Cy-17 the as-found test result was 60 sccm, which is below its administrative limit. In Cy-18 the as-found test result was 5.1 sccm. Since the as-found leak rate was again under its administrative limit, the frequency is reset to the 60 month extended frequency.

The following Type C tested valves are not eligible for extended test frequency based on RG 1.163, Section C.2 and NEI 94-01, Revision 3, Section 10.2.

B21-AOVF022A, B, C, D	Main Steam Inboard Isolation Valves
B21-AOVF028A, B, C, D	Main Steam Ouboard Isolation Valves
B21-MOVF016	Inboard MSIV Drain Valve
B21-MOVF019	Inboard MSIV Drain Valve
B21-MOVF067A, B, C, D	Outboard MSIV Drain Valves
B21-VF010A, B	Feedwater Inboard Isolation Valves
B21-AOVF032A, B	Feedwater Outboard Isolation Valves
FWS-MOV7A, B	Feedwater Outboard Isolation Valves

HVR-AOV123	Containment Purge Inboard Isolation Valve
HVR-AOV128	Containment Purge Inboard Isolation Valve
HVR-AOV165	Containment Purge Outboard Isolation Valve
HVR-AOV166	Containment Purge Outboard Isolation Valve
CPP-MOV104	Containment Purge Inboard Isolation Valve
CPP-MOV105	Containment Purge Outboard Isolation Valve
CPP-SOV140	Containment Purge Outboard Isolation Valve

Type B and C Testing Summation Margin

The as-found minimum pathway summation results and as-left maximum pathway summation results for Type B and Type C testing from the November 2004 outage and each successive outage are provided below. The Percentage of Regulatory Limit illustrates the margin between the Type B and Type C leakage rate summation and its acceptance criterion.

Table 4.0-5

Date	As-Found Min Path Leak Rate (sccm)	Percentage of Regulatory Limit	As-Left Max Path Leak Rate (sccm)	Percentage of Regulatory Limit	Regulatory Limit - .6 La (sccm)
March 2015	2,909.5	3.5%	12,144.5	14.6%	83,061
March 2013	6,701	8.1%	23,784	28.6%	83,061
February 2011	4,078	4.9%	16,889.9	20.3%	83,061
October 2009	5,034	6.1%	12,653	15.2%	83,061
February 2008	7,346	8.8%	15,983	19.2%	83,061
May 2006	7,363	8.9%	18,981	22.9%	83,061
November 2004	7,574	9.1%	17,842	21.5%	83,061

Secondary Containment Bypass Testing Summation Margin

The minimum pathway summation results and as-left maximum pathway summation results for Secondary Containment Bypass testing from the November 2004 outage and each successive outage are provided below. The Percentage of Regulatory Limit illustrates the margin between the Secondary Containment Bypass leakage rate summation and its acceptance criterion.

Table 4.0-6

Date	As-Found Min Path Leak Rate (sccm)	Percentage of Regulatory Limit	As-Left Max Path Leak Rate (sccm)	Percentage of Regulatory Limit	Regulatory Limit - TS (sccm)
March 2015	974.8	6.7%	2,853.2	19.5%	14,600

Table 4.0-6					
March 2013	1,113	7.7%	2,777	19.0%	14,600
February 2011	1,536	10.5%	4,055.6	27.8%	14,600
October 2009	1,427	9.8%	8,653	59.3%	14,600
February 2008	1,803	12.3%	4,783	32.8%	14,600
May 2006	2,180	14.9%	9,857	67.5%	14,600
November 2004	1,373	9.4%	9,400	64.4%	14,600

Regulatory Requirements

As required by 10 CFR 50.54(o), the RBS containment is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the 10 CFR 50, Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0.

The proposed amendment would revise TS 5.5.13, "Containment Leakage Rate Testing Program," by replacing the reference to RG 1.163, "Performance-Based Containment Leak Test Program," with NEI topical report NEI 94-01, "Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012. This implementation document will be used by Entergy to develop the RBS performance-based leakage rate testing program in accordance with 10 CFR 50, Appendix J, Option B.

Limitations and Conditions

In the June 25, 2008 SER, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SER.

In the June 8, 2012 SER, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Option B (Type B and C tests were addressed), and found that NEI 94-01, Revision 3, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SER.

The 2008 NRC SER Limitations and Conditions address the extension of the ILRT to 15 years and the 2012 SER Limitations and Conditions address increasing the test frequency of Type C LLRTs to 75 months. The 2008 SER Limitations and Conditions are addressed in Attachment 1 of this submittal. The following addresses the 2012 SER Limitations and Conditions as well as compliance.

Condition 1

NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months. This is Topical Report Condition 1.

Response to Condition 1:

1. The RBS post-outage report will include the margin between the Type B and Type C minimum pathway leak rate summation value and Secondary Containment Bypass minimum pathway leakage rate summation value, as adjusted to include the estimate of applicable understatement and their regulatory limit.

2. When the potential leakage understatement adjusted Type B and Type C, or Secondary Containment Bypass (SCB) minimum pathway leakage rate total is equal to or greater than the administrative leakage summation limit of 0.50 La (Type B & C), or 13,000 sccm (SCB), but less than the regulatory limit of 0.60 L_a, or 14,600 sccm, then an analysis and determination of a corrective action plan will be prepared to restore the leakage summation margin to less than the administrative limit. The corrective action plan will focus on those components that have contributed the most to the increase in the leakage summation value and corrective actions will be developed, focusing on prevention of future component leakage performance issues in order to maintain an acceptable level of margin.

3. RBS will apply the 9 month grace period only to eligible Type C components and only for non-routine emergent conditions.

Note: The regulatory limit for the Type B & C summation is .6 L_a . The regulatory limit for the Secondary Containment Bypass summation is the TS value of 14,600 sccm.

Condition 2

The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time.

The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRT's being performed during

plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential.

Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1.

When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations. This is Topical Report Condition 2.

Response to Condition 2

The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25% in the LLRT periodicity. Based on this, RBS will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the as-found Type C minimum pathway leak rate summation value for all type C components currently on the 75-month extended test interval. When the potential leakage understatement adjusted leak rate total for those Type C components being tested on a 75-month extended interval is summed with the non-adjusted total of those Type C components being tested at a less than 75-month interval and the total of the Type B components equals, or is greater than the RBS administrative leakage summation limit of 0.50 L_a, or 13,000 sccm (SCB), but less than the regulatory limit of 0.60 L_a, or 14,600 sccm (SCB), then an analysis and corrective action plan will be prepared to restore the leakage summation value to less than the RBS administrative summation leakage limit. The corrective action plan will focus on those components that have contributed the most to the increase in the leakage summation value and timely corrective actions will be developed in an effort to prevent future component leakage performance issues.

If the potential leakage understatement adjusted minimum pathway leak rate is less than the RBS administrative leakage summation limit of 0.50 L_a, or 13,000 sccm (SCB), then the acceptability of the 75-month LLRT extension for all Type C components has been

adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

The post-outage report, required by NEI-94-01, Revision 3, Section 12.1, will include the following in addition to the items listed in the response to Condition 1:

1 Type A, B & C test results performed during the cycle (including the concluding refueling outage). The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and the report is available on-site for NRC review. The report shows that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable. The report includes the Type B and C leakage summation, the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C will be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

   In the event an adverse trend in the aforementioned potential leakage understatement adjusted Type B and C, or secondary containment bypass summations is identified, then an analysis and determination of a corrective action plan shall be prepared to restore the trend and associated margin to an acceptable level.

2 An estimate of the potential leakage understatement will be made by applying an adjustment factor of 1.25 to the as-found Type C minimum pathway leak rate summation value for all type C components currently on the 75-month extended test interval. The understatement will be included in the Type B & C and Secondary Containment Bypass minimum pathway leakage rate summation to represent the actual leakage potential of the penetrations.

3 The reasoning and determination of the acceptability for interval extensions, demonstrating that the LLRT totals calculated represent the actual leakage potential for the penetrations.

Conclusion

NEI 94-01, Revision 3-A describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type C testing intervals to 75 months. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, B, and C containment leakage rate surveillance test frequencies. RBS is proposing to adopt the guidance of NEI 94-01, Revision 3-A for the 10 CFR 50, Appendix J, testing program plan.

Based on the previous Type B & C Local Leak rate (LLRT) tests, it may be concluded that extension of the containment Type C LLRT intervals from 60 to 75 months represents minimal risk when performed in accordance with 10 CFR 50, Appendix J, Option B and under the guidance of NEI-94-01, Revision 3A.

5.0 REGULATORY EVALUATION

Entergy has evaluated the safety significance of the proposed change to the RBS TS which revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program" to allow a permanent extension to the maximum frequency of the Type C tests to 75 months. The proposed changes have been evaluated according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject change does not involve a Significant Hazards Consideration, as discussed below

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The accidents previously evaluated in the USAR that could be potentially impacted by the proposed amendment change Loss-of-Coolant Accident (LOCA) Inside Containment, Fuel Handling Accident,

   In order to affect the frequency of occurrence of an accident, the change has to affect an accident initiator. There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause an accident LOCA, however, it is evaluated without the cause being identified because it provides an upper limit estimate to the effect of pipe breaks. A fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto stored fuel bundles.

   The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. Changing the frequency of leakage rate testing has no impact upon the likelihood of a LOCA or of a Fuel Handling Accident. The testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

   The proposed amendment adopts the NRC accepted guidelines of NEI 94-01, Revision 3-A for development of the RBS performance-based testing program for the Type C testing. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components would limit leakage rates to less than the values assumed in the plant safety analyses.

   Based on the above discussion, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The use of the NRC accepted NEI 94-01, Revision 3-A does not alter any conditions such that any other accidents previously not considered credible should become credible. No new structures, systems, or components are being added or altered by the proposed amendment, therefore no new failure modes can be postulated that would create the possibility of a different type of accident. There are no accident initiators created or altered by the proposed amendment. The proposed changes would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses.

   Based on the above discussion, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A for the development of the RBS performance-based leakage rate testing program, and establishes a maximum 75 month interval for the performance of the Type C LLRT tests. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the containment local leakage rate testing program, as defined in the, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage rate tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A.

   Based on the above discussion, the proposed amendment does not involve a significant reduction in a margin of safety.

Entergy concludes that the proposed amendment to the RBS TS presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of 'no significant hazards consideration' is justified.

6.0 Environmental Considerations

The proposed changes to the RBS TS do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

This request is similar in nature to the following license amendment, which was approved by the NRC:

1. Surry Power Station, Units 1 and 2 - Issuance of Amendment Regarding the Containment Type A and Type C Leak Rate Tests (TAC Nos. MF2612 and MF2613, Accession Number ML14148A235) approved July 3, 2014.
2. Beaver Valley Power Station, Units 1 and 2 - Issuance of Amendment Regarding License Amendment Request to Extend Containment Leakage Rate Test Frequency (TAC Nos. MF3985 and MF3986, Accession Number MIL15078A058) approved April 8, 2015.

Attachment 3

RBG-47620

Risk Analysis

ERIN
an SKF Group Company

RISK IMPACT ASSESSMENT FOR RIVER BEND STATION REGARDING THE ILRT (TYPE A) AND DWBT PERMANENT EXTENSION REQUEST

Prepared for:

Entergy

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September 2014

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1.0 PURPOSE OF ANALYSIS

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a permanent extension of the River Bend Station (RBS) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology outlined in EPRI TR-104285 [2], the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage going undetected during the extended test interval [5]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the October 2008 EPRI final report [3]. Additionally, consistent with other previous ILRT extension requests for BWR Mark III containments, the risk assessment also includes an assessment for extending the Drywell Bypass Test (DWBT) interval from ten years to fifteen years. The DWBT has been historically associated with the ILRT frequency because the plant line-ups are similar and the same equipment is used to perform both tests.

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than the normal containment leakage of 1.0La (allowable leakage).

The basis for a 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [6], "Performance-Based Containment Leak Test Program," provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [2].

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for RBS.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 [2] methodology to perform the risk assessment. In October 2008, EPRI 1018243 [3] was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174 [4]. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose. This ILRT/DWBT interval extension risk assessment for RBS employs the EPRI 1018243 methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary.

1.3 ACCEPTANCE CRITERIA

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 1.0E-06 per reactor year and increases in large early release frequency (LERF) less than 1.0E-07 per reactor year. Note that CDF is not impacted by the proposed change for RBS. Therefore, since the Type A test does not impact CDF for RBS, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1.0E-06 per reactor year, provided that the total LERF from all contributors (including external events) can be reasonably shown to be less than 1.0E-05 per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) is also calculated to help ensure that the defense-in-depth philosophy is maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of [3]) indicate a range of incremental increases in population dose[1] that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, the NRC SER on this issue [7] defines a small increase in population dose as an increase of ≤ 1.0 person-rem per year, or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This definition has been adopted for the RBS analysis.

The acceptance criteria are summarized below.

1. The estimated risk increase associated with permanently extending the ILRT/DWBT surveillance interval to 15 years must be demonstrated to be small. (Note that Regulatory Guide 1.174 defines very small changes in risk as increases in CDF less than 1.0E-6 per reactor year and increases in LERF less than 1.0E-7 per reactor year. Since the type A ILRT and the DWBT are not expected to impact CDF for RBS, the relevant risk metric is the change in LERF. Regulatory Guide 1.174 also defines small risk increase as a change in LERF of less than 1.0E-6 reactor year.) Therefore, a small change in risk for this application is defined as a LERF increase of less than 1.0E-6.
2. Per the NRC SE, a small increase in population dose is also defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive.
3. In addition, the SE notes that a small increase in Conditional Containment Failure Probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests (typically about 1% or less, with the largest increase being 1.2%). This would require that the increase in CCFP be less than or equal to 1.5 percentage points.

[1] The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, whereas this analysis uses 100La.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI methodology is used for evaluating the change in risk associated with increasing the test interval to fifteen years [3]. The analysis uses results from the core damage and large early release scenarios from the current RBS PRA analyses of record and the subsequent containment responses to establish the various fission product release categories including the release size.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report [3].
2. Develop plant-specific population dose rates (person-rem per reactor year) for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT/DWBT interval to fifteen years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 and compare this change with the acceptance guidelines of RG 1.174 [4].
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)
6. Evaluate the sensitivity of the results to assumptions in the steel corrosion analysis and to variations in the fractional contributions of large isolation failures (due to corrosion) to LERF.

Furthermore,

- Consistent with the previous industry containment leak risk assessments, the RBS assessment uses population dose as one of the risk measures. The other risk measures used in the RBS assessment are the conditional containment failure probability (CCFP) for defense-in-depth considerations, and change in LERF to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for RBS uses ground rules and methods to calculate changes in the above risk metrics that are consistent with those outlined in the current EPRI methodology [3].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The RBS Level 1 and LERF internal events PRA models provide representative core damage frequency and release category frequency distributions to be utilized in this analysis. The Level 1 and LERF PRA models are supplemented with simple extended logic expressions and cutset calculations to fill out the necessary radionuclide release end states consistent with the EPRI methodology [3] as summarized in Section 4.2.
- It is appropriate to use the RBS internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT/DWBT extension. It is reasonable to assume that the impact from the ILRT/DWBT extension (with respect to percent increases in population dose) will not substantially differ if other hazard groups were to be included in the calculations; however, other hazard groups have been accounted for in the analysis based on the available information for RBS [8, 9, 10, 11] as described in Section 5.7.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [17]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and Tech Spec leak rate differences for River Bend compared to the NUREG/CR-4551 Mark III reference plant, Grand Gulf.
- The use of the estimated 2030 population data from RBS USAR [18] is appropriate for this analysis.
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a is 10 La and for Class 3b sequences is 100La, based on the recommendations in the latest EPRI report [3] and as recommended in the NRC SE on this topic [7]. It should be noted that this is more conservative than the earlier previous industry ILRT extension requests, which utilized 35La for the Class 3b sequences.
- Based on the EPRI methodology and the NRC SE, the Class 3b sequences are categorized as LERF and the increase in Class 3b sequences is used as a surrogate for the ΔLERF metric.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [12].
- The methodology to evaluate the impact of concurrently extending the DWBT interval is performed consistent with previous one-time ILRT/DWBT extensions for BWR Mark III containment types [19, 20, 21].

4.0 INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2). Section 4.3 then provides the derivation of the RBS base population dose risk used as a starting point for this assessment. Section 4.4 provides more details on the EPRI methodology that is followed, Section 4.5 discusses the details of the Calvert Cliffs corrosion analysis method that is also used for this assessment, and Section 4.6 discusses the details of the analysis performed on the available Mark III DWBT data to estimate the likelihood and magnitude of DWBT leakage rates that may occur due to extending the DWBT interval in addition to the ILRT interval.

4.1 GENERAL RESOURCES AVAILABLE

Various industry studies on containment leakage risk assessment are briefly summarized here:

- NUREG/CR-3539 [13]
- NUREG/CR-4220 [14]
- NUREG-1273 [15]
- NUREG/CR-4330 [16]
- EPRI TR-105189 [12]
- NUREG-1493 [6]
- EPRI TR-104285 [2]
- Calvert Cliffs corrosion analysis [5]
- EPRI 1018243 [3]
- NRC Final Safety Evaluation [7]
- Prior Mark III ILRT/DWBT Extension Risk Assessments [19, 20, 21]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study

addresses the impact of age-related degradation of the containment on ILRT evaluations. EPRI 1018243 complements the previous EPRI report and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude. The NRC Safety Evaluation (SE) documents the acceptance by the NRC of the proposed methodology with a few exceptions. These exceptions (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 and the final version of the updated EPRI report [3], which was used for this application. Finally, reference is made to other extension requests for Mark III containments that considered extensions to the ILRT interval and the DWBT interval.

NUREG/CR-3539 [13]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [22] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [14]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. It assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [15]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [16]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals.

However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [12]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1.0E-8/yr to 1.0E-7/yr) is realized from extending the test intervals from 3 per 10 years to 1 per 10 years.

NUREG-1493 [6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing frequencies and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 [17] Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 [6] in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight categories of containment response to a core damage accident:

1. Containment intact and isolated

Containment isolation failures due to support system or active failures

Type A (ILRT) related containment isolation failures

Type B (LLRT) related containment isolation failures

Type C (LLRT) related containment isolation failures

Other penetration related containment isolation failures

Containment failure due to core damage accident phenomena

Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "These study results show that the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms..."

Release Category Definitions

Table 4.1-1 defines the accident classes used in the ILRT/DWBT extension evaluation, which is consistent with the EPRI methodology [3]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A ILRT and DWBT intervals as described in Section 5 of this report.

Table 4.1-1
EPRI/NEI Containment Failure Classifications

CLASS	DESCRIPTION
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.

Table 4.1-1
EPRI/NEI Containment Failure Classifications

CLASS	DESCRIPTION
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

Calvert Cliffs Steel Corrosion Analysis [5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting steel liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. RBS has a cylindrical, freestanding steel primary containment structure. The primary containment vessel is enveloped by an open annulus area and a concrete shield building. The shield building serves as a secondary containment, but has no pressure control function. The corrosion analysis for Calvert Cliffs is used for the RBS containment vessel with slight variations made to account for the design differences.

EPRI 1018243 [3]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years. This risk impact assessment complements the previous EPRI report, TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals. The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment considers the change in risk based on population dose, whereas the revision considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry and regulatory guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffrey's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the RBS analysis presented here in Section 6.2.

NRC Safety Evaluation Report [7]

This SE documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, subject to the limitations and conditions identified in the SE and summarized in Section 4.0 of the SE. These limitations (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 which are also included in Revision 3-A of NEI 94-01 [1] and the final version of the updated EPRI report [3]. Additionally, the SE clearly defined the acceptance criteria to be used in future Type A ILRT extension risk assessments as delineated previously in the end of Section 1.3.

Previous ILRT/DWBT Extension Risk Assessments for Mark III Plants [19, 20, 21]

Consistent with other previous ILRT extension requests for BWR Mark III containments, the risk assessment also includes an assessment for extending the Drywell Bypass Test (DWBT) interval from ten years to fifteen years. The DWBT has been historically associated with the ILRT frequency because the plant line-ups are similar and the same equipment is used to perform both tests. The DWBT is to verify that pre-existing drywell bypass leakage does not exceed the minimum requirements. The DWBT thus affects the likelihood of a suppression pool bypass in the Level 1 and 2 PRA analyses. The methodology for extending the DWBT has previously been accepted by the NRC for analysis of Clinton, Grand Gulf, and River Bend [19, 20, 21].

The DWBT verifies that pre-existing drywell bypass leakage does not exceed the maximum allowed leakage. For RBS, the DWBT acceptance criterion in the Tech Specs is <10% of the analyzed design limit. The design bypass limit is used to establish the timing of automatic initiation of the containment unit coolers following a LOCA. The unit coolers effectively control the containment pressure to its design limit (15 psi) by suppressing the steam from the drywell that bypasses the suppression pool. The DWBT thus affects the likelihood of suppression pool bypass in the Level 1 and Level 2 PRA analyses.

Even though the methodologies used for the ILRT extension do not directly address the DWBT, it is judged that the ILRT methodology can be used to address the impact of extending both

the ILRT and DWBT with a few additional considerations and assumptions. The primary difference in the methodology used to evaluate the extension of the DWBT is in the determination of the conditional probability of an existing drywell leak. In the base case analysis, the same release categories, consequence calculations, and acceptance criteria are used. The analysis will be performed assuming that both the ILRT and the DWBT are on the same frequencies. The impact of drywell leakage is to allow drywell atmosphere, including fission products, to be passed at some rate directly to the containment, without benefit of quenching and fission product retention in the suppression pool. The key augmentation needed to the ILRT method is to determine the impact of this suppression pool bypass on the containment and on the radionuclide releases.

It is assumed in this augmented methodology that the special leakage categories established by EPRI for use in ILRT risk assessments can also be applied to the drywell for the DWBT risk assessment. The Mark III containment has a different arrangement from BWR Mark I/II containments or PWR containments. The difference is that the drywell which includes the RPV is completely enclosed by the outer containment. As such, the drywell leakage does not leak directly to the environment but is further mitigated by the outer containment leakage barrier. Because of this "dual" containment, there are several possible leakage path combinations that must be considered. The drywell can be intact (base leakage assumed), it can have a small pre-existing failure (10 times base leakage), or it can have a large pre-existing failure (100 times base leakage). The probability of each of these drywell failure categories is taken from the industry experience evaluation of drywell bypass test data. As further discussed below, this leads to at least nine combinations of drywell and containment leakage sizes (refer to Figure 4.1-1). Each combination will have an impact on radionuclide releases that corresponds approximately to one of the original containment failure categories. An additional sensitivity case is also performed based on a more conservative evaluation of historical DWBT data for the Mark III plants. However, it should be noted that the number of DWBT data points in the industry is limited compared to the ILRT data available from all of the plants.

The different combinations of the drywell and containment leakage sizes can result in different accident classes. To address this issue, the Grand Gulf Nuclear Station (GGNS) and the Clinton Power Station (CPS) DWBT methods have some slight differences, which are discussed in more detail in the following two sub-sections. Following that, the approach used for RBS is presented. The Mark III and RBS plant-specific data utilized for the DWBT portion of this risk assessment is then provided in Section 4.6.

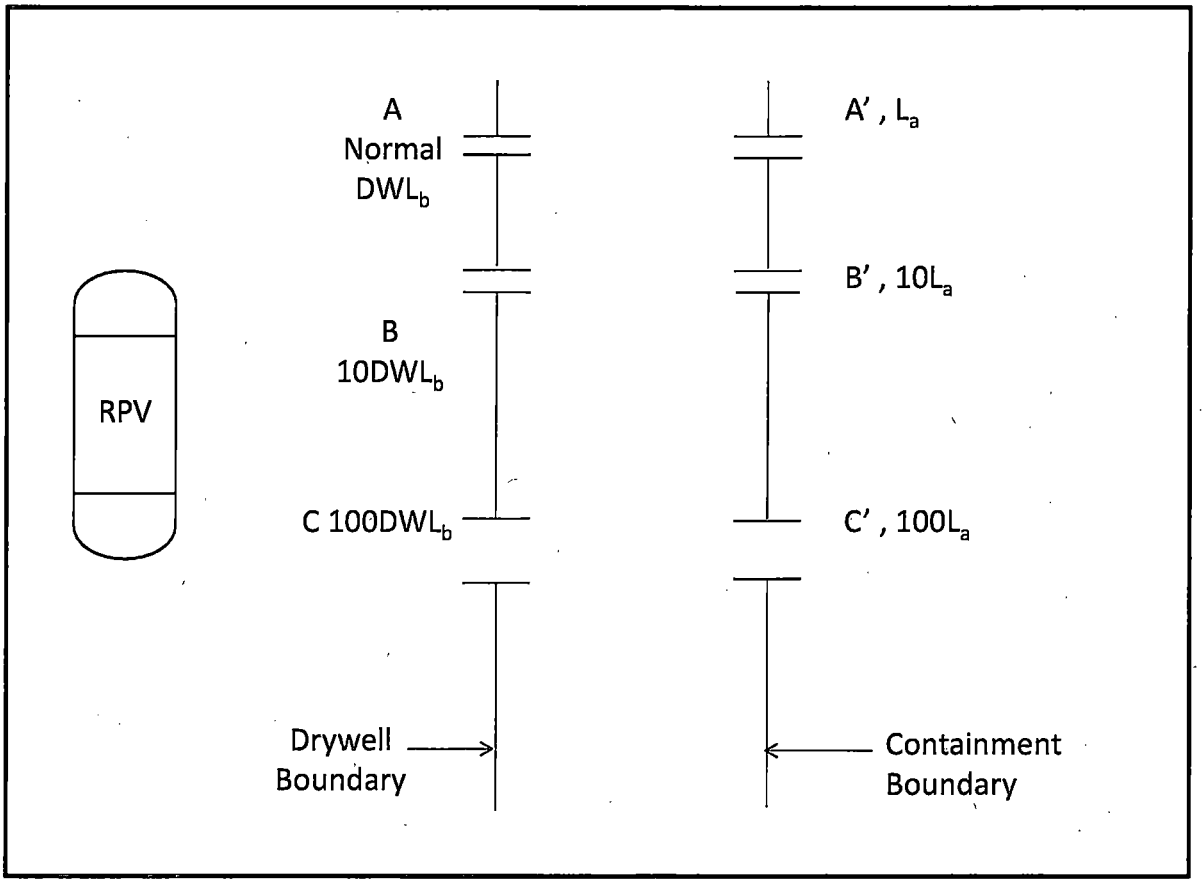

Figure 4.1-1
RBS Drywell and Containment Leakage Categories

4.1.1 GGNS DWBT Method

In the GGNS assessment [20], the assignment of each of these combinations to an original containment failure category depends on the consideration of the availability of the containment spray system, which has the similar effects in reducing the containment pressure as the containment unit coolers at RBS. If containment sprays are available, the combination of drywell and containment leakage is categorized based on the containment leakage category. If containment sprays are not available, the combination of drywell and containment leakage is assumed to result in containment failure (Class 7) except for the combinations with base drywell bypass leakage. The combinations with base drywell leakage (DWL_b) are assumed to have the same categories as the base case ILRT evaluation. Table 4.1-2 summarizes the classification of combinations into the EPRI accident classes.

Table 4.1-2
GGNS DWBT and ILRT Leakage Combination Accident Classes

Leakage Combinations	DW Bypass Leakage	Containment Leakage	EPRI Classification Assignment
AA'	1 DWL_b	1 L_a	1
AB'	1 DWL_b	10 L_a	3a
AC'	1 DWL_b	35 L_a (2)	3b
BA'1 CS Available	10 DWL_b	1 L_a	1
BA'2 CS Not Available	CF (1)	CF	7
BB'1 CS Available	10 DWL_b	10 L_a	3a
BB'2 CS Not Available	CF	CF	7
BC'1 CS Available	10 DWL_b	35 L_a	3b
BC'2 CS Not Available	CF	CF	7
CA'1 CS Available	35 DWL_b (2)	1 L_a	1
CA'2 CS Not Available	CF	CF	7
CB'1 CS Available	35 DWL_b	10 L_a	3a
CB'2 CS Not Available	CF	CF	7
CC'1 CS Available	35 DWL_b	35 L_a	3b
CC'2 CS Not Available	CF	CF	7

(1) CF = Containment failure assumed to occur.

(2) Note that 35 La was used in the prior assessments, but per the updated EPRI guidance as approved by the NRC, 100 La is now used for EPRI Class 3b.

The probability for each combination in Table 4.1-2 is determined by multiplying the conditional probabilities for DWBT and ILRT category by each other. For those cases where containment spray is a factor the probability of the combination of DWBT and ILRT is multiplied by the probability that containment spray is available or is not available as applicable.

The other change in the methodology to address the DWBT is the need to increase the containment failure due to phenomenology class (Class 7) frequency for the extended test frequencies. This is done in a manner similar to the method applied to Class 3a and 3b. That is, the Class 1 frequency is also adjusted downward for the Class 7 frequency increase in order to maintain the same total CDF.

The remaining portions of the DWBT methodologies are identical to that of the previously approved and alternate ILRT methodologies.

4.1.2 CPS DWBT Method

The CPS DWBT methodology [19] is very similar to the GGNS one except the assignment of the nine combinations of drywell and containment leakage sizes to an original containment failure category. Unlike the GGNS approach which conservatively assumed all combinations without containment spray available would contribute to accident class 7, CPS used a Clinton specific MAAP 4.0 model to determine the compound effects of the increased drywell leakages that could lead to higher containment pressure and increased concentration of fission products in containment. By examining CsI concentrations in containment and the amount of CsI released, one can assign each combination to one of the original containment failure classes. Once this is done, the calculation process is identical to that used for ILRT interval extension.

Based on the MAAP runs, CPS assigned the equivalent EPRI category and LERF characterization as shown in Table 4.1-3.

Table 4.1-3
CPS DWBT and ILRT Leakage Combination Accident Classes

Leakage Combinations	DW Bypass Leakage	Containment Leakage	EPRI Classification Assignment
AA'	1 DWL_b	1 L_a	1 (Non-LERF)
AB'	1 DWL_b	10 L_a	3a (Non-LERF)
AC'	1 DWL_b	35 L_a[(1)]	3b (LERF)
BA'1	10 DWL_b	1 L_a	1 (Non-LERF)
BB'1	10 DWL_b	10 L_a	3a (Non-LERF)
BC'1	10 DWL_b	35 L_a	3b (LERF)
CA'1	35 DWL_b[(1)]	1 L_a	1 (Non-LERF)
CB'1	35 DWL_b	10 L_a	3a (LERF)
CC'1	35 DWL_b	35 L_a	3b (LERF)

(1) Note that 35 La was used in the prior assessments, but per the updated EPRI guidance as approved by the NRC, 100 La is now used for EPRI Class 3b.

Note that no credit for the availability of containment spray was taken in the CPS analyses. Similar to GGNS method, all EPRI class 1 and 3a were categorized as Non-LERF while 3b was categorized as LERF. The CPS MAAP runs did not result in any EPRI class 7 leakage combinations, which were treated as LERF in the GGNS method.

4.1.3 RBS DWBT Method

For the most part, the GGNS methodology for DWBT extension evaluation was previously used for RBS [21]. The main modifications to the GGNS methodology were as follows:

- RBS credited the containment unit coolers to mitigate the adverse effects of the increased drywell leakages instead of the containment spray credited in the GGNS evaluation. Containment spray has dual functions by reducing the containment pressure and scrubbing the fission products from the containment atmosphere while containment unit coolers were designed mainly to reduce containment pressure. However, the GGNS method does not credit the containment spray for scrubbing. Thus the effects of crediting containment unit coolers and containment spray are the same.
- The RBS base cases for DWBT extension evaluation use EPRI Class 1 frequency to calculate the Class 3a, Class 3b and additional Class 7 frequencies. The GGNS method base cases used the total CDF for the calculation, which was conservative since more Class 1 frequencies would be re-categorized into Class 3a, 3b or Class 7 frequencies. Such a conservative approach was not appropriate for the RBS evaluation since the RBS Class 1 frequency only consisted of about 10% of the total CDF, and as such, the calculated Class 3a, 3b and additional Class 7 frequencies could exceed the Class 1 frequency if using the total CDF for calculations. Therefore, only the CDF portion that does not lead to a more severe release category in the Level 2 analysis is re-categorized to Class 3a, 3b, or 7. This exclusion of a portion of the CDF that is impacted by the DWBT extension is similar to the allowed exclusion of LERF contributors per the accepted EPRI methodology for ILRT extension assessments.

A similar approach to the previous RBS method to account for the DWBT will be used for this assessment. That is, the availability of containment unit coolers will be used to preclude additional Class 7 sequences due to increased drywell leakage, and the portion of CDF adversely impacted by the DWBT extension will only be that which does not already end up in a more severe Level 2 release category.

4.2 PLANT-SPECIFIC INPUTS

The RBS specific information used to perform this ILRT/DWBT interval extension risk assessment includes the following:

- PRA model Level 1 and LERF quantification results [24, 25]
- Population within a 50-mile radius [18]
- Reactor Power Level [18]
- Allowable Containment Leakage [18]

RBS Internal Events Core Damage Frequency

The current RBS Internal Events PRA analysis of record is an event tree / linked fault tree model characteristic of the as-built, as-operated plant. Based on the subsumed merged sequence cutset file results reported in the RBS PRA Summary Report for PRA Rev. 5 [24], the mean value of the internal events core damage frequency (CDF) is 2.60E-06/yr.

RBS Internal Events Release Category Frequencies

The RBS PRA Level 2 Model [25] is used to develop the initial set of internal events release categories for use in this analysis. The mean value of the internal events large early release frequency (LERF) is 2.48E-08/yr. Table 4.2-1 summarizes the pertinent RBS results in terms of the LERF end-states where a representative release category is assigned for each end-state. The individual release category frequencies are utilized here to provide the necessary delineation for the ILRT/DWBT risk assessment with the corresponding EPRI class for each release category being listed in Table 4.2-1. A discussion of the release categories follows this table.

Table 4.2-1
LERF Release Category Frequencies for RBS

Sequence	Description	Frequency / Year	EPRI Classification
@RXRUP	Vessel Rupture Marker	1.30E-08	8
@ATWS	ATWS Sequence Marker	6.28E-09	8
@ISL	ISLOCA Sequence Marker	5.33E-11	8
@BOC	Break Outside Containment Marker	3.27E-13	8
Remaining L2-39[(1)]	Containment Isolation Failure	6.64E-11	2
Other LERF	Remainder of LERF	5.35E-09	7-LERF
Total		**2.48E-08**	

(1) Sequence L2-39 includes all of the containment bypass sequences including @RXRUP, @ATWS, @ISL, and @BOC. The sum of those four frequencies is subtracted from the sequence frequency of 1.94E-08 to obtain the containment isolation failure frequency of 6.64E-11.

As RBS does not currently maintain a detailed Level 2 analysis, a review of the Level 1 event tree sequences was performed to determine the containment status if available. Table 4.2-2 provides a list of the un-truncated sequence values from the merged and subsumed model of record cutset file (RBS_R5_rec_merged_subsm_E-12.cut) and lists whether the containment status is known or not. The containment status in Table 4.2-2 is listed as bypassed, failed (late), not failed where W4 is known to have succeeded, or unknown. This information can be used as a starting point for determining the remaining release categories needed for this assessment. The known sequence frequencies can be assigned directly to one of the EPRI release categories. The bypassed frequencies are already assigned as noted in Table 4.2-1. The late containment failure sequences are assigned to EPRI Class 7 (non-LERF), and the no containment failure sequences are assigned to EPRI Class 1 (containment intact). An evaluation of the unknown sequence frequencies is then performed to determine if it should be assigned to EPRI Class 1 or EPRI Class 7. This analysis uses additional simplified cutset quantifications to determine the EPRI release category as described following Table 4.2-2.

Table 4.2-2
Level 1 Sequence Frequencies and Containment Status

L1 Sequence	Containment Status			
	Bypassed (1)	Failed Late	Not Failed (W4 OK)	Unknown
RPVRUPTURE	1.30E-08			
A-3		9.96E-12		
A-4				7.91E-11
A-13				1.99E-12
A-14	1.61E-12			
MSLB07	5.87E-12			
S1-4				5.62E-09
S1-13				1.43E-10
S1-14				6.04E-09
S1-15	9.51E-11			
S2A-6		2.36E-07		
S2A-9				5.76E-11
S2A-16				1.35E-09
S2A-17				3.58E-09
S2A-24				4.62E-09
S2A-25				1.59E-09
T-14		3.10E-07		
T-16		7.60E-10		
T-21				1.12E-11
T-22				1.32E-09
T-29		3.06E-07		
T-31			1.42E-09	

Table 4.2-2
Level 1 Sequence Frequencies and Containment Status

L1 Sequence	Containment Status			
	Bypassed [1]	Failed Late	Not Failed (W4 OK)	Unknown
T-33		2.41E-08		
T-46				2.72E-08
T-51				4.81E-08
T-TB-1		1.28E-07		
T-TB-10		2.82E-10		
T-TB-12				1.64E-09
T-TB-14				2.37E-08
T-TB-2		6.68E-07		
T-TB-3				2.78E-07
T-TB-4				1.86E-10
T-TB-5		6.28E-09		
T-TB-6				6.92E-08
T-TB-7				7.31E-10
T-TB-9				4.46E-08
T1-13			3.43E-09	
T1-14		8.85E-10		
T1-15			1.94E-10	
T1-24		1.74E-09		
T1-28		2.13E-10		
T1-33				3.96E-09
T1-34				4.29E-08
T1-4		3.21E-07		
T1-6		1.40E-10		
TC-11	1.14E-11			
TC-22	4.41E-10			
TC-25	8.06E-12			
TC-26	1.31E-11			
TC-27	2.86E-10			
TC-28	5.33E-09			
TC-29	3.72E-10			
TL-S1-17				3.95E-09
TL-S1-18				3.92E-11
TL-S1-4		7.29E-09		
TL-S1-8		1.53E-12		
V-2	1.18E-11			
V-4	1.79E-11			
V-5	2.27E-12			
Total (2.60E-06)	**1.98E-08**	**2.01E-06**	**5.05E-09**	**5.68E-07**

[1] The bypassed values do not agree exactly with that reported as LERF with the corresponding marker tags in Table 4.2-1 due to truncation and other assumption differences, but since the total LERF value is later utilized in the calculations, this has a negligible impact on the results of this risk assessment.

To determine if the "Unknown" sequences lead to containment intact (EPRI Class 1) or late containment failure (EPRI Class 7), each of the "Unknown" sequences was appended with logic expressions for the suppression pool cooling (W1) and containment fan cooler (W4) functions, and the sequences were re-quantified, concatenated, and subsumed into one cutset file. Failure of the W1 and W4 functions is consistent with the Level 1 PRA model success criteria for avoiding containment failure, and is used to approximate the late containment failure sequences if a more detailed Level 2 analysis were to be developed. When doing this, the resultant "Unknown*W1*W4" sequence frequency is 4.62E-07. However, many of the cutsets include early failure to recover off-site power terms due to the Level 1 event tree sequence structure. These cases can be assumed to not lead to containment failure if power is recovered by about 12 hours. Table 4.2-3 shows the initial contribution from the fail to recover off-site power cutsets from the merged "Unknown*W1*W4" cutset file, and the result of adjusting those cutsets by prorating the containment failure contribution by the conditional off-site power recovery by 12 hours. As can be seen in the bottom row of Table 4.2-3, of the original frequency of 4.62E-07, 4.28E-07 involves fail-to-recover off-site power events, and after accounting for the conditional fail to recover by 12 hours, a frequency of just 9.49E-08 would result in late containment failure (assuming power is restored and W1 or W4 succeeds).

Table 4.2-3
Unknown LOSP Sequence Frequencies Leading to Late Containment Failure

Event Name	Description	Prob	F-V	CDF*W1 *W4(1)	Cond. LOSP(2)	Cont. Failure(3)
ZHE-FO-T12-0RUN	Failure to Recover OSP before 12 Hours (0 run failures)	3.49E-02	4.25E-04	1.96E-10	1	1.96E-10
ZHE-FO-T12-1RUN	Failure to Recover OSP before 12 Hours (1 run failure)	8.15E-03	4.68E-06	2.16E-12	1	2.16E-12
ZHE-FO-T12-2RUN	Failure to Recover OSP before 12 Hours (2 run failures)	3.01E-03	0	0	1	0
ZHE-FO-T12-3RUN	Failure to Recover OSP before 12 Hours (3 run failures)	1.31E-03	0	0	1	0
ZHE-FO-T30-0RUN	Failure to Recover OSP before 30 Minutes (0 run failures)	7.28E-01	1.24E-01	5.74E-08	4.79E-02	2.75E-09
ZHE-FO-T1-0RUN	Failure to Recover OSP before 1 Hour (0 run failures)	6.29E-01	1.17E-01	5.42E-08	5.55E-02	3.01E-09
ZHE-FO-T1-1RUN	Failure to Recover OSP before 1 Hour (1 run failure)	9.38E-02	1.71E-02	7.91E-09	8.69E-02	6.87E-10
ZHE-FO-T1-2RUN	Failure to Recover OSP before 1 Hour (2 run failures)	3.58E-02	1.58E-03	7.27E-10	8.41E-02	6.11E-11
ZHE-FO-T1-3RUN	Failure to Recover OSP before 1 Hour (3 run failures)	2.16E-02	1.53E-04	7.05E-11	6.06E-02	4.28E-12
ZHE-FO-T2-0RUN	Failure to Recover OSP before 2 Hours (0 run failures)	4.25E-01	3.15E-03	1.45E-09	8.21E-02	1.19E-10
ZHE-FO-T2-1RUN	Failure to Recover OSP before 2 Hours (1 run failure)	8.99E-02	3.97E-04	1.83E-10	9.07E-02	1.66E-11
ZHE-FO-T2-2RUN	Failure to Recover OSP before 2 Hours (2 run failures)	3.57E-02	9.73E-05	4.49E-11	8.43E-02	3.79E-12
ZHE-FO-T2-3RUN	Failure to Recover OSP before 2 Hours (3 run failures)	2.16E-02	3.34E-05	1.54E-11	6.06E-02	9.35E-13

Table 4.2-3
Unknown LOSP Sequence Frequencies Leading to Late Containment Failure

Event Name	Description	Prob	F-V	CDF*W1 *W4(1)	Cond. LOSP(2)	Cont. Failure(3)
ZHE-FO-T4-0RUN	Failure to Recover OSP before 4 Hours (0 run failures)	1.78E-01	2.50E-02	1.15E-08	1.96E-01	2.26E-09
ZHE-FO-T4-1RUN	Failure to Recover OSP before 4 Hours (1 run failure)	6.68E-02	2.07E-02	9.56E-09	1.22E-01	1.17E-09
ZHE-FO-T4-2RUN	Failure to Recover OSP before 4 Hours (2 run failures)	3.32E-02	3.90E-03	1.80E-09	9.07E-02	1.63E-10
ZHE-FO-T4-3RUN	Failure to Recover OSP before 4 Hours (3 run failures)	2.13E-02	7.50E-04	3.46E-10	6.15E-02	2.13E-11
ZHE-FO-T6-0RUN	Failure to Recover OSP before 6 Hours (0 run failures)	1.11E-01	5.54E-01	2.56E-07	3.14E-01	8.04E-08
ZHE-FO-T6-1RUN	Failure to Recover OSP before 6 Hours (1 run failure)	5.29E-02	5.55E-02	2.56E-08	1.54E-01	3.95E-09
ZHE-FO-T6-2RUN	Failure to Recover OSP before 6 Hours (2 run failures)	3.03E-02	2.08E-03	9.58E-10	9.93E-02	9.51E-11
ZHE-FO-T6-3RUN	Failure to Recover OSP before 6 Hours (3 run failures)	2.06E-02	2.14E-05	9.86E-12	6.36E-02	6.27E-13
Total			**0.927**	**4.28E-07**		**9.49E-08**

(1) The CDF*W1*W4 contribution is determined from the product of the F-V value and the total "Unknown * W1 * W4" frequency of 4.62E-07.

(2) The conditional loss of offsite power probability is determined by the ratio of the OSP failure probability at 12 hours divided by the initial OSP failure probability at the specified time interval.

(3) The late containment failure probability is determined from the product of the CDF column and the conditional LOSP column values.

In summary, since many of the "Unknown*W1*W4" cutsets involve failure to recover off-site power terms (comprising 4.28E-07 of the 4.62E-07 total), a separate calculation was performed to determine how much of that frequency would result in continued off-site power unavailability such that late containment failure cannot be precluded. Based on the results shown in Table 4.2-3, a late containment failure frequency of 9.49E-08 is derived assuming late containment failure occurs if power is not recovered by 12 hours. The cutsets which did not include off-site power recovery terms, but did contribute to the 4.62E-07 "Unknown*W1*W4" total can also be assumed to result in late containment failure. This amounts to a 3.40E-08 late containment failure frequency (derived from 4.62E-07 – 4.28E-07). Therefore, the total additional late containment failure sequences from the "Unknown" containment status Level 1 sequences is 9.49E-08 + 3.40E-08, or 1.29E-07. The remainder of the "Unknown" sequences can then be assumed to result in an intact containment (not yet accounting for the small fraction of those scenarios which would lead to LERF). This means that the intact containment frequency from the "Unknown" containment status Level 1 sequences is 5.68E-07 – 1.29E-07, or 4.39E-07. Since the total LERF value is 2.48E-08, and only 1.98E-08 is accounted for by the Level 1 sequences shown in Table 4.2-2,

about 5.0E-09 of this contribution would go to LERF. So the total additional intact containment frequency is 4.39E-07 – 5.0E-09, or 4.34E-07.

Based on the Level 1 and LERF PRA model results described above, Table 4.2-4 lists the relevant EPRI release category frequencies pertinent for the ILRT/DWBT extension risk assessment, including the delineation of LERF and non-LERF frequencies for class 7.

Table 4.2-4
Relevant Level 2 Release Category Frequencies for RBS

EPRI RELEASE CATEGORY	FREQUENCY/YR	SOURCE
1: No Containment Failure	5.05E-09 + 4.34E-07 = 4.39E-07	Table 4.2-2 Table 4.2-3 discussion
2: Containment Isolation Failure	6.64E-11	Table 4.2-1
7: Phenomena-induced containment failures (LERF)	5.35E-09	Table 4.2-1
7: Phenomena-induced containment failures (non-LERF)	2.01E-06 + 1.29E-07 = 2.14E-06	Table 4.2-2 Table 4.2-3 discussion
8: Containment Bypass	1.30E-08 + 6.28E-09 + 5.33E-11 + 3.27E-13 = 1.93E-08	Table 4.2-1
Total:	**2.60E-06**	

4.3 RBS POPULATION DOSE DERIVATION

Since RBS does not maintain a detailed Level 3 PRA model, the approach recommended in EPRI 1018243 [3] is utilized. From the EPRI guidance it is noted that for the cases where plant-specific PRA dose information is not available, a representative population dose can be calculated using other references, such as NUREG/CR-4551 [17]. To develop a representative population dose, the NUREG/CR-4551 plant that most closely resembles the analysis plant is chosen and the following steps are performed.

- Relate the NUREG/CR-4551 accident progression bins (APBs), EPRI Accident Classes, and plant-specific plant damage states (PDSs) based on the definitions contained in NUREG/CR-4551, and plant-specific PDSs.
- Adjust the resulting EPRI Accident Class 1, 2, 7, and 8 population doses to account for substantial differences in reactor power level, population density, allowable containment leak rate (La), and other plant-specific factors that may affect population dose as follows:
  - Population density adjustment = (population within 50 miles of the plant ÷ population within 50 miles of the reference plant, from reference document)
  - Power level adjustment = (rated power level of plant (MWt) ÷ rated power level of reference plant)
  - La adjustment= La of plant (%wt/day) ÷ La of reference plant

Note that the population density and power level adjustments are applicable to all EPRI accident classes; however, the La adjustment should be made only to intact containment end states.

Reference Plant Population Dose Information

Based on this guidance, the ex-plant consequence analysis for Grand Gulf is used as the reference plant for RBS since Grand Gulf is also a BWR Mark III containment. Table 4.3-1 reproduces the APB descriptions for Grand Gulf provided in NUREG/CR-4551, and Table 4.3-2 provides a calculation to determine the relevant population dose associated with each APB. Note that Table 4.3-2 is consistent with the calculations previously performed for the Clinton ILRT/DWBT interval extension submittal [19].

Table 4.3-1
Collapsed Accident Progression Bin Descriptions for Grand Gulf

COLLAPSED APB NUMBER	DESCRIPTION
1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.
2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.

Table 4.3-1
Collapsed Accident Progression Bin Descriptions for Grand Gulf

COLLAPSED APB NUMBER	DESCRIPTION
3	CD, vessel breach, Early CF, Late SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.
4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.
5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.
6	CD, vessel breach, Vent This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.
7	CD, VB, No CF Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.
8	CD, No vessel breach Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.

Legend for Table 4.3-1 and Table 4.3-2

CCI = Core Concrete Interaction

CD = Core Damage

CF = Containment Failure

CS = Containment Sprays

SP = Suppression Pool

VB = Vessel Breach

Table 4.3-2
Grand Gulf NUREG/CR-4551 50-Mile Radius Population Dose

APB #(1)	APB Definition	APB Frequency (per year)(2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year)(4)	APB 50-Mile Radius Dose (Person-rem)(5)
1	CD, VB, Early CF, Early SP Bypass, CS Not Available	6.46E-7	.268	0.139	2.2E+5
2	CD, VB, Early CF, Early SP Bypass, CS Available	2.00E-7	.056	0.029	1.5E+5
3	CD, VB, Early CF, Late SP Bypass	2.86E-8	.011	5.7E-3	2.0E+5
4	CD, VB, Early CF, No SP Bypass	8.92E-7	.267	0.139	1.6E+5
5	CD, VB, Late CF	1.16E-6	.281	0.146	1.3E+5
6	CD, VB, Vent	1.55E-7	.039	0.0203	1.3E+5
7	CD, VB, No CF	2.05E-7	3E-4	1.56E-4	7.6E+2
8	CD, No VB	7.36E-7	.077	0.040	5.4E+4
Total		**4.09E-6**	**1.0**	**0.52**	

(1) This table is presented in the form of a calculation because NUREG/CR-4551 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose risk results in NUREG/CR-4551.

(2) The total (i.e., internal accident sequences) CDF of 4.09E-6/yr and the CDF subtotals by APB are taken from Figure 2.5-7 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1.

(3) The individual APB contributions to total (i.e., internal accident sequences) 50-mile radius dose rate are taken from Table 5.1-3 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1.

(4) The APB 50-mile dose risk is calculated by multiplying the individual APB dose risk contributions (column 4) by the total 50-mile radius dose risk of 0.52 person-rem/yr (taken from Table 5.1-1 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1).

(5) The individual APB doses are calculated by dividing the individual APB dose risk by the APB frequencies.

The APBs described above can then be assigned to one of the EPRI release categories for the RBS assessment. These assignments and their basis are provided in Table 4.3-3.

Table 4.3-3
Assigned APB for each of the Relevant Level 2 Release Categories for RBS

EPRI RELEASE CATEGORY	ASSIGNED APB	BASIS
1: No Containment Failure	7	The intact containment case with release limited to leakage is represented by APB 7 in the Grand Gulf assessment.
2: Containment Isolation Failure	4	APB 4 is chosen as the most likely representative case since the pool is not likely to be bypassed. This has minimal impact on the results for RBS due to the low frequency.
7: Phenomena-induced containment failures (LERF)	1	APB 1 w/o containment sprays available is chosen for RBS since RBS has containment fan coolers and not containment sprays.
7: Phenomena-induced containment failures (non-LERF)	5	For RBS, this release category is dominated by late containment failure events. As such, APB 5 is chosen from the Grand Gulf assessment.
8: Containment Bypass	1	The containment bypass case is selected as APB 1 from the Grand Gulf assessment. It results in the highest population dose.

Adjustments to Ex-Plant Consequence Calculations

The next step per the EPRI guidance is to adjust the resulting EPRI Accident Class 1, 2, 7, and 8 population doses from the reference plant to account for substantial differences in reactor power level, population density, and allowable containment leak rate (La).

The 50-mile radius population used in the Grand Gulf NUREG/CR-4551 consequence calculations is 3.4E+5 persons. This is based on an estimate provided in a previous ILRT extension submittal for Clinton [19]. The 50-mile radius population dose for RBS is based on the 2030 population estimate provided in the RBS USAR of 1.49E+6 persons [18]. Therefore, the ratio of the population surrounding RBS to that in the Grand Gulf analysis results in a factor increase of:

1.49E+6 persons / 3.4E+5 persons = 4.38

The Grand Gulf reactor power level used in the NUREG/CR-4551 consequence calculations is 3833 MWt. The current RBS reactor power level is 3091 MWt [18]. Therefore, the ratio of the RBS reactor power to that used in the Grand Gulf analysis results in a multiplication factor of:

3091 MWt / 3833 MWt = 0.81

The assumed containment leakage used in the NUREG/CR-4551 consequence calculations for Grand Gulf is 0.5 %wt/day. The current RBS allowable leakage is 0.325 %wt/day [18]. Because the leakage rates are a function of the containment volume, these plant characteristics are also needed:

- Grand Gulf Containment Volume [29] = 1.40E+6 ft^3
- RBS Containment Volume [18] = 1.19E+6 ft^3

Therefore, the ratio of the RBS allowable leakage and containment volume to that used in the Grand Gulf analysis results in a multiplication factor of:

$$(0.325\% * 1.19E+6) / (0.5\% * 1.40E+6) = 0.55$$

As stated previously, this final adjustment factor is only applied to the intact containment case. Table 4.3-4 provides a summary of each of the adjustment factors used for each APB to estimate the population doses for RBS that can be used in this assessment.

Table 4.3-4
RBS Adjusted 50-Mile Radius Population Dose

APB #	Grand Gulf 50-Mile Radius Dose (Person-rem)[1]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	RBS Population Dose Adjusted 50-Mile Radius Dose (Person-rem)
1	2.2E+05	4.38	0.81	N/A	7.81E+05
2	1.5E+05	4.38	0.81	N/A	5.32E+05
3	2.0E+05	4.38	0.81	N/A	7.10E+05
4	1.6E+05	4.38	0.81	N/A	5.68E+05
5	1.3E+05	4.38	0.81	N/A	4.61E+05
6	1.3E+05	4.38	0.81	N/A	4.61E+05
7	7.6E+02	4.38	0.81	0.55	1.48E+03
8	5.4E+04	4.38	0.81	N/A	1.92E+05

(1) The NUREG/CR-4551 evaluation of Grand Gulf is used as input to the assessment of population dose for RBS. Refer to Table 4.3-2.

Population Dose Risk Calculations

The next step is to take the frequency information from Table 4.2-4 for each relevant EPRI release category class from Table 4.1-1, and then associate a representative population dose from Table 4.3-4 for each release category based on the APB assignments made in Table

4.3-3. Table 4.3-5 lists the population dose risk organized by EPRI release category for RBS, including the delineation of LERF and non-LERF frequencies for Class 7. Note that the population dose risk (Column 4 of Table 4.3-5) was found by multiplying the release category frequency (Column 2 of Table 4.3-5) by the associated population dose (Column 3 of Table 4.3-5). Also note that only the applicable EPRI release categories at this point are shown in the tables (i.e., the Class 3 frequencies are derived later and the Class 4, 5, and 6 frequencies are not utilized in the EPRI methodology for the ILRT extension risk assessment).

Table 4.3-5
RBS Population Dose and Dose Risk Organized by EPRI Release Category

EPRI RELEASE CATEGORY	FREQUENCY/YR	POPULATION DOSE (PERSON-REM)	POPULATION DOSE RISK (PERSON-REM/YR)
1: No Containment Failure	4.39E-07	1.48E+03	6.51E-04
2: Containment Isolation Failure	6.64E-11	5.68E+05	3.77E-05
7: Phenomena-induced containment failures (LERF)	5.35E-09	7.81E+05	4.18E-03
7: Phenomena-induced containment failures (non-LERF)	2.14E-06	4.61E+05	9.87E-01
8: Containment Bypass	1.93E-08	7.81E+05	1.51E-02
Total:	**2.60E-06**		**1.007**

4.4 IMPACT OF EXTENSION ON DETECTION OF COMPONENT FAILURES THAT LEAD TO LEAKAGE (SMALL AND LARGE)

The ILRT can detect a number of component failures such as breach and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the latest EPRI guidance [3], as the mean failure estimated from the available data (i.e., 2 "small" failures that could only have been discovered by the ILRT in 217 tests leads to a 2/217=0.0092 mean value). For Class 3b, consistent with latest available EPRI data [3], a non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

The EPRI methodology contains information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This information includes a discussion of conservatisms in the quantitative guidance for delta LERF. EPRI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The methodology states:

> "The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage."

The application of this additional guidance to the analysis for RBS (as detailed in Section 5) means that the Class 2, Class 7-LERF, and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Note that Class 2 events refer to sequences with a large pre-existing containment isolation failure that lead to LERF, a subset of Class 7

events are LERF sequences due to an early containment failure from energetic phenomena, and Class 8 event are containment bypass events that contribute to LERF.

Consistent with the EPRI methodology [3], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing, given a 10-year vs. a 3-yr interval. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

RBS Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart) where the calculated performance leakage rate was less than 1.0La, and in compliance with the performance factors in NEI 94-01, Section 11.3. Based on the successful completion of two consecutive ILRTs at RBS, the current ILRT interval is once per ten years. Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry-wide historical results as noted in the EPRI guidance document [3].

EPRI Methodology

This analysis uses the approach outlined in the EPRI Methodology [3]. The six steps of the methodology are:

1. Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.
2. Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.
3. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
4. Determine the risk impact in terms of the change in LERF and the change in CCFP.
5. Consider both internal and external events.
6. Evaluate the sensitivity of the results to assumptions in the steel corrosion analysis.

The first three steps of the methodology deal with calculating the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF for RBS, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The fourth step of the methodology calculates the change in containment failure probability, referred to as the conditional containment failure probability, CCFP. The NRC has identified a CCFP of less than 1.5% as the acceptance criteria for extending the Type A ILRT test intervals as the basis for showing that the proposed change is consistent with the defense in depth philosophy [7]. As such, this step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174. Step 5 takes into consideration the additional risk due to external events, and Step 6 investigates the impact on results due to varying the assumptions associated with the liner corrosion rate and failure to visually identify pre-existing flaws.

4.5 IMPACT OF EXTENSION ON DETECTION OF STEEL CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. RBS has a cylindrical, freestanding steel primary containment structure. The primary containment vessel is enveloped by an open annulus area and a concrete shield building. The shield building serves as a secondary containment, but has no pressure control function. The corrosion analysis for Calvert Cliffs is used for the RBS containment vessel with slight variations made to account for the design differences.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel structure. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment cylinder and dome
- The historical flaw likelihood due to concealed corrosion
- The impact of aging

- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- The two corrosion events over a 5.5 year data period are used to estimate the flaw probability in the Calvert Cliffs analysis and are assumed to be applicable to the RBS containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner. It is noted that two additional events have occurred in recent years (based on a data search covering approximately 9 years documented in Reference [26]). In November 2006, the Turkey Point 4 containment building liner developed a hole when a sump pump support plate was moved. In May 2009, a hole approximately 3/8" by 1" in size was identified in the Beaver Valley 1 containment liner. For risk evaluation purposes, these two more recent events occurring over a 9 year period are judged to be adequately represented by the two events in the 5.5 year period of the Calvert Cliffs analysis incorporated in the EPRI guidance (See Table 4.5-1, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel ages (See Table 4.5-1, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every two years and every ten years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a flaw exists in the steel was estimated as 1.1% for the cylinder and dome region, and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 50 psig. For RBS, the ILRT target pressure is less (8.2 to 8.6 psig), but the containment failure probability at 37 psig of 5% [27] is used in this assessment as a conservative approximation to represent the unique free-standing shell containment configuration at RBS. The probabilities of 5% for the cylinder and dome, and 0.5% for the basemat are used in this analysis. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4.5-1, Step 4).
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used for the containment cylinder and dome. For the containment basemat, 100% is assumed unavailable for visual inspection. To date, all corrosion events have been detected through visual inspection (See Table 4.5-1, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 4.5-1
Steel Corrosion Base Case

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME		CONTAINMENT BASEMAT	
1	**Historical Steel Flaw Likelihood** Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	Events: 2 2/(70 * 5.5) = **5.2E-3**		Events: 0 (assume half a failure) 0.5/(70 * 5.5) = **1.3E-3**	
2	**Age Adjusted Steel Flaw Likelihood** During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year 1 avg 5-10 15	Failure Rate 2.1E-3 5.2E-3 1.4E-2	Year 1 avg 5-10 15	Failure Rate 5.0E-4 1.3E-3 3.5E-3
		15 year average = 6.27E-3		**15 year average = 1.57E-3**	
3	**Flaw Likelihood at 3, 10, and 15 years** Uses age adjusted flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [19]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, the values are calculated based on the 3, 10, and 15 year intervals.)		**0.18% (1 to 3 years)** **1.02% (1 to 10 years)** **2.35% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, values are calculated based on the 3, 10, and 15 year intervals.)	

Table 4.5-1
Steel Corrosion Base Case

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
4	**Likelihood of Breach in Containment Given Steel Flaw** The failure probability of the containment cylinder and dome is assumed to be 5% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.5% (compared to 0.11% in the Calvert Cliffs analysis).	**5%**	**0.5%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.0036% (at 3 years)** =0.71% * 5% * 10% **0.0203% (at 10 years)** =4.06% * 5% * 10% **0.0470% (at 15 years)** =9.40% * 5% * 10%	**0.0009% (at 3 years)** =0.18% * 0.5% * 100% **0.0051% (at 10 years)** =1.02% * 0.5% * 100% **0.0118% (at 15 years)** =2.35% * 0.5% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome, and the containment basemat:

At 3 years : 0.0036% + 0.0009% = 0.0045%

At 10 years: 0.0203% + 0.0051% = 0.0254%

At 15 years: 0.0470% + 0.0118% = 0.0588%

4.6 IMPACT OF DWBT INTERVAL EXTENSION ON RELEASE CATEGORIES

Similar to the prior RBS ILRT/DWBT interval extension risk assessments, Table 4.6-1 provides the release categories that are utilized in this assessment for the different combinations of drywell bypass leakage and containment leakages.

Table 4.6-1
RBS DWBT and ILRT Leakage Combination Accident Classes

Leakage Combinations	DW Bypass Leakage	Containment Leakage	EPRI Classification Assignment
AA'	1 DWL_b	1 L_a	1
AB'	1 DWL_b	10 L_a	3a
AC'	1 DWL_b	100 L_a	3b
BA'1 UC Available	10 DWL_b	1 L_a	1
BA'2 UC Not Available	CF [(1)]	CF	7
BB'1 UC Available	10 DWL_b	10 L_a	3a
BB'2 UC Not Available	CF	CF	7
BC'1 UC Available	10 DWL_b	100 L_a	3b
BC'2 UC Not Available	CF	CF	7
CA'1 UC Available	100 DWL_b	1 L_a	1
CA'2 UC Not Available	CF	CF	7
CB'1 UC Available	100 DWL_b	10 L_a	3a
CB'2 UC Not Available	CF	CF	7
CC'1 UC Available	100 DWL_b	100 L_a	3b
CC'2 UC Not Available	CF	CF	7

(1) CF = Containment failure assumed to occur.

Again, consistent with the prior assessments, the probability for each combination in Table 4.6-1 is determined by multiplying the conditional probabilities for DWBT and ILRT category by each other. For those cases where the availability of the unit coolers (UCs) is a factor, the probability of the combination of DWBT and ILRT leakage probabilities is multiplied by the probability that the UCs are available or are not available as applicable. Section 4.6.1 provides an analysis of available Mark III DWBT data to estimate the likelihood of the different DW bypass leakage categories. Section 4.6.2 provides a plant-specific assessment to determine the fraction of cases with UCs available to be applied in this analysis.

4.6.1 DWBT Data Analysis

Table 4.6-2 summarizes the available DWBT results for the Mark III containment types previously reported [21]. In the prior RBS DWBT extension analysis, 800 SCFM was used as the reference leakage for the risk assessment. This will also be used in this assessment for the base drywell leakage rate, L_b. Although a few data points are available with more recent successful test data[1], these points will not be included since the intent is to establish the baseline failure frequency consistent with the original testing frequency, and consistent with the accepted ILRT methodology, the failure frequency is then assumed to grow as the testing interval is extended.

Table 4.6-2
Mark III Drywell Bypass Test Results

Site	Test Date	Leakage Rate (SCFM)	Actual Leakage / 800 SCFM
Clinton	Jan-86	273	0.34
	Nov-86	20.8	0.03
	Apr-89	18.8	0.02
	Mar-91	21.9	0.03
	May-92	18	0.02
	Nov-93	30.2	0.04
Grand Gulf	Nov-85	2315	2.89
	Nov-86	1568	1.96
	Dec-87	1500	1.88
	Apr-89	1631	2.04
	Nov-90	1591	1.99
	May-92	618	0.77
	Nov-93	869	1.09
Perry	Aug-87	124	0.16
	Jul-89	123	0.15
	Dec-90	797	1.00
	May-92	253	0.32
	Jun-94	2450	3.06
	Jul-94	111	0.14
River Bend	Dec-87	602	0.75
	May-89	141	0.18
	Nov-90	345	0.43
	Aug-92	754	0.94
	Jun-94	421	0.53

[1] Drywell bypass tests at River Bend were successfully completed in 2000 and again in 2008 [30, 31]

Figure 4.6-1 then shows a scatter plot of this data compared to the reference assumed base leakage value, L_b, of 800 SCFM. (Note that the assumed base drywell leakage value of 800 SCFM is less than the allowable drywell bypass leakage for RBS of 3035 SCFM at 3.0 psid [31]). The 800 SCFM base case drywell leakage therefore represents a conservative assumption, but is used for consistency with the previously accepted ILRT/DWBT extension requests for RBS.

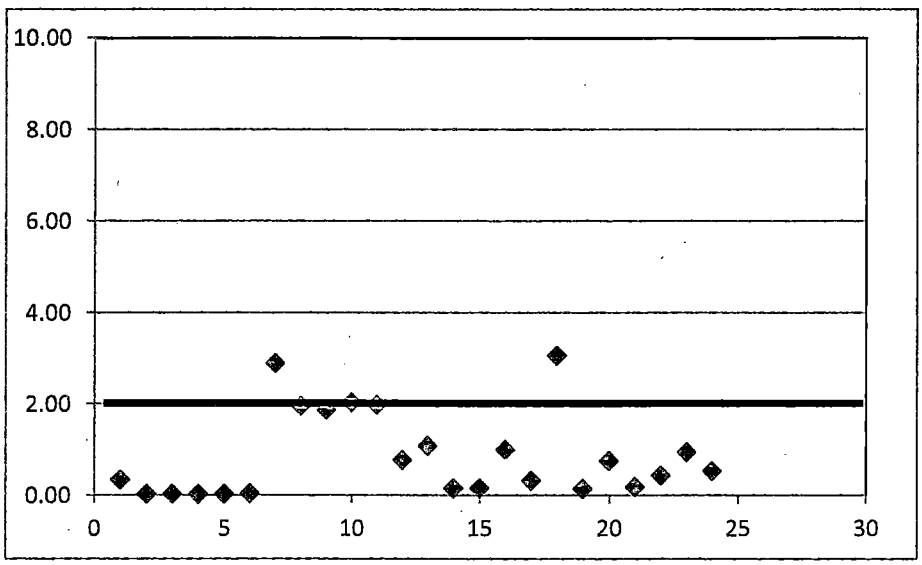

Figure 4.6-1 Mark III DWBT Results Compared to 800 SCFM

Although 7 of the data points are greater than 800 SCFM, only 2 of the data points are measurably in the 2-10 range, and none of them are close to 10 L_b leakage rate assumed for the intermediate category in this assessment. Therefore, the likelihood of the 10 L_b leakage rate is conservatively assumed to be represented by two events over the total number of representative test results.

Likelihood of 10 DWL_b = 2 / 24 = 0.083 = 8.3%

For the large leakage rate represented by a 100 L_b DW leakage rate, similar to the prior DWBT risk assessments, a Jeffrey's non-informative prior will be used to determine the likelihood of occurrence.

Likelihood of 100 DWL_b = (0 + ½) / (24 + 1) = 0.02 = 2.0%

When the three data points (i.e., > $1L_b$, $10L_b$, and $100L_b$) are plotted on a curve, the trend appears reasonable as shown in Figure 4.6-2. These values are therefore used for the base case assessment to represent the DW bypass leakage behavior. Increases to these values are assumed to occur for the different test intervals consistent with the ILRT methodology. A sensitivity case is also included regarding the likelihood of the 10 DWL_b leakage rate used in the assessment as described in Section 6.3.

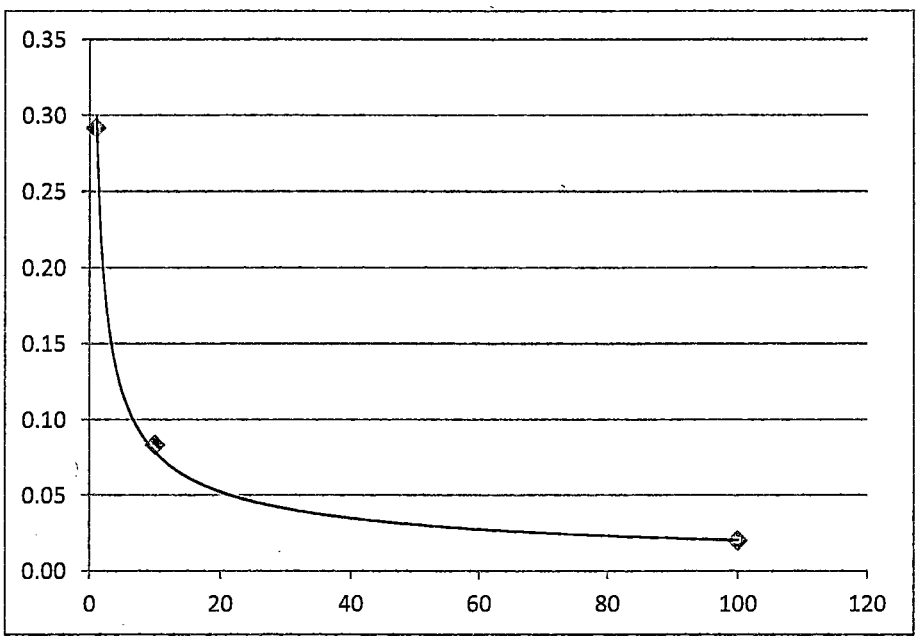

Figure 4.6-2 Estimated Mark III DWBT Leakage Probability Compared to 800 SCFM

4.6.2 Availability of Containment Unit Coolers

As previously mentioned, the availability of the containment unit coolers will be utilized to preclude additional Class 7 sequences due to increased drywell leakage, and the portion of CDF adversely impacted by the DWBT extension will only be that which does not already end up in a more severe Level 2 release category (i.e., the no containment failure release category). Section 4.2 described the process utilized to determine the intact containment failure frequency. A similar process is used to determine the fraction of the intact containment category that also includes success of the containment unit coolers. That is, the following steps were taken to determine the availability of the containment unit coolers using the RBS Level 1 Revision 5 PRA model [24].

1. Append logic for the unit coolers (i.e., the "W4" node) to the un-truncated Level 1 sequences whose containment status is "Unknown" (refer to Table 4.2-2).

2. Solve those new sequences and merge and subsume the cutsets into one cutset file.
3. Examine that portion of the cutset file representing failure to recover off-site power events.
4. Differentiate the likelihood of late off-site power recovery.
5. Apply random UC failure probabilities to the off-site power recovered portion.
6. Sum the UC failure probability contributions.
7. Assign the remainder to UC success.

To determine if the "Unknown" sequences lead to UC available or not, each of those sequences was appended with logic expressions for the containment unit cooler (W4) function, and the sequences were re-quantified, concatenated, and subsumed into one cutset file. When doing this, the resultant "Unknown*W4" sequence frequency is 4.93E-07. This means that of the total "Unknown" frequency of 5.68E-07 reported in Section 4.2, **7.5E-08** (5.68E-07 – 4.93E-07) can be stated as having no containment failure with UCs available. However, of the 4.93E-07 representing W4 node failures, many of the cutsets include early failure to recover off-site power terms due to the Level 1 event tree sequence structure. These cases can be assumed to not lead to UC failure if power is recovered by about 12 hours and the UCs do not fail randomly. Table 4.6-3 shows the initial contribution from the fail to recover off-site power cutsets from the merged "Unknown*W4" cutset file, and the result of adjusting those cutsets by prorating the failure contribution by the conditional off-site power recovery by 12 hours. As can be seen in the bottom row of Table 4.6-3, of the original frequency of 4.93E-07, 4.59E-07 involves fail-to-recover off-site power events, and after accounting for the conditional fail to recover by 12 hours, a frequency of just 1.00E-07 would result in late UC failure (because off-site power is not restored), and 3.59E-07 could have UC available if random failures do not occur. Since the W4 node by itself evaluates to approximately 2.0E-02, the sequences where off-site power is recovered and UC fails is represented by 3.59E-07 * 2.0E-02 = 7.2E-09. Subtracting this value of 7.2E-09 from the original frequency of 3.59E-07 results in a frequency of **3.52E-07** where it can be stated that containment failure does not occur because the UCs are available.

The total frequency of the known states derived above representing no containment failure and UCs available (7.5E-08 + 3.52E-07 = 4.27E-07) can be compared to the total no containment failure frequency of 4.39E-07 from Table 4.2-4.

4.39E-07 – 4.27E-07 = 1.2E-08

The value of 1.2E-08 is slightly higher than the simple calculation of applying the random W4 failure probability to the total initial intact containment release category frequency of 4.39E-7. That is, 1.2E-08 is slightly higher than 4.39E-7 * 2.0E-2, or approximately 8.8E-09. The slight higher value is due to dependencies which exist between other failures in the cutsets and the W4 failures. The frequency of 1.2E-08 represents 2.73% of the total intact containment frequency of 4.39E-07.

Table 4.6-3
Unknown LOSP Sequence Frequencies Leading to Late UC Failure

Event Name	Description	Prob	F-V	CDF *W4(1)	Cond. LOSP(2)	UC Failure(3)
ZHE-FO-T12-0RUN	Failure to Recover OSP before 12 Hours (0 run failures)	3.49E-02	4.36E-04	2.15E-10	1	2.15E-10
ZHE-FO-T12-1RUN	Failure to Recover OSP before 12 Hours (1 run failure)	8.15E-03	2.31E-04	1.14E-10	1	1.14E-10
ZHE-FO-T12-2RUN	Failure to Recover OSP before 12 Hours (2 run failures)	3.01E-03	1.88E-04	9.28E-11	1	9.28E-11
ZHE-FO-T12-3RUN	Failure to Recover OSP before 12 Hours (3 run failures)	1.31E-03	5.82E-05	2.87E-11	1	2.87E-11
ZHE-FO-T30-0RUN	Failure to Recover OSP before 30 Minutes (0 run failures)	7.28E-01	1.16E-01	5.74E-08	4.79E-02	2.75E-09
ZHE-FO-T1-0RUN	Failure to Recover OSP before 1 Hour (0 run failures)	6.29E-01	1.11E-01	5.47E-08	5.55E-02	3.04E-09
ZHE-FO-T1-1RUN	Failure to Recover OSP before 1 Hour (1 run failure)	9.38E-02	1.63E-02	8.04E-09	8.69E-02	6.98E-10
ZHE-FO-T1-2RUN	Failure to Recover OSP before 1 Hour (2 run failures)	3.58E-02	1.54E-03	7.57E-10	8.41E-02	6.37E-11
ZHE-FO-T1-3RUN	Failure to Recover OSP before 1 Hour (3 run failures)	2.16E-02	1.43E-04	7.05E-11	6.06E-02	4.28E-12
ZHE-FO-T2-0RUN	Failure to Recover OSP before 2 Hours (0 run failures)	4.25E-01	2.95E-03	1.45E-09	8.21E-02	1.19E-10
ZHE-FO-T2-1RUN	Failure to Recover OSP before 2 Hours (1 run failure)	8.99E-02	3.71E-04	1.83E-10	9.07E-02	1.66E-11
ZHE-FO-T2-2RUN	Failure to Recover OSP before 2 Hours (2 run failures)	3.57E-02	9.11E-05	4.49E-11	8.43E-02	3.79E-12
ZHE-FO-T2-3RUN	Failure to Recover OSP before 2 Hours (3 run failures)	2.16E-02	3.13E-05	1.54E-11	6.06E-02	9.35E-13
ZHE-FO-T4-0RUN	Failure to Recover OSP before 4 Hours (0 run failures)	1.78E-01	5.73E-02	2.82E-08	1.96E-01	5.53E-09
ZHE-FO-T4-1RUN	Failure to Recover OSP before 4 Hours (1 run failure)	6.68E-02	4.12E-02	2.03E-08	1.22E-01	2.48E-09
ZHE-FO-T4-2RUN	Failure to Recover OSP before 4 Hours (2 run failures)	3.32E-02	8.21E-03	4.05E-09	9.07E-02	3.67E-10
ZHE-FO-T4-3RUN	Failure to Recover OSP before 4 Hours (3 run failures)	2.13E-02	9.11E-04	4.49E-10	6.15E-02	2.76E-11
ZHE-FO-T6-0RUN	Failure to Recover OSP before 6 Hours (0 run failures)	1.11E-01	5.19E-01	2.56E-07	3.14E-01	8.04E-08
ZHE-FO-T6-1RUN	Failure to Recover OSP before 6 Hours (1 run failure)	5.29E-02	5.21E-02	2.57E-08	1.54E-01	3.96E-09
ZHE-FO-T6-2RUN	Failure to Recover OSP before 6 Hours (2 run failures)	3.03E-02	1.97E-03	9.73E-10	9.93E-02	9.67E-11
ZHE-FO-T6-3RUN	Failure to Recover OSP before 6 Hours (3 run failures)	2.06E-02	2.00E-05	9.86E-12	6.36E-02	6.27E-13
Total			**0.930**	**4.59E-07**		**1.00E-07**

(1) The CDF*W4 contribution is determined from the product of the F-V value and the total "Unknown * W4" frequency of 4.93E-07.

(2) The conditional loss of offsite power probability is determined by the ratio of the OSP failure probability at 12 hours divided by the initial OSP failure probability at the specified time interval.

(3) The late UC failure probability is determined from the product of the CDF column and the conditional LOSP column values.

5.0 RESULTS

The application of the approach based on EPRI Guidance [3] has led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

Table 5.0-1
Accident Classes

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures
3b	Large Isolation Failures
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Containment Bypass
CDF	**All CET End states (including very low and no release)**

The analysis performed examined RBS-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the categorization of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage, if applicable. (EPRI Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI Class 6 sequences). Consistent with the EPRI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypass (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5.0-1.
Step 2 Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.
Step 3 Evaluate risk impact of extending Type A test interval from 3 to 15 and 10 to 15 years.
Step 4 Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.
Step 5 Determine the impact on the Conditional Containment Failure Probability (CCFP).

5.1 STEP 1 – QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

The RBS PRA Level 2 Model [25] is used to develop the initial set of internal events release categories for use in this analysis. As described in Section 4.2, the release categories were assigned to the EPRI classes as shown in Table 4.2-4. This application combined with the RBS dose risk (person-rem/yr) as shown in Table 4.3-5 forms the basis for estimating the increase in population dose risk.

For the assessment of the impact on the risk profile due to the ILRT/DWBT extension, the potential for pre-existing leaks is included in the model. These pre-existing leak events are represented by the Class 3 sequences in EPRI TR-1018243 [3]. Two failure modes were considered for the Class 3 sequences, namely Class 3a (small breach) and Class 3b (large breach).

The determination of the frequencies associated with each of the EPRI categories listed in Table 5.0-1 is presented next. Since the Class 1 frequency is determined based on remaining contribution not assigned to other classes, the discussion appears in reverse order starting with EPRI Class 8 and ending with EPRI Class 1. However, EPRI Class 2 is discussed prior to Class 3 since its value is used in the final determination of the Class 3 frequencies.

Class 8 Sequences

This group represents sequences where containment bypass occurs. The failure frequency for Class 8 sequences is 1.93E-08/yr. Refer to Table 4.2-4.

Class 7 Sequences

This group represents containment failure induced by severe accident phenomena. For RBS, the frequency for non-LERF Class 7 representing late containment failure sequences is 2.14E-06/yr, and for LERF Class 7 sequences, the total is 5.35E-09/yr. Refer to Table 4.2-4. However, for the RBS BWR Mark III DWBT assessment, an additional adjustment is made to the non-LERF Class 7 sequences. That is, those scenarios which previously resulted in no containment failures, but now may involve large DWBT failures, are assumed to lead to additional late containment failure sequences if the containment unit coolers are not available. As described in Section 4.1.3, this is consistent with the previous assumptions utilized in ILRT/DWBT submittals for RBS and GGNS, but where the containment spray function was applicable in the GGNS assessment.

As described in Section 4.6.1, it can be estimated that about a 8.3% likelihood exists that DWBT would result in a flow rate greater than 2x the base DWBT leakage rate for RBS in the initial assessment. Additionally, it is conservatively assumed that there is a 2% likelihood that a 100x base leakage rate could occur in the initial assessment. This likelihood is combined with the analysis presented in Section 4.6.2 which showed that about 2.73% of the no containment failure scenarios involve failure of the containment unit coolers for RBS. By conservatively assuming that these DWBT leakage rates could lead to containment failure if unit coolers are unavailable in core damage scenarios, the following increase in the Class 7 sequences can be derived given that 4.39E-07/yr is the base case intact containment frequency as shown in Table 4.2-4.

Additional late containment failure frequency due to drywell bypass leakage

$$\text{Class } 7_{DWBT} = 4.39E\text{-}07/y * (8.3\% + 2.0\%) * 2.73\%$$

$$\text{Class } 7_{DWBT} = 1.24E\text{-}09/yr$$

This additional contribution to the late containment failure scenarios is added to Class 7 (non-LERF) release category for this assessment. Increases to this frequency are assumed to be at the same rate as that approved for the ILRT extension assessment (i.e., 3.33x for a 1-in-10 year interval, and 5x for a 1-in-15 year interval).

Class 6 Sequences

These are sequences that involve core damage with a failure-to-seal containment leakage due to failure to isolate the containment. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the EPRI guidance, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 5 Sequences

This group represents containment isolation failure-to-seal of Type C test components. Because these failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 4 Sequences

This group represents containment isolation failure-to-seal of Type B test components. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 2 Sequences

This group consists of large containment isolation failures. For RBS, this frequency is 6.64E-11/yr. Refer to Table 4.2-4. Note that this frequency is not affected by the ILRT/DWBT interval change.

Class 3 Sequences

This group represents pre-existing leakage in the containment structure. The containment leakage for these sequences can be either small (2La to 100La) or large (>100La). In this analysis, a value of 10La was used for small pre-existing flaws and 100La for relatively large flaws.

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment leakage
= 0.0092 (see Section 4.4)

$PROB_{Class\_3b}$ = probability of large pre-existing containment leakage
= 0.0023 (see Section 4.4)

As described in Section 4.3, additional consideration is made to not apply these failure probabilities to those cases that are already classified as LERF (i.e., the Class 2 and Class 7

and Class 8 LERF contributions), or do not lead to a larger population dose due to the accident sequence progression (i.e., the increase from the DWBT extension assigned to Class 7).

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7 LERF + Class 8 + Class 7_{DWBT})]
= 0.0092 * [2.60E-06 – (6.64E-11 + 5.35E-09 + 1.93E-08 + 1.24E-09)]
= 2.37E-08/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 7 LERF + Class 8 + Class 7_{DWBT})]
= 0.0023 * [2.60E-06 – (6.64E-11 + 5.35E-09 + 1.93E-08 + 1.24E-09)]
= 5.91E-09/yr

For this analysis, the associated containment leakage for Class 3a is 10La and 100La for Class 3b, which is consistent with the latest EPRI methodology [3].

Class 1 Sequences

This group represents the frequency when the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 4.08E-07/yr for RBS and is determined by subtracting all containment failure end states, including the EPRI/NEI Class 3a and 3b frequencies calculated below and the Class 7 increase associated with the DWBT interval extension, from the total CDF. For this analysis, the associated maximum containment leakage for this group is 1La, consistent with an intact containment evaluation. Note that the value for this Class reported in Table 5.1-1 is slightly lower than that reported in Tables 4.2-4 since the 3a and 3b frequencies, and increases in Class 7 are now subtracted from Class 1.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI TR-1018243 [3] and are shown in Table 5.1-1.

Table 5.1-1
Radionuclide Release Frequencies As A Function Of Accident Class (RBS Base Case)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)
1	No Containment Failure	4.08E-07
2	Large Isolation Failures (Failure to Close)	6.64E-11
3a	Small Isolation Failures	2.37E-08
3b	Large Isolation Failures	5.91E-09
4	Small Isolation Failures (Failure to seal –Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7 LERF	Failures Induced by Phenomena (LERF)	5.35E-09
7 non-LERF	Failures Induced by Phenomena (non-LERF)	2.14E-06
8	Containment Bypass	1.93E-08
CDF	**All CET End states (including intact case)**	**2.60E-06**

5.2 STEP 2 – DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the weighted average person-rem doses to the population within a 50-mile radius from the plant. The releases are based on the assessment provided in Section 4.3 for RBS (see Table 4.3-5 of this analysis). The results of applying these releases to the EPRI containment failure classifications are summarized as follows:

Class 1 = 1.48E+03 person-rem (at 1.0La)
Class 2 = 5.68E+05 person-rem
Class 3a = 1.48E+03 person-rem x 10La = 1.48E+04 person-rem
Class 3b = 1.48E+03 person-rem x 100La = 1.48E+05 person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7 LERF = 7.81E+05 person-rem
Class 7 non-LERF = 4.61E+05 person-rem
Class 8 = 7.81E+05 person-rem

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [3] for all EPRI classes are provided in Table 5.2-1, which includes the values previously presented in Table 4.3-5 as well as the Class 3a and 3b population doses calculated above.

Table 5.2-1
RBS Population Dose
for Population Within 50 Miles

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)
1	No Containment Failure (1 La)	1.48E+03
2	Large Isolation Failures (Failure to Close)	5.68E+05
3a	Small Isolation Failures	1.48E+04
3b	Large Isolation Failures	1.48E+05
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7 LERF	Failures Induced by Phenomena (LERF)	7.81E+05
7 non-LERF	Failures Induced by Phenomena (non-LERF)	4.61E+05
8 LERF	Containment Bypass	7.81E+05

The above population dose, when multiplied by the frequency results presented in Table 5.1-1, yields the RBS baseline mean dose risk for each EPRI accident class. These results are presented in Table 5.2-2.

Table 5.2-2
RBS Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 3 in 10 Year ILRT/DWBT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION [2]		CHANGE DUE TO CORROSION OR DWBT [3] EXTENSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1[1]	No Containment Failure	1.48E+03	4.08E-07	6.05E-04	4.08E-07	6.05E-04	-1.70E-07
2	Large Isolation Failures (Failure to Close)	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	--
3a	Small Isolation Failures	1.48E+04	2.37E-08	3.51E-04	2.37E-08	3.51E-04	--
3b	Large Isolation Failures	1.48E+05	5.91E-09	8.77E-04	6.03E-09	8.94E-04	1.70E-05
7 LERF	Failures Induced by Phenomena (LERF)	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	4.61E+05	2.14E-06	9.88E-01	2.14E-06	9.88E-01	5.70E-04
8	Containment Bypass	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	--
CDF	**All CET end states**		**2.60E-06**	**1.009**	**2.60E-06**	**1.009**	**5.87E-4**

(1) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.
(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.
(3) The DWBT leakage cases of 10x and 100x with unit coolers unavailable are assumed to lead to an increased frequency of Class 7 (non-LERF).

5.3 STEP 3 – EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 year interval).

Risk Impact Due to 10-year Test Interval

As previously stated, ILRT Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). However, as noted previously, the DWBT tests also impact the Class 7 sequences. Thus, the frequency of Class 3a, 3b, and Class 7 sequences are impacted by the ILRT/DWBT interval extension. The risk contribution is changed based on the EPRI guidance as described in Section 4.4 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1 for RBS.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of not detecting a leak in Classes 3a and 3b for the ILRT Type A tests, and for Class 7 for the DWBT tests. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.4. The results for this calculation are presented in Table 5.3-2.

Table 5.3-1
RBS Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 10 Year ILRT/DWBT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION [2]		CHANGE DUE TO CORROSION OR DWBT [3] EXTENSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1[1]	No Containment Failure	1.48E+03	3.36E-07	4.99E-04	3.36E-07	4.98E-04	-9.745E-07
2	Large Isolation Failures (Failure to Close)	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	--
3a	Small Isolation Failures	1.48E+04	7.88E-08	1.17E-03	7.88E-08	1.17E-03	--
3b	Large Isolation Failures	1.48E+05	1.97E-08	2.92E-03	2.04E-08	3.02E-03	9.75E-05
7 LERF	Failures Induced by Phenomena (LERF)	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	4.61E+05	2.14E-06	9.89E-01	2.14E-06	9.89E-01	1.90E-03
8	Containment Bypass	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	--
CDF	**All CET end states**		**2.60E-06**	**1.013**	**2.60E-06**	**1.013**	**1.99E-03**

(1) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.
(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.
(3) The DWBT leakage cases of 10x and 100x with unit coolers unavailable are assumed to lead to an increased frequency of Class 7 (non-LERF).

Table 5.3-2
RBS Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 15 Year ILRT/DWBT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION [2]		CHANGE DUE TO CORROSION OR DWBT [3] EXTENSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1[1]	No Containment Failure	1.48E+03	2.85E-07	4.23E-04	2.83E-07	4.20E-04	-2.26E-06
2	Large Isolation Failures (Failure to Close)	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	--
3a	Small Isolation Failures	1.48E+04	1.18E-07	1.75E-03	1.18E-07	1.75E-03	--
3b	Large Isolation Failures	1.48E+05	2.96E-08	4.39E-03	3.11E-08	4.61E-03	2.26E-04
7 LERF	Failures Induced by Phenomena (LERF)	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	4.61E+05	2.15E-06	9.90E-01	2.15E-06	9.90E-01	2.85E-03
8	Containment Bypass	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	--
CDF	**All CET end states**		**2.60E-06**	**1.016**	**2.60E-06**	**1.016**	**3.07E-03**

(1) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.
(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.
(3) The DWBT leakage cases of 10x and 100x with unit coolers unavailable are assumed to lead to an increased frequency of Class 7 (non-LERF).

5.4 STEP 4 – DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr, and small changes in LERF as below 1E-6/yr. Because the ILRT/DWBT interval extension does not impact CDF, the relevant metric is LERF.

For RBS, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on the original 3-in-10 year test interval assessment from Table 5.2-2, the Class 3b frequency is 6.03E-09/yr, which includes the corrosion effect of containment. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 2.04E-08/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 3.11E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years (including corrosion effects) is 2.51E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years (including corrosion effects) is 1.07E-08/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is within Region III of Figure 4 of Reference [4] (very small changes in LERF) when comparing the 15 year results to the original 3-in-10 year requirement.

Also note that the increase in the DWBT interval results in an increase in late containment failure sequences which are not considered LERF. However, even if these sequences were considered LERF, the associated sequence frequency increase from the base case of 1.24E-9/yr for a 3-in-10 year interval to 6.18E-09/yr for a 1-in-15 year interval would only change the increase in LERF from 2.51E-08/yr to 3.00E-08/yr.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT/DWBT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of

the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI methodology [3]. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The following table shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects in which the flaw rate is assumed to double every five years, and also includes the increase to the Class 7 frequency resulting from the proposed DWBT interval extension.

CCFP 3 IN 10 YRS	**CCFP 1 IN 10 YRS**	**CCFP 1 IN 15 YRS**	**$\Delta CCFP_{15-3}$**	**$\Delta CCFP_{15-10}$**
83.42%	84.08%	84.57%	1.15%	0.49%

CCFP = [1 – (Class 1 frequency + Class 3a frequency)/CDF] x 100%

The change in CCFP of approximately 1% as a result of extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be relatively insignificant.

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

Table 5.6-1 summarizes the internal events results of this ILRT extension risk assessment for RBS.

Table 5.6-1
RBS ILRT/DWBT Cases:
Base, 3 to 10, and 3 to 15 Yr Extensions
(Including Age Adjusted Steel Corrosion Likelihood)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	1.48E+03	4.08E-07	6.05E-04	3.36E-07	4.98E-04	2.83E-07	4.20E-04
2	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	6.64E-11	3.77E-05
3a	1.48E+04	2.37E-08	3.51E-04	7.88E-08	1.17E-03	1.18E-07	1.75E-03
3b	1.48E+05	6.03E-09	8.94E-04	2.04E-08	3.02E-03	3.11E-08	4.61E-03
7 LERF	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	5.35E-09	4.18E-03
7 non-LERF	4.61E+05	2.14E-06	9.88E-01	2.14E-06	9.89E-01	2.15E-06	9.90E-01
8	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	1.93E-08	1.51E-02
Total		2.60E-06	1.009	2.60E-06	1.013	2.60E-06	1.016
ILRT Dose Rate from 3a and 3b		1.24E-03		4.19E-03		6.37E-03	
DWBT Dose Rate from 7 non-LERF		5.70E-04		1.90E-03		2.85E-03	
Delta Total Dose Rate(1)	From 3 yr	---		4.16E-03		7.22E-03	
	From 10 yr	---		---		3.06E-03	
3b Frequency (LERF)		6.03E-09		2.04E-08		3.11E-08	
Delta 3b LERF	From 3 yr	---		1.43E-08		2.51E-08	
	From 10 yr	---		---		1.07E-08	
CCFP %		83.42%		84.08%		84.57%	
Delta CCFP %	From 3 yr	---		0.66%		1.15%	
	From 10 yr	---		---		0.49%	

1. The overall difference in total dose rate is less than the difference of only the 3a, 3b, and 7 categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT/DWBT frequency.

5.7 CONTRIBUTIONS FROM OTHER HAZARD GROUPS

Since the risk acceptance guidelines in RG 1.174 are intended for comparison with a full-scope assessment of risk, including internal and external events, a bounding analysis of the potential impact from external events and other hazard groups is presented here.

Seismic Risk [8]

The recent U.S. Nuclear Regulatory Commission (NRC) Safety/Risk Assessment (SRA) of U.S. Nuclear Plant Seismic Core Damage Frequencies (SCDFs) based on the 2008 US Geological Survey (USGS) seismic hazard curves used the River Bend Station (RBS) Individual Plant Examination of External Events (IPEEE) information to perform risk assessments in support of reaching a resolution to Generic Issue (GI) 199. The report identified RBS as one of the plants requiring further review due to the increase in the calculated SCDF. Since the RBS IPEEE work included conservatisms, Entergy assembled a Seismic Review Team (SRT) tasked with developing an SCDF estimate that more closely reflects the robustness of RBS.

Although the NRC-estimated SCDF was below the target goal of 1.0E-04 per year, thus acceptable in terms of the plant seismic risk, the Seismic Review Team re-assessed the SCDF to remove excessive conservatism. The Seismic Review Team demonstrated a larger plant-level seismic capacity than that used in the NRC assessment. The NRC used a very conservative value for plant capacity. This resulted in the NRC determining a conservative SCDF estimate of 2.5E-05 per year, or 1 in 40,000 reactor-years. Using the improved plant-level capacity developed by the team, a re-assessment of the SCDF estimate was performed. This resulted in a SCDF of 2.5E-06 per year, or 1 in 400,000 reactor-years, using the same USGS hazard curves relied upon by NRC for the seismic risk assessment.

Internal Fire Risk [9]

RBS does not maintain a living Fire PRA model. Therefore, the information provided in the RBS IPEEE is deemed to provide a representative estimate of total fire risk for use in this assessment. The total Fire CDF reported in the RBS IPEEE is 2.25E-05/yr. In addition to modeling limitations, the Fire PRA may be subject to more modeling uncertainty than the internal events PRA evaluations. While the Fire PRA is generally self-consistent within its calculational framework, the Fire PRA CDF results do not compare well with internal events PRAs because of the number of conservative assumptions that have been included in the Fire PRA process. Therefore, direct use of the Fire PRA results as a reflection of CDF may be

inappropriate, and the actual fire CDF based on the IPEEE may be overestimated. In any event, the reported Fire CDF value from the IPEEE is used as a bounding value for this calculation.

Internal Flood Risk [10]

The most recent evaluation of internal flood risk at RBS was completed in 2012. This model utilized the Rev. 5 PRA model for the assessment, and addressed many of the findings from the 2011 BWROG peer review. This model is used to support Risk-Informed In Service Inspection activities at the site using ASME Code Case N-716. Due to the large number of flooding scenarios involved, bounding assumptions are developed and applied to multiple scenarios resulting in an overestimate of the CDF contribution. Although many conservative treatments remain in the model, the reported CDF value of 4.97E-06/yr will be used for the ILRT/DWBT interval extension risk assessment.

High Winds [11]

In 2004, an assessment of high winds and tornado analysis was completed for RBS. This included effects from tornado-generated missiles, wind and tornado structural loading, and tornado vacuum. The reported CDF value of 1.81E-07/yr is used for the ILRT/DWBT interval extension risk assessment.

Other External Events [9]

The RBS Individual Plant Examination of External Events (IPEEE) concluded for "Other" external events, including external floods and transportation and nearby facility accidents, that no undue risks are present that might contribute to CDF with a predicted frequency in excess of 1.0E-06/yr. As these events are not dominant contributors to external event risk and quantitative analysis of these events is not practical, they are considered negligible in estimation of the external events impact on the ILRT/DWBT extension assessment.

Other Hazard Group Contributor Summary

The method chosen to account for external events contributions is similar to that used in the other analysis in which a multiplier was applied to the internal events results [28]. The contributions of the external events from various RBS analysis are summarized in Table 5.7-1. Note that internal flooding is also included in Table 5.7-1 since internal flooding is not maintained as part of the regular internal events model for RBS.

Table 5.7-1
Other Hazard Group Contributor Summary

OTHER HAZARD INITIATOR GROUP	CDF (1/YR)
Seismic [8]	2.50E-06
Internal Fire [9]	2.25E-05
Internal Flood [10]	4.97E-06
High Winds [11]	1.81E-07
External Floods [9]	Screened
Transportation and Nearby Facility Accidents [9]	Screened
Total (for initiators with CDF available)	**3.02E-05/yr**
Internal Events CDF	2.60E-06
External Events Multiplier	**11.60**

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT/DWBT intervals are shown in Table 5.6-1 as 6.03E-09/yr, 2.04E-08/yr, and 3.11E-08/yr, respectively. Using the other hazard group multiplier of 11.60 for RBS, the change in the LERF risk measure due to extending the ILRT/DWBT from 3-per-10 years to 1-per-15 years, including both internal events and other measurable hazard groups hazards risk, is estimated as shown in Table 5.7-2.

Table 5.7-2
RBS 3b (LERF) as a Function of ILRT/DWBT Frequency for Internal and External Events
(Including Age Adjusted Steel Corrosion Likelihood)

	3B FREQUENCY (3-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-15 YEAR ILRT/DWBT)	LERF INCREASE[1]
Internal Events Contribution	6.03E-09	2.04E-08	3.11E-08	2.51E-08
Other Hazard Group Contribution (Internal Events CDF x 11.60)	6.99E-08	2.36E-07	3.61E-07	2.91E-07
Combined	7.60E-08	2.56E-07	3.92E-7	3.16E-07

(1) Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

Thus, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution is estimated as 3.2E-07/yr, which includes the age adjusted steel corrosion likelihood.

The other acceptance criteria for the ILRT/DWBT interval extension risk assessment can be similarly derived using the multiplier approach. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are shown in Table 5.7-3. As can be seen, the impact from including the other hazard group contributors would not change the conclusion of the risk assessment. That is, the acceptance criteria are all met such that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years has been demonstrated to be small. Note that a bounding analysis for the total LERF contribution follows Table 5.7-3 to demonstrate that the total LERF value for RBS is less than 1.0E-5/yr consistent with the requirements for a "Small Change" in risk of the RG 1.174 acceptance guidelines.

Table 5.7-3
Comparison to Acceptance Criteria Including Other Hazard Groups Contribution for RBS

Contributor	ΔLERF	ΔPerson-rem/yr	ΔCCFP
RBS Internal Events	2.51E-8/yr	7.22E-03/yr (0.72%)	1.15%
RBS Other Hazard Groups	2.91E-7/yr	8.37E-02/yr (0.72%)	1.15%
RBS Total	**3.16E-7/yr**	**9.09E-02/yr (0.72%)**	**1.15%**
Acceptance Criteria	**<1.0E-6/yr**	**<1.0 person-rem/yr or <1.0%**	**≤1.5%**

The 3.2E-07/yr increase in LERF due to the combined hazard events from extending the RBS ILRT/DWBT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1E-7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines.

Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-1, the total LERF due to postulated internal event accidents is the sum of the LERF release categories, which is 2.48E-08/yr. Although some of the LERF contributors may not be applicable to other hazard group initiators (e.g., ISLOCAs or Reactor Vessel Rupture events), the base LERF due to external events is assumed to be in the same proportion as the internal events contribution. The total LERF value for RBS is then shown in Table 5.7-4.

Table 5.7-4
Impact of 15-yr ILRT Extension on LERF (3b) for RBS

Internal Events LERF	2.48E-08/yr
Other Hazard Group LERF (Internal Events LERF x 11.60)	2.87E-07/yr
Internal Events LERF due to ILRT (at 15 years) (1)	3.11E-08/yr
Other Hazard group LERF due to ILRT (at 15 years) (1)	3.61E-07/yr
Total	**7.04E-07/yr**

(1) Including age adjusted steel corrosion likelihood.

As can be seen, the estimated upper bound LERF for RBS is estimated as 7.0E-07/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1E-5/yr.

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2, 5.3-1, and 5.3-2 show that including corrosion effects calculated using the assumptions described in Section 4.5 does not significantly affect the results of the ILRT/DWBT extension risk assessment. For RBS, note that only the free-standing steel containment failure probabilities from the ILRT interval extension are adjusted as the drywell bypass test does not involve any steel structures or structures with steel liners. In any event, sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder, dome and basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case, the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 4.48E-8/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1-1
Steel Corrosion Sensitivity Cases

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT/DWBT EXTENSION FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (5.0% Cylinder-Dome, 0.6% Basemat)	Base Case (10% Cylinder-Dome, 100% Basemat)	2.51E-08	1.41E-09
Doubles every 2 yrs	Base	Base	2.69E-08	3.20E-09
Doubles every 10 yrs	Base	Base	2.48E-08	1.18E-09

Table 6.1-1
Steel Corrosion Sensitivity Cases

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT/DWBT EXTENSION FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base	Base	15% Cylinder-Dome	2.56E-08	1.97E-09
Base	Base	5% Cylinder-Dome	2.45E-08	8.49E-10
Base	50% Cylinder-Dome, 5% Basemat	Base	3.78E-08	1.41E-08
Base	0.5% Cylinder-Dome, 0.05% Basemat	Base	2.38E-08	1.41E-10
LOWER BOUND				
Doubles every 10 yrs	0.5% Cylinder-Dome, 0.05% Basemat	5% Cylinder-Dome 100% Basemat	2.37E-08	7.09E-11
UPPER BOUND				
Doubles every 2 yrs	50% Cylinder-Dome, 5% Basemat	15% Cylinder-Dome 100% Basemat	6.85E-08	4.48E-08

6.2 EPRI EXPERT ELICITATION SENSITIVITY

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leaks within containment [3]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as a function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of failure mechanisms, the potential for undiscovered mechanisms, inaccessible areas of the containment as well as the potential for detection by alternate means. The expert elicitation

process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage within containment. The base case methodology uses the Jeffrey's non-informative prior for the large leak size and the expert elicitation sensitivity study uses the results from the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the base case methodology (i.e., 10La for small and 100La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the base case and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, including the input to expert elicitation as well as the results of the expert elicitation, are available in the various appendices of EPRI TR-1018243 [3].

Table 6.2-1
EPRI Expert Elicitation Results

LEAKAGE SIZE (LA)	BASE CASE	EXPERT ELICITATION MEAN PROBABILITY OF OCCURRENCE [3]	PERCENT REDUCTION
10	9.2E-03	3.88E-03	58%
100	2.3E-03	2.47E-04	89%

The summary of results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the base case evaluation (10La for small and 100La for large). The expert elicitation process produces a relationship between probability and leakage magnitude in which it is possible to assess higher leakage magnitudes that are more reflective of large early releases; however, these evaluations are not performed in this particular study.

The net effect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall value for LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 3.95E-09/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.93E-09/yr. As such, if the expert elicitation mean probabilities of occurrence are used instead of the non-informative prior estimates, the change in LERF for RBS is much further within the range of a "very small" change in risk when compared to the current 1-in-10, or baseline 3-in-10 year requirement. The results of this sensitivity study are judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the values from the EPRI methodology [3], and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

6.3 DWBT DATA SENSITIVITY

For RBS, one additional sensitivity is included related to the interpretation of the DWBT data used for the 10 L_b base case assessment. If two additional data points are considered to fall in the 2-10 range even though none of them are close to 10 L_b leakage rate assumed for the intermediate category in this assessment, the likelihood of the 10 L_b leakage rate can be conservatively assumed to be represented by four events over the total number of representative test results from the time frame of interest used to establish the base case leakage likelihoods.

Likelihood of 10 DWL_b = 4 / 24 = 0.167 = 16.7%

Note that since the 10 DWL_b probability of occurrence is assumed to grow by a factor of 3.33 and 5.0 per the accepted methodology, then this is the maximum base case failure probability that can be assumed since 5 * 16.7% = 83.3% is the failure rate used to represent the 15 year extended test interval. (If an additional failure is considered for the base case, then 5x that value would exceed 100% likelihood of occurrence.) For this sensitivity case, no changes to the 100 L_b DW leakage rate are made since this is already considered to be bounded by the use of a Jeffrey's non-informative prior to determine the likelihood of occurrence. The summary of results using the revised values for probability of 10 L_b drywell bypass leakage is provided in Table 6.2-3. The results indicate increases to the population dose and to the CCFP values compared to the base risk assessment, but the results are all still within the acceptance criteria of less than 1.0 person-rem/yr or less than 1.0% person-rem/yr, and less than 1.5% change in CCFP.

Table 6.2-2
RBS ILRT/DWBT Cases:
3 in 10 (Base Case), 1 in 10, and 1 in 15 Yr intervals
(ILRT Leakage Based on EPRI Expert Elicitation Probabilities)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	1.48E+03	4.27E-07	6.33E-04	3.99E-07	5.92E-04	3.78E-07	5.61E-04
2	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	6.64E-11	3.77E-05
3a	1.48E+04	9.98E-09	1.48E-04	3.32E-08	4.93E-04	4.99E-08	7.40E-04
3b	1.48E+05	7.50E-10	1.11E-04	2.77E-09	4.11E-04	4.70E-09	6.97E-04
7 LERF	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	5.35E-09	4.18E-03
7 non-LERF	4.61E+05	2.14E-06	9.88E-01	2.14E-06	9.89E-01	2.15E-06	9.90E-01
8	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	1.93E-08	1.51E-02
Total		2.60E-06	1.008	2.60E-06	1.010	2.60E-06	1.011
ILRT Dose Rate from 3a and 3b		2.59E-04		9.04E-04		1.44E-03	
DWBT Dose Rate from 7 non-LERF		5.70E-04		1.90E-03		2.85E-03	
Delta Total Dose Rate[1]	From 3 yr	---		1.94E-03		3.39E-03	
	From 10 yr	---		---		1.45E-03	
3b Frequency (LERF)		7.50E-10		2.77E-09		4.70E-09	
Delta 3b LERF	From 3 yr	---		2.02E-09		3.95E-09	
	From 10 yr	---		---		1.93E-09	
CCFP %		83.22%		83.40%		83.56%	
Delta CCFP %	From 3 yr	---		0.19%		0.34%	
	From 10 yr	---		---		0.15%	

(1) The overall difference in total dose rate is less than the difference of only the 3a, 3b, and 7 categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT/DWBT frequency.

Table 6.2-3
RBS ILRT/DWBT Cases:
3 in 10 (Base Case), 1 in 10, and 1 in 15 Yr intervals
(DWBT 10 L_b Based on More Conservative Interpretation of Data)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	1.48E+03	4.07E-07	6.04E-04	3.33E-07	4.93E-04	2.79E-07	4.13E-04
2	5.68E+05	6.64E-11	3.77E-05	6.64E-11	3.77E-05	6.64E-11	3.77E-05
3a	1.48E+04	2.36E-08	3.50E-04	7.86E-08	1.17E-03	1.18E-07	1.75E-03
3b	1.48E+05	6.02E-09	8.92E-04	2.03E-08	3.01E-03	3.10E-08	4.60E-03
7 LERF	7.81E+05	5.35E-09	4.18E-03	5.35E-09	4.18E-03	5.35E-09	4.18E-03
7 non-LERF	4.61E+05	2.14E-06	9.88E-01	2.15E-06	9.90E-01	2.15E-06	9.92E-01
8	7.81E+05	1.93E-08	1.51E-02	1.93E-08	1.51E-02	1.93E-08	1.51E-02
Total		2.60E-06	1.009	2.60E-06	1.014	2.60E-06	1.018

ILRT Dose Rate from 3a and 3b		1.24E-03	4.18E-03	6.35E-03
DWBT Dose Rate from 7 non-LERF		1.03E-03	3.44E-03	5.17E-03
Delta Total Dose Rate[1]	From 3 yr	---	5.23E-03	9.05E-03
	From 10 yr	---	---	3.82E-03
3b Frequency (LERF)		6.02E-09	2.03E-08	3.10E-08
Delta 3b LERF	From 3 yr	---	1.43E-08	2.50E-08
	From 10 yr	---	---	1.07E-08
CCFP %		83.46%	84.21%	84.76%
Delta CCFP %	From 3 yr	---	0.75%	1.30%
	From 10 yr	---	---	0.56%

(1) The overall difference in total dose rate is less than the difference of only the 3a, 3b, and 7 categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT/DWBT frequency.

7.0 CONCLUSIONS

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency and the DWBT frequency to fifteen years:

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines "very small" changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. "Small" changes in risk are defined as increases in CDF below 10^{-5}/yr and increases in LERF below 10^{-6}/yr. Since the ILRT extension has no impact on CDF for RBS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT interval and the DWBT interval for the base case with corrosion included is 2.51E-08/yr (see Table 6.1-1), which falls within the "very small" change region of the acceptance guidelines in Reg. Guide 1.174.
  - If the EPRI Expert Elicitation methodology Class 3a and Class 3b failure probabilities are used, the change is estimated as 3.95E-09/yr (see Table 6.2-2), which falls further within the very small change region. If more conservative assumptions are utilized regarding the interpretation of the DWBT data, there is a negligible impact on LERF as this change mostly impacts the change in population dose and CCFP probabilities (see Table 6.2-3). However, even if the increase in the late containment failure cases from the DWBT extension were considered as LERF scenarios, the change in LERF would only be 3.40E-08/yr, which still falls within the "very small" change region of the acceptance guidelines in Reg. Guide 1.174.
- The change in dose risk for changing the Type A ILRT interval and the DWBT interval from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all accident sequences, is 7.22E-03 person-rem/yr using the EPRI guidance with the base case corrosion included (see Table 5.6-1). This change meets both of the related acceptance criteria identified in Section 1.3 for change in population dose of less than 1.0 person-rem/ year or less than 1% person-rem/yr.
  - The change in dose risk drops to 3.39E-03 person-rem/yr when using the EPRI Expert Elicitation methodology (see Table 6.2-2). The change in dose risk increases to 9.05E-03 person-rem/yr when a more conservative interpretation of the historical DWBT data is implemented (see Table 6.2-3), but this value still meets both of the related acceptance criteria identified in Section 1.3 for change in population dose of less than 1.0 person-rem/ year or less than 1% person-rem/yr.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance (see Table 5.6-1) is 1.15%, which is below the acceptance criteria of 1.5% identified in the NRC SE on the issue [7] as discussed in Section 1.3.

  - The increase in CCFP drops to about 0.34% using the EPRI Expert Elicitation methodology (see Table 6.2-2). The change in CCFP increases to 1.30% when a more conservative interpretation of the historical DWBT data is implemented (see Table 6.2-3), but this value still meets both of the related acceptance criteria identified in Section 1.3 for change in CCFP of less than 1.5%.

- To determine the potential impact from other hazard groups, an additional bounding assessment from the risk associated with the other relevant hazard groups for RBS utilizing the latest information from various sources was performed. As shown in Table 5.7-3, the total increase in LERF due to internal events and other hazard groups is 3.16E-07/yr, which is in Region II of the Reg. Guide 1.174 acceptance guidelines. As also shown in Table 5.7-3, the other acceptance criteria for change in population dose and change in CCFP are also still met when the other hazard groups are considered in the analysis.
- Finally, as shown in Table 5.7-4, a similar bounding analysis for the other hazard groups indicates that the total LERF from both internal events and the other hazard groups is 7.04E-07/yr, which is less than the Reg. Guide 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II (small change in risk).

Therefore, increasing the ILRT and DWBT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be significant since it represents only a small change in the RBS risk profile.

Previous Assessments

The NRC in NUREG-1493 [6] has previously concluded the following:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for RBS confirm these general findings on a plant specific basis for the ILRT/DWBT interval extension considering the severe accidents evaluated for RBS, the RBS containment failure modes, and the local population surrounding RBS.

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Appendix A to Attachment 3

RBG-47620

PRA Technical Adequacy

A.1 OVERVIEW

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this report to help support an extension of the RBS containment Type A integrated leak rate test (ILRT) and drywell bypass test (DWBT) intervals to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 2 [A.1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed to perform this study:

1. Identify the parts of the PRA used to support the application
   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model.
   - A definition of the acceptance criteria used for the application.
2. Identify the scope of risk contributors addressed by the PRA model
   - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.
   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 are covered in the main body of this report. The purpose of this appendix is to address the requirements identified in item 4 above. Each of these items (plant changes not yet incorporated into the PRA model, relevant peer review findings, consistency with applicable PRA standards and the identification of key assumptions) are discussed in the following sections.

The risk assessment performed for the ILRT/DWBT extension request is based on the current Level 1 and Level 2 PRA models of record. These models comprise Revision 5 of the RBS PRA, which was released in April 2011. Note that for this application, the accepted methodology involves a bounding approach to estimate the change in the LERF and population dose from extending the ILRT/DWBT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability

Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

As discussed, the PRA models used for this application are the latest Revision implemented at River Bend. There are no significant plant changes (design or operational practices) that have not yet been incorporated in those PRA models, other than those associated with FLEX. RBS completed plant modifications associated with the FLEX initiative (NEI-12-06) during the Spring 2015 refueling outage, and is in compliance with the NRC FLEX order; these changes will result in improvements in plant mitigating capability with resultant decreases in CDF, adding conservatism to the ILRT risk assessment for River Bend.

A discussion of the Entergy model update process, the peer reviews performed on the RBS model, the results of those peer reviews and the potential impact of peer review findings on the ILRT/DWBT extension risk assessment are provided in Section A.2. Section A.3 provides an assessment of key assumptions and approximations used in this assessment and Section A.4 briefly summarizes the results of the PRA technical adequacy assessment with respect to this application.

A.2 PRA Update Process and Peer Review Results

A.2.1 Introduction

The RBS Probabilistic Risk Assessment (PRA) models used for this application [A.2 and A.3] are the most recent evaluations of the RBS risk profiles for internal event challenges. The RBS PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause failure events. The PRA model quantification process is based on the event tree and fault tree methodology, which is a well-known methodology in the industry.

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

A.2.2 PRA Maintenance and Update

The Entergy risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the Entergy fleet procedure EN-DC-151, "PSA Maintenance and Update" [A.4]. This procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Entergy nuclear power plants. In addition, the procedure also defines the process for implementing regularly scheduled and interim PRA model updates, and for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.). To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. Potential PRA model changes resulting from these reviews are entered into the Model Change Request (MCR) database, and a determination is made regarding the significance of the change with respect to current PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.

- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated regularly. The corporate procedure is being revised to suggest an update frequency of approximately every five years.
- Industry standards, experience, and technologies are periodically reviewed to ensure that any changes are appropriately incorporated into the models.

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

A.2.3 Regulatory Guide 1.200 PWROG Peer Review of the RBS Internal Events PRA Model

The RBS internal events model went through a Regulatory Guide 1.200 BWR Owners Group peer review using the NEI 05-04 process. The RBS PRA internal events model peer review performed in April 2011, used RA-Sa-2009 (the American Society of Mechanical Engineers / American Nuclear Society Combined PRA Standard) and Regulatory Guide 1.200 Revision 2.

The RBS PRA peer review addressed all the technical elements of the internal events, at-power PRA:

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

During the RBS PRA model peer reviews, the technical elements identified above were assessed with respect to Capability Category II criteria to better focus the Supporting Requirement assessments.

A.2.4 Peer Review Results

The ASME PRA standards used for the RBS peer reviews contained a total of 325 numbered supporting requirements. A number of the supporting requirements were determined to be not applicable to the RBS PRA (e.g., PWR related, multi-site related). Of the applicable supporting requirements, more than 85% were satisfied at Capability Category II or greater for RBS with the majority of the supporting requirements not meeting Supporting Requirements related to the Internal Flooding elements. The Peer Review Team generated a total of 59 Findings which are provided in the peer review report [A.5]. Seven of these Findings were against LERF-related Supporting Requirements, as the River Bend LERF model was developed as a NUREG/CR-6595 simplified LERF model, which is defined as meeting Capability Category I of the Standard. The River Bend Internal Flooding Analysis was subsequently revised in 2012 to fulfill a commitment established via Entergy letter RBG-47029 dated 5/14/2010 to update the basis to PRA Revision 5 and to resolve many of the Findings related to Internal Flooding elements of the Standard.

River Bend is in the process of compiling an Interim Revision 5-A to its PRA that will incorporate resolution of selected Findings. Remaining open Findings against the Internal Events model will be addressed or incorporated as part of future full model update under Revision 6 to the PRA.

As a result of the Regulatory Guide 1.200 BWROG peer reviews, all the Facts and Observations (F&Os) (other than best practices) identified as potential improvements to the RBS PRA models or documentation were entered into the Entergy Model Change Request (MCR) database. Table A.2-1 contains the remaining 29 open findings resulting from the peer review for RBS, the status of the resolution for each finding and the potential impact of each finding on this application. This includes the findings that will be closed with implementation of PRA Revision 5A. In summary, a majority of the findings were related to documentation and have no material impact. Resolution of the remaining open peer review findings is not expected to significantly change the total internal events CDF for RBS such that there would be no changes to the conclusions of this application.

A.2.5 External Events

Although EPRI report 1018243 [A.6] recommends a quantitative assessment of the contribution of external events (for example, fire and seismic) where a model of sufficient quality exists, it also recognizes that the external events assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval. Since the most current external events models for RBS have not been peer reviewed, a multiplier was applied to the internal events results based on the available information, similar to that used in other ILRT analyses. This is further discussed in Section 5.7 of the risk assessment.

A.2.6 Summary

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

Resolution of Findings.

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
1	DA-C8 (Cat.I)	Cat.I: When required, ESTIMATE the time that components were configured in their standby status. Cat.II/III: When required, USE PLANT-SPECIFIC OPERATIONAL RECORDS TO DETERMINE the time that components were configured in their standby status.	This is a finding since the technical requirements of the SR were not met for Capability Category II	Table C-2A of PRA-RB-01-002S05 discusses the rationale used to determine run times. While a few components (e.g., SW pumps) appear to be collect actual run vs. standby time in the supporting spreadsheets, the standby time for most components in running systems was estimated (e.g., 1/2, 1/3, etc.). Therefore Category I is met.	To meet Category II, actual plant experience concerning standby/run fractions needs to be collected and used in the PRA.	Use of actual vice estimated availability for other components would be expected to have very small impact on PRA results. Actual availabilities are used for components for which this information is tracked, which includes those components monitored by MSPI, which tend to have higher PRA importance (e.g., diesel generators, RHR pumps, Standby Service Water pumps). This Finding will be addressed as a possible enhancement to the next periodic PRA model update. No impact on the RBS ILRT extension request.
2	AS-B3 (Not Met)	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	This is a finding because the requirements of the SR are not met.	There is not a specific discussion of the phenomenological impacts of each initiator upon the mitigating systems in the AS notebook (PRA-RB-01-002S01). One specific exception to this is the impacts of debris entrainment for ECCS following LOCA, which is discussed for the Large and Intermediate LOCA event trees. It appears that phenomenological impacts are addressed in the AS logic for all initiators. However, documentation of other impacts (or noting the absence of any impacts) should be provided.	Include an explicit discussion of phenomenological impacts (or lack thereof) for each event tree.	This is considered a documentation issue, as noted in the review comment. For the example of the MSPI application, Table G.5 of NEI 99-02 provides comments to focus on credit for injection post-venting (NPSH issues, environmental survivability) which are thoroughly addressed for the RBS PRA. The environmental effects of containment failure are explicitly considered to result in failure of Auxiliary Building equipment credited for core damage mitigation. As discussed in the PRA Success Criteria calculation, debris effects are considered for Medium Break LOCA and Large Break LOCA resulting in a more restrictive success criteria for those events; this is accounted for via the PRA Event Tree and through recovery rules. Environmental phenomena are thoroughly considered, as documented for the case of internal flooding in Att.2 to letter RBG-46944 dated August 11, 2009. Systems credited for BOC scenarios are unaffected by those breaks. Room heatup effects are fully considered, as documented in the Success Criteria calculation. No impact on the RBS ILRT extension request.
3	SY-A24 (Not Met) (DA-C15: Met))	SY-A24: DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an adequate analysis or examination of data. (See DA-C15.) DA-C15: For each SSC for which repair is to be modeled (see SY-A22), IDENTIFY instances of plant-specific or applicable industry experience and for each repair, COLLECT the associated repair time with the repair time being the period from identification of the component failure until the component is returned to service.	SY-A24: Several equipment recovery events are included in the models. These are a diesel recovery (ZHE-FO-DGN1HRS) and a decay heat removal recovery (ZRCXHE-FO-DHRLT). While a basis for these events is provided in the HRA notebook (PRA-RB-01-002S03), the bases may not justify the use of the specific data values used, as required by this SR. For the diesel repair, the value for the non-repair probability is based on a generic analysis using industry data and various assumptions. No documentation was provided to demonstrate that the data would be applicable to River Bend as required by this SR. For the DHR repair, the data used to develop this non-recovery probability is based on an EPRI report concerning recovery of loss of DHR cooling during shutdown conditions. As shutdown conditions can vary significantly from at-power systems (in terms of factors such as number of cooling water trains available, etc.), the data may not be applicable to the situation being evaluated in the River Bend	Several equipment recovery events are included in the models. These are a diesel recovery (ZHE-FO-DGN1HRS) and a decay heat removal recovery (ZRCXHE-FO-DHRLT). While a basis for these events is provided in the HRA notebook (PRA-RB-01-002S03), the bases may not justify the use of the specific data values used, as required by this SR. For the diesel repair, the value for the non-repair probability is based on a generic analysis using industry data and various assumptions. No documentation was provided to demonstrate that the data would be applicable to River Bend as required by this SR. For the DHR repair, the data used to develop this non-recovery probability is based on an EPRI report concerning recovery of loss of DHR cooling during shutdown conditions. As shutdown conditions can vary significantly from at-power systems (in terms of factors such as number of cooling water trains available, etc.), the data may not be applicable to the situation being evaluated in the River Bend PRA.	As required by this SR, evaluate the repair data used for the diesel and DHR repair terms for applicability to River Bend and document this evaluation. If necessary, select more representative data sources for these repair events.	Diesel Generator recovery ZHE-FO-DGN1HRS has been reviewed and confirmed to be specifically applicable to River Bend Station. A detailed study has been performed for the long-term decay heat removal recovery, ZRC-XHE-FO-DHRLT. This was conducted in tandem with a detailed review of the modeling of the Loss of Normal Service water initiating event. The basis for the long-term decay heat removal recovery was improved and made plant specific, based upon industry actuarial data. When refining the modelling for this recovery, the recovery was modelled in more detail. Entergy is incorporating this enhanced modeling into the River Bend PRA as Interim Revision 5A. Use of the Revision 5 model for ILRT extension risk assessment is considered acceptable. Diesel Generator recovery modeling is concluded to be appropriate or slightly conservative for River Bend Station. The long term decay heat removal recovery does not affect short-term sequences, thus does not impact LERF. These long term decay heat removal recoveries involve prevention of containment failure late into events (e.g., 16 hours or longer) where containment failure which results in core damage is prevented by recovering the ability to remove decay heat from the containment. Such failures would correspond to ILRT non-LERF accident Class 7. Class 7 sequences are not impacted by ILRT Type A tests. Enhancements to the Loss of Normal Service water initiating event model credit the ability of the plant to operate with a

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			PRA. DA-C15: Repair is considered for diesel generators and decay heat removal. However, the generic data used for these events may not be applicable (see SY-A24)			combination of Normal Service Water and Standby Service Water pumps, resulting in a decreased frequency for this initiator. The overall effect of the combination of these model improvements would be to reduce LERF contributions and increase late non-LERF contributions. Additionally, FLEX modifications implemented at River Bend in 2015 provide additional means of preventing long-term containment overpressurization by providing alternative means of containment and suppression pool cooling. It is thus concluded that PRA Rev.5 provides an appropriate basis for the risk assessment in support of the ILRT extension request.
5 4	AS-A11 (Met)	Transfers between event trees may be used to reduce the size and complexity of individual event trees. DEFINE any transfers that are used and the method that is used to implement them in the qualitative definition of accident sequences and in their quantification. USE a method for implementing an event tree transfer that preserves the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, and spatial or environmental dependencies.	This is a finding as the PRA does not appear to properly perform the noted event tree transfer	Transfers between event trees is accomplished through a direct transfer of the sequence logic through the top logic model. This approach generally retains the dependencies through the quantification process. However, in the case of transfers between the LOSP and ATWS event trees, it does not appear that sequence information is properly transferred, as no LOSP/ATWS cutsets appear in the results.	Review the fault tree and event tree logic associated with the LOOP to ATWS Sequence transfer and modify as necessary to obtain proper results.	A sensitivity analysis was performed to determine the impact of this Finding. Since the initiating event frequency for Loss of Offsite Power (LOOP) is relatively small compared to other initiators, and since RPS has high reliability, only a small increase in CDF results. Requantifying the model resulted in an increase in CDF of 5.9E-11, or about a 1% increase in the ATWS-only CDF. Overall CDF remained unchanged to four significant figures. Fault tree changes to incorporate resolution of this item have been incorporated into the RBS PRA model. Because this logic error had so little impact on calculated CDF, it does not impact the ability of the RBS Rev.5 PRA model to be applied for purposes of ILRT. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This resolved Finding has no impact on the RBS ILRT extension request.
6 5	HR-D3 (Cat.I)	Cat.I: No requirement for evaluating the quality of written procedures, administrative controls, or human-machine interfaces. Cat. II/III: For each detailed human error probability assessment, INCLUDE in the evaluation process the following plant-specific relevant information: (a) the quality of written procedures (for performing tasks) and administrative controls (for independent review) (b) the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout	This is a finding since the technical requirements of this SR are not met.	Since HLR-HR-D concerns pre-initiating events, No evidence of an evaluation process for the quality of pre-initiator written procedures and the quality of the pre-initiator human-machine interface could be found anywhere in the River Bend PRA documentation. Note that Post Initiator procedures have been evaluated for quality (Section 1.4.1) as well as the quality of the man machine interface (Section 1.4.3.) in the RBS HRA/Rule Recovery Work Package, Calculation PRA-RB-01-002S03	Perform an assessment of the quality of pre-initiator procedures and man-machine interface.	This is considered to be primarily an issue of increasing the robustness of PRA model documentation. Only negligible or very slight changes in PRA would be expected as a result of the review of pre-initiator procedures. Any inadequacy in the procedures associated with pre-initiator human failure events would be evidenced during the construction of the detailed spreadsheet calculations for these probabilities. These spreadsheets include documentation and review of the procedure references for each individual pre-initiator event, as well as review of the procedures and nature of indications for the calculation of the basic human error probability. The Man-machine interface quality discussion of section 1.4.3 is also generally applicable to pre-accident initiator actions as well as post-accident actions. No procedural inadequacies were noted during the development of these HRA calculations. Procedure RBNP-001, "Development and Control of RBS Procedures," governs plant operations procedures. RBNP-001 includes requirements for Technical Verification and Validation of procedures to ensure procedure quality. Thus, the intent of the SR is fulfilled through the HRA calculation process. This Finding remains open as a documentation enhancement to consider for the next PRA update. This finding has no impact on the ILRT Extension Request.
7 6	IFQU-A7	PERFORM internal flood sequence quantification	This is a finding because a significant	Quantification of the flooding model is documented in	Re-perform the flooding	At the time of the RBS Rev.5 PRA peer review, the Internal

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	(Not Met)	in accordance with the applicable requirements described in 2-2.7.	number of technical issues are noted with the flood quantification process	PRA-RB-01-006. The quantification process is carried out separately from the internal events model, and is based on a prior version of the internal events model. The approach used is that, for each flood scenario, to modify the internal events model to fail the components impacted by the flood and setting the initiator to which the flood is mapped to true (while setting all others to false). This computes a CCDP and CLERP, which are then multiplied by the scenario frequency to obtain a sequence frequency. The base model quantification is performed using appropriate codes, truncation levels, recovery and dependency rules, etc. The following specific issues are noted with respect to complying with the QU requirements of the Standard at the Category II level: - the internal events model used is inconsistent with the model used for non-flood initiators - evaluation of the flooding results is not performed to the same level as the non-flood initiators (e.g., significant components, human actions, etc.) - since scenario-by scenario numerical results are being summed to compute a total flood CDF and LERF (using the rare event approximation), the computed total frequencies may be conservative - since the model is not integrated with the other initiators, it is not possible to compute an overall importance for components and human actions - parametric uncertainty analysis is not performed on the flooding results - Reviews of the results for reasonableness are not documented, nor are reviews of non-significant cutsets.	quantification using the current internal events model, using methods that allow for an integrated quantification with the non-flood events. Perform a parametric uncertainty analysis. Perform and document reviews of the flooding results in a manner consistent with that used for the non-flood initiators.	Flooding PRA remained based the previous Rev.4 PRA. RBS has subsequently reperformed the internal flooding quantification using Revision 5 of the RBS PRA in 2012. RBS had committed via letter RBG-47029 dated 5/14/2010 to requantify the Internal Flooding PRA using PRA Revision 5. This requantification addressed many of the findings from the 2011 peer review. The RBS internal flood model is used in support of Risk-Informed In-Service Inspection (RIISI) activities at the site using ASME Code Case N-716. Although conservatisms remain in the model, it is used in the Section 5.7 bounding analysis of potential impact of ILRT extension from external events and other hazard groups, which shows the estimated upper bound LERF for RBS is less than RG 1.174 requirements. Thus, the Flooding model was made consistent with the Internal Events model and evaluated in a consistent manner, albeit including conservatisms for simplicity due to the large number (~500) of flooding scenarios. While it is not possible to capture component and human action importances because the model cannot be integrated, the conclusions in the flooding quantification calculation discuss relative contributions from different buildings, systems, pipe failure sizes, etc. Results, including cutsets, have been reviewed for reasonableness as documented in the flooding quantification calculation. Thus, any impacts associated with this finding have been accounted for in the ILRT Extension Request report..
11 7	DA-C10 (Cat.I)	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. IF THE COMPONENT FAILURE MODE IS DECOMPOSED INTO SUBELEMENTS (OR CAUSES) THAT ARE FULLY TESTED, THEN USE TESTS THAT EXERCISE SPECIFIC SUBELEMENTS IN THEIR EVALUATION. THUS, ONE SUBELEMENT SOMETIMES HAS MANY MORE SUCCESSES THAN ANOTHER. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid test would be significantly decreased.]	This is a finding since the Category II requirements for this SR are not met. This was judged to meet Category I of the Standard: "When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operations."	Surveillance tests are not decomposed into sub-elements.	Decompose surveillance tests into subelements.	Only slight or negligible changes to plant specific data would be expected to result from consideration of subelements of surveillance test procedures. The River Bend PRA does not decompose failure modes into subelements. This element was judged as acceptably meeting the PRA Standard (Category I). Documentation to address this finding will be added to the Data Analysis workbook as part of the next periodic PRA Revision update. This finding does not impact the ILRT Extension Request.
13 8	MU-C1 (Met)	The PRA configuration control process shall consider the cumulative impact of pending changes in the performance of risk applications.	This is a finding because the guideline being used is not mandatory and the cumulative impact of pending model changes is not tracked or measured for their impact on each specific applications.	In Engineering Guide EN-NE-G-026, Revision 0, 'Probabilistic Safety Assessment Applications', all open F&Os, MCRs, and gaps impacting an application are reviewed against a specific application. Justification is provided as to why open items in the model are acceptable for the application or why they do not impact the results. However, it is	Make the evaluation process used to assess impacts on risk applications a formal procedure. Also, implement a method that considers the cumulative	The ILRT Extension Request uses Revision 5 of the RBS PRA, the Model of Record compiled and released in early 2011. Miscellaneous changes have been incorporated in the model used for on-line risk assessment, which have resulted in a small decrease in Core Damage Frequency. Interim Revision 5A is scheduled for implementation in late 2015 or early 2016; the expected impact of Revision 5A is discussed with relation to the

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				not mandatory to follow this guideline; this needs to be made mandatory. In accordance with EN-DC-151 Revision 2, the cumulative impact of pending model changes is not tracked. Per the guidance, only when a model change request for an implemented change is graded A, or there are over 25 open model change requests that are graded B for a particular model will an interim PRA update be implemented. However, a method to measure the cumulative impact of pending changes particularly on the particular applications of concern should be implemented to fully meet the intent of this SR.	impact of pending changes with respect to the specific applications that are in effect for the plant.	finding for SY-A24 elsewhere in this table. That impact has been assessed and determined to not appreciably impact the ILRT Extension Request. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This finding thus does not appreciably impact the ILRT Extension Request.
15 9	SC-A3 (Not Met)	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	This is a finding since the requirements of the SR are not met for all initiating events.	Based on information in PRA-RBS-01-002S14, Although Section 4.0 provide success criteria relevant to the equipment needed for all key safety functions that involve LOCAs and general transients, success criteria for support system initiators, LOCAs outside containment, and ISLOCAs are missing.	Include appropriate success criteria for support system initiators, LOCA's outside containment, and ISLOCA's.	This Finding is considered to involve documentation of success criteria. Scenario specific success criteria has been considered in the development of RBS Accident Sequence and Success Criteria calculations. Much of the discussion of success criteria is implicit and included under discussion of Success Criteria for individual top events in the RBS Event Trees. Success criteria has been explicitly considered in the development of the Event Trees and in treatment of the support systems for the Event Tree top events. Support system initiating events have the same success criteria as other RBS transients. Specifically, conservative assumptions regarding potential environmental and inventory effects are included in the treatment of Interfacing Systems LOCA (ISLOCA) and Breaks Outside Containment (BOC), which are only small contributors to RBS CDF. The success criteria for each event tree top for BOC is documented in the BOC calculations. Success criteria for ISLOCA are the same as for LOCA, except only limited top gates (depressurization and Standby Service Water cross-tie through RHR) are credited to prevent core damage for ISLOCA. Documentation in this area will be enhanced as part of the next periodic PRA update (Rev.6) for River Bend. This Finding has no impact upon the ILRT Extension Request.
16 10	SY-A4 (Met)	PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	This is a finding because the intent of this SR was not met, since the degree of documentation is insufficient.	PRA-RB-01-002S11, Based on information provided by the PSA group, plant walkdowns have been conducted to ensure the system model correctly reflects the as-built, as-operated plant. However, limited evidence exists that interviews have been conducted to ensure the system model correctly reflects the as-built, as-operated plant.	Provide solid evidence and documentation that interviews/reviews with knowledgeable plant personnel (i.e., system engineers) occurred to document that the system model correctly reflects the as-built as-operated plant.	. System engineers participated in the Expert Panel review documented in the Integration & Quantification package. The PRA model is continually subject to discussion with system engineers as part of the Maintenance Rule Expert Panel and as periodic plant issues arise. System Engineering also reviews risk information related to PRA model revisions (e.g., documentation of risk ranking for Revision 5). The site PRA engineer also reviews the Maintenance Rule Basis Documents, providing further interaction between PRA and System Engineers on PRA assumptions for plant systems. Thus, numerous opportunities exist and have been utilized for review of the RBS PRA by knowledgeable plant personnel, including system engineers. Documentation in this area will be enhanced as part of the next periodic PRA update (Rev.6) for River Bend. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This documentation related finding has no impact upon the ILRT Extension Request.

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17 11	SY-A4 (Met) SY-B8 (Not Met)	A4: PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant. B8: Identify spatial and environmental hazards that may impact multiple systems or redundant components in the same system, and ACCOUNT for them in the system fault tree or the accident sequence evaluation.	This is a finding because information gathered from the system walkdowns are not reflected in the system notebooks documentation. Hence, a review of spatial dependencies and harsh environment operation with a potential to impact system PSA function cannot be adequately ascertained.	PRA-RB-01-002S11 R1 states that plant walkdowns were used to identify spatial and environmental hazards. Attachment A of that document contains a set of completed walkdown forms. A sample of those forms were reviewed; many of the forms indicated the existence of some kind of spatial or environmental hazard for the walkdown area. Review of several system notebooks did not reveal any indication that the identified spatial and environmental hazards identified in the walkdowns were reviewed for inclusion in or exclusion from the system models. No evidence was found that identified hazards were accounted for in the system or integrated fault tree model. (SY-A4) PRA-RB-01-002S11, Based on documentation provided by the PSA group during the Peer Review, walkdowns have been performed, however, these walkdowns do not discussed spatial and environmental hazards that may impact multiple systems or redundant components in the same system in the system notebooks.	Incorporate information from the walkdowns in the system notebooks documentation and reflect in models, as appropriate.	Spatial and environmental hazards that may impact multiple systems or redundant components are addressed in the Internal Flooding PRA. There are no impacts of this documentation issue upon the results of the Internal Events PRA. Additional walkdown information is documented in the Internal Flooding Analysis document. Also, SR SY-A4 was addressed for RBS in the 11 August 2009 submittal of supplementary information for adoption of ASME code case N-716 for Risk-Informed In-Service Inspection. Many of the environmental conditions documented in the walkdown notes in the Systems Analysis package are conditions which do not impact equipment operation and/or would be accounted for in any HRA calculations. For example, high temperatures were noted for many locations, but these would have been temperatures in the 90's since the walkdowns were conducted in the summer; these temperatures do not impact equipment performance and are considered in the overall assessment of HRA calculations. This finding is concluded to be a documentation issue. Documentation will be enhanced to address this as part of the next periodic update of the RBS PRA. This documentation finding has no impact on the ILRT Extension Request.
18 12	LE-A5 (Not Met)	DEFINE plant damage states in a manner consistent with LE-A1, LE-A2, LE-A3, and LE-A4.	This is a finding because use of NUREG/CR-6595 methodology is used to transfer results from Level 1 directly into the LERF model. This method is adequate for Capability Category I.	Plant damage states are not defined in a manner which accounts for both physical and sequence characteristics. The interface between the Level 1 and containment event tree is based on NUREG/CR-6595 and does not adequately account for all potential dependencies between the systems.	Strongly suggest the development of plant damage states or equivalent (i.e., core damage accident classes). The definition of plant damage states allows for additional containment event tree modeling to account for a more refined Mark III severe accident phenomena behavior.	This finding is documentation in nature and has no impact on LERF results. While the RBS LERF model does not define Plant Damage States, this does not impact the calculation of LERF. This only results in increased difficulty in extracting LERF-related risk insights from the model. SR's LE-A1 through LE-A4 which provide the input for SR LE-A5 were all characterized as "Met" for the RBS PRA Peer Review. RBS plans to document Plant Damage States as part of the LERF calculation for the next regular PRA update. This documentation finding does not impact the ILRT Extension Request.
27 13	IE-A2 (Met) IE-A5 (Met)	IE-A2: INCLUDE in the spectrum of internal-event challenges considered at least the following general categories: (a) Transients. INCLUDE among the transients both equipment and human-induced events that disrupt the plant and leave the primary system pressure boundary intact. (b) LOCAs. INCLUDE in the LOCA category both equipment and human-induced events that disrupt the plant by causing a breach in the core coolant system with a resulting loss of core coolant inventory. DIFFERENTIATE the LOCA initiators, using a defined rationale for the differentiation. Examples of LOCA types include (1) Small LOCAs. Examples: reactor coolant pump seal LOCAs, small pipe breaks (2) Medium LOCAs. Examples: stuck open safety or relief valves (3) Large LOCAs. Examples: inadvertent ADS, component ruptures (4) Excessive LOCAs (LOCAs that cannot be	IE-A2: Based on information in PRA-RB-01-002S06, Section 4.0, Appendix C, D, E, F, G, H, K, and I general spectrum of internal-event challenges have been considered as potential initiating events. The IE notebook includes: (a) transients, except LOSP, (b) (1) Small, (2) Medium, (3) Large LOCAs, (4) vessel rupture, and (e) special initiators including loss of RPCCW, TPCCW, NSW, loss of a single DC bus, loss of a single non-safety bus. (a) LOSP is included in a notebook specific to that initiator. (b)(5) LOCAs outside containment are included in a notebook specific to breaks outside containment. (c) SGTR is not applicable. The spectrum of LOOP events is broken down into the four generally accepted subsets consistent with NUREG-6890 (grid	This is a finding because there is at least one case of a unique initiating event not being considered. The systematic process by which plant systems are reviewed for potential to cause an initiating event is not described. Some of the results of the system screening do not appear to be complete. LPCS pipe break would constitute a unique type of LOCA with failure of a mitigating system.	Document the systematic approach used. That systematic approach should consider all credible system failure modes. Rescreen systems in accordance with the systematic approach.	As noted in the recommendations related to this finding, resolution to this Finding is considered to be documentation in nature. The specific example cited of a LPCS pipe break inside containment is a scenario that has a negligible risk impact. LPCS piping subject to vessel pressure is already considered in determination of LOCA initiating event frequencies. The behavior of LPCS piping maintained at low pressure standby conditions would be similar to that of a LPCS discharge line break in the auxiliary building, which has been assessed in the Internal Flooding Analysis. This line is above the level of the suppression pool and would result in a maximum sustained leak rate of 50 gpm, the capacity of the LPCS keep-fill pump, prior to operator action to terminate the event. This would not be expected to be a challenge to plant operation. LPCS pipe failures in containment would be expected to have an initiating event frequency in the E-06/year range based on EPRI pipe failure frequencies and thus would be negligible contributors to plant risk. Note the applicable Supporting Requirements from the Standard were judged to be Met.

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		mitigated by any combination of engineered systems). Example: reactor pressure vessel rupture (5) LOCAs Outside Containment. Example: primary system pipe breaks outside containment (BWRs). (c) SGTRs. INCLUDE spontaneous rupture of a steam generator tube (PWRs). (d) ISLOCAs. INCLUDE postulated events in systems interfacing with the reactor coolant system that could fail or be operated in such a manner as to result in an uncontrolled loss of core coolant outside the containment [e.g., interfacing systems LOCAs (ISLOCAs)]. (e) Special initiators (e.g., support systems failures, instrument line breaks) [Note (1)]. IE-A5 Cat.II : PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE A STRUCTURED APPROACH [SUCH AS A SYSTEM-BY-SYSTEM REVIEW OF INITIATING EVENT POTENTIAL, OR A FAILURE MODES AND EFFECTS ANALYSIS (FMEA), OR OTHER SYSTEMATIC PROCESS] TO ASSESS AND DOCUMENT THE POSSIBILITY OF AN INITIATING EVENT RESULTING FROM INDIVIDUAL SYSTEMS OR TRAIN FAILURES.	centered, plant centered, switchyard centered, and weather related). Consequential LOOP initiating events are also assessed. (PRA-RB-01-002S09 revision 1). Therefore this SR is met. IE-A5: Calculation PRA-RB-01-002S06, Appendix I provides a system-by-system evaluation to determine possible support system IE's. However, details of the screening process are not provided. In addition, some systems were screened for one failure mode (for example, LPCS inadvertent start) but not other failure modes (for example, LPCS pipe break inside containment).			Documentation will be enhanced to address this Finding as part of the next periodic PRA update (Revision 6). This documentation related Finding has negligible impact upon the ILRT Extension Request.
28 14	IE-A6 (Met)	When performing the systematic evaluation required in IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	Calculation PRA-RB-01-002S06, Section 4.9, Table 5, examines common cause failure of multiple AC or DC buses and eliminates them from consideration. System-by-system screening in Appendix I, considers system level multiple failures. Initiating event fault trees considered multiple failures by design.	There is not evidence presented in the IE notebook that multiple failures (for CCF) were considered in the development of the IE list.	Evaluate the potential for CCF failures causing an initiating event.	Common Cause events are included in calculating Initiating Event frequencies (e.g., event SWP-MDP-C2-NSWRA for CCF of Normal Service Water pumps; event CCP-MDP-C2-FTRA for CCF of Primary Component Cooling Water pumps). Such events are relatively minor contributors to Initiating Event frequencies. Plant alignments are also considered in the evaluation of Initiating Event fault trees; the appendices in the Initiating Event calculation provide quantification of initiating event fault trees based on various system alignments. The impact of these plant alignments on IE frequency is also captured in the EOOS on-line risk assessment monitor. Thus, resolution of this Finding is expected to result in only negligible or slight changes to PRA results. This Finding remains open to address as an enhancement to the next full PRA model update. Note the applicable Supporting Requirement from the Standard was judged to be Met. This Finding does not impact the ILRT Extension Request.
31 15	IE-C4 (Met) DA-D4 (Met)	C4: When combining evidence from generic and plant-specific data, USE a Bayesian update process or equivalent statistical process. JUSTIFY the selection of any informative prior distribution used on the basis of industry experience (see reference [2-2]).	IE-C4: PRA-RB-01-002S06, Section 5.1 and Appendix B, describe the Bayesian update process used to combine data for RBS PSA. IE notebook assumption 5 states that all generic data was considered to be lognormal, but no basis was provided for	For those IE frequencies for which generic data was updated by plant specific data, a Bayesian process was used. IE notebook assumption 5 states that all generic data was considered to be lognormal, but no basis was provided.	Revise IE notebook assumption 5 to provide the required justification.	The generic data for the Internal Events (IE) notebook came from NUREG CR-6928 and was developed using beta or gamma distributions. However, NUREG CR-6928 also provides mean and error factor (EF) parameters, which are used for developing lognormal distributions. Thus, it is acceptable to use the data to develop lognormal distributions. Converting from gamma or beta distribution to lognormal does cause some loss of fidelity. However, the numbers provided in NUREG CR-6928 are best

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		DA-D4: When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	that assumption. For LOSP initiating events, no Bayesian updating was applied as River Bend has no plant specific data relating to these events. (PRA-RB-01-002S09 section 4.2.1.1) DA-D4: Document PRA-RB-01-002S05 provides the results of the Bayesian approach and the cumulative distribution, but does not provide evidence of a review of the distributions confirming the Bayesian updating was appropriate. BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard Also, as noted in IE-C4, log normal prior distributions were assumed for the generic failure data although the generic data was based on beta and gamma distributions. The basis for this transformation to lognormal should be discussed in the notebook.			estimates based on the quality of the data provided by each plant. Therefore a small loss of fidelity would be insignificant because of the uncertainty in the base numbers. Entergy Data Analysis guidelines allow assuming lognormal distributions based on its simplicity of use, general application and because it closely approximates the observed variability in component failure rates. DG information from the Data calculation clearly shows an appropriate overlap between plant specific and generic data, for both Standby DG's and the SBO DG. While the plant specific 50th percentile value is about the 95th percentile of the generic FTR data, the plant specific FTS distribution is entirely bounded by the generic distribution and the plant specific F1 (failure to run first hour) data is smaller than the generic data (50th percentile of plant distribution is about 5th percentile of generic distribution) This meets the standard of reasonableness for application of Bayesian updating. There is no known significant different in unreliability amongst RBS diesel generators that would impact the calculation of failure rates using Bayesian updating. Also, the Station Blackout diesel generator was replaced in 2010 with a new unit to improve reliability. Note the design and the function of the SBO DG differs markedly from the three Divisional DG's. The SBO DG is a small portable 200 KW unit used to maintain DC power under station blackout conditions. The applicable Supporting Requirement from the Standard was judged to be Met.. In conclusion, the resolution of this Finding has negligible to minimal impact on the ILRT Extension Request.
33 16	AS-A3 (Met)	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A3), IDENTIFY the systems that can be used to mitigate the initiator. [See Note (1).]	The AS discussion in PRA-RB-01-002S03 and the related notebooks for ISLOCA, ATWS, and Breaks outside containment discuss the success criteria for each event tree node at a relative high level. The specific criteria for each node is more specifically discussed in the Success Criteria Notebook (PRA-RB-01-002S14). The ATWS event analyses, documented in PRA-RB-01-002S07 revision 1, table 1 identifies the systems associated with each safety function. That table also identifies safety function success criteria in most cases. The success criteria for RPS-mechanical was found in a notebook assumption. However, there is at least one instance in which success criteria is not documented in table 1. An example is SLC.	The ATWS event analyses, documented in PRA-RB-01-002S07 revision 1, table 1 identifies the systems associated with each safety function. That table also identifies safety function success criteria in most cases. The success criteria for RPS-mechanical was found in a notebook assumption. However, there is at least one instance in which success criteria is not documented in table 1. An example is SLC.	Update the documentation to specify the missing success criteria or provide a pointer to where that criteria is located. In addition, update table 1 to include all success criteria.	As discussed in the associated recommendation, this finding is documentation in nature and its resolution does not impact RBS PRA results. Appropriate system related success criteria are documented in System Notebooks. Documentation in the success criteria notebook for the Rev.6 PRA update will be expanded to include success criteria specific to ATWS, ISLOCA, and BOC events, which are the only events for which accident sequences are developed in notebooks separate from the Accident Sequence notebook. Note the applicable Supporting Requirement from the Standard was judged to be Met. This documentation related Finding has no impact on the ILRT Extension Request.
34 17	SY-A19 (Met)	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, in a manner consistent with the actual practices and history of the plant for removing equipment from service. (a) INCLUDE (1) unavailability caused by testing when a component or system train is reconfigured from	This is a finding because one instance was identified in which the requirement is not met.	PRA-RB-01-002S11, LPCI and CCP system models contain basic events for unavailability of the RHR pumps and CCP pumps at the component level. Basic events for maintenance unavailability are indicated by 'MA' in the basic event name. Section 1.7 of the system notebooks documents the review of test and maintenance applicability	Add maintenance unavailability to the feedwater and condensate systems analysis.	Consideration of unavailability of feedwater and condensate systems would be expected to result in only very slight changes to RBS PRA results, since only one of three pumps for each system are required to meet system success criteria for event mitigation. This is consistent with risk ranking results for the Feedwater and Condensate pumps from the RBS PRA Summary Calculation; no events associated with the feedwater pumps appears in the cutsets generated at an E-13 truncation limit for the risk ranking;

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
		its required accident mitigating position such that the component cannot function as required (2) maintenance events at the train level when procedures require isolating the entire train for maintenance (3) maintenance events at a sub-train level (i.e., between tagout boundaries, such as a functional equipment group) when directed by procedures (b) Examples of out-of-service unavailability to be modeled are as follows: (1) train outages during a work window for preventive/corrective maintenance (2) a functional equipment group (FEG) removed from service for preventive/corrective maintenance (3) a relief valve taken out of service		associated with a given system/train/component. PRA-RB-01-002S11 R1 documents the feedwater and condensate system analysis. PRA-RB-01-002S11 R1 documents the feedwater and condensate system analysis. Unavailability of a feedwater or condensate pump due to maintenance is not included in the analysis. There are 3 40% feedwater pumps and 3 50% condensate pumps. Therefore, maintenance of a single feedwater pump or a single condensate pump during power operations is possible.		the maximum RAW of 1.016 and maximum FV of 3.21E-05 for the individual Condensate pumps demonstrate very low risk significance. Thus, only very small if any impact on PRA results would be expected associated with resolution of this finding. This Finding remains open for consideration as an enhancement to add to the model for the next full model update, Revision 6. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This Finding has negligible impact on the ILRT Extension Request.
36 18	HR-E4 (Cat.I)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	This is considered to be a finding as validation/input has not been obtained to validate proper modeling and timing of operator response.	No documentation of simulator observations were identified. While Appendix C to PRA-RB-01-002S03 documents operator input for the HRA analysis, no documented talk throughs or review by either Operations Staff or Operations Training Staff with respect to the response modeling (accident sequence progression) was identified.	Involve/conduct and document Operations / Operations Training review of the response models to insure timing and operator response modeling is correct.	There has been extensive discussion regarding operator actions modeled in the RBS PRA over the years. Discussions regarding operator actions arise during Expert Panel meetings, PRA training for operations, and regular observations of simulator training and scenarios by PRA staff. Scenarios are also discussed as part of routine support for on-line maintenance issues and when risk assessments are performed for plant conditions. More explicit documentation of interactions between the RBS PRA staff and Operations will be incorporated in the next PRA update. Thus, this finding is by nature a documentation issue. This Finding has no impact on the ILRT Extension Request.
37 19	HR-E4 (Cat.I)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	Coupling of two separate operator responses, that may be performed by different individuals, may mask insights related to operator significance, in addition to, masking dependencies	B21-XHE-FO-INHIB was identified which coupled two separate operator actions (inhibit ADS and terminate and prevent HPCS) in a single action. Both of these actions may or may not be performed by the same individual. Spreadsheet HFE_CP.xls only evaluates inhibit ADS with no execution probability based on a simple action (agree), however, terminate and prevent of HPCS is performed via a hardcard (several actions).	Model the inhibit ADS during an ATWS and terminate and prevent HPCS as separate operator actions (separate actions on EOP-01A in step RCA-3).	The suggested change to break B21-XHE-FO-INHIB into two separate operator actions is being incorporated into PRA Revision 5A. This change has negligible impact on CDF. This finding has been resolved and has negligible impact on the ILRT Extension Request.
38 20	HR-E4 (Cat.I) HR-E2 (Met)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled. E2 (Cat.I/II/III): IDENTIFY those actions (a) required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating core damage as defined by the success criteria (e.g., operator initiates RHR) (b) performed by the control room staff either in response to procedural direction or as skill-of-the-craft to diagnose and then recover a failed function, system, or component that is used in the performance of a response action as identified in HR-H1.	This is a finding because this is a potentially significant operator action, that if failure were to occur, could lead to an uncontrolled injection / power excursion during an ATWS. HR-E2: The HRA development has modeled HEPs for operator response in regards to automatic failure of systems such as ECCS. Review has noted that an HEP for the termination and prevention of the low pressure ECCS systems as directed per EOP-01A step RLA-13 was not developed/modeled.	An operator action was not identified/modeled for the termination and prevention of the low pressure injection systems (EOP-01A RLA-13). Following an emergency depressurization, if this action was not performed, a substantial uncontrolled injection could occur resulting in a power excursion. Operator review/walk through/talk through the specific accident sequence may have identified this detail.	Model the low pressure terminate and prevent operator action and validate with Operations / Operations Training.	The River Bend model includes the operator action to terminate and prevent injection as part of the action for controlling RPV power and level during ATWS, event B21-XHE-FO-LVCTL. The human error probability for this operator action encompasses the action to terminate ECCS injection per RBS Emergency Operating Procedures (EOP's). This finding has been resolved and does not impact the ILRT Extension Request.
40 21	SY-A20	INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity (see DA-C14).	This is considered a finding as the requirement is to include these types of maintenance practices (as-operated plant).	DA-C14 basis for assessment notes that Appendix H of PRA-RB-01-002S05 Rev1 EC22619.pdf discusses 'super outages' that occur usually twice a year on	Incorporate these events into the model.	These events were intended for incorporation into the model; failure to do so was an oversight. Inclusion of these events is has only a minor impact on the model, based on the 1.52E-04

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				various components belonging to intersystem trains. Basic events CONCRNT-MA-DIV01, CONCRNT-MA-DIV02, and CONCRNT-MA-DIV03 were developed for these outages. Although developed to represent actual maintenance practices, these basic events do not appear in the PRA fault model. Also, the CRD System and SW fault tree models do not include the simultaneous unavailability of redundant equipment due to technical specification constraints for planned activities.	Evaluate CRD and SW for possible simultaneous maintenance conditions.	probability associated with these events and the fact that unavailability of the opposite train would be a mutually exclusive event. There is no connection between the CRD system and the Service Water system which would require modeling as a simultaneous maintenance condition. Changes in response to this Finding have been incorporated into the RBS PRA model used with EOOS and for PRA Revision 5A.. Thus, this Finding has been resolved and does not impact the ILRT Extension Request.
41 22	QU-D4 (Met) LE-F2 (Met)	QU-D4: COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another? LE-F2: REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant-specificity is appropriate for significant contributors, etc.).	QU-D4: This is considered a finding as an opportunity for a checks and balances may be missed. While this SR is administratively met, further depth in understanding the differences would help in strengthening the model. LE-F2: PRA-RB-01-002S12 revision 1, Attachment 10 documents meeting minutes/notes associated with LERF cutset reviews. Attachment 11 also captures review comments and resolution. A comparison among the various BWR/6 designs was also provided, albeit at an administrative level. An approach similar to that recommended for QU-D4 for CDF should be used for the LERF as well.	A comparison among the BWR/6 population was conducted and is documented in PRA-RB-01-002 revision 1. Observations of differences were noted, however, additional depth as to the differences may be required establish more credible explanations. For example, the loss of the power conversion system differences may to more attributable to the additional containment heat removal capability at RB verses high pressure injection when compared to Plant "B" (Plant B also has a motor driven feedwater pump). Additionally, higher SBO contributions were attributed to the dependence upon electrical switchgear room cooling (RCIC is not dependent upon electrical switchgear room cooling). This SR is marked as met, however, a finding has been given to establish more credible explanations of the deltas. An in-depth comparison may also provide feedback insights.	Provide a more detailed / credible comparative analysis of significant deltas identified.	While additional insights would be obtained from a deeper and more detailed review of differences between plants, the level of detail at which River Bend has performed this comparison is judged to be better than average. RBS participates in the monthly BWR6 PRA conference call, which includes discussions of the various plant system models to allow for understanding of differences due to plant designs and modeling. Additional insights have been gained through participation of Entergy PRA engineers in the Perry Level 2 focused scope peer review and through support work for the ongoing Grand Gulf PRA Revision 4. The additional insights would be of value but would not result in changes to the results of the plant PRA, thus this Finding is considered to be documentation in nature and will be closed as part of the future Revision 6 PRA update. Note the applicable Supporting Requirements from the Standard was judged to be Met. This documentation related Finding has no impact upon the ILRT Extension Request.
44 23	DA-D4 (Met)	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Document PRA-RB-01-002S05 provides the results of the Bayesian approach and the cumulative distribution, but does not provide evidence of a review of the distributions confirming the Bayesian updating was appropriate. BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. Also, as noted in IE-C4, log normal prior distributions were assumed for the generic failure data although the generic data was based on beta and gamma distributions. The basis for this transformation to lognormal should be discussed in the notebook. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard.	BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard.	Review the updated results and correct any values that do not meet the acceptance criteria. Document the review that the posterior distributions to confirm the Bayesian updates were appropriate.	See discussion related to Finding #15 (SR's IE-C4 and DA-D4) which establishes that the Entergy data Bayesian update process meets the reasonableness criteria of the Standard and that the Bayesian update results were appropriate. This finding has been resolved. As discussed therein, this finding has no impact on the ILRT Extension Request.
45 24	DA-C1	OBTAIN generic parameter estimates from recognized sources. ENSURE that the parameter	The generic parameter estimates are	The generic data document PRA-ES-01-003 has a	Resolve the differences between the	System Analysis Guide EN-NE-G-010 provides the event and component type identifiers and standard failure modes codes for

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	(Met)	definitions and boundary conditions are consistent with those established in response to DA-A1 to DA-A4. (Example: some sources include the breaker within the pump boundary, whereas others do not.) DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data is consistent with the test and maintenance philosophies for the subject plant. Examples of parameter estimates and associated sources include (a) component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20] (b) common cause failures: NUREG/CR-5497 [2-8], NUREG/CR-6268 [2-9] (c) AC off-site power recovery: NUREG/CR-5496 [2-10], NUREG/CR-5032 [2-11] (d) component recovery See NUREG/CR-6823 [2-1] for a listing of additional data sources.	generally taken from NUREG/CR-6928, as documented in the generic data notebook PRA-ES-01-003. However, the naming conventions between the generic data and the River Bend actual basic event nomenclatures differ. Common cause failure data is obtained from NUREC/CR-5497 and the NRC CCFWIN database as documented in the CCF notebook (PRA-RB-01-002S04). AC OSP recovery is documented in PRA-RB-01-002S09 and is based on data from NUREG/CR-5032, NUREG/CR-6890 and EPRI LOSP reports. Component recovery is not credited in the PRA. Unavailability estimates are based on plant specific date.	type code of BUS FTOP where the database has a type code of BAC NO for bus failure to operate. There is no direct tie from the database to the documentation. Also, the ASL DN type appears to be STL FTOP in the generic data document	documentation and the contents of the PRA database.	Entergy PRA's. The Tables in the data calculation show the mapping of failure modes, per EN-NE-G-010, to the NUREG/CR-6928 failure modes. This confirms, for example, that BUS FTOP from the NUREG is mapped to BAC LP or BDC LP, bus fail to operate events for AC or DC busses. The RBS CAFTA database file includes in the Type Code data window the corresponding NUREG/CR-6928 type codes. Thus, documentation exists to provide the direct tie from the source NUREG to the RBS type codes. Note the applicable Supporting Requirement from the Standard was judged to be Met. This finding has been resolved and has no impact on the ILRT Extension Request.
46 25	DA-D6 (Met) SY-B4 (Met)	DA-D6: USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities in a manner consistent with the component boundaries. SY-B4: INCORPORATE common cause failures into the system model in a manner consistent with the common cause model used for data analysis. (See DA-D6.)	DA-D6: Generic common cause failure factors are used, as documented in PRA-RB-01-002S04. The CCFWIN database is used, which includes all industry data. No specific discussion of the applicability of the generic data to River Bend is provided, however. The common cause factors boundaries were intended to match the independent data from document PRA-ES-01-003. There is not a strong documentation link between it and the CCF information found in PRA-RB-01-002S04, Rev. 2 since the factors were not used. The EDG's independent run basic events have been split into first hour and fails to continue running events (e.g. EGS-DGN-F1-EG01A and EGS-DGN-FR-EG01A). The CCF basic events have combined failure to start and failure to run for the first hour (e.g. EGS-DGN-C2-DGFR), but this is not well documented, and the BE name is misleading. Did not find CCFs for EDG FO transfer pump check valves (e.g. EGF-CKV-CC-V33) although CCFs exist for pumps. This omission was self-identified by the PRA staff, but has not yet been incorporated into the model. SY-B4: 'Table B-1. RBS CCF Basic Events and their associated Independent Basic Events' and 'Table B-2. RBS CCF Basic Events and their Associated CCF Multipliers' in Calculation PRA-RB-01-002S04, Rev. 2 lists all CCF events incorporated in the model. A list of common cause failures basic events are presented in Section 2.1.3 of	Found in FPW-ENG-C2-2FTR that it appears to use a generic independent event as described in PRA-RB-01-002S04 and not the calculated independent event in the database that matches the document PRA-RB-01-002S05. (2.07E-03 EF 9.8 vs. 3.75E-04 EF 18). Checked the spreadsheet RBSCCF that was used to develop the CCFs and found the 2.07E-3 event was used. The cutset file provided (RBS-R5_rec_merged....cut contains the 3.75E-4 value.	Update the CCF BE calculation to use the current independent BE failure values and consistent component boundaries.	The issues related to Common Cause Failure of diesel fuel oil transfer pumps and Fire Protection Water pumps identified in this Finding have been corrected and incorporated into the RBS PRA model for EOOS and for Rev.5A. These corrections have minimal impact on results; CCF of the Fuel Oil Transfer Pumps have a FV risk importance of less than 0.01. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This Finding has been resolved and has minimal impact on the ILRT Extension Request.
47 26	DA-D6 (Met) SY-B4 (Met)			The EDG's independent run basic events have been split into first hour and fails to continue running events (e.g. EGS-DGN-F1-EG01A and EGS-DGN-FR-EG01A). The CCF basic events have combined failure to start and failure to run for the first hour (e.g. EGS-DGN-C2-DGFR), but this is not well documented, and the BE name is misleading. Did not find CCFs for EDG FO transfer pump check valves (e.g. EGF-CKV-CC-V33) although CCFs exist for pumps. This omission was self-identified by the PRA staff, but has not yet been incorporated into the model.	Update the CCF analysis to include the missing CCF events for the transfer pumps. Enhance the documentation of the DG CCF events to better explain the FTS and FTR modeling.	The CCF analysis and the PRA model for EOOS and for Rev.5A have been updated to include the missing CCF events for the Diesel Fuel Oil system. Note the applicable Supporting Requirements from the Standard were judged to be Met. This Finding has been resolved and has minimal impact on the ILRT Extension Request.

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
			each system notebook. Checking of a sample CCF basic events from Appendix B1 Table B-1 showed them to be incorporated in the model in accordance with the CCF method and the CCF boundaries			
49 27	IFEV-B3 IFPP-B3 IFQU-B3 IFSN-B3 IFSO-B3 (Not Met)	IFEV-B3 Document sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood-induced initiating events. IFPP-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood plant partitioning. IFQU-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood accident sequences and quantification. IFSN-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood scenarios. IFSO-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood sources.	This is a finding since an assessment of the sources of uncertainty is required by the standard.	No assessment of the sources of uncertainty was documented.	Perform and document an assessment of flood-related sources of uncertainty.	This finding is considered documentation in nature, since performance of an uncertainty study would not impact CDF results. All of the SR's associated with this finding are considered Documentation requirements in the Standard. The Revision of the Internal Flooding PRA subsequent to the peer review did review the results to obtain insights into importance of system and location contributors to the Internal Flooding risk, which does permit judgments concerning the impact of uncertainties. The Section 5.7 bounding analysis of potential impact of ILRT extension from external events and other hazard groups fully accounts for Internal Flooding. This finding does not impact the ILRT Extension Request.
51 28	IFSO-A4 (Not Met)	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a release. INCLUDE (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow-through openings created to perform maintenance; inadvertent actuation of fire-suppression system (c) other events resulting in a release into the flood area	This is a finding since the requirements of this SR are not met. Identification of mechanisms is required by the SR. Missing failure mechanisms could impact the overall results.	Flooding mechanisms are not identified in the analysis. Although calculation PRA-RB-01-004 Rev. 0 states in Section 3.4 that all mechanisms were considered, this does not appear to be the case. For example, section 4.2.5.8 states that the area is not considered because the only source is a pre-action fire system. However, inadvertent actuation of this system should be addressed. Other instances exist.	Document the potential failure mechanism included in each flood area.	This finding is considered primarily documentation in nature, as discussed in the finding and recommendation. The EPRI pipe failure data used in this analysis encompasses all pipe failure mechanisms; there is no readily available data that allows distinguishing between different failure mechanisms. Since the failure rate data used in the analysis encompasses the various failure mechanisms, there would be no change to the results associated with identifying specific failure mechanisms. Thus, this documentation related finding does not impact the ILRT Extension Request.
52 29	IFSO-A5 (Met)	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) flow rate (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source	Inclusion of this information is required by the SR; hence this is a finding.	The characteristics of each source are documented for each scenario developed in Section 4.2 of Calculation PRA-RB-01-004 Rev. 0. These scenarios identify the flow rate by evaluating a complete rupture of the line analyzed. The capacity of the source is considered for finite-volume systems, however, no volume information for these systems was identified in the documentation. Pressure and temperature were not identified in the documentation.	Document release characterization for each source.	Characterization of failures and flow rates are included in the scenario descriptions in the revision to Internal Flooding Analysis calculation, including documentation for the scenario in an Appendix. As stated, source capacities have been considered in detailed scenario development. System information, including volumes and pump flow, has been added as part of the subsequent Rev.1 to the calculation. System pressures are used to calculate flow rates using spreadsheets. Systems which are potential HELB sources are identified. Much of this information had been included in the original documentation but was not well organized. Since failure flow rates have been appropriately developed and other characteristics documented through the

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						documentation would not impact the results, this documentation finding does not impact IFPRA results. Note the applicable Supporting Requirement from the Standard was judged to be Met. This documentation Finding does not impact the ILRT Extension Request.

A.3 Identification of Key Assumptions

The methodology employed in this risk assessment followed the EPRI guidance [A.6] as approved by the NRC. The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analysis indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.4 Summary

A PRA technical adequacy evaluation was performed consistent with the requirements of RG 1.200, Revision 2. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed extension to the ILRT/DWBT intervals for RBS to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.5 References

[A.1] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 2, March 2009.

[A.2] Entergy Calculation, *RBS PRA Summary Report (PRA Rev. 5)*, PRA-RB-01-002, Revision 1, March 2011.

[A.3] Entergy Calculation, *RBS PRA LERF Model*, PRA-RB-01-002S12, Revision 1, March 2011.

[A.4] Entergy Fleet Procedure EN-DC-151, Revision 2, *PSA Maintenance and Update*, January 2011.

[A.5] BWR Owners Group, *River Bend Station PRA Peer Review Report Using ASME/ANS PRA Standard Requirements,* July 2011.

[A.6] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325*, EPRI, Palo Alto, CA: 2008. 1018243.

Appendix A to Attachment 3

RBG-47620

PRA Technical Adequacy

A.1 OVERVIEW

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this report to help support an extension of the RBS containment Type A integrated leak rate test (ILRT) and drywell bypass test (DWBT) intervals to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 2 [A.1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed to perform this study:

1. Identify the parts of the PRA used to support the application
   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model.
   - A definition of the acceptance criteria used for the application.
2. Identify the scope of risk contributors addressed by the PRA model
   - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.
   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 are covered in the main body of this report. The purpose of this appendix is to address the requirements identified in item 4 above. Each of these items (plant changes not yet incorporated into the PRA model, relevant peer review findings, consistency with applicable PRA standards and the identification of key assumptions) are discussed in the following sections.

The risk assessment performed for the ILRT/DWBT extension request is based on the current Level 1 and Level 2 PRA models of record. These models comprise Revision 5 of the RBS PRA, which was released in April 2011. Note that for this application, the accepted methodology involves a bounding approach to estimate the change in the LERF and population dose from extending the ILRT/DWBT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability

Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

As discussed, the PRA models used for this application are the latest Revision implemented at River Bend. There are no significant plant changes (design or operational practices) that have not yet been incorporated in those PRA models, other than those associated with FLEX. RBS completed plant modifications associated with the FLEX initiative (NEI-12-06) during the Spring 2015 refueling outage, and is in compliance with the NRC FLEX order; these changes will result in improvements in plant mitigating capability with resultant decreases in CDF, adding conservatism to the ILRT risk assessment for River Bend.

A discussion of the Entergy model update process, the peer reviews performed on the RBS model, the results of those peer reviews and the potential impact of peer review findings on the ILRT/DWBT extension risk assessment are provided in Section A.2. Section A.3 provides an assessment of key assumptions and approximations used in this assessment and Section A.4 briefly summarizes the results of the PRA technical adequacy assessment with respect to this application.

A.2 PRA Update Process and Peer Review Results

A.2.1 Introduction

The RBS Probabilistic Risk Assessment (PRA) models used for this application [A.2 and A.3] are the most recent evaluations of the RBS risk profiles for internal event challenges. The RBS PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause failure events. The PRA model quantification process is based on the event tree and fault tree methodology, which is a well-known methodology in the industry.

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

A.2.2 PRA Maintenance and Update

The Entergy risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the Entergy fleet procedure EN-DC-151, "PSA Maintenance and Update" [A.4]. This procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Entergy nuclear power plants. In addition, the procedure also defines the process for implementing regularly scheduled and interim PRA model updates, and for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.). To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. Potential PRA model changes resulting from these reviews are entered into the Model Change Request (MCR) database, and a determination is made regarding the significance of the change with respect to current PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.

- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated regularly. The corporate procedure is being revised to suggest an update frequency of approximately every five years.
- Industry standards, experience, and technologies are periodically reviewed to ensure that any changes are appropriately incorporated into the models.

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

A.2.3 Regulatory Guide 1.200 PWROG Peer Review of the RBS Internal Events PRA Model

The RBS internal events model went through a Regulatory Guide 1.200 BWR Owners Group peer review using the NEI 05-04 process. The RBS PRA internal events model peer review performed in April 2011, used RA-Sa-2009 (the American Society of Mechanical Engineers / American Nuclear Society Combined PRA Standard) and Regulatory Guide 1.200 Revision 2.

The RBS PRA peer review addressed all the technical elements of the internal events, at-power PRA:

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

During the RBS PRA model peer reviews, the technical elements identified above were assessed with respect to Capability Category II criteria to better focus the Supporting Requirement assessments.

A.2.4 Peer Review Results

The ASME PRA standards used for the RBS peer reviews contained a total of 325 numbered supporting requirements. A number of the supporting requirements were determined to be not applicable to the RBS PRA (e.g., PWR related, multi-site related). Of the applicable supporting requirements, more than 85% were satisfied at Capability Category II or greater for RBS with the majority of the supporting requirements not meeting Supporting Requirements related to the Internal Flooding elements. The Peer Review Team generated a total of 59 Findings which are provided in the peer review report [A.5]. Seven of these Findings were against LERF-related Supporting Requirements, as the River Bend LERF model was developed as a NUREG/CR-6595 simplified LERF model, which is defined as meeting Capability Category I of the Standard. The River Bend Internal Flooding Analysis was subsequently revised in 2012 to fulfill a commitment established via Entergy letter RBG-47029 dated 5/14/2010 to update the basis to PRA Revision 5 and to resolve many of the Findings related to Internal Flooding elements of the Standard.

River Bend is in the process of compiling an Interim Revision 5-A to its PRA that will incorporate resolution of selected Findings. Remaining open Findings against the Internal Events model will be addressed or incorporated as part of future full model update under Revision 6 to the PRA.

As a result of the Regulatory Guide 1.200 BWROG peer reviews, all the Facts and Observations (F&Os) (other than best practices) identified as potential improvements to the RBS PRA models or documentation were entered into the Entergy Model Change Request (MCR) database. Table A.2-1 contains the remaining 29 open findings resulting from the peer review for RBS, the status of the resolution for each finding and the potential impact of each finding on this application. This includes the findings that will be closed with implementation of PRA Revision 5A. In summary, a majority of the findings were related to documentation and have no material impact. Resolution of the remaining open peer review findings is not expected to significantly change the total internal events CDF for RBS such that there would be no changes to the conclusions of this application.

A.2.5 External Events

Although EPRI report 1018243 [A.6] recommends a quantitative assessment of the contribution of external events (for example, fire and seismic) where a model of sufficient quality exists, it also recognizes that the external events assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval. Since the most current external events models for RBS have not been peer reviewed, a multiplier was applied to the internal events results based on the available information, similar to that used in other ILRT analyses. This is further discussed in Section 5.7 of the risk assessment.

A.2.6 Summary

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

Resolution of Findings.

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
1	DA-C8 (Cat.I)	Cat.I: When required, ESTIMATE the time that components were configured in their standby status. Cat.II/III: When required, USE PLANT-SPECIFIC OPERATIONAL RECORDS TO DETERMINE the time that components were configured in their standby status.	This is a finding since the technical requirements of the SR were not met for Capability Category II	Table C-2A of PRA-RB-01-002S05 discusses the rationale used to determine run times. While a few components (e.g., SW pumps) appear to be collect actual run vs. standby time in the supporting spreadsheets, the standby time for most components in running systems was estimated (e.g., 1/2, 1/3, etc.). Therefore Category I is met.	To meet Category II, actual plant experience concerning standby/run fractions needs to be collected and used in the PRA.	Use of actual vice estimated availability for other components would be expected to have very small impact on PRA results. Actual availabilities are used for components for which this information is tracked, which includes those components monitored by MSPI, which tend to have higher PRA importance (e.g., diesel generators, RHR pumps, Standby Service Water pumps). This Finding will be addressed as a possible enhancement to the next periodic PRA model update. No impact on the RBS ILRT extension request.
2	AS-B3 (Not Met)	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	This is a finding because the requirements of the SR are not met.	There is not a specific discussion of the phenomenological impacts of each initiator upon the mitigating systems in the AS notebook (PRA-RB-01-002S01). One specific exception to this is the impacts of debris entrainment for ECCS following LOCA, which is discussed for the Large and Intermediate LOCA event trees. It appears that phenomenological impacts are addressed in the AS logic for all initiators. However, documentation of other impacts (or noting the absence of any impacts) should be provided.	Include an explicit discussion of phenomenological impacts (or lack thereof) for each event tree.	This is considered a documentation issue, as noted in the review comment. For the example of the MSPI application, Table G.5 of NEI 99-02 provides comments to focus on credit for injection post-venting (NPSH issues, environmental survivability) which are thoroughly addressed for the RBS PRA. The environmental effects of containment failure are explicitly considered to result in failure of Auxiliary Building equipment credited for core damage mitigation. As discussed in the PRA Success Criteria calculation, debris effects are considered for Medium Break LOCA and Large Break LOCA resulting in a more restrictive success criteria for those events; this is accounted for via the PRA Event Tree and through recovery rules. Environmental phenomena are thoroughly considered, as documented for the case of internal flooding in Att.2 to letter RBG-46944 dated August 11, 2009. Systems credited for BOC scenarios are unaffected by those breaks. Room heatup effects are fully considered, as documented in the Success Criteria calculation. No impact on the RBS ILRT extension request.
3	SY-A24 (Not Met) (DA-C15: Met))	SY-A24: DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an adequate analysis or examination of data. (See DA-C15.) DA-C15: For each SSC for which repair is to be modeled (see SY-A22), IDENTIFY instances of plant-specific or applicable industry experience and for each repair, COLLECT the associated repair time with the repair time being the period from identification of the component failure until the component is returned to service.	SY-A24: Several equipment recovery events are included in the models. These are a diesel recovery (ZHE-FO-DGN1HRS) and a decay heat removal recovery (ZRCXHE-FO-DHRLT). While a basis for these events is provided in the HRA notebook (PRA-RB-01-002S03), the bases may not justify the use of the specific data values used, as required by this SR. For the diesel repair, the value for the non-repair probability is based on a generic analysis using industry data and various assumptions. No documentation was provided to demonstrate that the data would be applicable to River Bend as required by this SR. For the DHR repair, the data used to develop this non-recovery probability is based on an EPRI report concerning recovery of loss of DHR cooling during shutdown conditions. As shutdown conditions can vary significantly from at-power systems (in terms of factors such as number of cooling water trains available, etc.), the data may not be applicable to the situation being evaluated in the River Bend	Several equipment recovery events are included in the models. These are a diesel recovery (ZHE-FO-DGN1HRS) and a decay heat removal recovery (ZRCXHE-FO-DHRLT). While a basis for these events is provided in the HRA notebook (PRA-RB-01-002S03), the bases may not justify the use of the specific data values used, as required by this SR. For the diesel repair, the value for the non-repair probability is based on a generic analysis using industry data and various assumptions. No documentation was provided to demonstrate that the data would be applicable to River Bend as required by this SR. For the DHR repair, the data used to develop this non-recovery probability is based on an EPRI report concerning recovery of loss of DHR cooling during shutdown conditions. As shutdown conditions can vary significantly from at-power systems (in terms of factors such as number of cooling water trains available, etc.), the data may not be applicable to the situation being evaluated in the River Bend PRA.	As required by this SR, evaluate the repair data used for the diesel and DHR repair terms for applicability to River Bend and document this evaluation. If necessary, select more representative data sources for these repair events.	Diesel Generator recovery ZHE-FO-DGN1HRS has been reviewed and confirmed to be specifically applicable to River Bend Station. A detailed study has been performed for the long-term decay heat removal recovery, ZRC-XHE-FO-DHRLT. This was conducted in tandem with a detailed review of the modeling of the Loss of Normal Service water initiating event. The basis for the long-term decay heat removal recovery was improved and made plant specific, based upon industry actuarial data. When refining the modelling for this recovery, the recovery was modelled in more detail. Entergy is incorporating this enhanced modeling into the River Bend PRA as Interim Revision 5A. Use of the Revision 5 model for ILRT extension risk assessment is considered acceptable. Diesel Generator recovery modeling is concluded to be appropriate or slightly conservative for River Bend Station. The long term decay heat removal recovery does not affect short-term sequences, thus does not impact LERF. These long term decay heat removal recoveries involve prevention of containment failure late into events (e.g., 16 hours or longer) where containment failure which results in core damage is prevented by recovering the ability to remove decay heat from the containment. Such failures would correspond to ILRT non-LERF accident Class 7. Class 7 sequences are not impacted by ILRT Type A tests. Enhancements to the Loss of Normal Service water initiating event model credit the ability of the plant to operate with a

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
			PRA. DA-C15: Repair is considered for diesel generators and decay heat removal. However, the generic data used for these events may not be applicable (see SY-A24)			combination of Normal Service Water and Standby Service Water pumps, resulting in a decreased frequency for this initiator. The overall effect of the combination of these model improvements would be to reduce LERF contributions and increase late non-LERF contributions. Additionally, FLEX modifications implemented at River Bend in 2015 provide additional means of preventing long-term containment overpressurization by providing alternative means of containment and suppression pool cooling. It is thus concluded that PRA Rev.5 provides an appropriate basis for the risk assessment in support of the ILRT extension request.
5 4	AS-A11 (Met)	Transfers between event trees may be used to reduce the size and complexity of individual event trees. DEFINE any transfers that are used and the method that is used to implement them in the qualitative definition of accident sequences and in their quantification. USE a method for implementing an event tree transfer that preserves the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, and spatial or environmental dependencies.	This is a finding as the PRA does not appear to properly perform the noted event tree transfer	Transfers between event trees is accomplished through a direct transfer of the sequence logic through the top logic model. This approach generally retains the dependencies through the quantification process. However, in the case of transfers between the LOSP and ATWS event trees, it does not appear that sequence information is properly transferred, as no LOSP/ATWS cutsets appear in the results.	Review the fault tree and event tree logic associated with the LOOP to ATWS Sequence transfer and modify as necessary to obtain proper results.	A sensitivity analysis was performed to determine the impact of this Finding. Since the initiating event frequency for Loss of Offsite Power (LOOP) is relatively small compared to other initiators, and since RPS has high reliability, only a small increase in CDF results. Requantifying the model resulted in an increase in CDF of 5.9E-11, or about a 1% increase in the ATWS-only CDF. Overall CDF remained unchanged to four significant figures. Fault tree changes to incorporate resolution of this item have been incorporated into the RBS PRA model. Because this logic error had so little impact on calculated CDF, it does not impact the ability of the RBS Rev.5 PRA model to be applied for purposes of ILRT. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This resolved Finding has no impact on the RBS ILRT extension request.
6 5	HR-D3 (Cat.I)	Cat.I: No requirement for evaluating the quality of written procedures, administrative controls, or human-machine interfaces. Cat. II/III: For each detailed human error probability assessment, INCLUDE in the evaluation process the following plant-specific relevant information: (a) the quality of written procedures (for performing tasks) and administrative controls (for independent review) (b) the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout	This is a finding since the technical requirements of this SR are not met.	Since HLR-HR-D concerns pre-initiating events, No evidence of an evaluation process for the quality of pre-initiator written procedures and the quality of the pre-initiator human-machine interface could be found anywhere in the River Bend PRA documentation. Note that Post Initiator procedures have been evaluated for quality (Section 1.4.1) as well as the quality of the man machine interface (Section 1.4.3.) in the RBS HRA/Rule Recovery Work Package, Calculation PRA-RB-01-002S03	Perform an assessment of the quality of pre-initiator procedures and man-machine interface.	This is considered to be primarily an issue of increasing the robustness of PRA model documentation. Only negligible or very slight changes in PRA would be expected as a result of the review of pre-initiator procedures. Any inadequacy in the procedures associated with pre-initiator human failure events would be evidenced during the construction of the detailed spreadsheet calculations for these probabilities. These spreadsheets include documentation and review of the procedure references for each individual pre-initiator event, as well as review of the procedures and nature of indications for the calculation of the basic human error probability. The Man-machine interface quality discussion of section 1.4.3 is also generally applicable to pre-accident initiator actions as well as post-accident actions. No procedural inadequacies were noted during the development of these HRA calculations. Procedure RBNP-001, "Development and Control of RBS Procedures," governs plant operations procedures. RBNP-001 includes requirements for Technical Verification and Validation of procedures to ensure procedure quality. Thus, the intent of the SR is fulfilled through the HRA calculation process. This Finding remains open as a documentation enhancement to consider for the next PRA update. This finding has no impact on the ILRT Extension Request.
7 6	IFQU-A7	PERFORM internal flood sequence quantification	This is a finding because a significant	Quantification of the flooding model is documented in	Re-perform the flooding	At the time of the RBS Rev.5 PRA peer review, the Internal

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
	(Not Met)	in accordance with the applicable requirements described in 2-2.7.	number of technical issues are noted with the flood quantification process	PRA-RB-01-006. The quantification process is carried out separately from the internal events model, and is based on a prior version of the internal events model. The approach used is that, for each flood scenario, to modify the internal events model to fail the components impacted by the flood and setting the initiator to which the flood is mapped to true (while setting all others to false). This computes a CCDP and CLERP, which are then multiplied by the scenario frequency to obtain a sequence frequency. The base model quantification is performed using appropriate codes, truncation levels, recovery and dependency rules, etc. The following specific issues are noted with respect to complying with the QU requirements of the Standard at the Category II level: - the internal events model used is inconsistent with the model used for non-flood initiators - evaluation of the flooding results is not performed to the same level as the non-flood initiators (e.g., significant components, human actions, etc.) - since scenario-by scenario numerical results are being summed to compute a total flood CDF and LERF (using the rare event approximation), the computed total frequencies may be conservative - since the model is not integrated with the other initiators, it is not possible to compute an overall importance for components and human actions - parametric uncertainty analysis is not performed on the flooding results - Reviews of the results for reasonableness are not documented, nor are reviews of non-significant cutsets.	quantification using the current internal events model, using methods that allow for an integrated quantification with the non-flood events. Perform a parametric uncertainty analysis. Perform and document reviews of the flooding results in a manner consistent with that used for the non-flood initiators.	Flooding PRA remained based the previous Rev.4 PRA. RBS has subsequently reperformed the internal flooding quantification using Revision 5 of the RBS PRA in 2012. RBS had committed via letter RBG-47029 dated 5/14/2010 to requantify the Internal Flooding PRA using PRA Revision 5. This requantification addressed many of the findings from the 2011 peer review. The RBS internal flood model is used in support of Risk-Informed In-Service Inspection (RIISI) activities at the site using ASME Code Case N-716. Although conservatisms remain in the model, it is used in the Section 5.7 bounding analysis of potential impact of ILRT extension from external events and other hazard groups, which shows the estimated upper bound LERF for RBS is less than RG 1.174 requirements. Thus, the Flooding model was made consistent with the Internal Events model and evaluated in a consistent manner, albeit including conservatisms for simplicity due to the large number (~500) of flooding scenarios. While it is not possible to capture component and human action importances because the model cannot be integrated, the conclusions in the flooding quantification calculation discuss relative contributions from different buildings, systems, pipe failure sizes, etc. Results, including cutsets, have been reviewed for reasonableness as documented in the flooding quantification calculation. Thus, any impacts associated with this finding have been accounted for in the ILRT Extension Request report..
11 7	DA-C10 (Cat.I)	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. IF THE COMPONENT FAILURE MODE IS DECOMPOSED INTO SUBELEMENTS (OR CAUSES) THAT ARE FULLY TESTED, THEN USE TESTS THAT EXERCISE SPECIFIC SUBELEMENTS IN THEIR EVALUATION. THUS, ONE SUBELEMENT SOMETIMES HAS MANY MORE SUCCESSES THAN ANOTHER. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid test would be significantly decreased.]	This is a finding since the Category II requirements for this SR are not met. This was judged to meet Category I of the Standard: "When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operations."	Surveillance tests are not decomposed into sub-elements.	Decompose surveillance tests into subelements.	Only slight or negligible changes to plant specific data would be expected to result from consideration of subelements of surveillance test procedures. The River Bend PRA does not decompose failure modes into subelements. This element was judged as acceptably meeting the PRA Standard (Category I). Documentation to address this finding will be added to the Data Analysis workbook as part of the next periodic PRA Revision update. This finding does not impact the ILRT Extension Request.
13 8	MU-C1 (Met)	The PRA configuration control process shall consider the cumulative impact of pending changes in the performance of risk applications.	This is a finding because the guideline being used is not mandatory and the cumulative impact of pending model changes is not tracked or measured for their impact on each specific applications.	In Engineering Guide EN-NE-G-026, Revision 0, 'Probabilistic Safety Assessment Applications', all open F&Os, MCRs, and gaps impacting an application are reviewed against a specific application. Justification is provided as to why open items in the model are acceptable for the application or why they do not impact the results. However, it is	Make the evaluation process used to assess impacts on risk applications a formal procedure. Also, implement a method that considers the cumulative	The ILRT Extension Request uses Revision 5 of the RBS PRA, the Model of Record compiled and released in early 2011. Miscellaneous changes have been incorporated in the model used for on-line risk assessment, which have resulted in a small decrease in Core Damage Frequency. Interim Revision 5A is scheduled for implementation in late 2015 or early 2016; the expected impact of Revision 5A is discussed with relation to the

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
				not mandatory to follow this guideline; this needs to be made mandatory. In accordance with EN-DC-151 Revision 2, the cumulative impact of pending model changes is not tracked. Per the guidance, only when a model change request for an implemented change is graded A, or there are over 25 open model change requests that are graded B for a particular model will an interim PRA update be implemented. However, a method to measure the cumulative impact of pending changes particularly on the particular applications of concern should be implemented to fully meet the intent of this SR.	impact of pending changes with respect to the specific applications that are in effect for the plant.	finding for SY-A24 elsewhere in this table. That impact has been assessed and determined to not appreciably impact the ILRT Extension Request. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This finding thus does not appreciably impact the ILRT Extension Request.
15 9	SC-A3 (Not Met)	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	This is a finding since the requirements of the SR are not met for all initiating events.	Based on information in PRA-RBS-01-002S14, Although Section 4.0 provide success criteria relevant to the equipment needed for all key safety functions that involve LOCAs and general transients, success criteria for support system initiators, LOCAs outside containment, and ISLOCAs are missing.	Include appropriate success criteria for support system initiators, LOCA's outside containment, and ISLOCA's.	This Finding is considered to involve documentation of success criteria. Scenario specific success criteria has been considered in the development of RBS Accident Sequence and Success Criteria calculations. Much of the discussion of success criteria is implicit and included under discussion of Success Criteria for individual top events in the RBS Event Trees. Success criteria has been explicitly considered in the development of the Event Trees and in treatment of the support systems for the Event Tree top events. Support system initiating events have the same success criteria as other RBS transients. Specifically, conservative assumptions regarding potential environmental and inventory effects are included in the treatment of Interfacing Systems LOCA (ISLOCA) and Breaks Outside Containment (BOC), which are only small contributors to RBS CDF. The success criteria for each event tree top for BOC is documented in the BOC calculations. Success criteria for ISLOCA are the same as for LOCA, except only limited top gates (depressurization and Standby Service Water cross-tie through RHR) are credited to prevent core damage for ISLOCA. Documentation in this area will be enhanced as part of the next periodic PRA update (Rev.6) for River Bend. This Finding has no impact upon the ILRT Extension Request.
16 10	SY-A4 (Met)	PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	This is a finding because the intent of this SR was not met, since the degree of documentation is insufficient.	PRA-RB-01-002S11, Based on information provided by the PSA group, plant walkdowns have been conducted to ensure the system model correctly reflects the as-built, as-operated plant. However, limited evidence exists that interviews have been conducted to ensure the system model correctly reflects the as-built, as-operated plant.	Provide solid evidence and documentation that interviews/reviews with knowledgeable plant personnel (i.e., system engineers) occurred to document that the system model correctly reflects the as-built as-operated plant.	. System engineers participated in the Expert Panel review documented in the Integration & Quantification package. The PRA model is continually subject to discussion with system engineers as part of the Maintenance Rule Expert Panel and as periodic plant issues arise. System Engineering also reviews risk information related to PRA model revisions (e.g., documentation of risk ranking for Revision 5). The site PRA engineer also reviews the Maintenance Rule Basis Documents, providing further interaction between PRA and System Engineers on PRA assumptions for plant systems. Thus, numerous opportunities exist and have been utilized for review of the RBS PRA by knowledgeable plant personnel, including system engineers. Documentation in this area will be enhanced as part of the next periodic PRA update (Rev.6) for River Bend. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This documentation related finding has no impact upon the ILRT Extension Request.

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
17 11	SY-A4 (Met) SY-B8 (Not Met)	A4: PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant. B8: Identify spatial and environmental hazards that may impact multiple systems or redundant components in the same system, and ACCOUNT for them in the system fault tree or the accident sequence evaluation.	This is a finding because information gathered from the system walkdowns are not reflected in the system notebooks documentation. Hence, a review of spatial dependencies and harsh environment operation with a potential to impact system PSA function cannot be adequately ascertained.	PRA-RB-01-002S11 R1 states that plant walkdowns were used to identify spatial and environmental hazards. Attachment A of that document contains a set of completed walkdown forms. A sample of those forms were reviewed; many of the forms indicated the existence of some kind of spatial or environmental hazard for the walkdown area. Review of several system notebooks did not reveal any indication that the identified spatial and environmental hazards identified in the walkdowns were reviewed for inclusion in or exclusion from the system models. No evidence was found that identified hazards were accounted for in the system or integrated fault tree model. (SY-A4) PRA-RB-01-002S11, Based on documentation provided by the PSA group during the Peer Review, walkdowns have been performed, however, these walkdowns do not discussed spatial and environmental hazards that may impact multiple systems or redundant components in the same system in the system notebooks.	Incorporate information from the walkdowns in the system notebooks documentation and reflect in models, as appropriate.	Spatial and environmental hazards that may impact multiple systems or redundant components are addressed in the Internal Flooding PRA. There are no impacts of this documentation issue upon the results of the Internal Events PRA. Additional walkdown information is documented in the Internal Flooding Analysis document. Also, SR SY-A4 was addressed for RBS in the 11 August 2009 submittal of supplementary information for adoption of ASME code case N-716 for Risk-Informed In-Service Inspection. Many of the environmental conditions documented in the walkdown notes in the Systems Analysis package are conditions which do not impact equipment operation and/or would be accounted for in any HRA calculations. For example, high temperatures were noted for many locations, but these would have been temperatures in the 90's since the walkdowns were conducted in the summer; these temperatures do not impact equipment performance and are considered in the overall assessment of HRA calculations. This finding is concluded to be a documentation issue. Documentation will be enhanced to address this as part of the next periodic update of the RBS PRA. This documentation finding has no impact on the ILRT Extension Request.
18 12	LE-A5 (Not Met)	DEFINE plant damage states in a manner consistent with LE-A1, LE-A2, LE-A3, and LE-A4.	This is a finding because use of NUREG/CR-6595 methodology is used to transfer results from Level 1 directly into the LERF model. This method is adequate for Capability Category I.	Plant damage states are not defined in a manner which accounts for both physical and sequence characteristics. The interface between the Level 1 and containment event tree is based on NUREG/CR-6595 and does not adequately account for all potential dependencies between the systems.	Strongly suggest the development of plant damage states or equivalent (i.e., core damage accident classes). The definition of plant damage states allows for additional containment event tree modeling to account for a more refined Mark III severe accident phenomena behavior.	This finding is documentation in nature and has no impact on LERF results. While the RBS LERF model does not define Plant Damage States, this does not impact the calculation of LERF. This only results in increased difficulty in extracting LERF-related risk insights from the model. SR's LE-A1 through LE-A4 which provide the input for SR LE-A5 were all characterized as "Met" for the RBS PRA Peer Review. RBS plans to document Plant Damage States as part of the LERF calculation for the next regular PRA update. This documentation finding does not impact the ILRT Extension Request.
27 13	IE-A2 (Met) IE-A5 (Met)	IE-A2: INCLUDE in the spectrum of internal-event challenges considered at least the following general categories: (a) Transients. INCLUDE among the transients both equipment and human-induced events that disrupt the plant and leave the primary system pressure boundary intact. (b) LOCAs. INCLUDE in the LOCA category both equipment and human-induced events that disrupt the plant by causing a breach in the core coolant system with a resulting loss of core coolant inventory. DIFFERENTIATE the LOCA initiators, using a defined rationale for the differentiation. Examples of LOCA types include (1) Small LOCAs. Examples: reactor coolant pump seal LOCAs, small pipe breaks (2) Medium LOCAs. Examples: stuck open safety or relief valves (3) Large LOCAs. Examples: inadvertent ADS, component ruptures (4) Excessive LOCAs (LOCAs that cannot be	IE-A2: Based on information in PRA-RB-01-002S06, Section 4.0, Appendix C, D, E, F, G, H, K, and I general spectrum of internal-event challenges have been considered as potential initiating events. The IE notebook includes: (a) transients, except LOSP, (b) (1) Small, (2) Medium, (3) Large LOCAs, (4) vessel rupture, and (e) special initiators including loss of RPCCW, TPCCW, NSW, loss of a single DC bus, loss of a single non-safety bus. (a) LOSP is included in a notebook specific to that initiator. (b)(5) LOCAs outside containment are included in a notebook specific to breaks outside containment. (c) SGTR is not applicable. The spectrum of LOOP events is broken down into the four generally accepted subsets consistent with NUREG-6890 (grid	This is a finding because there is at least one case of a unique initiating event not being considered. The systematic process by which plant systems are reviewed for potential to cause an initiating event is not described. Some of the results of the system screening do not appear to be complete. LPCS pipe break would constitute a unique type of LOCA with failure of a mitigating system.	Document the systematic approach used. That systematic approach should consider all credible system failure modes. Rescreen systems in accordance with the systematic approach.	As noted in the recommendations related to this finding, resolution to this Finding is considered to be documentation in nature. The specific example cited of a LPCS pipe break inside containment is a scenario that has a negligible risk impact. LPCS piping subject to vessel pressure is already considered in determination of LOCA initiating event frequencies. The behavior of LPCS piping maintained at low pressure standby conditions would be similar to that of a LPCS discharge line break in the auxiliary building, which has been assessed in the Internal Flooding Analysis. This line is above the level of the suppression pool and would result in a maximum sustained leak rate of 50 gpm, the capacity of the LPCS keep-fill pump, prior to operator action to terminate the event. This would not be expected to be a challenge to plant operation. LPCS pipe failures in containment would be expected to have an initiating event frequency in the E-06/year range based on EPRI pipe failure frequencies and thus would be negligible contributors to plant risk. Note the applicable Supporting Requirements from the Standard were judged to be Met.

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		mitigated by any combination of engineered systems). Example: reactor pressure vessel rupture (5) LOCAs Outside Containment. Example: primary system pipe breaks outside containment (BWRs). (c) SGTRs. INCLUDE spontaneous rupture of a steam generator tube (PWRs). (d) ISLOCAs. INCLUDE postulated events in systems interfacing with the reactor coolant system that could fail or be operated in such a manner as to result in an uncontrolled loss of core coolant outside the containment [e.g., interfacing systems LOCAs (ISLOCAs)]. (e) Special initiators (e.g., support systems failures, instrument line breaks) [Note (1)]. IE-A5 Cat.II : PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE A STRUCTURED APPROACH [SUCH AS A SYSTEM-BY-SYSTEM REVIEW OF INITIATING EVENT POTENTIAL, OR A FAILURE MODES AND EFFECTS ANALYSIS (FMEA), OR OTHER SYSTEMATIC PROCESS] TO ASSESS AND DOCUMENT THE POSSIBILITY OF AN INITIATING EVENT RESULTING FROM INDIVIDUAL SYSTEMS OR TRAIN FAILURES.	centered, plant centered, switchyard centered, and weather related). Consequential LOOP initiating events are also assessed. (PRA-RB-01-002S09 revision 1). Therefore this SR is met. IE-A5: Calculation PRA-RB-01-002S06, Appendix I provides a system-by-system evaluation to determine possible support system IE's. However, details of the screening process are not provided. In addition, some systems were screened for one failure mode (for example, LPCS inadvertent start) but not other failure modes (for example, LPCS pipe break inside containment).			Documentation will be enhanced to address this Finding as part of the next periodic PRA update (Revision 6). This documentation related Finding has negligible impact upon the ILRT Extension Request.
28 14	IE-A6 (Met)	When performing the systematic evaluation required in IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	Calculation PRA-RB-01-002S06, Section 4.9, Table 5, examines common cause failure of multiple AC or DC buses and eliminates them from consideration. System-by-system screening in Appendix I, considers system level multiple failures. Initiating event fault trees considered multiple failures by design.	There is not evidence presented in the IE notebook that multiple failures (for CCF) were considered in the development of the IE list.	Evaluate the potential for CCF failures causing an initiating event.	Common Cause events are included in calculating Initiating Event frequencies (e.g., event SWP-MDP-C2-NSWRA for CCF of Normal Service Water pumps; event CCP-MDP-C2-FTRA for CCF of Primary Component Cooling Water pumps). Such events are relatively minor contributors to Initiating Event frequencies. Plant alignments are also considered in the evaluation of Initiating Event fault trees; the appendices in the Initiating Event calculation provide quantification of initiating event fault trees based on various system alignments. The impact of these plant alignments on IE frequency is also captured in the EOOS on-line risk assessment monitor. Thus, resolution of this Finding is expected to result in only negligible or slight changes to PRA results. This Finding remains open to address as an enhancement to the next full PRA model update. Note the applicable Supporting Requirement from the Standard was judged to be Met. This Finding does not impact the ILRT Extension Request.
31 15	IE-C4 (Met) DA-D4 (Met)	C4: When combining evidence from generic and plant-specific data, USE a Bayesian update process or equivalent statistical process. JUSTIFY the selection of any informative prior distribution used on the basis of industry experience (see reference [2-2]).	IE-C4: PRA-RB-01-002S06, Section 5.1 and Appendix B, describe the Bayesian update process used to combine data for RBS PSA. IE notebook assumption 5 states that all generic data was considered to be lognormal, but no basis was provided for	For those IE frequencies for which generic data was updated by plant specific data, a Bayesian process was used. IE notebook assumption 5 states that all generic data was considered to be lognormal, but no basis was provided.	Revise IE notebook assumption 5 to provide the required justification.	The generic data for the Internal Events (IE) notebook came from NUREG CR-6928 and was developed using beta or gamma distributions. However, NUREG CR-6928 also provides mean and error factor (EF) parameters, which are used for developing lognormal distributions. Thus, it is acceptable to use the data to develop lognormal distributions. Converting from gamma or beta distribution to lognormal does cause some loss of fidelity. However, the numbers provided in NUREG CR-6928 are best

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		DA-D4: When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	that assumption. For LOSP initiating events, no Bayesian updating was applied as River Bend has no plant specific data relating to these events. (PRA-RB-01-002S09 section 4.2.1.1) DA-D4: Document PRA-RB-01-002S05 provides the results of the Bayesian approach and the cumulative distribution, but does not provide evidence of a review of the distributions confirming the Bayesian updating was appropriate. BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard Also, as noted in IE-C4, log normal prior distributions were assumed for the generic failure data although the generic data was based on beta and gamma distributions. The basis for this transformation to lognormal should be discussed in the notebook.			estimates based on the quality of the data provided by each plant. Therefore a small loss of fidelity would be insignificant because of the uncertainty in the base numbers. Entergy Data Analysis guidelines allow assuming lognormal distributions based on its simplicity of use, general application and because it closely approximates the observed variability in component failure rates. DG information from the Data calculation clearly shows an appropriate overlap between plant specific and generic data, for both Standby DG's and the SBO DG. While the plant specific 50th percentile value is about the 95th percentile of the generic FTR data, the plant specific FTS distribution is entirely bounded by the generic distribution and the plant specific F1 (failure to run first hour) data is smaller than the generic data (50th percentile of plant distribution is about 5th percentile of generic distribution) This meets the standard of reasonableness for application of Bayesian updating. There is no known significant different in unreliability amongst RBS diesel generators that would impact the calculation of failure rates using Bayesian updating. Also, the Station Blackout diesel generator was replaced in 2010 with a new unit to improve reliability. Note the design and the function of the SBO DG differs markedly from the three Divisional DG's. The SBO DG is a small portable 200 KW unit used to maintain DC power under station blackout conditions. The applicable Supporting Requirement from the Standard was judged to be Met.. In conclusion, the resolution of this Finding has negligible to minimal impact on the ILRT Extension Request.
33 16	AS-A3 (Met)	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A3), IDENTIFY the systems that can be used to mitigate the initiator. [See Note (1).]	The AS discussion in PRA-RB-01-002S03 and the related notebooks for ISLOCA, ATWS, and Breaks outside containment discuss the success criteria for each event tree node at a relative high level. The specific criteria for each node is more specifically discussed in the Success Criteria Notebook (PRA-RB-01-002S14). The ATWS event analyses, documented in PRA-RB-01-002S07 revision 1, table 1 identifies the systems associated with each safety function. That table also identifies safety function success criteria in most cases. The success criteria for RPS-mechanical was found in a notebook assumption. However, there is at least one instance in which success criteria is not documented in table 1. An example is SLC.	The ATWS event analyses, documented in PRA-RB-01-002S07 revision 1, table 1 identifies the systems associated with each safety function. That table also identifies safety function success criteria in most cases. The success criteria for RPS-mechanical was found in a notebook assumption. However, there is at least one instance in which success criteria is not documented in table 1. An example is SLC.	Update the documentation to specify the missing success criteria or provide a pointer to where that criteria is located. In addition, update table 1 to include all success criteria.	As discussed in the associated recommendation, this finding is documentation in nature and its resolution does not impact RBS PRA results. Appropriate system related success criteria are documented in System Notebooks. Documentation in the success criteria notebook for the Rev.6 PRA update will be expanded to include success criteria specific to ATWS, ISLOCA, and BOC events, which are the only events for which accident sequences are developed in notebooks separate from the Accident Sequence notebook. Note the applicable Supporting Requirement from the Standard was judged to be Met. This documentation related Finding has no impact on the ILRT Extension Request.
34 17	SY-A19 (Met)	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, in a manner consistent with the actual practices and history of the plant for removing equipment from service. (a) INCLUDE (1) unavailability caused by testing when a component or system train is reconfigured from	This is a finding because one instance was identified in which the requirement is not met.	PRA-RB-01-002S11, LPCI and CCP system models contain basic events for unavailability of the RHR pumps and CCP pumps at the component level. Basic events for maintenance unavailability are indicated by 'MA' in the basic event name. Section 1.7 of the system notebooks documents the review of test and maintenance applicability	Add maintenance unavailability to the feedwater and condensate systems analysis.	Consideration of unavailability of feedwater and condensate systems would be expected to result in only very slight changes to RBS PRA results, since only one of three pumps for each system are required to meet system success criteria for event mitigation. This is consistent with risk ranking results for the Feedwater and Condensate pumps from the RBS PRA Summary Calculation; no events associated with the feedwater pumps appears in the cutsets generated at an E-13 truncation limit for the risk ranking;

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		its required accident mitigating position such that the component cannot function as required (2) maintenance events at the train level when procedures require isolating the entire train for maintenance (3) maintenance events at a sub-train level (i.e., between tagout boundaries, such as a functional equipment group) when directed by procedures (b) Examples of out-of-service unavailability to be modeled are as follows: (1) train outages during a work window for preventive/corrective maintenance (2) a functional equipment group (FEG) removed from service for preventive/corrective maintenance (3) a relief valve taken out of service		associated with a given system/train/component. PRA-RB-01-002S11 R1 documents the feedwater and condensate system analysis. PRA-RB-01-002S11 R1 documents the feedwater and condensate system analysis. Unavailability of a feedwater or condensate pump due to maintenance is not included in the analysis. There are 3 40% feedwater pumps and 3 50% condensate pumps. Therefore, maintenance of a single feedwater pump or a single condensate pump during power operations is possible.		the maximum RAW of 1.016 and maximum FV of 3.21E-05 for the individual Condensate pumps demonstrate very low risk significance. Thus, only very small if any impact on PRA results would be expected associated with resolution of this finding. This Finding remains open for consideration as an enhancement to add to the model for the next full model update, Revision 6. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This Finding has negligible impact on the ILRT Extension Request.
36 18	HR-E4 (Cat.I)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	This is considered to be a finding as validation/input has not been obtained to validate proper modeling and timing of operator response.	No documentation of simulator observations were identified. While Appendix C to PRA-RB-01-002S03 documents operator input for the HRA analysis, no documented talk throughs or review by either Operations Staff or Operations Training Staff with respect to the response modeling (accident sequence progression) was identified.	Involve/conduct and document Operations / Operations Training review of the response models to insure timing and operator response modeling is correct.	There has been extensive discussion regarding operator actions modeled in the RBS PRA over the years. Discussions regarding operator actions arise during Expert Panel meetings, PRA training for operations, and regular observations of simulator training and scenarios by PRA staff. Scenarios are also discussed as part of routine support for on-line maintenance issues and when risk assessments are performed for plant conditions. More explicit documentation of interactions between the RBS PRA staff and Operations will be incorporated in the next PRA update. Thus, this finding is by nature a documentation issue. This Finding has no impact on the ILRT Extension Request.
37 19	HR-E4 (Cat.I)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	Coupling of two separate operator responses, that may be performed by different individuals, may mask insights related to operator significance, in addition to, masking dependencies	B21-XHE-FO-INHIB was identified which coupled two separate operator actions (inhibit ADS and terminate and prevent HPCS) in a single action. Both of these actions may or may not be performed by the same individual. Spreadsheet HFE_CP.xls only evaluates inhibit ADS with no execution probability based on a simple action (agree), however, terminate and prevent of HPCS is performed via a hardcard (several actions).	Model the inhibit ADS during an ATWS and terminate and prevent HPCS as separate operator actions (separate actions on EOP-01A in step RCA-3).	The suggested change to break B21-XHE-FO-INHIB into two separate operator actions is being incorporated into PRA Revision 5A. This change has negligible impact on CDF. This finding has been resolved and has negligible impact on the ILRT Extension Request.
38 20	HR-E4 (Cat.I) HR-E2 (Met)	Cat.I: No requirement for using simulator observations or talkthroughs with operators to confirm response models. Cat.II: USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled. E2 (Cat.I/II/III): IDENTIFY those actions (a) required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating core damage as defined by the success criteria (e.g., operator initiates RHR) (b) performed by the control room staff either in response to procedural direction or as skill-of-the-craft to diagnose and then recover a failed function, system, or component that is used in the performance of a response action as identified in HR-H1.	This is a finding because this is a potentially significant operator action, that if failure were to occur, could lead to an uncontrolled injection / power excursion during an ATWS. HR-E2: The HRA development has modeled HEPs for operator response in regards to automatic failure of systems such as ECCS. Review has noted that an HEP for the termination and prevention of the low pressure ECCS systems as directed per EOP-01A step RLA-13 was not developed/modeled.	An operator action was not identified/modeled for the termination and prevention of the low pressure injection systems (EOP-01A RLA-13). Following an emergency depressurization, if this action was not performed, a substantial uncontrolled injection could occur resulting in a power excursion. Operator review/walk through/talk through the specific accident sequence may have identified this detail.	Model the low pressure terminate and prevent operator action and validate with Operations / Operations Training.	The River Bend model includes the operator action to terminate and prevent injection as part of the action for controlling RPV power and level during ATWS, event B21-XHE-FO-LVCTL. The human error probability for this operator action encompasses the action to terminate ECCS injection per RBS Emergency Operating Procedures (EOP's). This finding has been resolved and does not impact the ILRT Extension Request.
40 21	SY-A20	INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity (see DA-C14).	This is considered a finding as the requirement is to include these types of maintenance practices (as-operated plant).	DA-C14 basis for assessment notes that Appendix H of PRA-RB-01-002S05 Rev1 EC22619.pdf discusses 'super outages' that occur usually twice a year on	Incorporate these events into the model.	These events were intended for incorporation into the model; failure to do so was an oversight. Inclusion of these events is has only a minor impact on the model, based on the 1.52E-04

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				various components belonging to intersystem trains. Basic events CONCRNT-MA-DIV01, CONCRNT-MA-DIV02, and CONCRNT-MA-DIV03 were developed for these outages. Although developed to represent actual maintenance practices, these basic events do not appear in the PRA fault model. Also, the CRD System and SW fault tree models do not include the simultaneous unavailability of redundant equipment due to technical specification constraints for planned activities.	Evaluate CRD and SW for possible simultaneous maintenance conditions.	probability associated with these events and the fact that unavailability of the opposite train would be a mutually exclusive event. There is no connection between the CRD system and the Service Water system which would require modeling as a simultaneous maintenance condition. Changes in response to this Finding have been incorporated into the RBS PRA model used with EOOS and for PRA Revision 5A.. Thus, this Finding has been resolved and does not impact the ILRT Extension Request.
41 22	QU-D4 (Met) LE-F2 (Met)	QU-D4: COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another? LE-F2: REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant-specificity is appropriate for significant contributors, etc.).	QU-D4: This is considered a finding as an opportunity for a checks and balances may be missed. While this SR is administratively met, further depth in understanding the differences would help in strengthening the model. LE-F2: PRA-RB-01-002S12 revision 1, Attachment 10 documents meeting minutes/notes associated with LERF cutset reviews. Attachment 11 also captures review comments and resolution. A comparison among the various BWR/6 designs was also provided, albeit at an administrative level. An approach similar to that recommended for QU-D4 for CDF should be used for the LERF as well.	A comparison among the BWR/6 population was conducted and is documented in PRA-RB-01-002 revision 1. Observations of differences were noted, however, additional depth as to the differences may be required establish more credible explanations. For example, the loss of the power conversion system differences may to more attributable to the additional containment heat removal capability at RB verses high pressure injection when compared to Plant "B" (Plant B also has a motor driven feedwater pump). Additionally, higher SBO contributions were attributed to the dependence upon electrical switchgear room cooling (RCIC is not dependent upon electrical switchgear room cooling). This SR is marked as met, however, a finding has been given to establish more credible explanations of the deltas. An in-depth comparison may also provide feedback insights.	Provide a more detailed / credible comparative analysis of significant deltas identified.	While additional insights would be obtained from a deeper and more detailed review of differences between plants, the level of detail at which River Bend has performed this comparison is judged to be better than average. RBS participates in the monthly BWR6 PRA conference call, which includes discussions of the various plant system models to allow for understanding of differences due to plant designs and modeling. Additional insights have been gained through participation of Entergy PRA engineers in the Perry Level 2 focused scope peer review and through support work for the ongoing Grand Gulf PRA Revision 4. The additional insights would be of value but would not result in changes to the results of the plant PRA, thus this Finding is considered to be documentation in nature and will be closed as part of the future Revision 6 PRA update. Note the applicable Supporting Requirements from the Standard was judged to be Met. This documentation related Finding has no impact upon the ILRT Extension Request.
44 23	DA-D4 (Met)	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Document PRA-RB-01-002S05 provides the results of the Bayesian approach and the cumulative distribution, but does not provide evidence of a review of the distributions confirming the Bayesian updating was appropriate. BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. Also, as noted in IE-C4, log normal prior distributions were assumed for the generic failure data although the generic data was based on beta and gamma distributions. The basis for this transformation to lognormal should be discussed in the notebook. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard.	BYS-EG1 FTR distribution shows a prior of 8E-4 while the plant data is near 1E-2. Concern is that generic data from reliable components is applied to an unreliable component. Found the same concern with EGS-EG1A,B and E22-* on page 65. As a follow-up, the utility stated the first distribution did not meet their acceptance criteria, but the second did (not less than 5 percentile of generic mean). Plant criteria may not meet the reasonableness requirement of the standard.	Review the updated results and correct any values that do not meet the acceptance criteria. Document the review that the posterior distributions to confirm the Bayesian updates were appropriate.	See discussion related to Finding #15 (SR's IE-C4 and DA-D4) which establishes that the Entergy data Bayesian update process meets the reasonableness criteria of the Standard and that the Bayesian update results were appropriate. This finding has been resolved. As discussed therein, this finding has no impact on the ILRT Extension Request.
45 24	DA-C1	OBTAIN generic parameter estimates from recognized sources. ENSURE that the parameter	The generic parameter estimates are	The generic data document PRA-ES-01-003 has a	Resolve the differences between the	System Analysis Guide EN-NE-G-010 provides the event and component type identifiers and standard failure modes codes for

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
	(Met)	definitions and boundary conditions are consistent with those established in response to DA-A1 to DA-A4. (Example: some sources include the breaker within the pump boundary, whereas others do not.) DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data is consistent with the test and maintenance philosophies for the subject plant. Examples of parameter estimates and associated sources include (a) component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20] (b) common cause failures: NUREG/CR-5497 [2-8], NUREG/CR-6268 [2-9] (c) AC off-site power recovery: NUREG/CR-5496 [2-10], NUREG/CR-5032 [2-11] (d) component recovery See NUREG/CR-6823 [2-1] for a listing of additional data sources.	generally taken from NUREG/CR-6928, as documented in the generic data notebook PRA-ES-01-003. However, the naming conventions between the generic data and the River Bend actual basic event nomenclatures differ. Common cause failure data is obtained from NUREC/CR-5497 and the NRC CCFWIN database as documented in the CCF notebook (PRA-RB-01-002S04). AC OSP recovery is documented in PRA-RB-01-002S09 and is based on data from NUREG/CR-5032, NUREG/CR-6890 and EPRI LOSP reports. Component recovery is not credited in the PRA. Unavailability estimates are based on plant specific date.	type code of BUS FTOP where the database has a type code of BAC NO for bus failure to operate. There is no direct tie from the database to the documentation. Also, the ASL DN type appears to be STL FTOP in the generic data document	documentation and the contents of the PRA database.	Entergy PRA's. The Tables in the data calculation show the mapping of failure modes, per EN-NE-G-010, to the NUREG/CR-6928 failure modes. This confirms, for example, that BUS FTOP from the NUREG is mapped to BAC LP or BDC LP, bus fail to operate events for AC or DC busses. The RBS CAFTA database file includes in the Type Code data window the corresponding NUREG/CR-6928 type codes. Thus, documentation exists to provide the direct tie from the source NUREG to the RBS type codes. Note the applicable Supporting Requirement from the Standard was judged to be Met. This finding has been resolved and has no impact on the ILRT Extension Request.
46 25	DA-D6 (Met) SY-B4 (Met)	DA-D6: USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities in a manner consistent with the component boundaries. SY-B4: INCORPORATE common cause failures into the system model in a manner consistent with the common cause model used for data analysis. (See DA-D6.)	DA-D6: Generic common cause failure factors are used, as documented in PRA-RB-01-002S04. The CCFWIN database is used, which includes all industry data. No specific discussion of the applicability of the generic data to River Bend is provided, however. The common cause factors boundaries were intended to match the independent data from document PRA-ES-01-003. There is not a strong documentation link between it and the CCF information found in PRA-RB-01-002S04, Rev. 2 since the factors were not used. The EDG's independent run basic events have been split into first hour and fails to continue running events (e.g. EGS-DGN-F1-EG01A and EGS-DGN-FR-EG01A). The CCF basic events have combined failure to start and failure to run for the first hour (e.g. EGS-DGN-C2-DGFR), but this is not well documented, and the BE name is misleading. Did not find CCFs for EDG FO transfer pump check valves (e.g. EGF-CKV-CC-V33) although CCFs exist for pumps. This omission was self-identified by the PRA staff, but has not yet been incorporated into the model. SY-B4: 'Table B-1. RBS CCF Basic Events and their associated Independent Basic Events' and 'Table B-2. RBS CCF Basic Events and their Associated CCF Multipliers' in Calculation PRA-RB-01-002S04, Rev. 2 lists all CCF events incorporated in the model. A list of common cause failures basic events are presented in Section 2.1.3 of	Found in FPW-ENG-C2-2FTR that it appears to use a generic independent event as described in PRA-RB-01-002S04 and not the calculated independent event in the database that matches the document PRA-RB-01-002S05. (2.07E-03 EF 9.8 vs. 3.75E-04 EF 18). Checked the spreadsheet RBSCCF that was used to develop the CCFs and found the 2.07E-3 event was used. The cutset file provided (RBS-R5_rec_merged....cut contains the 3.75E-4 value.	Update the CCF BE calculation to use the current independent BE failure values and consistent component boundaries.	The issues related to Common Cause Failure of diesel fuel oil transfer pumps and Fire Protection Water pumps identified in this Finding have been corrected and incorporated into the RBS PRA model for EOOS and for Rev.5A. These corrections have minimal impact on results; CCF of the Fuel Oil Transfer Pumps have a FV risk importance of less than 0.01. Note the applicable Supporting Requirement from the Standard was judged to be Met.. This Finding has been resolved and has minimal impact on the ILRT Extension Request.
47 26	DA-D6 (Met) SY-B4 (Met)			The EDG's independent run basic events have been split into first hour and fails to continue running events (e.g. EGS-DGN-F1-EG01A and EGS-DGN-FR-EG01A). The CCF basic events have combined failure to start and failure to run for the first hour (e.g. EGS-DGN-C2-DGFR), but this is not well documented, and the BE name is misleading. Did not find CCFs for EDG FO transfer pump check valves (e.g. EGF-CKV-CC-V33) although CCFs exist for pumps. This omission was self-identified by the PRA staff, but has not yet been incorporated into the model.	Update the CCF analysis to include the missing CCF events for the transfer pumps. Enhance the documentation of the DG CCF events to better explain the FTS and FTR modeling.	The CCF analysis and the PRA model for EOOS and for Rev.5A have been updated to include the missing CCF events for the Diesel Fuel Oil system. Note the applicable Supporting Requirements from the Standard were judged to be Met. This Finding has been resolved and has minimal impact on the ILRT Extension Request.

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
			each system notebook. Checking of a sample CCF basic events from Appendix B1 Table B-1 showed them to be incorporated in the model in accordance with the CCF method and the CCF boundaries			
49 27	IFEV-B3 IFPP-B3 IFQU-B3 IFSN-B3 IFSO-B3 (Not Met)	IFEV-B3 Document sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood-induced initiating events. IFPP-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood plant partitioning. IFQU-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood accident sequences and quantification. IFSN-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood scenarios. IFSO-B3: DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood sources.	This is a finding since an assessment of the sources of uncertainty is required by the standard.	No assessment of the sources of uncertainty was documented.	Perform and document an assessment of flood-related sources of uncertainty.	This finding is considered documentation in nature, since performance of an uncertainty study would not impact CDF results. All of the SR's associated with this finding are considered Documentation requirements in the Standard. The Revision of the Internal Flooding PRA subsequent to the peer review did review the results to obtain insights into importance of system and location contributors to the Internal Flooding risk, which does permit judgments concerning the impact of uncertainties. The Section 5.7 bounding analysis of potential impact of ILRT extension from external events and other hazard groups fully accounts for Internal Flooding. This finding does not impact the ILRT Extension Request.
51 28	IFSO-A4 (Not Met)	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a release. INCLUDE (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow-through openings created to perform maintenance; inadvertent actuation of fire-suppression system (c) other events resulting in a release into the flood area	This is a finding since the requirements of this SR are not met. Identification of mechanisms is required by the SR. Missing failure mechanisms could impact the overall results.	Flooding mechanisms are not identified in the analysis. Although calculation PRA-RB-01-004 Rev. 0 states in Section 3.4 that all mechanisms were considered, this does not appear to be the case. For example, section 4.2.5.8 states that the area is not considered because the only source is a pre-action fire system. However, inadvertent actuation of this system should be addressed. Other instances exist.	Document the potential failure mechanism included in each flood area.	This finding is considered primarily documentation in nature, as discussed in the finding and recommendation. The EPRI pipe failure data used in this analysis encompasses all pipe failure mechanisms; there is no readily available data that allows distinguishing between different failure mechanisms. Since the failure rate data used in the analysis encompasses the various failure mechanisms, there would be no change to the results associated with identifying specific failure mechanisms. Thus, this documentation related finding does not impact the ILRT Extension Request.
52 29	IFSO-A5 (Met)	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) flow rate (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source	Inclusion of this information is required by the SR; hence this is a finding.	The characteristics of each source are documented for each scenario developed in Section 4.2 of Calculation PRA-RB-01-004 Rev. 0. These scenarios identify the flow rate by evaluating a complete rupture of the line analyzed. The capacity of the source is considered for finite-volume systems, however, no volume information for these systems was identified in the documentation. Pressure and temperature were not identified in the documentation.	Document release characterization for each source.	Characterization of failures and flow rates are included in the scenario descriptions in the revision to Internal Flooding Analysis calculation, including documentation for the scenario in an Appendix. As stated, source capacities have been considered in detailed scenario development. System information, including volumes and pump flow, has been added as part of the subsequent Rev.1 to the calculation. System pressures are used to calculate flow rates using spreadsheets. Systems which are potential HELB sources are identified. Much of this information had been included in the original documentation but was not well organized. Since failure flow rates have been appropriately developed and other characteristics documented through the

Finding	SR and Assessment	SR description	Basis for Peer Review Finding	Peer Review Comment	Possible Resolution	Disposition and Impact on ILRT
						documentation would not impact the results, this documentation finding does not impact IFPRA results. Note the applicable Supporting Requirement from the Standard was judged to be Met. This documentation Finding does not impact the ILRT Extension Request.

A.3 Identification of Key Assumptions

The methodology employed in this risk assessment followed the EPRI guidance [A.6] as approved by the NRC. The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analysis indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.4 Summary

A PRA technical adequacy evaluation was performed consistent with the requirements of RG 1.200, Revision 2. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed extension to the ILRT/DWBT intervals for RBS to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.5 References

[A.1] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 2, March 2009.

[A.2] Entergy Calculation, *RBS PRA Summary Report (PRA Rev. 5)*, PRA-RB-01-002, Revision 1, March 2011.

[A.3] Entergy Calculation, *RBS PRA LERF Model*, PRA-RB-01-002S12, Revision 1, March 2011.

[A.4] Entergy Fleet Procedure EN-DC-151, Revision 2, *PSA Maintenance and Update*, January 2011.

[A.5] BWR Owners Group, *River Bend Station PRA Peer Review Report Using ASME/ANS PRA Standard Requirements,* July 2011.

[A.6] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325*, EPRI, Palo Alto, CA: 2008. 1018243.

Attachment 4

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Proposed Technical Specification Changes (mark-up)

Note, markup deletions identified by strikethrough (~~delete~~) and additions identified by underline (<u>addition</u>).

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.3	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	--------NOTE------- SR 3.0.2 is not applicable for extensions > ~~12~~ 9 months ---------- 24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit AND 48 months following a test with bypass leakage greater than the bypass leakage limit AND ~~120~~ 180 months ~~except that the next drywell leak rate test performed after the June 24, 1994 test shall be performed no later than June 23, 2009.~~

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that do not meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 DELETED

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~, NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012. ~~except that the next Type A test performed after the August 15, 1992, Type A test shall be performed no later than April 14, 2008.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 7.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.325% of primary containment air weight per day.

The Primary Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

The provisions of SR 3.0.2 do not apply to test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.14 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an

Attachment 5

RBG-47620

Proposed Technical Specification Bases Changes (mark-up)

For Information Only

Note, markup deletions identified by strikethrough (~~delete~~) and additions identified by underline (addition).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.1.3

The analyses in Reference 1 are based on a maximum drywell bypass leakage. This Surveillance ensures that the actual drywell bypass leakage is less than or equal to the acceptable $A/\sqrt{k}$ design value of 0.81 ft^2. As left drywell bypass leakage, prior to the first startup after performing a required drywell bypass leakage test, is required to be ≤ 10% of the drywell bypass leakage limit. At all other times between required drywell leakage rate tests, the acceptance criteria is based on design $A/\sqrt{k}$. At the design $A/\sqrt{k}$ the containment temperature and pressurization response are bounded by the assumptions of the safety analysis. Due to NRC Generic Letter 96-06 concerns, integrity of the reactor recirculation flow control valve hydraulic power unit (HPU) penetrations cannot be assumed. For this reason, 0.0164 ft^2 is added to the drywell bypass leakage surveillance result (Ref. 3). This surveillance is performed at least once every ~~10~~ 15 years on a performance based frequency. ~~This frequency is modified on a one time basis until June 23, 2009.~~ The frequency is consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility that sufficient component failures will occur such that the drywell bypass leakage limit will be exceeded. If during the performance of this required Surveillance the drywell bypass leakage rate is greater than the drywell bypass leakage limit, the Surveillance Frequency is increased to every 48 months. If during the performance of the subsequent consecutive Surveillance the drywell bypass leakage rate is less than or equal to the drywell bypass leakage limit, the ~~10~~ 15 year Frequency may be resumed. If during the performance of two consecutive Surveillances the drywell bypass leakage is greater than the drywell bypass leakage limit, the Surveillance Frequency is increased to at least once every 24 months. The 24 month Frequency is maintained until during the performance of two consecutive Surveillances the drywell bypass leakage rate is less than or equal to the drywell bypass leakage limit, at which time the ~~10~~ 15 year Frequency may be resumed. For two Surveillances to be considered consecutive, the Surveillances must be performed at least 12 months apart. Since the frequency is performance based, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.1.4

The exposed accessible drywell interior and exterior surfaces are inspected to ensure there are no apparent physical defects that would prevent the drywell from

(continued)

Attachment 6

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List of Regulatory Commitments

List of Regulatory 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	
RBS will continue to perform this qualitative assessment of drywell leak tightness once per operating cycle to support a change to performing the DWBT every 15 years.		X	Upon Implementation