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CNS-16-001

January 18, 2016

10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station (CNS), Units 1 and 2
Docket Numbers 50-413 and 50-414
License Amendment Request (LAR) to Revise Technical Specifications (TS) Section 5.5.2, "Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

Pursuant to 10 CFR 50.90, Duke Energy requests an amendment to the CNS Unit 1 Renewed Facility Operating License (NPF-35) and the CNS Unit 2 Renewed Facility Operating License (NPF-52), by incorporating the attached proposed change into the Unit 1 and Unit 2 TS. Specifically, the proposed amendment is a request to revise TS 5.5.2 to allow the following:

- Increase in the existing Type A Integrated Leakage Rate Test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Topical Report NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A.

- Adopt an extension of the containment isolation valve leakage testing (Type C) frequency from the 60 months currently permitted by 10 CFR 50, Appendix J, Option B, to a 75-month frequency for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3-A.

- Adopt the use of ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements".

- Adopt a more conservative grace interval of 9 months for Type A, Type B, and Type C leakage tests in accordance with NEI 94-01, Revision 3-A.

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NRR

The proposed change to the TS contained herein would revise CNS TS 5.5.2 by replacing the references to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program" and 10 CFR 50, Appendix J, Option B with a reference to NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as the documents used by CNS to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. This LAR also proposes the following administrative changes to TS 5.5.2:

- Delete the information regarding the performance of containment visual inspections as required by RG 1.163, Regulatory Position C.3, as the containment inspections are addressed in TS Surveillance Requirement (SR) 3.6.1.1.

- Delete the information regarding the performance of the next CNS Unit 1 Type A test no later than November 13, 2015 and the next CNS Unit 2 Type A test no later than February 6, 2008, as both Type A tests have already occurred.

Duke Energy requests approval of the proposed LAR by August 31, 2016, to be implemented within 120 days of the issuance of the license amendment.

The contents of this amendment request package are as follows:

The enclosure to this letter provides an evaluation of the proposed change. Attachment 1 provides the marked-up TS pages showing the proposed changes. Attachment 2 provides retyped TS pages. Attachment 3 provides a marked-up TS Bases page for TS Bases B 3.6.3, "Containment Isolation Valves". This page is being provided for information only. Attachment 4 provides a retyped TS Bases page for TS Bases B 3.6.3. This page is being provided for information only. Attachment 5 provides an evaluation of the risk significance of the permanent ILRT extension.

Implementation of the approved amendments will require changes to the Updated Final Safety Analysis Report (UFSAR). Necessary UFSAR changes will be submitted to the NRC in accordance with 10 CFR 50.71(e), with approved exemptions.

There are no regulatory commitments being made in conjunction with this amendment request.

In accordance with Duke Energy administrative procedures and the Quality Assurance Program Topical Report, this amendment request has been previously reviewed and approved by the Catawba Plant Operations Review Committee.

Pursuant to 10 CFR 50.91, a copy of this amendment request is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to L.J. Rudy at (803) 701-3084.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on January 18, 2016.

Very truly yours,

Kelvin Henderson
Vice President, Catawba Nuclear Station

LJR/s

Enclosure and Attachments

xc (with enclosure and attachments):

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Evaluation of the Proposed Change

Subject: License Amendment Request - Revise Technical Specification Section 5.5.2 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

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Attachments:
1. Technical Specification Pages (Mark-up)
2. Technical Specification Pages (Retyped)
3. Technical Specification Bases Page Markup (For Information Only)
4. Retyped Technical Specification Bases Page (For Information Only)
5. Evaluation of Risk Significance of Permanent ILRT Extension

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy requests an amendment to the Catawba Nuclear Station (CNS) Unit 1, Renewed Facility Operating License (NPF-35), and Unit 2, Renewed Facility Operating License (NPF-52) (CNS), by incorporating the attached proposed change into the Unit 1 and Unit 2 Technical Specifications (TS). Specifically, proposed change is a request to revise TS 5.5.2 "Containment Leakage Rate Testing Program to allow the following:

- Increase in the existing Type A integrated leakage rate test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A.

- Adopt an extension of the containment isolation valve leakage testing (Type C) frequency from the 60 months currently permitted by 10 CFR 50, Appendix J, Option B, to a 75-month frequency for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3-A.

- Adopt the use of ANSI/ANS 56.8-2002, Containment System Leakage Testing Requirements.

- Adopt a more conservative grace interval of 9 months, for Type A, Type B and Type C leakage tests in accordance with NEI 94-01, Revision 3-A.

The proposed change to the TS contained herein would revise CNS TS 5.5.2, by replacing the references to Regulatory Guide (RG) 1.163, Performance-Based Containment Leak-Test Program, (Reference 1) and 10 CFR 50, Appendix J, Option B with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A (Reference 2), dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 3), dated October 2008, as the documents used by CNS to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. This license amendment request (LAR) also proposes the following administrative changes to TS 5.5.2:

- Deleting the information regarding the performance of containment visual inspections as required by Regulatory Position C.3 as the containment inspections are addressed in TS SR 3.6.1.1

- Deleting the information regarding the performance of the next CNS Unit 1 Type A test no later than November 13, 2015 and the next CNS Unit 2 Type A test no later than February 6, 2008, as both Type A tests have already occurred.

2.0 DETAILED DESCRIPTION

2.1 Current 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, as modified by the following exceptions:

a. The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test; and

b. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 14, 2000 (Unit 1) and February 7, 1993 (Unit 2) Type A test shall be performed no later than November 13, 2015 (Unit 1) and February 6, 2008 (Unit 2).

2.2 TS Change Description

The proposed change to the Technical Specifications (TS) contained herein would revise CNS TS 5.5.2, by replacing the reference to Regulatory Guide (RG) 1.163 (Reference 1) with a reference to NEI topical report NEI 94-01 Revision 3-A (Reference 2) and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 3), dated October 2008, as the specific documents used by CNS to implement the Unit 1 and Unit 2 performance-based leakage testing programs in accordance with Option B of 10 CFR 50, Appendix J.

The proposed change would allow an increase in the Integrated Leak Rate Test (ILRT) test interval from its current 10-year frequency to a maximum of 15 years and the extension of the containment isolation valves leakage test (Type C tests) from its current 60-month frequency to 75 months in accordance with NEI 94-01 Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A. This license amendment request (LAR) also proposes the following administrative changes to TS 5.5.2:

- Deleting the information regarding the performance of containment visual inspections as required by Regulatory Position C.3 as the containment inspections are already addressed in TS SR 3.6.1.1.

- Deleting the information regarding the performance of the next CNS Unit 1 Type A test no later than November 13, 2015 and the next CNS Unit 2 Type A test no later than February 6, 2008, as both Type A tests have already occurred.

The proposed change will revise TS 5.5.2 to state, in part:

> 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 NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

A markup of TS 5.5.2 is provided in Attachment 1. The retyped TS pages are provided in Attachment 2.

A markup of TS Bases for SR 3.6.3.6 is provided in Attachment 3 for informational purposes only. The retyped TS Bases page is provided in Attachment 4 for informational purposes only.

Approval of this proposed change is being requested in time to defer the performance of the next CNS Unit 2 ILRT from the Fall 2016 refueling outage (RFO) to a subsequent RFO no later than Fall 2022.

Attachment 5 contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of Nuclear Regulatory Commission (NRC) RG 1.174 (Reference 4) and NRC RG 1.200, Revision 2 (Reference 5). The risk assessment concluded that the increase in risk as a result of this proposed change is considered to be insignificant since it represents a very small change to the CNS risk profile.

3.0 TECHNICAL EVALUATION

3.1 Justification for the Technical Specification Change

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. 10 CFR 50, Appendix J, also ensures that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating primary containment. 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 (DBA). Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Types 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 Types 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.

Also in 1995, RG 1.163 was issued. The RG endorsed NEI 94-01, Revision 0, (Reference 6) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 7) and Electric Power Research Institute (EPRI) TR-104285 (Reference 8) both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months were considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but that this "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A, (Reference 3) was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC Safety Evaluation Report (SER) on NEI 94-01. The NRC SER was included in the front matter of this NEI report. NEI 94-01, Revision 2-A, includes provisions for extending Type A ILRT intervals to up to fifteen years and incorporates the regulatory positions stated in RG 1.163 (September 1995). It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

Acceptability for referencing by licensees proposing to amend their TS is provided in Section 5.0 of the SER and states the following:

> The NRC staff, therefore, finds that this guidance is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of this SE. In addition, in accordance with the NRC staff's resolution of the comments provided by NEI on the draft SE, the following changes will be made by NEI to the "-A" version of the TR. Therefore, consistent with the language in this final SE:
>
> A. NEI TR 94-01, Revision 2, will be revised in the "-A" version of the report, as discussed in the last paragraph of Section 3.1.2.2, "Extending Type B&C Test Intervals," to the final SE.
>
> B. EPRI Report No. 1009325, Revision 2, will be revised in the "-A" version of the report, to change the population dose acceptance guidelines and the CCFP guidelines. (As stated in Section 4.2 of the final SE Limitation and Condition #2).

In 2012, NEI 94-01, Revision 3-A, was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, and includes provisions for extending Type A ILRT intervals to up to fifteen years. NEI 94-01 has been endorsed by RG 1.163 and NRC SERs of June 25, 2008 (Reference 9) and June 8, 2012 (Reference 10) as an acceptable methodology

for complying with the provisions of Option B to 10 CFR Part 50. The regulatory positions stated in RG 1.163 as modified by NRC SERs of June 25, 2008, and June 8, 2012, are incorporated in this document. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic as-found tests where the results of each test are within a licensee's allowable administrative limits. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2.

Acceptability for referencing by licensees proposing to amend their TS is provided in Section 5.0 of the SER and states the following:

> The NRC staff, therefore, finds that this guidance, as modified to include two limitations and conditions, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing. Any applicant may reference NEI TR 94-01, Revision 3, as modified by this SE and approved by the NRC, in a licensing action to satisfy the requirements of Option B to 10 CFR Part 50, Appendix J. The NRC staff is not required to repeat its review of the matters described in the TR conditioned upon the changes described in this SE (Sections 3 and 4) to be incorporated when the report appears as a reference which was complied with a request for relief, or other related licensing actions.

NEI 94-01, Revision 3-A, Section 10.1 concerning the use of grace in the deferral of Type B and Type C LLRTs test intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing, states:

> "Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing given in this section may be extended by up to 25 percent of the test interval, not to exceed nine months.
>
> Notes: For routine scheduling of tests at intervals over 60 months, refer to the additional requirements of Section 11.3.2.
>
> Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. This provision (nine month extension) does not apply to valves that are restricted and/or limited to 30 month intervals in Section 10.2 (such as BWR MSIVs) or to valves held to the base interval (30 months) due to unsatisfactory LLRT performance."

3.1.1 Current CNS 10 CFR 50, Appendix J Requirements

Title 10 CFR Part 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance Based Requirements." CNS has implemented the requirements of 10 CFR Part 50, Appendix J, Option B for Types A, B and C testing. Current TS 5.5.2 requires the following:

> 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, as modified by the following exceptions:
>
> a. The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test; and
>
> b. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 14, 2000 (Unit 1) and February 7, 1993 (Unit 2) Type A test shall be performed no later than November 13, 2015 (Unit 1) and February 6, 2008 (Unit 2).

RG 1.163, Section C.1, states that licensees intending to comply with 10 CFR Part 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 6) rather than using test intervals specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994. NEI 94-01, Section 11.0, refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of the Option B performance-based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Type A tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 7). The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak tightness for differing types of containment types, including a reinforced, shallow domed concrete containment similar to CNS containment structures. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three tests per ten years to one test per twenty 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 Types 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, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

3.1.2 CNS 10 CFR 50, Appendix J, Option B Licensing History

SER dated February 29, 1988 – ML013040289 (Reference 11)
The Commission issued Amendment No. 41 to Facility Operating License No. NPF-35 and Amendment No. 34 to Facility Operating License NPF-52 for CNS, Units 1 and 2.

The amendments modified the TS to increase by 50% the allowed containment overall integrated leakage rate.

SER dated May 13, 1996 - ML013060529 (Reference 12)
The Commission issued Amendment No.144 to Facility Operating License No. NPF-35 and Amendment No. 138 to Facility Operating License NPF-52 for CNS, Units 1 and 2.

The amendments revised the TS so that the containment integrated leak rate Type A test could be performed consistent with the revised 10 CFR Part 50, Appendix J, Option B, by referring to RG 1.163, "Performance-Based Containment Leak-Test Program." No changes to implement Option B for the Type B and Type C tests were requested by the licensee at that time.

SER dated July 31, 2001 – ML012290327 (Reference 13)
The Commission issued Amendment No. 192 to Facility Operating License NPF-35 and Amendment No. 184 to Facility Operating License NPF-52 for CNS, Units 1 and 2.

The amendments permitted implementation of 10 CFR Part 50, Appendix J, Option B and reference RG 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specified a method acceptable to the NRC for complying with Option B. These changes related only to Type B and Type C (local) leakage rate testing (LLRT). In addition, the amendments revised Surveillance Requirement 3.6.3.8 by deleting the requirement for soap bubble testing of welded penetrations that are not individually testable and clarified the Bases for TS 3.6.2 pertaining to the containment air lock door.

SER dated March 12, 2003 – ML030760108 (Reference 14)
The Commission issued Amendment No. 205 to Facility Operating License NPF-35 and Amendment No. 198 to Facility Operating License NPF-52 for CNS, Units 1 and 2.

The amendments revised the TSs to allow a one-time change in the Appendix J, Type A Containment ILRT interval from the currently required 10-year interval to a test interval of 15 years.

3.1.3 Containment Building Description

Description of the Containment

The Containment consists of a Containment Vessel and a separate Reactor Building enclosing an annulus. The Containment Vessel is a freestanding welded steel structure with a vertical cylinder, hemispherical dome, and a flat circular base. The Reactor Building is a reinforced concrete structure, similar in shape to the Containment Vessel.

The containment vessel is a freestanding welded steel structure consisting of a vertical cylinder with a hemispherical dome and a flat circular base. The cylinder is stiffened by circumferential ring girders and vertical stringers welded to the exterior of the vessel. The containment shell is anchored to the Reactor Building foundation by means of anchor bolts around the perimeter of the cylinder base. The flat base of the containment is a 1/4 in. liner plate encased in concrete and anchored to the Reactor Building foundation. The base liner plate functions only as a leak-tight membrane and is not designed for structural capabilities. The containment vessel has a diameter of 115 ft. and overall height of 171 ft. 3 in.

The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal unit operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal unit operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal unit operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice bed consists of a minimum of 2,132,000 lbs. of ice stored within the ice condenser. The primary purpose of the ice bed is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice baskets contain the ice within the ice condenser. The ice bed is considered to consist of the total volume from the bottom elevation of the ice baskets to the top elevation of the ice baskets. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

The Reactor Building is a reinforced concrete structure composed of a right cylinder with a shallow dome roof and flat circular foundation slab. The cylinder has an inside radius of 63 ft. 6 in. and a wall thickness of 3 ft. The dome has an inside spherical radius of 87 ft and is 2 ft. 3 in. thick. The foundation slab is 137 ft. in diameter and 6 ft. thick.

The Reactor Building houses the steel containment vessel and is designed to provide environmental as well as missile protection for the steel shell.

A six-foot annular space is provided between the steel containment vessel and the Reactor Building for control of the containment external temperatures and pressures. The annular space also provides a controlled air volume for filtering and provides access to penetrations for testing and inspection. Following a loss-of-coolant accident (LOCA), the annular space is kept at a slightly negative pressure to control and filter radioactive leakage, if any, from the containment vessel and penetrations.

Containment Penetrations
Several penetrations are required through the containment vessel for personnel and equipment access, fuel transfer and various piping systems. The containment penetrations are:

Equipment Hatch
The equipment hatch is composed of a 20-foot cylindrical sleeve in the containment shell and a dished head with mating, bolted flanges. The flanged joint has double compressible seals with an annular space for pressurization and testing.

Personnel Locks
Two personnel locks are provided for each unit. Each lock consists of a cylindrical sleeve with a bulkhead at each end. Each bulkhead has double gasketed doors with an interlocking system to prevent both doors from being opened simultaneously. Instrumentation is provided to indicate the position of each door. Double, inflatable seals are provided on each door. The use of double inflatable seals allows testing of the annular space without the use of external strong-backs or other remote devices.

Fuel Transfer Penetration
The Fuel Transfer Penetration is a 30-inch diameter rolled sleeve. Passing through the Fuel Transfer Penetration is a 20-inch Fuel Transfer Tube. The assembly is used for transfer of fuel to and from the fuel pool and the containment fuel transfer canal. The fuel transfer penetration is provided with a double gasketed blind flange in the transfer canal and a gate valve in the fuel pool. Expansion bellows are provided to accommodate differential movement between the connecting buildings.

Spare Penetrations
Spare penetrations are provided to accommodate future mechanical and electrical penetrations. Typical spare penetrations consist of the penetration sleeve and a welded pipe cap.

Penetration Sleeves
All penetration sleeves are preassembled and welded into containment vessel shell plates. Each shell plate having penetration sleeves is stress relieved prior to installation into the containment.

Purge Penetrations
The purge penetrations have one interior and one exterior containment isolation valve. These isolation valves are locked closed in Modes 1 through 4 and verified sealed closed at least once every 31 days. These isolation valves will automatically actuate closed on containment high radiation in Modes 5, 6, and No Mode.

Electrical Penetrations
Medium voltage electrical penetrations for reactor coolant pump power (shown on UFSAR Figure 3-261) use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor.

Low voltage power, control and instrumentation cables enter the containment vessel through penetration assemblies which have been designed to provide two leak tight barriers in series with each conductor.

All electrical penetrations have been designed to maintain containment integrity for Design Basis Accident conditions including pressure, temperature and radiation. Double barriers permit testing of each assembly as required to verify that containment integrity is maintained.

Mechanical Penetrations
Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME Section III, Subsections NC and NE and which are assigned the same classification as the piping system that includes the assembly.

The process line making up the pressure boundary is consistent with the system piping materials, fabrication, inspection, and analysis requirements of ASME Section III, Subsection NC.

Critical high temperature lines and selected engineered safety system and auxiliary lines (regardless of temperature) require the "Hot Penetration" assembly which features the exterior guard pipe for the purpose of returning any fluid leakage to the Containment and for protection of other penetrations in the building annular space. Other lines are treated as cold penetrations since a leak into the annular space would not cause a personnel hazard or damage other penetrations in the immediate area.

Overpressure Protection
Although not required by specific Regulatory or Licensing commitments, where the potential exists to overpressurize containment penetration piping due to thermal expansion of the fluid trapped in the penetration piping, overpressure protection shall be provided. In otherwords, overpressure protection shall be provided to relieve the presssure buildup caused by the heatup of a trapped volume of incompressible fluid between two positively closing valves (due to containment temperature transient) back into containment where an open relief path exists. This open relief path could be the relief valve on any normally aligned component, or to the Reactor Coolant System (NC) itself.

3.1.4 Containment Valve Injection Water System (CVIWS)

The CVIWS is required by 10 CFR 50, Appendix A, General Design Criterion (GDC) 54, "Piping Systems Penetrating Containment," to ensure a water seal to a specific class of containment isolation valves (double disc gate valves) during a LOCA, to prevent leakage of containment atmosphere through the gate valves.

The CVIWS is designed to inject water between the two seating surfaces of double disc gate valves used for Containment isolation. The injection pressure is higher than

Containment design peak pressure during a LOCA. This will prevent leakage of the Containment atmosphere through the gate valves, thereby reducing potential offsite dose below the values specified by 10 CFR 50.67 limits following the postulated accident.

During normal power operation, the system is in a standby mode and does not perform any function. During accident situations the CVIWS is activated to perform its safety related function, thus limiting the release of containment atmosphere past specific containment isolation valves, in order to mitigate the consequences of a LOCA. Containment isolation valves, for systems which are not used to mitigate the consequences of an accident, will be supplied with CVIWS seal water upon receipt of a Phase A isolation signal. Containment isolation valves, for accident mitigating systems which are supplied with seal water from the CVIWS, have their seal water supplies actuated by a Containment Pressure - High-High signal.

The system consists of two independent, redundant trains; one supplying gate valves that are powered by the A train diesel and the other supplying gate valves powered by the B train diesel. This separation of trains prevents the possibility of both containment isolation valves not sealing due to a single failure.

Each train consists of a surge chamber which is filled with water and pressurized with nitrogen. One main header exits the chamber and splits into several headers. A solenoid valve is located in the main header before any of the branch headers which will open after a 60 second delay on a Phase A isolation signal. Each of the headers supply injection water to containment isolation valves located in the same general location, and close on the same engineered safety signal. A solenoid valve is located in each header which supplies seal water to valves closing on a Containment Pressure - High-High signal. These solenoid valves open after a 60 second delay on a Containment Pressure - High-High signal. Since a Phase A isolation signal occurs before a Containment Pressure - High-High signal, the solenoid valve located in the main header will already be injecting water to Containment isolation valves closing on a Phase A isolation signal. This leaves an open path to the headers supplying injection water on a Containment Pressure - High-High signal. The delay for the solenoid valves opening is to allow adequate time for the slowest gate valve to close, before water is injected into the valve seat.

Makeup water is provided from the Demineralized Water Storage Tank for testing and adding water to the surge chamber during normal plant operation. Assured water is provided from the essential header of the Nuclear Service Water System (NSWS). This supply is assured for at least 30 days following a postulated accident. If the water level in the surge chamber drops below the low-low level or if the surge chamber nitrogen pressure drops below the low-low pressure after a Phase A isolation signal, a solenoid valve in the supply line from the NSWS will automatically open and remains open, assuring makeup to the CVIWS at a pressure greater than 110% of peak Containment accident pressure.

Overpressure protection is provided to relieve the pressure buildup caused by the heatup of a trapped volume of incompressible fluid between two positively closing valves (due to containment temperature transient) back into containment where an open relief path exists.

Applicable Safety Analyses
The CVIWS design basis is established by the consequences of the limiting Design Bases Accident (DBA), which is a LOCA. The accident analysis contained in UFSAR Section 6.2, Containment Systems, assumes that only one train of the CVIWS is functional due to a single failure that disables the other train. Makeup water can be assured from the NSWS for 30 days following a postulated LOCA.

The CVIWS satisfies Criterion 3 of 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3.1.5 Containment Spray and Residual Heat Removal Pump Net Positive Suction Head (NPSH)

The available NPSH for the containment spray pumps and the residual heat removal pumps was calculated ignoring any pressure in the containment above atmospheric. Only a minimum amount of static head due to water level above the sump floor was included. This minimum level is the amount of water that must be present in order to transition into the recirculation mode of ECCS operation.

The design adequacy of the recirculation sump strainer has been verified as acceptable through Duke Energy's response to Generic Letter 04-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors". The Generic Letter response (including RAI responses and Supplemental Content Guide) includes evaluations pertaining to minimum sump inventory, NPSH margins, chemical effects, debris loading, etc.

3.2 Inspections

3.2.1 Primary Containment Coatings

Duke Energy complies with RG 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants.

The original coating materials and coating systems were specified by Engineering and applied by the Duke Power Construction Department to all structures within the containment and the Containment Vessel. The coating systems were qualified for radiation exposure, pressure, temperature, and water chemistry exposure during a DBA in accordance with ANSI N101.2.

Carboline coating materials are now used for maintenance of the existing coating systems and for any new applications. These coating systems are specified by Engineering and applied by the Duke Energy Maintenance Department. The Carboline coating materials have been qualified over the existing Mobil/Valspar coatings as a mixed system and as a new coating system for radiation exposure, pressure, temperature, and water chemistry exposure during a DBA in accordance with ANSI N101.2.

The original, maintenance, and new coating systems defining temperature limitations, surface preparation, type of coating, and dry film thickness are tabulated on UFSAR Table 6-135.

The elements of the Catawba Coatings Program are documented in a Nuclear Generation Department Directive. The Catawba Coatings Program includes periodic condition assessments of Service Level I coatings used inside containment. As localized areas of degraded coatings are identified, those areas are evaluated for repair or replacement, as necessary.

9,300 square feet of unqualified coatings inside the containment have been evaluated for impact on the ECCS Sump Strainer and its ability to support ECCS recirculation and containment spray functions.

Coatings inside the Containment are classified as either qualified or unqualified for the purposes of debris generation analysis, which encompasses all of the coating systems used within Containment. Unqualified coatings in Containment are assumed to fail as particulates, in accordance with NEI 04-07 guidance. Qualified coatings are assumed to fail as particulates within a 5D zone of influence (ZOI) based on the methodology outlined in WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of influence for DBA Qualified/Acceptable Coatings". For unqualified coatings, NEI 04-07 guidance directs utilities to assume 100% failure of unqualified coatings into transportable particulate. Catawba performed an alternative analysis utilizing EPRI OEM coatings failure data to refine the quantity of coatings assumed to fail. All of the failed coatings are assumed to transport to the Sump Strainer as particulate debris.

3.2.2 Inservice Inspection Program for Containment – IWE

In accordance with the requirements of Paragraph 50.55a(g), and as modified and supplemented by paragraph 50.55a(b)(2)(ix) of 10 CFR Part 50, the Inservice inspection of Class MC metal containments at Units 1 and 2 of the Catawba Nuclear Station will be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Division I, 2007 Edition with the 2008 Addenda (hereafter referred to as Section XI). The examinations will be performed to the extent practicable within the limitations of design, geometry and materials of construction of the component. Examinations were scheduled for the Third Inspection Interval in accordance with ASME Section XI IWE-2411.

ASME Boiler and Pressure Code Section XI Code Cases Used

- N-532-5, "Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000, Section XI, Division 1".

- N-765, "Alternative to Inspection Interval Scheduling Requirements of IWA-2430, Section XI, Division 1.

Component/System Boundaries Subject to Inspection and Examination

The boundaries of Class MC non-exempt components and their supports are shown on drawings listed in Section 2 of the Unit Specific Plan. The Class MC Pressure Test boundaries are not specifically identified on these drawings. Class MC Pressure Testing is performed in accordance with listed station procedures.

Note: Boundaries of piping penetration subassemblies (bellows) are also shown on drawings listed in Section 2 of the Unit Specific Plan. These boundaries include component parts classified as ASME Class NC and are indicated solely for convenience in performing examinations specified by the Owner and do not constitute the boundary for the Class MC containment vessel.

Components Exempted from Examination

Vessels, parts, and appurtenances that are outside the boundaries of the containment as defined in the Design Specifications. Mechanical process piping penetrations and subassemblies classified as ASME Code Class 2 (NC), including bellows assemblies (except surfaces of connecting welds to penetration sleeves) are exempt. However, examination of some piping penetration subassembly surfaces is included as Owner Specified Examinations, as indicated in 5.0 of this Plan. Also exempted from examination are flanges and covers attached to exterior ends of containment penetration sleeves for those penetrations where flanges and covers are attached to the interior end of the penetration sleeve.

Embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original Construction Code. Inaccessible portions of containment vessels, parts, and appurtenances are those with surfaces that cannot be visually examined by direct or remote methods on both sides, and which are embedded or otherwise obstructed from view by structures, components, or permanent plant equipment or materials.

Portions of containment vessels, parts, and appurtenances that become embedded or inaccessible as a result of vessel repair or replacement if the conditions of IWE-1232 and IWE-5220 are met.

Piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components shall be examined in accordance with the rules of IWB or IWC, as appropriate to the classification defined by the Design Specifications.

Pressure Testing Class MC Components

Except as noted in IWE-5224, a pneumatic leakage test shall be performed following repair/replacement activities performed by welding or brazing, prior to returning the component to service.

There are no periodic system pressure testing requirements for Class MC Components in Subsection IWE.

Examination Boundaries

Section 2 of the Unit Specific Plan contains a listing of drawings that identify examination areas subject to IWE examination. These drawings also show Class MC component supports subject to examination, and include portions of Class NC bellows surfaces which the Owner has elected to include in this program. Revisions to drawings are reviewed for additions/changes to the ISI boundaries. These additions/changes are

incorporated into the ISI Plan as necessary. The controlled drawings are maintained in accordance with applicable procedures and directives.

Inspection Interval and Inspection Periods

The third Inservice Inspection Intervals for Class MC metal containments and their supports are shown below. Please note that these intervals may not coincide with Inservice Inspection Intervals for ASME Class 1, 2, and 3 systems and components. The term "EOC" used below is an abbreviation for "End of Cycle" and the associated number indicates the sequential RFO following initial operation of the unit. The schedules for previous Inservice Inspection Intervals for Class MC metal containments and their supports are documented in File #CN-1042-CISI-0001, "First Interval Containment Inservice Inspection Plan" and File #CN-ISIC2-1042-0001, "Second Interval Containment Inservice Inspection Plan." Interval 3 Containment plan start/end dates are set at the maximum 12 month allowed in IWA-2430 and modified by RR #03-G0-010. Therefore, Interval end dates can be shortened but not extended.

Table 3.2.2-1, Second Containment Inservice Inspection Interval

Unit 1 (See Note 1)			
Start Date			**End Date**
07/15/2005	**07/15/2008**	**07/15/2011**	**07/14/2015**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 16)	Outage 4 (EOC 18)	Outage 6 (EOC 20)	
Outage 2 (EOC 17)	Outage 5 (EOC 19)	Outage 7 (EOC 21)	

Note 1:The schedule for Interval 2 is included in this discussion as it encompasses the previous Unit 1 ILRT.

Table 3.2.2-2, Third Containment Inservice Inspection Interval

Unit 1 (See Note 2)			
Start Date			**End Date**
07/15/2015	**07/15/2018**	**07/15/2022**	**07/14/2025**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 22)	Outage 3 (EOC 24)	Outage 6 (EOC 27)	
Outage 2 (EOC 23)	Outage 4 (EOC 25)	Outage 7 (EOC 28)	
	Outage 5 (EOC 26)		

Note 2: The start date for Interval 2 was delayed from 07/15/04 to 07/15/05, within the 12-month adjustment allowed by IWA-2430(d)(I) of the 1998 Edition with the 2000 Addenda. This adjustment was necessary because of the delay in obtaining NRC approval of Relief Request Serial #03-G0-010.

Table 3.2.2-3, Fourth Containment Inservice Inspection Interval			
	Unit 1 (See Note 3)		
Start Date			**End Date**
07/15/2025	**07/15/2028**	**07/15/2032**	**07/14/2035**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 29)	Outage 4 (EOC 31)	Outage 6 (EOC 33)	
Outage 2 (EOC 30)	Outage 5 (EOC 32)	Outage 7 (EOC 34)	

Note 3: The dates and outages for the Fourth Interval are projections and are tentative and subject to change as the Fourth Interval plan has yet to be developed.

Table 3.2.2-4, Second Containment Inservice Inspection Interval			
	Unit 2 (See Note 1)		
Start Date			**End Date**
07/15/2005	**07/15/2008**	**07/15/2011**	**07/14/2015**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 14)	Outage 3 (EOC 16)	Outage 6 (EOC 18)	
Outage 2 (EOC 15)	Outage 4 (EOC 17)	Outage 7 (EOC 19)	
		Outage 7 (EOC 20)	

Note 1:The schedule for Interval 2 is included in this discussion as it encompasses the previous Unit 2 ILRT.

Table 3.2.2-5, Third Containment Inservice Inspection Interval			
	Unit 2 (See Note 2)		
Start Date			**End Date**
07/15/2015	**07/15/2018**	**07/15/2022**	**07/14/2025**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 21)	Outage 3 (EOC 23)	Outage 6 (EOC 25)	
Outage 2 (EOC 22)	Outage 4 (EOC 24)	Outage 7 (EOC 26)	

Note 2: The start date for Interval 2 was delayed from 07/15/04 to 07/15/05, within the 12-month adjustment allowed by IW A-2430(d)(J) of the 1998 Edition with the 2000 Addenda. This adjustment was necessary because of the delay in obtaining NRC approval of Relief Request Serial #03-G0-010.

Table 3.2.2-6, Fourth Containment Inservice Inspection Interval			
	Unit 2 (See Note 3)		
Start Date			**End Date**
07/15/2025	**07/15/2028**	**07/15/2032**	**07/14/2035**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 27)	Outage 3 (EOC 29)	Outage 6 (EOC 32)	
Outage 2 (EOC 28)	Outage 4 (EOC 30)	Outage 7 (EOC 33)	
	Outage 5 (EOC 31)		

Note 3: The dates and outages for the Fourth Interval are projections and are tentative and subject to change as the Fourth Interval plan has yet to be developed.

Examination Categories and Requirements

The examination categories to be used are those listed in Table IWE-2500-1 of Section XI. Class MC Items to be inspected include the following:

Table 3.2.2-7, Category E-A: Containment Surfaces		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E1.10	Containment Vessel – Pressure Retaining Boundary	
E1.11	Accessible Surface Areas	General Visual, 100% each Inspection Period (See Note 1)
E1.12	Wetted Surfaces of Submerged Areas	(See Note 2)
E1.20	BWR Vent System-Accessible Surface Areas	(See Note 2)
E1.30	Moisture Barriers	General Visual, 100% each Inspection Period

Notes:

1. If this examination is to be credited towards satisfying the examinations required by 10CFR50, Appendix J, the examination shall be performed during the RFO in which a Type A test is to be performed, just prior to the start of the Type A test. Duke Energy intends to credit Item E1.11 visual exams towards satisfying the requirements of 10CFR50, Appendix J.

2. These examinations are applicable only to BWR containments and are not applicable at CNS.

Table 3.2.2-8, Category E-C: Containment Surfaces Requiring Augmented Examination		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E4.10	Containment Surface areas	
E4.11	Visible Surfaces	VT-1 Visual, 100% Each Period (Deferral Not Permissible)
E4.12	Surface Area Grid, Minimum Wall Thickness Location	Volumetric, Ultrasonic Thickness Measurement 100% Each Inspection Period (Deferral Not Permissible)

Table 3.2.2-9, Category E-G: Pressure Retaining Bolting		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E8.10	Bolted Connections	Visual VT-1, 100% End of Interval Deferral Permissible

Table 3.2.2-10, Category F-A: Supports		
Table IWF-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
F1.40	Supports Other Than Piping Supports (Class MC Airlock Supports)	Visual VT-3, 100% End of Interval (See Note 1)

Note 1: Although 10CFR50.55a does not include requirements for inservice or preservice examination of Class MC component supports; these supports shall be included within the Containment ISI Plan and shall be examined in accordance with Subsection IWF of the Code.

Owner Specified Examination Requirements

Owner specified examination requirements shall be performed as specified in this Plan. Specific examination listings and schedules are described in Part A, Section 6 of this Plan. Please note that the term "Augmented Examinations" is not used in this plan to describe examinations that are above and beyond those required by the Code. An alternative term "Owner Specified Examinations" is used to alleviate confusion with Subsection IWE, Category E-C Augmented Examinations. Owner specified examinations may include those which are the result of regulatory commitments, those required solely by regulation, and may include other examinations deemed appropriate by the Owner for inclusion in this program. Unless governed by regulatory requirements or commitments, the Owner reserves the right to remove items subject to "Owner Specified Examinations" from this inspection plan, and also reserves the right to add or modify examination requirements as desired.

Listing of Owner-Specified Examinations:

1. VT-3 Visual Examination of Fuel Transfer Tube penetration surfaces (including accessible surfaces of the Fuel Transfer Tube Penetration) on the exterior of the containment (Annulus side). These surfaces are not readily accessible because of lead shielding and locked access ports. These areas are posted as Very High Radiation (Grave Danger); however, these conditions exist only during fuel movement. These locations are identified because they are not routinely accessed for general visual examination in accordance with the Containment ISI Program. A review of similar industry practices confirmed that many licensees consider access to these types of areas to be sufficiently difficult as to justify exemption from the General Visual Examination requirements of the ASME Code, Subsection IWE. The operating experience at CNS has not identified any specific concerns with these areas that would warrant changing the Containment ISI Program to require General Visual Examination of these areas. However, it is deemed appropriate to require that these surfaces receive a VT-3 Visual Examination once during each ten-year interval.

2. Ultrasonic Thickness Measurements on selected surfaces opposite the ice condenser areas once every ten-year interval. The number and extent of these examinations shall be determined by Engineering (SXI). At a minimum, some examinations should be performed every ten years opposite the Ice Condenser Floor Slab/containment vessel interface. Additional UT examinations may be warranted at locations near the Ice Condenser Top Deck Doors where condensation has been occasionally observed on the Annulus side of the vessel during scheduled examinations. If conditions are detected during the performance of these examinations, a determination shall be made as to whether the conditions warrant examination of the affected surfaces under Category E-C, Item E4.12.

3. Additional Owner-Specified Examinations are described below.

Owner-Specified Examination Categories and Requirements

Owner-specified examinations are described in the following table.

Table 3.2.2-11, Owner Specified Examinations			
Item Subject to Owner Specified Examination	**Part to Be Examined**	**Examination Requirement(s)**	**Requirement / Comments**
E1.11	Accessible Surfaces of Process Piping Penetration Assemblies Classified as ASME Code Class NC as shown on in service inspection drawings. (See Note 1)	General Visual Examination Each Period (See Note 2)	None. These surfaces do not require examination in accordance with IWC or IWE. However, these surfaces form a part of the containment primary pressure boundary.
E4.12	Surface Area Grid, Minimum Wall Thickness Location	Volumetric (See Note 3)	None. (See Note 3)
F1.40	Class MC Component Supports	Visual, VT -3	None. However, Airlock supports at Catawba shall be treated as NF supports and shall be examined in accordance with IWF requirements at the Owner's discretion.
E1.30	Accessible Surfaces of 100% of Containment Leak Chase Channel Closures	General Visual Examination Each Inspection Period	See Note 4.

Notes:

1. Penetration subassemblies (bellows) are classified as Code Class NC at their welded connection to the first outboard circumferential weld on the containment vessel sleeve. However, the Containment ISI Plan drawings indicate that pressure retaining penetration subassembly surfaces are included within the scope of IWE Examination from the containment sleeve circumferential weld to

the bellows assembly seal weld at the flued head connection. These are Owner specified examinations and need not comply with applicable provisions of IWE or 10CFR50.55a(b)(2) and I0CFR50.55a(g)(4).

2. Visual Examination shall be performed on accessible surfaces of areas shown on the Containment ISI drawings, except that surfaces of subassemblies extending outside of the Reactor Shield Building need not be examined during general visual examinations specified for Item E1.11.

3. IWE-2500, Table IWE-2500-1, Category E-C, Item E4.12 examinations are performed to determine the minimum wall thickness location within each grid. The examinations specified herein as "Owner Specified" are in addition to those required by the Code and are performed within each grid subject to E4.12 examination when the minimum wall thickness measurement initially recorded lies within 4" of the centerline of any vertical or circumferential weld. These Owner specified examinations are not required by the Code, but have been added by the Owner to ensure that the minimum wall thickness measurements within each grid are not influenced by local plate thinning that typically can occur adjacent to welds. These examinations have been added to be consistent with procedure NDE-951, "Ultrasonic Thickness Measurement of Metallic Containment Structures".

4. A VT-3 visual examination in accordance with NDE-67, "Visual Examination (VT-1 and VT-3) of Metal and Concrete Containment," shall be specified to satisfy the General Visual Examination.

Additional Program Requirements

Additional programmatic requirements specified by 10CFR50.55a(b)(2)(ix) are described below:

10CFR50.55a(b)(2)(ix) (A) *Metal containment examinations: First provision.*
For Class MC applications, the following apply to inaccessible areas.

(1) The applicant or licensee must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas.

(2) For each inaccessible area identified for evaluation, the applicant or licensee must provide the following in the ISI Summary Report as required by IW A-6000:

    (i) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

    (ii) An evaluation of each area, and the result of the evaluation; and

    (iii) A description of necessary corrective actions.

10CFR50.55a(b)(2)(ix)(B) *Metal containment examinations: Second provision.*

When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

ASME Code 2007 Edition with 2008 Addenda has no Table IWA-2210-1. Duke Energy has elected to comply with the requirements of ASME Code 2007 Edition with 2008 Addenda Subsection IWA for performance of VT-1 and VT-3 visual examinations on metal containment.

Accessible Surface Areas

The provisions of IWE-1231(a)(3) shall be met. This paragraph describes how the total accessible surface area is to be determined to meet this provision.

The total surface area of the containment (in square feet) was computed for the First Interval Containment ISI Plan, File #CN-1042-CISI-0001. Based on the data and computations documented during ISI Interval I, it is clear that the requirements of IWE-1231(a) have been met during Interval and the examination surface areas were the same during Interval 2. Because the examination surface areas are the same during Interval 3, these computations need not be repeated in the Third Interval ISI Plan. It is also clear that, unless major modifications are made to the Containment Vessel, the minimum required areas will remain accessible for visual examination from at least one side of the vessel.

Inaccessible Surface Area

The inaccessible surface area is equal to the total surface area obstructed within the areas identified above such that direct or remote visual examination of that area is not possible from both the inside and outside surface. An area need not be considered inaccessible if access permits examination from at least one side of the vessel. However, the location and extent of all surface areas where access is not possible from one side shall be documented. The First Interval Containment ISI Plan did not identity a significant number of inaccessible surface areas. If additional inaccessible areas are identified during the third ISI Interval, the ISI Plan shall be updated to document this data and shall be revised as needed to demonstrate continued compliance with IWE-1231(a)(3). Data on the location and extent of inaccessible surface areas shall be of sufficient detail as to allow these areas to be identified and located on applicable inservice inspection drawings referenced in this Plan.

Requests for Relief from ASME Code and Regulatory Requirements

Currently there are no relief requests for the third interval plan.

3.2.3 License Renewal – IWE

The Containment Inservice Inspection Plan - IWE is credited in the joint McGuire and Catawba License Renewal Application (LRA) with managing loss of material due to corrosion of steel surfaces. The Containment Inservice Inspection Plan - IWE

implements the requirements of ASME Code Section XI Subsections IWE. The purpose of ASME Subsection IWE examination is to identify and correct degradation of the accessible steel surfaces of the containment liner prior to the loss of the essentially leak tight barrier.

Section 3.5 of the LRA identified the Containment ISI Plan - IWE as an aging management program for Reactor Building containment steel components. The discussion of the program and objective evidence associated with the effectiveness of the program were provided in LRA Appendix B.3.7 of the McGuire and Catawba application. More details for the basis and the determination of the adequacy of the program are documented in specification DPS-1274.00-00-0005, License Renewal Aging Management Programs and Activities.

3.2.4 Integrated Leakage Rate Testing (ILRT) History

Previous Type A tests confirmed that the CNS reactor containment structure has leakage well under acceptance limits and represents minimal risk to increased leakage. Continued Type B and Type C testing for direct communication with containment atmosphere minimize this risk. Also, the Inservice Inspection (IWE/IWL) program and maintenance rule monitoring provide confidence in containment integrity.

To date, five (5) operational Type A tests have been performed on CNS Unit 1 and four (4) on CNS Unit 2. There is considerable margin between these Type A test results and the TS 5.5.2 limit of 0.75 La (0.225 % Weight per Day), where La is equal to 0.3% by weight of the containment air per day at the peak accident pressure. These test results demonstrate that CNS has low leakage Containments.

Table 3.2.3-1, CNS Unit 1 ILRT Test Results

Test Date	As-Found Test Results (% Weight per Day)	As-Left Test Results (% Weight per Day)
June 2014	0.057135	0.060505
November 2000	0.0965	0.0965
March 1991	0.06745	0.06745
November 1987	0.06141	0.05216
January 1984	N/A	0.1115

Table 3.2.3-2, CNS Unit 2 ILRT Test Results

Test Date	As-Found Test Results (% Weight per Day)	As-Left Test Results (% Weight per Day)
November 2007	0.127999	0.127999
February 1993	0.1461	0.1461
March 1989	0.0243	0.0243
July 1985	N/A	0.1259

3.2.5 License Renewal - ILRT

The Containment Leak Rate Testing Program is credited in the McGuire and Catawba LRA with managing loss of material of steel components of the Reactor Building Containment and cracking of penetration bellows. The purpose of the Containment Leak Rate Testing Program is to assure that leakage through the containment and systems and components penetrating containment shall not exceed allowable leakage rate values specified in the Technical Specifications or associated bases and periodic surveillances of containment penetrations and isolation valves are performed.

Section 3.5 of the LRA identified the Containment Leak Rate Testing Program as an aging management program for steel components of the Reactor Building Containment. The discussion of the program and objective evidence associated with the effectiveness of the program were provided in LRA Appendix B.3.8 of the McGuire and Catawba application. More details for the basis and the determination of the adequacy of the program are documented in specification DPS-1274.00-00-0005, License Renewal Aging Management Programs and Activities.

3.3 Containment Leakage Rate Testing Program, Type B and Type C Testing

CNS Type B and C testing program currently requires testing of electrical penetrations, airlocks, hatches, flanges, bellows and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per TS 5.5.2, the allowable maximum pathway total for Type B and C leakage is 0.6 La where 0.6 La equals 82,979 sccm.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential Containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results from 2005 through 2014 for CNS Unit 1 and from 2006 through 2015 for CNS Unit 2 has shown an exceptional amount of margin between the actual As-Found (AF) and As-left (AL) outage summations and the regulatory requirements as described below:

- The As-Found minimum pathway leak rate average for CNS Unit 1 shows an average of 12.2% of 0.6 La with a high of 15.51% or 0.093 La.

- The As-Left maximum pathway leak rate average for CNS Unit 1 shows an average of 13.14% of 0.6 La with a high of 15.94% or 0.096 La.

- The As-Found minimum pathway leak rate average for CNS Unit 2 shows an average of 14.04% of 0.6 La with a high of 19.7% or 0.118 La.

- The As-Left maximum pathway leak rate average for CNS Unit 2 shows an average of 10.66% of 0.6 La with a high of 13.83% or 0.083 La.

Tables 3.3-1 and 3.3-2 provide LLRT data trend summaries for CNS since 2005 for Unit 1 and 2006 for Unit 2 and encompasses both previous ILRTs. This summary shows that there have been no As-Found failures that resulted in exceeding the Technical Specification 5.5.2 limit of 0.6 La (82,979 sccm) and demonstrates a history of successful tests.

This following demonstrate a history of satisfactory Type B and Type C tested component performance.

Table 3.3-1, CNS Unit 1 Type B and C LLRT Trend Summary

RFO	1EOC15 2005 (sccm)	1EOC16 2006 (sccm)	1EOC17 2008 (sccm)	1EOC18 2009 (sccm)	1EOC19 2011 (sccm)	1EOC20 2012 (sccm)	1EOC21 2014 (sccm)
As-Found Minimum Pathway	11708.7	11698.7	10604.1	12866.7	9728.9	9068.5	5360.7
Fraction of 0.6 La	0.141	0.141	0.128	0.155	0.117	0.109	0.065
As-Left Maximum Pathway	11682.2	11668.2	10415.2	12405.1	13231.6	8885.2	8030.9
Fraction of 0.6 La	0.141	0.141	0.126	0.149	0.159	0.107	0.097

Table 3.3-2, CNS Unit 2 Type B and C LLRT Trend Summary

RFO	2EOC14 2006 (sccm)	2EOC15 2007 (sccm)	2EOC16 2009 (sccm)	2EOC17 2010 (sccm)	2EOC18 2012 (sccm)	2EOC19 2013 (sccm)	2EOC20 2015 (sccm)
As-Found Minimum Pathway	10345.6	13715.0	12626.4	13087.5	16349.9	6290.1	9103.4
Fraction of 0.6 La	0.125	0.165	0.152	0.158	0.197	0.076	0.110
As-Left Maximum Pathway	8996.6	9981.4	10121.3	8952.7	6250.5	6114.9	11479.6
Fraction of 0.6 La	0.108	0.120	0.122	0.108	0.075	0.074	0.138

Tables 3.3-3 and 3.3-4 identify the components that have not demonstrated acceptable performance during the previous two outages for CNS Units 1 and 2:

Table 3.3-3, CNS Unit 1 Type B and C LLRT Program Implementation Review						
1EOC20 - 2012						
Component	**As-Found (sccm)**	**Admin Limit (sccm)**	**As-Left (sccm)**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval (Months)**
Penetration 1C100 1IAECK5370	1351	450	54	Foreign material (1)	Valve rebuilt (1)	30
Penetration M-235 1NM424	51,000	500	52.8	Foreign material (2)	Valve replaced (2)	30
1EOC21 - 2014						
Component	**As-Found (sccm)**	**Admin Limit (sccm)**	**As-Left (sccm)**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval (Months)**
None						

1. During 1EOC20, 1IAECK5370 failed the Type C Leak Rate Test per PT/1/A/4200/001 I. The leakage was recorded at 1351 sccm, and only exceeded the administrative leak limit of 450 sccm.

   Engineering inspected all check valve internals. The hinge pin and O-rings did not show signs of degradation or wear, nor was the soft seat degraded. However, new soft seats were installed in the valve. The valve was rebuilt and then bench tested and passed the reverse leakage test at 0 sccm and forward flow cracking pressure at 0.2 psig.

2. Valves 1NM6A and 1NM424 failed an As-Found Leak Rate Test (LRT) during the 2012 Catawba Unit 1 RFO. Leakage was determined to be approximately 51,000 sccm (Minimum pathway leakage was 10 sccm). Valve 1NM424 is the pressurizer sample line bypass check valve, while valve 1NM6A is the pressurizer steam sample line containment isolation valve.

   The cause is FME particulate being blown into the valves during the testing process. This FME prevents these valves from completely closing which has led to the high leakage rates. There have been repeated failures during Leak Rate Testing of these valves. During a previous RFO in May 2008, penetration M235 (1NM6A/1NM424) failed its as-found Type-C LRT with a measure leak rate of 19,841 sccm. In that instance, valve 1NM424 was replaced. During another RFO in May 2011, the measured leakage across this penetration was 46,700 sccm, once again failing to meet the acceptance criteria of the test procedure.

   Valve 1NM-424 has been replaced as a direct result of this event. Further, a modification has been completed to install a vent valve upstream of the check valve to preclude particulate from being blown into the check valve during testing. Installation of vent valve 1NM980 on December 12, 2012 removed the failure mechanism of this event. This modification was a corrective action coming from this valve's previous failure and will prevent recurrence of the issue during testing.

Table 3.3-4, CNS Unit 2 Type B and C LLRT Program Implementation Review						
2EOC19 - 2013						
Component	**As-Found (sccm)**	**Admin Limit (sccm)**	**As-Left (sccm)**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval (Months)**
Penetration M-374 Bellows	3.3	1	0	(1)	(1)	Each Refueling
Penetration M-323 2KC-47	35,328	450	15.6	Foreign material (2)	Penetration flushed, valve rebuilt (2)	30
Penetration M-316 2RF-392	100,000	2400	0	Foreign material (3)	Valve replaced (3)	30
2EOC20 - 2015						
Component	**As-Found (sccm)**	**Admin Limit (sccm)**	**As-Left (sccm)**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval (Months)**
Penetration M-307 2NI-495	73,281	300	0	Seat leakage	Replaced soft seat (4)	30
Penetration M-235 2NM-235	32,150	500	21.3	Foreign material (5)	Valve replaced (5)	30

1. Unit 2 Mechanical Penetration M-374 (WL SYSTEM / CONT. FLOOR SUMP & INCORE INST. SUMP PUMP DISCHG) failed during performance of periodic surveillance PT/2/A/4200/001 G Mechanical Penetration Bellows Integrity Test. During this periodic test, a low pressure (4 psig) is applied between the dual-ply bellows assembly to verify the structural integrity of this mechanical bellows piping penetration. The test acceptance criteria is < or = 1 sccm. During two consecutive tests, this acceptance criteria was NOT met with measured leak rates at 3.3 and 3.4 sccm.

   With confirmation of unacceptable leakage, the unit 2 containment penetration M-374 was next tested at the maximum expected accident pressure (Pa = 15 psig) for the postulated design basis accident (DBA/LOCA). For this third test, a MIRAPS assembly was installed to close and seal the opening on the containment side of the mechanical bellows. With this opening closed and sealed, the test pressure was applied to the containment penetration through the MIRAPS assembly to simulate the postulate accident pressure. The test connection on the outer shroud of the bellows assembly was open (uncapped); as such, the simulated post-accident pressure was applied only to the inner ply of this dual-ply mechanical bellows assembly. During this third test, the unit 2 mechanical penetration M-374 had no measured leakage (0 sccm).

   Upon inspection, the inner surface of this dual-ply bellows had an area of rust buildup at the inner-most connection at the mechanical bellow (SS) and the assembly end cap (CS) weld. The accident pressure (15 psig) test applies pressure to the containment side of the inner bellows and could effectively close

any leakage path that may exist at this rust build-up area. Future accident pressure (15 psig) tests will be performed consistent with the requirement of 10CFR50 Appendix J. These future tests will be effective at monitoring the performance (leak rate) of this rust buildup on this mechanical bellows assembly.

2. While performing a Type C containment penetration leak rate test of 2KC-47 (containment penetration M323) per PT/2/A/4200/001 I, leakage of 35,328 standard cubic centimeters per minute (sccm) was identified (Minimum pathway leakage was 10 sccm). Acceptable leakage for 2KC-47 is less than or equal to 450 sccm.

   On 10/7/2013, valve 2KC-47 was disassembled and inspected per WO #2120620. Upon disassembly of the valve, Maintenance noticed a black gritty substance settled on the bottom of valve body and around the disc seating area. There were also signs of rust inside piping and on portions of the valve disc assembly and seating area.

   On 10/11/13 the containment penetration piping was flushed using demineralized water to the "A" Containment Floor and Equipment Sump.

   After performing the flush, Maintenance performed inspections of the piping and valve 2KC-47 internals with a borescope to verify the piping and valve were free of any system particulate. Valve 2KC-47 was then re-assembled with a new disc assembly, spring and soft seat. On October 11, 2013, a retest was completed with successful results for the Type C leak rate test and the forward flow test. The As-Left leak rate test results were documented as 15.6 sccm. The rust that was identified on the valve seating area was determined to be the cause of the Type C LRT failure. The rust is believed to be from iron particulate from the KC System piping.

3. During as-found Appendix-J Type-C LLRT (Tech Spec 5.5.2), it appears that check valve (CIV) 2RF-392 experienced "excessive" seat leakage. The test boundary would not fully pressurize (achieved 9.9 psig vs. required 15 psig). The recorded value for measured leak rate (53,700 sccm) does NOT represent the actual leak rate at the required test pressure. Therefore, the leak rate monitor (LRM) maximum indication range will be assumed as the actual 2RF-392 leak rate, which is 100,000 sccm (Minimum pathway leakage was 0 sccm).

   CIV 2RF-392 is the inboard boundary of containment penetration 2M-316. The outboard boundary, CIV 2RF-389B is a gate valve that is sealed closed by injection water from the NW System. This outboard CIV successfully passed its NW Leak Rate Test during the 2EOC19 RFO (COMPLETE: September 26, 2013).

   The original 2RF-392 was a 4-inch swing check valve (Pacific Valve). The disc/hinge assembly was body hung, with hinge pins plugs on each side of the valve body. This valve had a soft seat O-ring installed in the valve body seat. The body seat face was perpendicular to its horizontal piping; as such, this valve design was dependent on system pressure to provide most of its disc-to-seat closing force. The disc/hinge assembly extends below the bottom of the piping,

which makes the seating surface susceptible to leakage due to system particulate.

During 2EOC19 RFO, 2RF-392 was replaced with a flanged wafer 4-inch swing check valve. This new valve is considered less likely to experience seat leakage due to system rust / particulate because the disc assembly is elevated above the bottom of the pipe and the design includes a spring (~ 5 psig) to assist with valve closure. Installing the new valve also replaces the aged elastomeric seat of the old valve, which provides additional assurance of improved performance and leak tight closure.

4. While performing a Type C containment penetration leak rate test of 2NI-495 (containment penetration M307) per PT/2/A/4200/001 I, 2NI-495 would not pressurize. Volumetrics pressure was increased and at 15.4 psig leakage was quantified at 73,281 sccm (minimum pathway leakage was 10 sccm). Acceptance criteria on 2NI-495 is less than or equal to 300 sccm.

   Valve was replaced per WO #02154787 in a configuration providing slight positive slope to assist in valve closure.

5. During Leak rate test of penetration M-235 (PT/2/A/4200/001I Encl. 3.12), 2NM-424 and 2NM-6A are leak rate tested together in section 3. The initial measured leak rate for these valves in step 3.1 was 0 SCCM. Step 3.2 performs forward flow testing of 2NM-424. The expected cracking pressure of 2NM-424 is 200 psid and the procedure allows applying pressure up to approximately 400 psig to crack the valve. The actual initial cracking pressure was 360 psig and per the PT the steps were repeated to confirm this pressure. The second performance of these steps yielded a cracking pressure of 60 psig which would indicate that 2NM-424 was not reseating properly. The next section of the procedure test 2NM-424 and 2NM-6A together as a final as found test. This test yielded a leakrate of 32,150 SCCM which indicates that 2NM-424 is leaking. The acceptable leak rate for 2NM-424 and 2NM-6A combined is 500 SCCM. (Minimum pathway leakage was 10 sccm).

   2NM-424 had a previous step change in performance. This valve was replaced during the 2EOC20 RFO.

   During the 2NM-424 cutout and replacement, boroscope inspections identified foreign material (appearing to be metal shavings) within the small bore piping on the containment penetration (upstream) side of check valve 2NM-424. Therefore, foreign material was present within this adjacent piping prior to this cutout and during the 2NM-424 Appendix J LLRT but was subsequently removed during valve replacement.

3.4 NRC Information Notices (INs)

3.4.1 IN 2010-12, "Containment Liner Corrosion"

This IN provides examples of containment liner degradation caused by corrosion. Concrete reactor containments are typically lined with a carbon steel liner to ensure a

high degree of leak tightness during operating and accident conditions. The reactor containment is required to be operable as specified in plant technical specifications to limit the leakage of fission product radioactivity from the containment to the environment. The regulations at 10 CFR 50.55a, "Codes and Standards," require the use of Subsection IWE of ASME Section XI to perform inservice inspections of containment components. The required inservice inspections include periodic visual examinations and limited volumetric examinations using ultrasonic thickness measurements. The containment components include the steel containment liner and integral attachments for the concrete containment, containment personnel airlock and equipment hatch, penetration sleeves, moisture barriers, and pressure-retaining bolting. The NRC also requires licensees to perform leak rate testing of the containment pressure-retaining components and isolation valves according to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as specified in plant technical specifications. This operating experience highlights the importance of good quality assurance, housekeeping and high quality construction practices during construction operations in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Corrosion to the containment liner is not a new industry issue. Programs and procedures are in-place to inspect the containment liner and would identify any areas subject to corrosion.

Containment Liner Corrosion Operating Experience Summary, Technical Letter Report Revision 1, dated August 2, 2011, Section 2.4, "Previous Assessments and Containment Operating Experience" provided the following descriptions of previously identified containment liner and moisture barrier degradation for CNS:

> Section 2.4.2, Corrosion with Coating or Moisture Barrier Degradation in PWRs
>
> In September 1989, Catawba Units 1 and 2 were found to have coating damage and base metal corrosion on the outer surfaces of the steel shells at the intersection of the shell and the concrete annulus floor. The damage was limited to a circumference of 4.6 m [15 ft.], a height of 2.5 cm [1 in] above the annulus floor and an average depth of 7.6 mm [300 mils]. The cause was believed to be attack by boric acid coolant that had leaked from instrument line compression fittings and condensed, and collected on the annulus floor (Ashar and Bagchi, 1995).

As a follow up to a McGuire steel containment vessel (SCV) inspection, a similar SCV inspection was performed at CNS by Design Engineering Personnel. Coating failures on the exterior face of the SCV above elevation 552' +0" were noted. These failures had allowed base metal corrosion. Corrective actions were implemented during the spring of 1990 for CNS Unit 1 and the fall of 1990 for CNS Unit 2 to correct the deficiencies.

All identified failed coatings and moisture barrier were removed, the base metal was cleaned and inspected, coatings were reapplied, concrete interface repaired and the removed moisture barrier caulking was replaced.

3.4.2 IN 2014-07, "Degradation of Leak Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner"

This IN was issued to inform addressees of issues identified by the NRC staff concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures.

IN 2014-07 described the leak chase channel system as follows:

> Consists of steel channel sections that are fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection. Each test connection consists of a small carbon or stainless steel tube (less than 1-inch (2.5 centimeters) diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel.

IN 2014-07 describes a recessed box with a cover plate at floor level that allows for water to pool inside the recessed box and cause degradation. The CNS system is the same as the cited systems.

NRC Information Notice 2014-07 addresses NRC expectations for examination of containment leak chase channel systems. Until the NRC mandates these examinations through conditions imposed by 10 CFR 50.55a, Duke Energy reserves the right to include these in the Containment ISI Plan as Elective Examinations. ASME Interpretation #XI-1-13-10 clarified that the ASME Code, Section XI does not require examination of these items in accordance with Subsection IWE, Category E-A, Item E1.30. Because of confusion regarding the legal imposition of examination requirements for these items, it is appropriate for Duke Energy to voluntarily add examination of these items in the Containment ISI Plan. If the NRC imposes any new 10 CFR 50.55a(b)(2) condition mandating these examinations, Duke Energy shall take action as needed to comply. For now, these items shall be added to the Containment ISI Plan as Elective Examinations to be performed 100% during each Inspection Interval (instead of the 100% each Inspection Period as indicated in IN 2014-07). Duke Energy shall elect to perform a VT-3 visual examination in accordance with procedure "Visual Examination (VT-1 and VT-3) of Metal and Concrete Containment." These examinations may be scheduled and performed as follows:

100% of the containment interior concrete floors shall be examined during each inspection interval. Approximately 1/3 of the floor surface areas shall be examined during each inspection period to determine the condition of all leak chase channel bronze caps and test channel drain plugs installed in the floor within the examination area.

Bronze cap and test channel drain plug details are shown on Containment Inservice Inspection Drawings.

The condition of the bronze caps and drain plugs shall be considered acceptable if there is no evidence of damage or degradation that could result in possible leakage of water into the leak chase channel systems. Evidence of boric acid in the vicinity of any bronze cap or plug shall require evaluation by Engineering to determine the acceptability of the item.

The specific location of each bronze cap and drain plug need not be identified on the examination record. However, if any condition is detected on any bronze cap or drain plug that requires evaluation by engineering, the specific location of the bronze cap or drain plug shall be identified and documented on the examination record. For conditions identified on the Pipe Chase Floor, location information shall include azimuth, and radial distance from either the Steel Containment Vessel shell plate or Crane Wall exterior surface. For conditions identified inside of the Crane Wall, location information shall include azimuth, and radial distance from the Crane Wall interior surface.

3.4.3 IN 92-20, Inadequate Local Leak Rate Testing

NRC IN 92-20 was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

All bellows expansion joints are of two-ply construction with a wire mesh between plys for testability of bellows and bellows weld to piping.

All bellows on mechanical penetrations are subjected to a local structural integrity test by pressurizing the volume between the two ply bellows to 3-5 psig to verify no detectable leakage. Otherwise, the assembly must be tested with the containment side of the bellows assembly pressurized to conditions representing DBA containment pressure. The bellows test frequency and other acceptance criteria are specified in the Containment Leak Rate Testing Program.

During the performance of each Type A test, the test connections on all bellows are uncapped. This assures that the volume between the two ply bellows is vented to the annulus during performance of this test. At the completion of this test, all of the test connections are capped except for the main steam and feedwater penetration outer bellows test connections which remain uncapped and vented to the annulus.

The testing program for bellows on mechanical penetrations is the same as was previously reviewed and approved by the NRC for McGuire (NUREG-0422, Supplement 1, May 1978).

TS SR 3.6.1.1 contains two Notes addressing bellows assemblies as follows:

1. The space between each dual-ply bellows assembly on penetrations between the Containment building and annulus shall be vented to the annulus during Type A tests.

2. The space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3 to 5 psig to verify no detectable leakage, or the assembly shall be subjected to a leak test with the pressure on the containment side of the assembly at Pa.

The following containment penetration bellows assemblies are currently subject to testing with the test pressure on the containment side of the assembly at Pa:

Unit 1, M110 and M113.

Unit 2, M113, M262, M310, M332, M374 and M422.

The results of the most recent performance of the "Mechanical Penetration Bellows Integrity Test" are as follows:

Unit 1, Completed June 14, 2014
All bellows were tested with zero measured leakage with the exception of Cold Penetration M-321 Bellows which had a measured leak rate of 0.66 sccm, which was below the acceptance criteria of ≤ 1 sccm.

Unit 2, Completed March 21, 2015
All bellows were tested with zero measured leakage with the exception of 1) Cold Penetration M-204 Bellows which had a measured leak rate of 0.5, and 2 sccm) Hot Penetration M-273 Bellows which had a measured leak rate of 0.5 sccm. Both were below the acceptance criteria of ≤ 1 sccm.

3.4.4 Results of Recent Containment Inspections – Problem Identification Reports

Unit 1 1EOC12 – 2000 - First Interval

Indication: #1-SCVI-002.2000.1
Location: Steel Containment Vessel Interior - Vent Unit A/D Room

Description: Moisture barriers #1-MBRI-006 and #1-MBRI-032 (See Drawing CN-1042-ISI.1-002 and -003) installed along vertical joints at each end of this room are cracking and separating. The condition of these moisture barriers makes them ineffective. No moisture was noted in the vicinity of these moisture barriers.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-002.2000.1. This WR generated WO #01002133, which removed moisture barriers 1-MBRI-006 and 1-MBRI-032. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002133.

Indication: #1-SCVI-002.2000.2
Location: Steel Containment Vessel Interior - Vent Unit A/D Room

Description: Between azimuths 330° and 0°, nearly all of the cork has been removed from between the floor and the steel containment vessel. Some cork material remains affixed to the concrete floor side of this joint. The condition of the containment vessel shell is similar to that noted at other lower containment rooms. These conditions include damaged coatings, staining, and minor corrosion.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-002.2000.2. This WR generated WO #01002112, which removed the remaining cork material. Additionally under WO #01002112, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell after cork material was removed.

Indication: #1-SCVI-003.2000.1
Location: Steel Containment Vessel Interior - Accumulator A Room, Elevation 565'+3"

Description: Moisture barrier material and cork expansion joint material has been removed from all of the joint between the Steel Containment Vessel and the floor of this room. However, because the grout pad at the base of the Accumulator butts against the steel containment vessel, it is difficult to determine whether all of the cork expansion joint material has been removed beneath the grout pad. Exposed surfaces of the Containment Vessel are degraded, with coatings damage, staining and minor corrosion at various locations behind the floor (mostly at the floor elevation). These conditions are similar to those found in other lower containment rooms where the cork expansion joint material has been removed.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-003.2000.1. This WR generated WO #01002112, which removed the remaining cork material. Additionally under WO #01002112, corrective coatings maintenance was performed on the newly exposed and degraded surfaces of the containment vessel shell.

Indication: #1-SCVI-003.2000.2
Location: Steel Containment Vessel Interior - Accumulator A Room

Description: Moisture barriers #1-MBRI-007 and #1-MBRI-008 (Reference Drawing CN-1042-ISI.1-002) installed along vertical joints at each end of this room are cracking and separating. The condition of these moisture barriers makes them ineffective. No moisture was noted in the vicinity of these moisture barriers.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-003.2000.2. This WR generated WO #01002133, which removed moisture barriers 1-MBRI-007 and 1-MBRI-008. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002133.

Indication: #1-SCVI-006.2000.1
Location: Steel Containment Vessel Interior - Accumulator B Room, Elevation 565'+3"

Description: Approximately 10 feet of sealant and cork expansion joint material have been removed from the joint between the containment vessel shell and the floor of this room. Conditions on the containment shell are similar to those reported in other lower containment rooms where cork has been removed. These conditions include damaged coatings, staining, and minor corrosion.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-006.2000.1. This WR generated WO #01039787, which removed moisture barriers 1-MBRI-017 and 1-MBRI-018. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01039787.

Indication: #1-SCVI-007.2000.2
Location: Steel Containment Vessel Interior - Vent Unit B/C Room, Elevation 565'+3"

Description: Standing water was observed on floor behind both fan units. The source of water is condensation from YV (containment chilled water system) and RN (nuclear service water system) piping in this area. Portions of this piping are not painted and are corroding. These conditions were noted during the previous inspection and are unchanged.

This condition was documented in PIP #C-00-05438 (AR #01406096).

This issue is not directly associated with the Containment Vessel examination boundary for the General Visual Examination. Subsequent General Visual Examinations have found this area to be acceptable. The responsible system engineer is currently monitoring the condition. Additionally, this area was found acceptable during the most recent General Visual Examination (1EOC22 - Third Containment ISI Interval).

Indication: #1-SCVI-007.2000.3
Location: Steel Containment Vessel Interior - Vent Unit B/C Room, Elevation 565'+3"

Description: Moisture barrier material has been removed from between the Steel Containment Vessel and the floor of the Vent Unit B/C Room. This moisture barrier is identified as Component I.D. 1-MBRI-041 on drawings CN-1042-ISI.1-002 and -003. Cork expansion joint material is not completely removed from between the floor and the containment vessel shell in this room. Where it has been removed, conditions were observed that are similar to that reported in other lower containment rooms. These conditions include damaged coatings, staining, and minor corrosion.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-007.2000.3. This WR generated WO #01002112, which removed moisture barrier 1-MBRI-041.

WR #0909613 was written as a result of indication #1-SCVI-007.2000.3. This WR generated WO #01719816, which performed corrective coatings maintenance on the newly exposed and degraded surfaces of the containment vessel shell.

Indication: #1-SCVI-008.2000.2
Location: Steel Containment Vessel Interior - Accumulator C Room, Elevation 565'+3"

Description: Moisture barrier material and cork expansion joint material has been removed from nearly all of the joint between the Steel Containment Vessel and the floor of this room, except behind the Accumulator tank. Exposed surfaces of the Containment Vessel are degraded, with coatings damage, staining and minor corrosion at various locations behind the floor (mostly at the floor elevation). Some cork material remains in the floor joint, but is attached only to the concrete side of the 2" wide joint. Because of this, debris has become trappéd between the Containment Vessel and the cork material behind the floor near azimuth 217°.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-008.2000.2. This WR generated WO #01002112, which removed cork expansion joint material behind the 'C' Accumulator Tank.

WR #0909613 was written as a result of indication #1-SCVI-008.2000.2. This WR generated WO #01719816, which performed corrective coatings maintenance on the newly exposed and degraded surfaces of the containment vessel shell.

Indication: #1-SCVI-009.2000.1
Location: Steel Containment Vessel Interior - "Nothing Room", Elevation 565'+3"

Description: Moisture Barrier #1-MBRI-029 at azimuth 305° (Reference Drawing CN-1042-ISI.1-003) has been almost entirely removed. At the bottom of the wall, some cork expansion joint material has been removed from behind the wall for approximately 10 feet. Where cork material has been removed, some coatings damage and minor corrosion was noted. Some minor pitting is beginning to occur. Moisture was observed along this joint and is coming from areas in Upper Containment above this area.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-009.2000.1. This WR generated WO #01002133, which removed cork expansion joint material.

WR #0909613 was written as a result of indication #1-SCVI-009.2000.1. This WR generated WO #01719816, which performed corrective coatings maintenance on the newly exposed and degraded surfaces of the containment vessel shell.

Indication: #1-SCVI-009.2000.3
Location: Steel Containment Vessel Interior - "Nothing Room", Elevation 565'+3"

Description: Moisture barrier material and cork expansion joint material has been removed from between the Steel Containment Vessel and the floor of this room, between azimuths 270° and 303°. Exposed surfaces of the Containment Vessel are degraded, with coatings damage, staining and minor corrosion at various locations behind the floor (mostly at the floor elevation). Near azimuth 280°, cork removal work

has resulted in some dings and minor gouging of the Steel Containment Vessel. At this time, the corrosion is not considered significant enough to warrant further evaluation.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-009.2000.3. This WR generated WO #01002112, which removed cork expansion joint material between the Steel Containment Vessel and the floor of the "Nothing Room".

WR #0909613 was written as a result of indication #1-SCVI-009.2000.3. This WR generated WO #01719816, which performed corrective coatings maintenance on the newly exposed and degraded surfaces of the containment vessel shell.

Indication: #1-SCVI-009.2000.7
Location: Steel Containment Vessel Interior - "Nothing Room", Elevation 565'+3"

Description: Moisture barrier #1-MBRI-024 (Reference Drawing CN-1042-ISI.1-003) still exists between the floor and the containment vessel between azimuths 235° and 247°. Cork has not been removed from between this floor and the containment vessel in this vicinity.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-009.2000.7. This WR generated WO #01002112, which removed moisture barrier 1-MBRI-024 and cork expansion joint material. Additionally under WO #01002112, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell after cork material was removed.

Indication: #1-SCVI-009.2000.8
Location: Steel Containment Vessel Interior - "Nothing Room", Elevation 565'+3"

Description: Moisture barrier (sealant) #1-MBRI-023 (Reference Drawing CN-1042-ISI.1-003) at azimuth 235° is degrading. Portions of the sealant are missing at several locations along this joint, and the remaining material is cracking and separating. No moisture was observed in this vicinity and no degradation of the containment shell surfaces was noted.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-009.2000.8. This WR generated WO #01002133, which removed moisture barrier 1-MBRI-023. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002133.

Indication: #1-SCVI-010.2000.1
Location: Steel Containment Vessel Interior - Accumulator D Room, Elevation 565'+3"

Description: Moisture barrier material and cork expansion joint material has been removed from between the Steel Containment Vessel and the floor of the Accumulator D

Room, except that cork still needs to be removed from behind the Accumulator tank. Exposed surfaces of the Containment Vessel are degraded, with coatings damage, staining and minor corrosion at various locations behind the floor. At this time, the corrosion is not considered significant enough to warrant further evaluation.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165871 was written as a result of indication #1-SCVI-010.2000.1. This WR generated WO #01002112, which removed cork expansion joint material between the Steel Containment Vessel and 'D' Accumulator Tank. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002112.

Indication: #1-SCVI-010.2000.2
Location: Steel Containment Vessel Interior - Accumulator D Room

Description: Moisture barrier (sealant) installed along vertical wall joint at azimuth 305° is cracked, separating, and has portions missing. In addition, moisture was observed in this area. No unacceptable conditions were observed on the containment vessel surfaces.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-010.2000.2. This WR generated WO #01002133, which removed the damaged moisture barrier. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002133.

Indication: #1-SCVI-010.2000.3
Location: Steel Containment Vessel Interior - Accumulator D Room

Description: Moisture barrier (sealant) installed along vertical wall joint at azimuth 326° is cracked, separating, and has portions missing. No unacceptable conditions were observed on the containment vessel surfaces and moisture was not observed at this location.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165854 was written as a result of indication #1-SCVI-010.2000.3. This WR generated WO #01002133, which removed the damaged moisture barrier. Additionally, corrective coatings maintenance was performed on the newly exposed surfaces of the containment vessel shell under WO #01002133.

Indication: #1-SCVI-013.2000.2
Location: Steel Containment Vessel Interior - Pipe Chase Quadrant 3 (180°-247°)

Description: Moisture barrier (sealant) does not appear to have been installed correctly and needs repair between the sump near azimuth 184° and column near azimuth 193°, along the base of the Containment Vessel at the Pipe Chase floor.

This condition was documented in PIP #C-00-05438 (AR #01406096).

This condition was identified in PIP #C-06-07708 (AR #01459162) and an associated engineering evaluation was completed and determined that the condition of the Containment Vessel is acceptable.

Indication: #1-SCVI-013.2000.3
Location: Pipe Chase Quadrant 3 (180°-247°)

Description: Cracked, broken, and loose paint flakes (as thick as ¼" in some places) were noted on the Pipe Chase floor near azimuth 218° at the base of the Containment Vessel.

This condition was documented in PIP #C-00-05438 (AR #01406096).

Loose cork material noted during the inspection was removed during the final walkdown of the Containment Building during 1EOC12.

Indication: #1-SCVI-013.2000.5
Location: Steel Containment Vessel Interior - Pipe Chase Quadrant 3 (180°-247°)

Description: Moisture barrier (sealant) between the Containment Vessel and Pipe Chase floor is separating from the Containment Vessel near azimuth 240°. It appears that the containment vessel shell surfaces were coated prior to placing sealant, but that the containment shell surfaces may not have been adequately cleaned prior to application of coatings. As a result, the coatings are delaminating, causing the sealant to separate. No moisture intrusion was noted at this location.

This condition was documented in PIP #C-00-05438 (AR #01406096).

This condition was identified in PIP #C-06-07708 (AR #01459162) and an associated engineering evaluation was completed and determined that the condition of the Containment Vessel is acceptable.

Indication: #1-SCVI-014.2000.1
Location: Pipe Chase Quadrant 4 (270°-360°)

Description: Coatings damage noted on Pipe Chase floor (near azimuth 300°). Thick flakes of paint were removed with putty knife. Also, moisture barrier (sealant) applied to base of containment vessel is damaged/degraded. As a result of further inspections, portions of the damaged sealant were removed and will now need repair. The joint between the containment vessel and the Pipe Chase floor remains sealed with coatings at this time.

This condition was documented in PIP #C-00-05438 (AR #01406096).

Loose cork material noted during the inspection was removed during the final walkdown of the Containment Building during 1EOC12.

Indication: #1-SCVI-014.2000.2
Location: Steel Containment Vessel Interior - Pipe Chase Quadrant 4 (270°-360°)

Description: Brown staining observed on containment shell, and brown water noted on Pipe Chase floor behind column at azimuth 303°. Moisture barrier (sealant) is also becoming degraded and is starting to separate from the containment vessel shell at this location. Portions of the damaged sealant were removed to confirm that material is still sealing the joint. As a result, the sealant material will now need repair.

This condition was documented in PIP #C-00-05438 (AR #01406096).

This condition remains unchanged from previous inspections. No evidence of damage or degradation was observed to coatings on the Containment Vessel wall. Catawba Nuclear Station continues to monitor during future inspections.

Indication: #1-SCVI-014.2000.3
Location: Steel Containment Vessel Interior - Pipe Chase Quadrant 4 (270°-360°)

Description: Moisture barrier (sealant) is not installed at the base of the containment vessel at 340°, beneath electric tray at this location. Beneath electric tray, debris and cork material was observed at the base of the containment vessel.

This condition was documented in PIP #C-00-05438 (AR #01406096).

This condition remains unchanged from previous inspections. No evidence of damage or degradation was observed to coatings on the Containment Vessel wall. Catawba Nuclear Station continues to monitor during future inspections.

Indication: #1-SCVI-016.2000.1
Location: Steel Containment Vessel Interior - VX Fan Pit Area

Description: Considerable standing water and glycol was observed on floor of VX Fan Pit at Elevation 593'+8½", between Azimuths 247° and 260°. This condition has been observed during several past inspections and continues to be the source of problems associated with moisture gaining access to Containment Vessel surfaces in this area. It is evident that water collecting in this area has contacted Containment Vessel surfaces behind the floor joint. Please note that inspections in Lower Containment areas confirm that water and staining has come from this area in Upper Containment. At locations along the floor joint in the VX Fan Pit, Containment coatings have become damaged and corrosion has been observed.

The condition of the Containment Vessel is considered acceptable, with maintenance required. Because these conditions have been observed during past inspections, the Containment ISI Plan considers the floor joint location along the VX Fan Pit as an area which warrants Augmented Examination in accordance with the ASME Code, Section XI, Subsection IWE. As a result, ultrasonic thickness measurements have been performed from the Annulus side of the vessel during this RFO (1EOC12) to confirm the acceptability of the Steel Containment Vessel behind the VX Fan Pit floor. These UT examination results indicated that the condition of the Containment shell is acceptable at this time. UT examinations will continue to be required until such time that these conditions have been eliminated. Please note that active corrosion is occurring and that corrective actions are urgently needed. Minor Modifications have been written to allow

the cork expansion joint material to be removed from between the containment vessel and the VX Fan Pit floor, and to repair coatings on exposed surfaces of the containment vessel behind the concrete floor. This work should be completed at the earliest opportunity to prevent continued degradation of the containment vessel in this area, and should be completed no later than the end of RFO 1EOC13. During 1EOC12, it was discovered that this work had been originally planned for this outage, but had been postponed. This work will be required to be completed during a RFO due to the geometry and access problems in this area. Also, please note that a VT-1 visual examination was also performed during 1EOC12 on containment surfaces within several inches of the VX Fan Pit floor.

The VT-1 examinations also confirmed that the condition of the containment vessel in this area is not good and is deteriorating. Reference PIP #1-C-99-1832 for additional information.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098079389 was written as a result of indication #1-SCVI-016.2000.1. This WR generated WO #00954958, which removed all remaining cork expansion joint material from behind the floor joint.

Indication: #1-SCVO-001.2000.5
Location: Steel Containment Vessel Exterior - 1st Stiffening Ring Area

Description: Flaking paint and minor corrosion was observed on stiffening ring web-to-shell weld and vertical stiffener area at the bottom of vertical stiffener located at Azimuth 345°, El. 555+4. Similar conditions were also observed at Azimuth 0° and 120°. These conditions have been identified during previous inspections and still exist. These conditions have been previously addressed in PIP #1-C-99-1771 which was generated during the previous inspection during 1EOC11 in 1999.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-001.2000.5. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-003.2000.4
Location: Steel Containment Vessel Exterior - 3rd Stiffening Ring Area

Description: Flaking paint and minor corrosion was observed on stiffening ring web-to-shell weld and vertical stiffener area at the bottom of vertical stiffener located at Azimuth 33°, El. 573+5. This problem was addressed previously in PIP #1-C-99-1771.

These conditions are unchanged from the previous inspection, but are still considered acceptable at this time. Coatings maintenance was performed to prevent further degradation.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-003.2000.4. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-006.2000.1
Location: Steel Containment Vessel Exterior - 6th Stiffening Ring Area

Description: Standing water, coatings damage and loss, staining and minor corrosion was observed on top of stiffening ring near Azimuth 233°, El. 599+5. These conditions were noted during the previous inspection during 1EOC11 in 1999. No corrective action has been taken to address this problem.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-006.2000.1. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-007.2000.2
Location: Steel Containment Vessel Exterior - 7th Stiffening Ring Area

Description: Standing water, coatings damage and loss, and some minor corrosion was observed on top of stiffening ring web adjacent to the Upper Personnel Airlock near Azimuth 248°, El. 609+5. Small area directly above this location on barrel has coatings loss and minor rusting. Rust staining was also observed on exterior of Airlock barrel adjacent to this area, but no corrosion was noted on barrel at this location. These conditions were noted during the previous inspection during 1EOC11 in 1999. No corrective actions have been taken since the last inspection.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-007.2000.2. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-007.2000.3
Location: Steel Containment Vessel Exterior - 7th Stiffening Ring Area

Description: Dirt, debris, and staining was also observed on the opposite side of the Upper Personnel Airlock barrel at Azimuth 259°, El. 609+5. These conditions were noted during the previous inspection during 1EOC11 in 1999. No corrective actions have been taken since the last inspection.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-007.2000.3. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-012.2000.1
Location: Steel Containment Vessel Exterior - 12th Stiffening Ring Area

Description: Conditions identified during the previous inspection during 1EOC11 in 1999 are unchanged. These conditions include the following:

Rust and damaged coatings were observed on Containment Vessel vertical weld seams and/or surrounding areas at the following locations:

- Azimuth 350° (approx.), between Elevations 659+1 and 669+1
- Azimuth 340° (approx.), between Elevations 659+1 and 669+1.
- Azimuth 139° (approx.), between Elevations 659+1 and 669+1.
- Azimuth 130° (approx.), between Elevations 659+1 and 669+1.
- Azimuth 120° (approx.), between Elevations 659+1 and 669+1.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-012.2000.1. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-SCVO-012.2000.2
Location: Steel Containment Vessel Exterior - 12th Stiffening Ring Area

Description: Conditions identified during the previous inspection during 1EOC11 in 1999 are unchanged. These conditions include coatings damage and base metal exposed on portion of Containment Vessel dome at azimuth 184° (approx.), Elevation 668+0.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-SCVO-012.2000.2. This WR generated WO #01002114, which re-coated the identified conditions.

Indication: #1-PENE-C100.2000.1
Location: Steel Containment Vessel Exterior - Lower Airlock (Annulus)

Description: Some coating degradation and minor rusting was observed on top of the Airlock Barrel on the Steel Containment Vessel exterior (Annulus side) at Azimuth 65°, El. 573+5.No coating repair has been performed on top of the Airlock Barrel since the previous inspection. The coating degradation and minor rusting are unchanged.

This condition was documented in PIP #C-00-05438 (AR #01406096).

WR #098165811 was written as a result of indication #1-PENE-C100.2000.1. This WR generated WO #01002114, which re-coated the identified conditions.

<u>Unit 1 1EOC21 – 2014 - Second Interval</u>

An inspection of the Unit 1 Steel Containment Vessel (SCV) moisture barriers1-MBRI-002, 1-MBRI-003 and SCV liner plate 1-SCVI-012, 1-SCVI-013 (Pipe Chase, ECCS sump Az. 180) was performed by a certified Quality Control (QC) inspector and RES Engineer on May 22, 2014. This inspection was performed in accordance with 10 CFR 50 and to satisfy the in-service visual examinations requirements in accordance with the ASME Section XI, Subsection IWE Category E-C, Item E4.11 and was documented in WO #02070341-1 and the Containment Structural Integrity Inspection procedure. This area was cleaned prior to this inspection (Ref. WO #02070341-7). No suspect boron

deposits were observed on the moisture barriers along the SCV/Pipe Chase Floor interface within the ECCS Sump at Penetrations Mk. M210 and Mk. M303, after the final cleanup. The inspection of the moisture barriers "1-MBRI-002 and 1-MBRI-003 and the inspection zones of the SCV liner plate "1-SCVI-012 and 1-SCVI-013" (CN-ISIC2-1042-0002 and 0003) by QC was found acceptable per the applicable ISI acceptance criteria requirements. There was no evidence of any SCV degradation/wastage or any problem with the moisture barrier sealant along the concrete floor and SCV interface. This area was clean and dry.

Based on the required Containment General Visual Examination performed during 1EOC21, condition of the SCV is acceptable.

Unit 2 2EOC13 – 2004 - First Interval

2-SCVO-001.2004.1
Moisture barrier at the containment vessel embedment zone was observed to be separated, with possible moisture intrusion at azimuth 300 degrees, Elevation 552'. Dried residue from boron leaks were present in this area at the time of inspection. Boron residue appears to have come from leakage associated with the ND Pump. A portion of the degraded moisture barrier was removed with a putty knife. Although some moisture was observed on cork expansion joint material directly beneath the damaged area, cork material was dry at the containment vessel surface. For this reason, degradation of the embedded containment shell plate is not suspected. However, given that there has been, and continues to be, boron present on the floor in this area, the degraded moisture barrier must be repaired immediately. Without immediate repair, borated water could gain access to inaccessible surfaces of the SCV.

AR #01431852-02: WR #98325209 (WO #98691237-01) was written to address indications #2-SCVO-001.2004.1 and #2-SCVO-003.2004.1 and was completed during 2EOC13.

2-SCVO-003.2004.1
Moisture barrier at azimuth 262°, El. 576' (approx.) (3" to left of vertical stiffener) is separated from steel containment vessel on top of Fuel Transfer Tube Shielding concrete. Moisture barrier is identified as 2-MBRO-002 on drawing CN-1042-ISI.2-005. Standing water was observed in this area, apparently from condensation on piping located above this area. During the inspection, a portion of the degraded moisture barrier material was removed and no evidence of moisture intrusion was observed at this time. However, because there is staining (evidence of standing water in this area), this moisture barrier shall be repaired immediately to prevent moisture intrusion onto inaccessible surfaces of the steel containment vessel beneath this area. Without immediate repair, water could gain access to inaccessible surfaces of the SCV.

AR #01431852-02: WR #98325209 (WO #98691237-01) was written to address indications #2-SCVO-001.2004.1 and #2-SCVO-003.2004.1 and was completed during 2EOC13.

2-SCVO-002.2004.2
Moisture barrier material not installed between the SCV and the Fuel Transfer Tube Shielding concrete at 270° between ring 2 (El. 563'+5") and ring 3 (El. 573'+5"), along

vertical interface. No evidence of moisture intrusion. This moisture barrier is identified as 2-MBRO-002 on drawing #CN-1042-ISI.2-005. Because this location is not one where moisture is a concern, the Containment ISI Plan File #CN-1042-CISI-0001 and drawing #CN1042-ISI.2-005 should be revised to indicate that moisture barrier material does not exist at this location. Installation of moisture barrier material is not recommended and is not required.

AR #01431852-01: The Containment ISI Plan and affected drawing were revised and transmitted.

2-SCVO-006.2004.1
Condensation from 8" NF piping is dripping onto containment ring stiffener web between azimuths 232 to 247 (approx.), El. 593'+5". This condensation, in addition to some coatings damage, is resulting in some minor surface corrosion at various locations in this area. Corrosion is superficial and does not require additional examination to confirm continued acceptability of the ring web. These conditions have been observed during several previous examinations. Corrective coatings maintenance should continue to prevent these conditions from worsening. This area is one which sees a lot of traffic. These conditions need not be repaired immediately, but should be corrected prior to RFO 2EOC14.

AR #01431852-02: WR #98325880 (WO #98692500-01) was written to address indications #2-SCVO-006.2004.1, #2-SCVO-006.2004.2, #2-SCVO-007.2004.1, and #2-SCVO-010.2004.1. Work complete per WR #01094406.

2-SCVO-006.2004.2
Corrosion identified behind lifting lug at azimuth 249° (approx.) El. 593'+5" at interface between containment shell and ring web. Corrosion is superficial and does not require additional examination to confirm continued acceptability of the ring web or shell plate. However, this condition shall be corrected as soon as possible to prevent continued corrosion. These conditions need not be repaired immediately, but should be corrected prior to RFO 2EOC14.

AR #01431852-02: WR #98325880 (WO #98692500-01) was written to address indications #2-SCVO-006.2004.1, #2-SCVO-006.2004.2, #2-SCVO-007.2004.1, and #2-SCVO-010.2004.1. Work complete per WR #01094406.

2-SCVO-007.2004.1
Moisture and corrosion observed on ring stiffener web near Upper Airlock at azimuth 249°, Elevation 609' (approx.). Moisture is due to 8" NF piping in this area. Flaking paint observed and minor corrosion, but no unacceptable wall thinning occurred at the time of this inspection. These conditions need not be repaired immediately, but should be corrected prior to RFO 2EOC14.

AR #01431852-02: WR #98325880 (WO #98692500-01) was written to address indications #2-SCVO-006.2004.1, #2-SCVO-006.2004.2, #2-SCVO-007.2004.1, and #2-SCVO-010.2004.1. Work complete per WR #01094406.

2-SCVO-010.2004.1
Minor rust observed on stiffener plate near azimuth 215° (Elevation 639'+5"). No immediate corrective action required, but the affected area should be recoated to prevent corrosion. It is recommended that this be corrected prior to RFO 2EOC14.

AR #01431852-02: WR #98325880 (WO #98692500-01) was written to address indications #2-SCVO-006.2004.1, #2-SCVO-006.2004.2, #2-SCVO-007.2004.1, and #2-SCVO-010.2004.1. Work complete per WR #01094406.

Unit 2 2EOC17 – 2010 - Second Interval

Indication Number: 2-SCVO-0001.2010.1
Area/Component I.D.: 2_SCVO-0001 (Ref. Drawings CN-ISIC2-2042-004 and 005)
Location/Azimuth: Containment Vessel Exterior Surface - Annulus Floor and Ring 1/0° through 360°.
Elevation: 552'+0" through 563'+5"

Description of Abnormality/Degradation: Small amount of white residue (appeared to be boron) was found on the floor and on the moisture barrier. No active leak at 244 degrees. Fluid Leak Management generated WO #001948592 to address. Area was cleaned per WO and determined not to be boron but ground water seeping through the outside concrete wall.
Action Required: Continue monitoring during future outages.

Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVO-0001.2010.2
Area/Component I.D.: 2_SCVO-0001 (Ref. Drawings CN-ISIC2-2042-004 and 005)
Location/Azimuth: Containment Vessel Exterior Surface - Annulus Floor and Ring 1/0° through 360°.
Elevation: 552'+0" through 563'+5"

Description of Abnormality/Degradation: Small amount of white residue (appeared to be boron) was found on the floor and on the moisture barrier. No active leak at 225 degrees. Fluid Leak Management generated WO #001948592 to address. Area was cleaned per WO and determined not to be boron but ground water seeping through the outside concrete wall.

Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVO-0003.2010.1
Area/Component I.D.: 2_SCVO-00031 (Ref. Drawings CN-ISIC2-2042-004 and 005)
Location/Azimuth: Containment Vessel Exterior Surface - Annulus Floor and Ring 1/0° through 360°.
Elevation: 573'to 575'

Description of Abnormality/Degradation: Moisture barrier 2MBRO-002 is separated from the SCV at various top and sides locations. WO #1017673 was written to correct this condition.

Action Required: Moisture barrier has been corrected per WR #01017673.

Indication Number: 2-SCVI-0002.2010.1
Area/Component I.D.: 2-SCVI-0002 (Ref. Drawings CN-ISIC2-2042-002 and -003)
Location: Containment Vessel Interior Surface - Vent Unit A/D Room.
Elevation: 565'+0" through 591'+0"

Description of Indication: Moistures barriers 2-MBRI-006 and 2-MBRI-032 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. Also, some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0003.2010.1
Area/Component I.D.: 2-SCVI-0003 (Ref. Drawing CN-ISIC2-2042-002)
Location/Azimuth: Containment Vessel Interior Surface - Accumulator A Room/34° through 54°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moistures barriers 2-MBRI-007 and 2-MBRI-008 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0004.2010.1
Area/Component I.D.: 2-SCVI-0004 (Ref. Drawing CN-ISIC2-2042-002)
Location/Azimuth: Containment Vessel Interior Surface - Lower Airlock Area/54° through 126°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barrier 2-MBRI-011 along vertical wall joint has not been completely removed. This item has been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0005.2010.1
Area/Component I.D.: 2-SCVI-0005 (Ref. Drawings CN-ISIC2-2042-002)
Location/Azimuth: Containment Vessel Interior Surface - Excess Letdown HX Area/ 106° through 126°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barrier 2-MBRI-014 along vertical wall joint has not been completely removed. This item has been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0006.2010.1
Area/Component I.D.: 2-SCVI-0006 (Ref. Drawings CN-ISIC2-2042-002)
Location/Azimuth: Containment Vessel Interior Surface - B Accumulator Room/126° through 146°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moistures barriers 2-MBRI-0017 and 2-MBRI-0018 (along vertical wall joints) have not been completely removed. Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0007.2010.1
Area/Component I.D.: 2-SCVI-0007 (Ref. Drawings CN-ISIC2-2042-002 and -003)
Location/Azimuth: Containment Vessel Interior Surface - Vent Unit B/C Room/146° through 214°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moistures barriers 2-MBRI-0019 and 2-MBRI-0020 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04 04768). WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0008.2010.1
Area/Component I.D.: 2-SCVI-0007 (Ref. Drawings CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - C Accumulator Room/214° through 234°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moistures barriers 2-MBRI-0021 and 2-MBRI-0022 (along vertical wall joints) have not been completely removed.
Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. These items have
been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0009.2010.1
Area/Component I.D.: 2-SCVI-0009 (Ref. Drawings CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - Nothing and Top of Fuel Transfer Canal Wall/Room/234° through 305°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barrier 2-MBRI-026 is intact but degraded. This moisture barrier was replaced per WO #986330. Moistures barriers 2-MBRI-0025, 2-MBRI-0027 and 2-MBRI-0029 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0010.2010.1
Area/Component I.D.: 2-SCVI-0010 (Ref. Drawings CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - D Accumulator Room/305° through 326°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moistures barriers 2-MBRI-0030 and 2-MBRI-0031 (along vertical wall joints) have not been completely removed. Also some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exist with no change. These items have been identified during previous inspections (Reference PIPs C-00-01239 and C-04-04768). WO #986330 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #986330 for 2EOC18 or issue new work request. Continue monitoring during future inspections.

Work was completed via WO #00986330-01. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0013.2010.1
Area/Component I.D.: 2-SCVI-0017 (Ref. Drawing CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - Sump Area/ 184° (approx.).
Elevation: 552'+0"

Description of Abnormality/Degradation: Pen. M303 Approx. 184 deg./Sump Area/Boron residue/staining. Indications 2-SCVI-013.2007.1, 2, and 3 identified during 2EOC15 were repaired and found to be acceptable. A small amount of boron residue was found in the trench below the recirculation sump pipe. This area was cleaned and re-inspected by QC and found to acceptable.

Action Required: Continue monitoring during future inspections.

Area is monitored via ongoing containment inspections.

<u>Unit 2 2EOC19 – 2013 - Second Interval</u>

Indication Number: 2-SCVI-0006.2013.1
Area/Component I.D.: 2-SCVI-0006 (Ref. Drawings CN-ISIC2-2042-002)
Location/Azimuth: Containment Vessel Interior Surface - B Accumulator Room/126° through 146°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barriers 2-MBRI-0017 and 2-MBRI 0018 (along vertical wall joints) have not been completely removed. Also, some light staining of the SCV liner plate below the concrete floor identified during previous inspection still exists with no change. These items have been identified during previous

inspections. WO #02036636 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #02036636 for 2EOC20. Continue monitoring during future inspections.

Work closed out per WO #02036636-02. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0007.2013.1
Area/Component I.D.: 2-SCVI-0007 (Ref. Drawings CN-ISIC2-2042-002 and -003)
Location/Azimuth: Containment Vessel Interior Surface - Vent Unit B/C Room/146° through 214°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barriers 2-MBRI-0019 and 2-MBRI-0020 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections. WO #02036636 is still open to complete this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #02036636 for 2EOC20. Continue monitoring during future inspections.

Work performed per WO #02036636-03. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0008.2013.1
Area/Component I.D.: 2-SCVI-0007 (Ref. Drawings CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - C Accumulator Room/214° through 234°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barriers 2-MBRI-0021 and 2-MBRI-0022 (along vertical wall joints) have not been completely removed. These items have been identified during previous inspections. WO #02036636 is still open to perform this work. The condition of the containment vessel is acceptable.

Action Required: Schedule WO #02036636 for 2EOC20. Continue monitoring during future inspections.

Work performed per WO #02036636-04. Area is monitored via ongoing containment inspections.

Indication Number: 2-SCVI-0009.2013.1
Area/Component I.D.: 2-SCVI-0009 (Ref. Drawings CN-ISIC2-2042-003)
Location/Azimuth: Containment Vessel Interior Surface - Nothing and Top of Fuel Transfer Canal Wall/Room/234° through 305°.
Elevation: 565'+3" through 591'+2-1/2"

Description of Abnormality/Degradation: Moisture barriers 2-MBRI-0027 and 2-MBRI-0029 (along vertical wall joints) have not been completely removed and repairs are in progress. These items have been identified during previous inspections. WO #02036636 is still open to perform this work. The condition of the containment vessel is acceptable.

Work performed per WO #02036636-05. Area is monitored via ongoing containment inspections.

Response to Conditions Identified During Containment Inspections

Conditions identified during previous examinations that require ongoing monitoring are re-inspected during successive General Visual Examinations performed in accordance with procedures PT/1/A/4200/078 "Containment Structural Integrity Inspection", Unit 1 and PT/2/A/4200/078 "Containment Structural Integrity Inspection", Unit 2. These examinations are performed during each inspection Period and are used to satisfy the requirements of the ASME Code, Section XI, Category E-A, Item E1.11 and E1.30.

3.5 Supplemental Inspections

In the Safety Evaluation Report for NEI 94-01 Revision 2-A, the NRC stated the following requirement for the performance of Supplemental Visual Inspections in SER Section 3.1.1.3, Adequacy of Pre-Test Inspections (Visual Examinations):

> NEI TR 94-01, Revision 2, Section 9.2.3.2, states that: "To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years." NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3. of RG 1.163, are acceptable considering the longer 15-year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.
>
> Subsections IWE and IWL of the ASME Code, Section XI, as incorporated by reference in 10 CFR 50.55a, require general visual examinations two times within a 10-year interval for concrete components (Subsection IWL), and three times within a 10-year interval for steel components (Subsection IWE). To avoid duplication or deletion of examinations, licensees using NEI TR 94-01, Revision 2, have to develop a schedule for containment inspections that satisfy the provisions of Section 9.2.3.2 of this TR and ASME Code, Section XI, Subsection IWE and IWL requirements.

As previously stated in LAR Table 3.2.2-5, Note 1, "If this examination (general visual inspection of the Containment Vessel) is to be credited towards satisfying the examinations required by 10CFR50, Appendix J, the examination shall be performed during the RFO in which a Type A test is to be performed, just prior to the start of the Type A test. Duke Energy intends to credit Item E1.11 visual exams towards satisfying the requirements of 10CFR50, Appendix J."

To this end, the general visual inspections of the SCV and the Reactor Shield Building are performed in accordance with the Containment Structural Integrity Inspection procedure. This inspection procedure is performed at the designated frequency to satisfy the following inspection requirements:

- 10 CFR 50, Appendix J, Option A, Paragraph V.A. requires that "A general inspection of the accessible interior and exterior surfaces of the Containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the Containment structural integrity or leak-tightness. If there is evidence of structural deterioration, Type A tests shall NOT be performed until corrective action is taken in accordance with repair procedures, non-destructive examinations, and tests as specified in the applicable code specified in 50.55a at the commencement of repair work. Such structural deterioration and corrective actions taken shall be included in the summary report required by V.B."

- 10 CFR 50, Appendix J, Option B, Paragraph III.A. requires that "A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system."

- The ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE requires that a general visual examination be performed on Class MC Components (Metal Containments) prior to each Type A Test. Conditions that may affect containment structural integrity or leak tightness shall be accepted by engineering evaluation or corrected by repair or replacement prior to proceeding with a Type A Test. A general visual examination shall be performed once each inspection period as required by the ASME Code, Section XI. Subsection IWE, 1998 Edition with 2000 Addenda. General visual examinations of the Steel Containment Vessel may be used to satisfy the inspection requirements of 10 CFR 50, Appendix J.

The performance of Supplemental Inspections in addition to the performance of the Containment Structural Integrity Inspection to satisfy the requirement for the performance of Supplemental Visual Inspections in SER Section 3.1.1.3 is not required.

3.6 Limitations and Conditions

3.6.1 Limitations and Conditions Applicable to NEI 94-01, Revision 2-A

The NRC staff found that the use of NEI TR 94-01, Revision 2, was acceptable for referencing by licensees proposing to amend their TS to permanently extend the ILRT surveillance interval to 15 years, provided the following conditions as listed in Table 3.6-1 were satisfied.

Table 3.6-1, NEI 94-01 Revision 2-A Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	CNS Response
For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1.)	CNS will utilize the definition in NEI 94-01 Revision 2-A, Section 5.0.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3.)	Reference Tables 3.2.2-1, 3.2.2-2, 3.2.2-3, 3.2.2-4, 3.2.2-5, and 3.2.2-6 and Section 3.5 of this submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3.)	Reference Section 3.2.2, Inaccessible Class MC Areas, of this submittal. Reference Section 3.2.2, Table 3.2.2-8, Category E-C: Containment Surfaces Requiring Augmented Examination and LAR Section 3.2.2, Owner Specified Examination Requirements, of this submittal.

Table 3.6-1, NEI 94-01 Revision 2-A Limitations and Conditions	
Limitation/Condition (From Section 4.0 of SE)	**CNS Response**
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4.)	The CNS Unit 1 steam generators were replaced in October 1996. The equipment hatch was utilized for this modification. For CNS, Unit 2, the steam generators that were installed during original construction have not been replaced. There are no planned modifications for CNS that will require a Type A test prior to the next scheduled Type A test proposed under this LAR. There is no anticipated addition or removal of plant hardware within the containment building, which could affect its leak-tightness.
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 TR 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. (Refer to SE Section 3.1.1.2.)	CNS will follow the requirements of NEI 94-01 Revision 2-A, Section 9.1. In accordance with the requirements of 94-01 Revision 2-A, SER Section 3.1.1.2, CNS will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.
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. CNS was not licensed under 10 CFR Part 52.

3.6.2 Limitations and Conditions Applicable to NEI 94-01 Revision 3-A

The NRC staff found that the guidance in NEI TR 94-01, Revision 3, was acceptable for referencing by licensees in the implementation for the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. However, the NRC staff

identified two conditions on the use of NEI TR 94-01, Revision 3 (Reference NEI 94-01 Revision 3-A, NRC SER 4.0, Limitations and Conditions):

Topical Report 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 TR 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.

Response to Condition 1

Condition 1 presents three (3) separate issues that are required to be addressed as follows:

- ISSUE 1 - The allowance of an extended interval for Type C LLRTs of 75 months carries 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.

- ISSUE 2 - In addition, a corrective action plan shall be developed to restore the margin to an acceptable level.

- ISSUE 3 - Use of the allowed 9-month extension for eligible Type C valves is only authorized for non-routine emergent conditions.

Response to Condition 1, Issue 1

The post-outage report shall include the margin between the Type B and Type C Minimum Pathway Leak Rate (MNPLR) summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 La.

Response to Condition 1, Issue 2

When the potential leakage understatement adjusted Type B and C MNPLR total is greater than the CNS administrative leakage summation limit of 0.50 La, but less than the regulatory limit of 0.6 La, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the CNS administrative limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the

prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

Response to Condition 1, Issue 3

CNS will apply the 9-month grace period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

Topical Report 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 RFO, nearly all LLRTs 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 RFO's. 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 TR 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 and 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.

Response to Condition 2

Condition 2 presents two (2) separate issues that are required to be addressed as follows:

- ISSUE 1 - 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 TR 94-01, Revision 3, Section 12.1.

- ISSUE 2 - 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 and 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.

Response to Condition 2, Issue 1

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. As such, CNS will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the As-Left leakage total for each Type C component currently on greater than a 60-month test interval up to the 75-month extended test interval. This will result in a combined conservative Type C total for all 75-month LLRTs being "carried forward" and will be included whenever the total leakage summation is required to be updated (either while on line or following an outage).

When the potential leakage understatement adjusted leak rate total for those Type C components being tested on greater than a 60-month test interval up to the 75-month extended test interval is summed with the non-adjusted total of those Type C components being tested at less than or equal to a 60-month test interval, and the total of the Type B tested components, if the MNPLR is greater than the CNS administrative leakage summation limit of 0.50 La, but less than the regulatory limit of 0.6 La, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the CNS administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

Response to Condition 2, Issue 2

If the potential leakage understatement adjusted leak rate MNPLR is less than the CNS administrative leakage summation limit of 0.50 La, then the acceptability of the greater than a 60-month test interval up to the 75-month LLRT extension for all affected Type C components has been adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

In addition to Condition 1, Parts 1and 2, which deal with the MNPLR Type B and C summation margin, NEI 94-01, Revision 3-A also has a margin related requirement as contained in Section 12.1, Report Requirements.

A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that

outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and 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 leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

At CNS, in the event an adverse trend in the aforementioned potential leakage understatement adjusted Type B and C summation is identified, and then an analysis and determination of a corrective action plan shall be prepared to restore the trend and associated margin to an acceptable level. The corrective action plan shall focus on those components which have contributed the most to the adverse trend in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

At CNS an adverse trend is defined as three (3) consecutive increases in the final pre-RCS Mode Change Type B and C MNPLR leakage summation values, as adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of La.

3.7 Evaluation of Risk Impact

3.7.1 Methodology

The purpose of this analysis is to provide a risk assessment of permanently extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled RFO's for the Catawba Nuclear Station (CNS). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A (Reference 2), the methodology used in EPRI TR-104285 (Reference 8), the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 18), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Catawba to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 19), and the methodology used in EPRI 1018243, Revision 2-A of EPRI 1009325 (Reference 17).

In the SER issued by NRC letter dated June 25, 2008 (Reference 9), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. Table 3.7.1-1 addresses each of the four limitations and conditions for the use of EPRI 1009325, Revision 2.

Table 3.7.1-1, EPRI Report No.TR-1009325 Revision 2 Limitations and Conditions	
Limitation/Condition (From Section 4.2 of SE)	**CNS Response**
1.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.	The technical adequacy of the CNS PRA is consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension, as detailed in Attachment 5 of this submittal, "PRA Risk Assessment for Extending ILRT Interval to 15 Years," Attachment 1.
2. 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 SE.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and CCFP acceptance guidelines, and these guidelines have been used for the CNS plant specific assessments.
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.	The increase in population dose is 0.026 person-rem/year.
In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	The increase in CCFP is 0.502%. The increase proved to be below 1.5 percentage points and thus is considered to be small.
3. 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 accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 La as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the CNS plant specific risk assessment. Reference Attachment 5, Section 5.1.2 of the submittal.

4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.	For CNS, containment over-pressure is NOT relied upon for emergency core cooling system (ECCS) performance. Reference Section 3.1.5 of the submittal.

3.7.2 Summary of Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension

Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension

The CNS internal events PRA model (Revision 3b) is used to calculate CDF and LERF for the permanent 15-year ILRT extension. Any elements of the supporting requirements detailed in ASME/ANS RA-Sa-2009 that could be significantly affected by the application are required to meet Capability Category II requirements.

The internal events PRA provides an adequate base model for the development of the permanent 15-year ILRT extension. In accordance with RG 1.200, the most recent full scope CNS Internal Events PRA Peer Review was performed in March 2002 using the peer review process described in NEI 00-02 (Attachment U of Reference 15). More recently, focused scope peer reviews have been conducted on the CNS LERF PRA model and the CNS Internal Flooding PRA model. The results from these focus scope peer reviews are discussed in LAR Attachment 5, section A.1.1 for LERF and A.1.2 for Internal Flooding.

In March 2002, the CNS internal events PRA model received a peer review to certify the acceptability of PRAs before a consensus PRA Standard was available. The industry-developed process and methodology outlined in NEI 00-02 was used for the peer review. The review process was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the NEI Risk Applications Task Force.

Revision 2b of the CNS internal events PRA was the model of record at the time of the peer review. The Revision 3 model was used as the basis for the Fire PRA model which supports the NFPA 805 transition. Revision 4 of the internal events PRA is currently under development.

The NEI 00-02 Peer Review process used grades to assess the relative technical merits and capabilities of each sub-element reviewed. The grades provide guidance on appropriate use of the information covered by the sub-element for risk-informed applications. Per NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard", in general, the following approximate correspondence exists between the NEI 00-02 grading system and the ASME/ ANS PRA Standard RA-Sa-2009:

NEI 00-02	ASME PRA Standard
Grade 1	No equivalent "grade"
Grade 2	Capability Category I
Grade 3	Capability Category II
Grade 4	Capability Category III

Approximately 73% of the graded sub-elements received grades of 3 or higher. None of the sub-elements received a grade of 1 (or contingent 2), and 27% of the sub-elements received a grade of 2 or contingent 3 (roughly 75% of this group was contingent grade 3).

F&Os from the 2002 peer review were assigned a significance level of A, B, C, D, or S based on guidance in NEI 00-02. Significance level A and B are equivalent to "Findings" in NEI 05-04 Revision 2. There were no level A F&Os; there were 32 level B F&Os, and 1 superior notation. In the time since the NEI 00-02 peer review, focused peer reviews have been performed for the internal flood and LERF models, which supersede one of the 32 F&Os.

In 2008, Duke Energy performed a self-assessment that evaluated the differences between the original peer review against NEI 00-02 and RA-S-2008 of the ASME/ANS PRA Standard, as endorsed by Regulatory Guide 1.200, Revision 1.

In 2013, Duke Energy performed a self-assessment against the ASME/ANS PRA Standard RA-Sa-2009 supporting requirements, as endorsed by Reg. Guide 1.200 Revision 2.

Attachment 5, Table A-1 presents an assessment of all ASME/ANS PRA Standard RA-Sa-2009 supporting requirements that were assessed to be "Not Met" at the equivalent of Capability Category II in the 2002 peer review, were not assessed in the 2002 peer review (no equivalent NEI 00-02 sub-elements), or were assessed to be "Met" but had related Findings. Regulatory Guide 1.200, Appendix B was used to correlate NEI 00-02 sub-elements to ASME/ANS PRA Standard RA-Sa-2009 supporting requirements for the assessments. F&Os from the 2002 peer review are dispositioned for the applicable ASME/ANS PRA Standard RA-Sa-2009 SRs.

All changes to the CNS internal events PRA model since the last full-scope peer review have been reviewed and, with the exception of the LERF and Internal Flood PRA models for which focused-scope peer reviews were performed, there are no changes that are considered PRA upgrades as defined in ASME/ANS-RA-Sa-2009, as endorsed by Regulatory Guide 1.200 Revision 2. The CNS Internal Events PRA was judged to meet Capability Category II consistent with RG 1.200 guidance.

LERF PRA Quality Statement

In December 2012, a focused scope peer review was performed of the CNS LERF PRA against selected requirements of the ASME/ANS PRA Standard RA-Sa-2009, and any Clarifications and Qualifications provided in the NRC endorsement of the Standard contained in Revision 2 to RG 1.200. The peer review was performed using the process defined in NEI 05-04. The scope of the review was limited to the High Level Requirements and SRs in Part 2, Requirements for Internal Events At-Power PRA,

Tables 2-2.8-1 and 2-2.8-2 through 2-2.8-8, of the ASME/ANS PRA Standard. The model reviewed was the LERF portion of CNS Internal Events PRA Model.

The ASME/ANS PRA Standard contains a total of 41 numbered SRs for the LERF portion of the internal events standard requirements. Two of the LERF SRs were determined to be not applicable to the CNS LERF PRA. Of the 39 applicable SRs, 26 SRs, or 67%, were rated as SR Met, Capability Category I/II, or greater. Only two SRs were not met. However, 11, or 28%, of the SRs were assessed at Capability Category I. CNS uses a LERF model based on the simplified LERF model in NUREG/CR-6595. While a NUREG/CR-6595 model is classified as Capability Category I, the NRC has determined this to be of sufficient capability to support risk-informed applications.

In the course of this review, 9 new F&Os were prepared, including 6 suggestions and 3 findings. Attachment 5, Table A-2 lists the 13 SRs that were assessed at Capability Category I or Not Met and the related findings, including the peer review assessment comments, the disposition and status for each of the findings, and an assessment of the impact on the Fire PRA and NFPA 805 application. The LERF Analysis peer review report is available upon request.

Internal Flood PRA Quality Statement

In September 2012, a focused scope peer review was performed of the CNS Internal Flood PRA using the NEI 05-04 process and the ASME PRA Standard ASME/ANS RA-Sa- 2009, along with the NRC clarifications provided in Regulatory Guide 1.200, Revision 2. The peer review concluded that 56 of the total 62 numbered SRs outlined within the 2009 ASME PRA Standard for At-Power Internal Flood met Capability Category II or greater. Five of the SRs were rated as Not Met and 1 was rated as CC I.

The independent peer review identified 17 new F&Os which are comprised of 9 findings, 7 suggestions, and 1 best practice. Table A-3 presents the SRs and related F&O findings, including the peer review assessment comments, the disposition and status for each of the findings, and an assessment of the impact on the Fire PRA and NFPA 805 application. The Flooding Analysis peer review report is available upon request.

Fire PRA Quality Statement for Permanent 15-Year ILRT Extension

In accordance with RG 1.205 position 4.3: "The licensee should submit the documentation described in Section 4.2 of Regulatory Guide 1.200 to address the baseline PRA and application-specific analyses. For PRA Standard "supporting requirements" important to the NFPA 805 risk assessments, the NRC position is that Capability Category II is generally acceptable."

The CNS Fire PRA Peer Review was performed on July 12-16, 2010 using RG 1.200, Revision 2, the combined PRA standard, ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2, and the NEI 07-12 Fire PRA peer review process (Attachment V of Reference 15). The purpose of this review was to provide a method for establishing the technical quality and adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The peer review findings were addressed and the dispositions reviewed to validate that no changes were made which meet the definition of a PRA model upgrade per RG 1.200.

Therefore, no additional peer reviews, partial scope or focused scope, were required to be conducted for the CNS Fire PRA.

The CNS Fire PRA was judged to meet Capability Category II consistent with RG 1.205 guidance. A total of twenty (20) F&O findings and twenty-nine (29) F&O suggestions (plus 1 best practice F&O) were generated. The capability categories are defined in ASME/ANS RA-Sa-2009, Part 4, "Requirements for Fires At-Power PRA." The peer review report noted that there were 13 SRs where the standard was not met. Sixteen F&Os were issued against SRs which met Capability Category I (some classified as "findings" and some addressed via "suggestions"). The findings have been resolved with the dispositions summarized in Attachment 5, Table A-4. The impact of those areas where only the Capability Category I requirement was met is summarized in Attachment 5, Table A-5. All F&Os that were defined as suggestions have been dispositioned. No changes were made in the resolution of the findings that meet the definition of a model upgrade as defined by RG 1.200; therefore, a follow-up peer review is not required. The Fire PRA is judged to be adequate to support the ILRT extension.

<u>High Wind PRA (HWPRA) Quality Statement for Permanent 15-Year ILRT Extension</u>

The HWPRA was assessed by a peer team against ASME/ANS PRA standard with RG 1.200 Revision 2 clarifications in August of 2013. The peer team documented the Facts and Observations (F&Os) that pertain to the CNS HWPRA in LTR-RAM-II-13-077 (Reference 16). Each of these F&Os are resolved or dispositioned in order to ensure the capability category of each individual Standard Requirement is met so that the CNS HWPRA can be used to support risk-informed applications. Attachment 5, Table A-6 shows the findings and resolutions.

3.7.3 Summary of Plant-Specific Risk Assessment Results

The risk impact of permanently extending the Type A ILRT test frequency to once in fifteen years is as follows:

- Regulatory Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF less than 1.0E-06/year and increases in LERF less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 1.13E-7/year using the EPRI guidance (this value increases slightly to 1.14E-7/year if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included), and baseline LERF is 1.12E-6. As such, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Regulatory Guide 1.174.

- The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.026 person-rem/year. EPRI Report No. 1009325, Revision 2-A (Reference 17) states that a very small population dose is defined 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. The results of this calculation meet these criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the 3 in 10-year interval to 1 in 15-year interval is 0.502%. EPRI Report No. 1009325, Revision 2-A (Reference 17) states that increases in CCFP of ≤ 1.5% is very small. Therefore, this increase is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the CNS risk profile.

3.7.4 Previous Assessments

The NRC in NUREG-1493 (Reference 7) has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 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 or Type 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 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for CNS confirm these general findings on a plant-specific basis considering the severe accidents evaluated for CNS, the CNS containment failure modes, and the local population surrounding CNS.

Details of the CNS risk assessment are contained in Attachment 5 of this submittal.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/ Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. 10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR Part 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type B and Type C test frequency will not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2, provided a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94 01, Revision 3-A, Section 9.2.3.1 states that Type A ILRT intervals of up to 15 years are allowed by this guideline. The Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI Report 1018243 (Formerly TR-1009325, Revision 2) indicates that, in general, the risk impact associated with ILRT interval extensions for intervals up to 15 years is small. However, plant-specific confirmatory analyses are required.

The NRC staff reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01, Revision 2, the NRC staff determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. This guidance includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC staff finds that the Type A testing methodology as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by NEI TR 94- 01, Revision 2, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight.

In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

For EPRI Report No. 1009325, Revision 2, a risk-informed methodology using plant specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC staff finds that the proposed methodology satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174. The NRC staff, therefore, found that this guidance was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.2 of the Safety Evaluation Report (SER).

The NRC staff reviewed NEI TR 94-01, Revision 3, and determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J, as modified by the conditions and limitations summarized in Section 4.0 of the associated Safety Evaluation. This guidance included provisions for extending Type C LLRT intervals up to 75 months. Type C testing ensures that individual containment isolation valves are essentially leak tight. In addition,

aggregate Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. The NRC staff, therefore, found that this guidance, as modified to include two limitations and conditions, was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing. Any applicant may reference NEI TR 94-01, Revision 3, as modified by the associated SER and approved by the NRC, and the conditions and limitations specified in NEI 94- 01, Revision 2-A, dated October 2008, in a licensing action to satisfy the requirements of Option B to 10 CFR Part 50, Appendix J.

4.2 Precedent

This license amendment request is similar in nature to the following license amendments previously authorized by the NRC to extend the Type A test frequency to 15 years and the Type C test frequency to 75 months:

- Surry Power Station, Units 1 and 2 (Reference 21)
- Donald C. Cook Nuclear Plant, Units 1 and 2 (Reference 22)
- Beaver Valley Power Station, Unit Nos. 1 and 2 (Reference 23)
- Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (Reference 24)
- Peach Bottom Atomic Power Station, Units 2 and 3 (Reference 25)

4.3 Significant Hazards Consideration

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," 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 proposed amendment to the Technical Specifications (TS) involves the extension of the Catawba Nuclear Station (CNS) Type A containment integrated leak rate test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. The containment and the testing requirements invoked 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 the prevention or identification of any precursors of an accident. The change in dose risk for

changing the Type A test frequency from three-per-ten years to once-per-fifteen years, measured, as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.026 person-rem/year. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined 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. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

As documented in NUREG-1493, Type B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The CNS Type A test history supports this conclusion.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and; (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes an exception previously granted to allow one-time extensions of the Unit 1 and Unit 2 ILRT test frequency for CNS. This exception was for activities that have already taken place; therefore, their deletion is solely an administrative action that has no effect on any component and no impact on how the units are operated.

Therefore, the proposed change does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment to the TS involves the extension of the CNS Type A containment integrated leak rate test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. The containment and 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 do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

The proposed amendment also deletes an exception previously granted to allow one-time extensions of the Unit 1 and Unit 2 ILRT test frequency for CNS. This exception was for activities that have already taken; therefore, their deletion is solely an administrative action that does not result in any change in how the units are operated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed amendment to TS 5.5.2 involves the extension of the CNS Type A containment integrated leak rate test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. 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 TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

The proposed change involves only the extension of the interval between Type A containment leak rate tests, and Type C tests for CNS. The proposed surveillance interval extension is bounded by the 15-year ILRT interval, and the 75-month Type C test interval currently authorized within NEI 94-01, Revision 3-A. Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, TS and the Maintenance Rule serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A, and Type C test intervals.

The proposed amendment also deletes an exception previously granted to allow one-time extensions of the Unit 1 and Unit 2 ILRT test frequency for CNS. This exception was for activities that have already taken place; therefore, their deletion is solely an administrative action and does not change how the units are operated and maintained. Thus, there is no reduction in any margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment does not involve a 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.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR Part 50, Appendix J, Option B. It incorporated the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years, and Type C test intervals to 75 months. 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. CNS is adopting the guidance of NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, for CNS, 10 CFR Part 50, Appendix J testing program plan.

Based on the previous ILRT tests conducted at CNS, it may be concluded that the permanent extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR Part 50, Appendix J and the overlapping inspection activities performed as part of the following CNS inspection programs:

- Containment Inservice Inspection Program (IWE/IWL)
- Periodic Condition Assessments of Service Level I Coatings
- Containment Structural Integrity Inspection

This experience is supplemented by risk analysis studies, including the CNS risk analysis provided in Attachment 5. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 15 years results in a very small change to the CNS risk profiles.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does 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.

6.0 REFERENCES

1. Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, September 1995.

2. NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 2012.

3. NEI 94-01, Revision 2-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, October 2008.

4. Regulatory Guide 1.174, Revision 2, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, May 2011.

5. Regulatory Guide 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, March 2009.

6. NEI 94-01, Revision 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 1995.

7. NUREG-1493, Performance-Based Containment Leak-Test Program, January 1995.

8. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

9. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (TAC No. MC9663), dated June 25, 2008.

10. Letter from S. Bahadur (NRC) to B. Bradley (NEI), Final Safety Evaluation of Nuclear Energy Institute (NEI) Report 94-01, Revision 3, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J (TAC No. ME2164), dated June 8, 2012.

11. ML013040289, Letter from K. Jabbour (NRC) to H. Tucker (Duke), Issuance of Amendment No. 41 to Facility Operating License NPF-35 and Amendment No. 34 to Facility Operating License NPF-52 -Catawba Nuclear Station, Units I and 2 (TACS 66755/66756) dated February 29, 1988.

12. ML013060529, Letter from R. Martin (NRC) to W. McCollum (Duke), Issuance Of Amendments - Containment Leak Rate Testing Consistent With Revised Appendix J, Option B - Catawba Nuclear Station, Units I And 2 (TAC NOS. M94444 and M94445) dated May 13, 1996.

13. ML012290327, Letter from C. Patel (NRC) to G. Peterson (Duke), Catawba Nuclear Station, Units 1 And 2 Re: Issuance Of Amendments (TAC NOS. MB1383 and MB1384) dated July 31, 2001.

14. ML030760108, Letter from R. Martin (NRC) to G. Peterson (Duke), Catawba Nuclear Station, Units 1 And 2 Re: Issuance Of Amendments (TAC NOS. MB5254 and MB5255) dated March 12, 2003.

15. Transition Report, "Transition to 10 CFR 50.48(c) - NFPA 805: Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," ML13276A503, ML13276A504, and ML13276A505, September 25, 2013.

16. Letter from Yan Gao (Westinghouse) to U.S. Nuclear Regulatory Commission, LTR-RAM-II-13-077, "Catawba Nuclear Plants RG 1.200 High Wind PRA Peer Review Report," Revision 0, January 2014.

17. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.

18. Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

19. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC (Document Control Desk), Docket No. 50-317, dated March 27, 2002.

20. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA: 2008.

21. ML14148A235, Letter to D. Heacock from S. Williams (NRC) dated July 3, 2014. Surry Power Station, Units 1 And 2 - Issuance of Amendment Regarding the

Containment Type A And Type C Leak Rate Tests.

22. ML15072A264, Letter to L. Weber from A. Dietrich (NRC) dated March 30, 2015. Donald C. Cook Nuclear Plant, Units 1 And 2 - Issuance of Amendments Re: Containment Leakage Rate Testing Program.

23. ML15078A058, Letter to E. Larson from T. Lamb (NRC) dated April 8, 2015. Beaver Valley Power Station, Unit Nos. 1 And 2 - Issuance of Amendment Re: License Amendment Request to Extend Containment Leakage Rate Test Frequency.

24. ML15154A661, Letter to G. Gellrich from A. Chereskin (NRC) dated July 16, 2015. Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 And 2 - Issuance of Amendments Re: Extension Of Containment Leakage Rate Testing Frequency.

25. ML15196A559, Letter to B. Hanson from R. Ennis (NRC) dated September 8, 2015. Peach Bottom Atomic Power Station, Units 2 And 3 - Issuance of Amendments Re: Extension of Type A and Type C Leak Rate Test Frequencies (TAC NOS. MF5172 AND MF5173).

Attachment 1

Technical Specification Pages

(Mark-up)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Plant Manager or Radiation Protection Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A dated July 2012, and the conditions and limitations specified in NEI 94-01 Revision 2-A, dated October 2008.

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, as modified by the following exceptions:~~

~~a. The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test; and~~

(continued)

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

~~b. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 14, 2000 (Unit 1) and February 7, 1993 (Unit 2) Type A test shall be performed no later than November 13, 2015 (Unit 1) and February 6, 2008 (Unit 2).~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 14.68 psig. The containment design pressure is 15 psig. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.3% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests and $< 0.6\ L_a$ for Type B and Type C tests.

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$. For each door, the leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 14.68 psig.

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, Chemical and Volume Control, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.4 DELETED

(continued)

Attachment 2

Technical Specification Pages

(Retyped)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Plant Manager or Radiation Protection Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 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 NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A dated July 2012, and the conditions and limitations specified in NEI 94-01 Revision 2-A, dated October 2008.

(continued)

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 14.68 psig. The containment design pressure is 15 psig. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.3% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests and $< 0.6\ L_a$ for Type B and Type C tests.

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$. For each door, the leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 14.68 psig.

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, Chemical and Volume Control, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.4 DELETED

(continued)

Attachment 3

Technical Specification Bases Page Markup

(For Information Only)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and the Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.6

For the Containment Purge System valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. The measured leakage rate for the Containment Purge System and Hydrogen Purge System valves must be $\leq$ 0.05 L_a when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), these valves will not be placed on the maximum extended test interval. Therefore, these valves will be tested in accordance with ~~Regulatory Guide 1.163, which allows~~ a maximum test interval of 30 months.

Insert: NEI 94-01 with

The Containment Air Release and Addition System valves have a demonstrated history of acceptable leakage. The measured leakage rate for containment air release and addition valves must be $\leq$ 0.01 L_a when pressurized to P_a. These valves will be tested in accordance with ~~Regulatory Guide 1.163, which allows~~ a maximum test interval of 30 months.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment

Attachment 4

Retyped Technical Specification Bases Page

(For Information Only)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and the Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.6

For the Containment Purge System valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. The measured leakage rate for the Containment Purge System and Hydrogen Purge System valves must be $\leq$ 0.05 L_a when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), these valves will not be placed on the maximum extended test interval. Therefore, these valves will be tested in accordance with NEI 94-01 with a maximum test interval of 30 months.

The Containment Air Release and Addition System valves have a demonstrated history of acceptable leakage. The measured leakage rate for containment air release and addition valves must be $\leq$ 0.01 L_a when pressurized to P_a. These valves will be tested in accordance with NEI 94-01 with a maximum test interval of 30 months.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment

Attachment 5

Evaluation of Risk Significance of Permanent ILRT Extension

JENSEN HUGHES

Advancing the Science of Safety

Catawba Nuclear Station:
Evaluation of Risk Significance of Permanent ILRT Extension

54003-CALC-02

Prepared for:

Catawba Nuclear Station

Project Title: Permanent ILRT Extension

Revision: 2

		Name and Date
Preparer: Justin Sattler		Digitally signed by Justin Sattler Date: 2015.11.12 11:08:24-06'00'
Reviewer: Kelly Wright		Digitally signed by Kelly Wright Date: 2015.11.12 11:09:45-06'00'
Review Method	Design Review ☒ Alternate Calculation ☐	
Approved by: Matthew Johnson		Digitally signed by Matt Johnson Date: 2015.11.12 12:33:13-06'00'

REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue
1	Updated report, per Duke request, for updated Internal Flood and Fire PRA results.
2	Updated report, per minor Duke comments.

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of permanently extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Catawba Nuclear Station (CNS). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], the methodology used in EPRI TR-104285 [Reference 2], the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [Reference 3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [Reference 4], the methodology used for Catawba to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [Reference 5], and the methodology used in EPRI 1018243, Revision 2-A of EPRI 1009325 [Reference 24].

2.0 SCOPE

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in 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 rate was less than limiting containment leakage rate of $1L_a$.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and 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, "Performance-Based Containment Leak Test Program," September 1995 [Reference 6], 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 assessment 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 TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

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 that for a representative PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1% to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for CNS.

NEI 94-01 Revision 2-A contains a Safety Evaluation Report that supports using EPRI Report No. 1009325 Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," for performing risk impact assessments in support of ILRT extensions [Reference 24]. The Guidance provided in Appendix H of EPRI Report No. 1009325 Revision 2-A builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service

inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

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 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} 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), which helps ensure the defense-in-depth philosophy is maintained, is also calculated.

Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of ILRT intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, a change in the CCFP of up to 1.5% is assumed to be small.

In addition, the total annual risk (person rem/year population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports (SER) for one-time interval extension (summarized in Appendix G of Reference 24) indicate a range of incremental increases in population dose that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/year and/or 0.002% to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [Reference 6], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase from the baseline interval (3 tests per 10 years) dose of ≤1.0 person-rem per year or 1% of the total baseline dose, whichever is less restrictive for the risk impact assessment of the proposed extended ILRT interval.

For those plants that credit containment overpressure for the mitigation of design basis accidents, a brief description of whether overpressure is required should be included in this section. In addition, if overpressure is included in the assessment, other risk metrics such as CDF should be described and reported.

3.0 REFERENCES

The following references were used in this calculation:

1. *Revision 3-A to Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 2012.
2. *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.
4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, May 2011.
5. *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.
7. *Evaluation of Severe Accident Risks: Surry Unit 1*, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.
8. Letter from R. J. Barrett (Entergy) to U. S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
11. *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.
13. *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Volume 2, June 1986.
14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA, TR-105189, Final Report, May 1995.
15. *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, NUREG-1150, December 1990.
16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
17. Calculation CNC-1535.00-00-0131, Revision 2, Catawba Nuclear Station, "Catawba PRA Rev. 3b."

18. Calculation CNC-1535.00-00-0192, Revision 1, Catawba Nuclear Station, "Catawba Nuclear Station PRA RAI 03 Response Documentation."

19. Calculation CNC-1535.07-00-0020, Catawba Nuclear Station, "Catawba Nuclear Station Severe Accident Mitigation Design Alternatives (SAMDAs) Analysis for License Renewal."

20. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.

21. Letter from J. A. Hutton (Exelon, Peach Bottom) to U. S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

22. *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.

23. Letter from D. E. Young (Florida Power, Crystal River) to U. S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

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

25. Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.

26. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002.

27. Catawba Nuclear Station Procedure, PT/1/A/4200/01A, "Containment Integrated Leak Rate Test."

28. Letter L-14 -121, ML14111A291, FENOC Evaluation of the Proposed Amendment, Beaver Valley Power Station, Unit Nos. 1 and 2, April 2014.

29. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.

30. ML021580235, Duke Energy Corporation, "One-Time Extension of Integrated Leak Rate Testing (ILRT) Interval," May 29, 2002.

31. Armstrong, J., Simplified Level 2 Modeling Guidelines: WOG PROJECT: PA-RMSC-0088, Westinghouse, WCAP-16341-P, November 2005.

32. IPEEE, "Catawba Nuclear Station: IPEEE Submittal Report," June 21, 1994.

33. Transition Report, "Transition to 10 CFR 50.48(c) - NFPA 805: Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," ML13276A503, ML13276A504, and ML13276A505, September 25, 2013.

34. Calculation CNC-1535.00-00-0154, Revision 1, Catawba Nuclear Station, "CNS High Wind Probabilistic Risk Assessment (HWPRA)."

35. NFPA 805 Implementation Management Update, June 29, 2015.

36. Westinghouse Attachment to LTR-RAM-13-01: "Focused Scope RG 1.200 PRA Review Against ASME/ANS PRA Standard Requirements for the Catawba and McGuire Large Early Release Frequency Probabilistic Risk Assessments," January 2013.

37. Catawba Nuclear Station Internal Flooding PRA Focused Peer Review Report, AREVA NP Inc., February 2013

38. Internal Flooding PRA Peer Review Facts and Observations Resolutions for Catawba Nuclear Station Units 1 and 2, LTR-RAM-II-13-008, June 2013.

39. Calculation CNC-1535.00-00-0151, Revision 1, Catawba Nuclear Station, "Flood PRA Modeling and Quantification for Catawba Nuclear Station Units 1 & 2."

40. Calculation RWA-1430-001, Revision 0, Catawba Nuclear Station, "Catawba Battery Room Temperature Response Following Loss of Ventilation," March 2015.

41. Calculation RWA-1430-002, Revision 0, Catawba Nuclear Station, "Catawba Control Room Temperature Response Following Loss of Ventilation," March 2015.

42. Calculation RWA-1430-003, Revision 0, Catawba Nuclear Station, "Catawba Switch Gear Room Temperature Response Following Loss of Ventilation," February 2015.

43. Calculation CNC-1535.00-00.0118, Revision 0, Catawba Nuclear Station, "Catawba Nuclear Station Success Criteria Notebook," 2010.

44. Letter from Yan Gao (Westinghouse) to U.S. Nuclear Regulatory Commission, LTR-RAM-II-13-077, "Catawba Nuclear Plants RG 1.200 High Wind PRA Peer Review Report," Revision 0, January 2014.

45. SAA Short Form #336, Revision 0, Prepared by A. Mironenko, December 2014.

46. EPRI Technical Report 1016741, "Support System Initiating Events," December 2008.

47. Work Request 20008836, "Replace Fasteners Turbine Building Walls," Unit 1, October 2015.

48. Work Order Package 02118879, "Resecure Fasteners A-Train T.B. W. Wall During OTG," Unit 2, Finished October 7, 2013.

49. Work Order Package 02165104, "2ST: (A-Train) TB Bldg Fastener Securement Project," Unit 2, Finished March 9, 2015.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The technical adequacy of the CNS PRA is either consistent with the requirements of Regulatory Guide 1.200 or where gaps exist, the gaps have been addressed, as is relevant to this ILRT interval extension, as detailed in Attachment 1.
- The CNS Level 1 and Level 2 internal events PRA models provide representative results. The current internal events PRA model (Revision 3b) does not contain a full Level 2 PRA, but previous models contain a full Level 2 PRA. Where detail is needed from a Level 2 PRA, the results from the previous revisions are scaled using the current revision's total risk. It is a reasonable assumption that this scaling does not significantly affect the conclusions of this analysis.
- Even though CNS has two units, there is only one internal events PRA model because the two units are very similar. It is assumed that the two units are similar enough that the one internal events PRA model accurately models both units.
- It is appropriate to use the CNS internal events PRA model to effectively describe the risk change attributable to the ILRT extension. An extensive sensitivity study is done in Section 5.3.1 to show the effect of including external event models for the ILRT extension. The Seismic PRA and Fire PRA (model fire_cr3a_r3v14) are used for this sensitivity analysis.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 2].
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$ based on the previously approved methodology performed for Indian Point Unit 3 [Reference 8, Reference 9].
- The representative containment leakage for Class 3b sequences is $100L_a$ based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (EPRI 1018243) [Reference 24].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [Reference 8, Reference 9].
- 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 in 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.

5.0 METHODOLOGY AND ANALYSIS

5.1 Inputs

This section summarizes the general resources available as input (Section 5.1.1) and the plant specific resources required (Section 5.1.2).

5.1.1 General Resources Available

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

1. NUREG/CR-3539 [Reference 10]
2. NUREG/CR-4220 [Reference 11]
3. NUREG-1273 [Reference 12]
4. NUREG/CR-4330 [Reference 13]
5. EPRI TR-105189 [Reference 14]
6. NUREG-1493 [Reference 6]
7. EPRI TR-104285 [Reference 2]
8. NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]
9. NEI Interim Guidance [Reference 3, Reference 20]
10. Calvert Cliffs liner corrosion analysis [Reference 5]
11. EPRI Report No. 1009325, Revision 2-A (EPRI 1018243), Appendix H [Reference 24]

This 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 is 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 local leak rate test (LLRT) intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the basis for the consequence analysis of the ILRT interval extension for CNS. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

<u>NUREG/CR-3539 [Reference 10]</u>

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [Reference 16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

<u>NUREG/CR-4220 [Reference 11]</u>

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.

NUREG-1273 [Reference 12]

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 [Reference 13]

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 [Reference 14]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains 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 conclusion from the study is that a small, but measurable, safety benefit is realized from extending the test intervals.

NUREG-1493 [Reference 6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals 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 [Reference 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 ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures

4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

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

"...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. For example, for the PWR analyzed, the change is about 0.04 person-rem per year..."

NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec Leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the CNS Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent CNS. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [Reference 3, Reference 20]

The guidance provided in this document builds on the EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [Reference 5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting 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.

EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [Reference 24]

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the CNS assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis, as described in Section 5.2.

5.1.2 Plant Specific Inputs

The plant-specific information used to perform the CNS ILRT Extension Risk Assessment includes the following:

- Level 1 Model results [Reference 17]
- Release category definitions used in the Level 2 Model [Reference 19]
- Dose within a 50-mile radius [Reference 19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 30]

CNS Model

The Internal Events PRA Model that is used for CNS is characteristic of the as-built plant. The current Level 1 model (CNS PRA Model Version cr3b) [Reference 17] is a linked fault tree model. The Internal Flood PRA was updated; the CDF is 3.92E-5/year, and the LERF is 5.42E-7/year [Reference 39]. The total CDF is 5.48E-5/year, and the total LERF is 1.75E-6/year with the updated Internal Flooding analysis [Reference 39]. The results documented in Reference 17 include CDF and LERF contributors from legacy fire and tornado PRAs. These contributors have been superseded by updated Fire PRA and High Wind PRA models (see section 5.3.1 for discussion of the Fire and High Wind PRA results). Therefore, the CDF and LERF from the legacy fire and tornado PRAs are not applicable and have been removed from the total CDF and LERF values that are used in this analysis. See Section 5.2.1 for details of this CDF and LERF removal. Table 5-1 and Table 5-2 provide a summary of the Internal Events CDF and LERF results for CNS PRA Model Version cr3b with the legacy fire and tornado risk removed.

The total Fire CDF is 3.57E-5/year for Unit 1 and 3.64E-5/year for Unit 2; the total Fire LERF is 3.41E-6/year for Unit 1 and 3.48E-6 for Unit 2 [Section 3.0 of Reference 18]. Refer to Section 5.3.1 for further details on external events as they pertain to this analysis.

Table 5-1 – Internal Events CDF (CNS PRA Model Version cr3b)

Internal Events	Frequency (per year)
Internal Floods	3.92E-05
Transients	1.01E-05
LOCAs	2.48E-06
SGTR	4.63E-07
RPV	4.55E-08
ISLOCA	4.36E-07
Total Internal Events CDF	5.27E-05
Total Internal Events CDF (Excluding ISLOCA & SGTR)	5.18E-05

Table 5-2 – Internal Events LERF (CNS PRA Model Version cr3b)

Internal Events	Frequency (per year)
Internal Floods	5.58E-07
Transients	2.40E-07
LOCAs	3.30E-08
SGTR	4.00E-07
RPV	4.73E-10
ISLOCA	4.44E-07
Total Internal Events LERF	1.67E-06

Population Dose Calculations

The population dose calculation was reported in the CNS SAMDA [Reference 19]. Table 5-3 presents dose exposures calculated from methodology described in Reference 1 and data from Reference 30. Reference 30 provides the population dose (person-rem) for Classes 1, 2, 6, 7, and 8; Class 3a and 3b population dose values are calculated from the Class 1 population dose and represented as $10L_a$ and $100L_a$, respectively, as guidance in Reference 1 dictates.

Table 5-3 – Population Dose

Accident Class	Description	Release (person-rem)
1	Containment Remains Intact	1.72E+03
2	Containment Isolation Failures	9.41E+04
3a	Independent or Random Isolation Failures SMALL	1.72E+04[1]
3b	Independent or Random Isolation Failures LARGE	1.72E+05[2]
4	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type B test Failures	n/a
5	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type C test Failures	n/a
6	Isolation Failure that can be verified by IST/IS or surveillance	n/a
7	Containment Failure induced by severe accident	7.69E+05
8	Accidents in which containment is by-passed	8.08E+06[3]

1. $10 * L_a$
2. $100 * L_a$
3. The Class 8 dose value differs from the value presented in Reference 30 because the dose is weighted based on frequency of the two Class 8 contributors: ISLOCA and SGTR.

Release Category Definitions

Table 5-4 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval, as described in Section 5.2 of this report.

Table 5-4 – EPRI Containment Failure Classification [Reference 2]

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 Individual Plant Examinations) including 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 including 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 test and their potential failures.
6	Containment isolation failures including 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.

5.1.3 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 liner breach, failure of certain bellows arrangements, 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 addressed, the EPRI Class 3 accident class, as defined in Table 5-3, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures respectively.

The probability of the EPRI Class 3a and Class 3b failures is determined consistent with the EPRI Guidance [Reference 24]. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to "large" failures in 217 tests (i.e., 2 / 217 = 0.0092). For Class 3b, the probability is based on the Jeffreys non-informative prior (i.e., 0.5 / 218 = 0.0023).

In a follow-up letter [Reference 20] to their ILRT guidance document [Reference 3], NEI issued additional 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 [Reference 4]. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for ΔLERF. NEI describes ways to demonstrate that, using plant-specific calculations, the ΔLERF is smaller than that calculated by the simplified method.

The supplemental information 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 CNS, as detailed in Section 5.2, involves subtracting the LERF 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.

Consistent with the NEI Guidance [Reference 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 years / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 years / 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. Correspondingly, an extension of the ILRT interval to 15 years can be estimated to lead to a factor of 5 ((15/2)/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [Reference 9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.2 Analysis

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H [Reference 24], EPRI TR-104285 [Reference 2] and previous risk assessment submittals on this subject [References 5, 8, 21, 22, and 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report, as described in Table 5-5.

The analysis performed examined CNS-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown 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 TR-104285, Class 1 sequences [Reference 2]).
- 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 bellow leakage (EPRI TR-104285, Class 3 sequences [Reference 2]).
- Accident sequences involving containment bypassed (EPRI TR-104285, Class 8 sequences [Reference 2]), large containment isolation failures (EPRI TR-104285, Class 2 sequences [Reference 2]), and small containment isolation "failure-to-seal" events (EPRI TR-104285, Class 4 and 5 sequences [Reference 2]) 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.

Table 5-5 – EPRI Accident Class Definitions

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
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	Bypass (Interfacing System LOCA)
CDF	All CET End States (Including Very Low and No Release)

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

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the accident classes presented in Table 5-5.

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 in 10 years to 1 in 15 years and 1 in 10 years to 1 in 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [Reference 4].

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP).

5.2.1 Step 1 – Quantify the Baseline Risk in Terms of Frequency per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285 [Reference 2].) The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-5 were developed for CNS by first determining the frequencies for Classes 1, 2, 6, 7, and 8. Table 5-6 presents the grouping of each release category in EPRI Classes based on the associated description. Table 5-7 presents the frequency and EPRI category for each sequence and the totals of each EPRI classification. Table 5-8 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the definitions of accident classes defined in EPRI TR-104285 [Reference 2], the NEI Interim Guidance [Reference 3], and guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24]. Adjustments were made to the Class 3b and hence Class 1 frequencies to account for the

impact of undetected corrosion of the steel liner per the methodology described in Section 5.2.6. Note: calculations were performed with more digits than shown in this section. Therefore, minor differences may occur if the calculations in these sections are followed explicitly.

The Catawba PRA Rev. 3b model [Reference 17] contains risk contribution from internal events and external events (fire and tornadoes). For the baseline analysis in Section 5.2, only internal events will be addressed. External events (fire, tornadoes, and seismic) will be addressed in a sensitivity in Section 5.3.1 using updated PRA models. Using the total CDF of 5.48E-5 and finding the contribution from the Fussell-Vesely (FV) importance measure for fire (initiators %FCBLR, %FCR, %FDG, %FETB, and %FKC) to total 3.1% and tornadoes (initiator %TORNSW) to be 0.8% for CDF, the fire CDF contribution is 1.72E-6 and the tornado CDF contribution is 4.20E-7. Using the total LERF of 1.75E-6, the total fire FV of 2.2%, and the tornado FV of 2.2%, the fire LERF contribution is 3.79E-8 and the tornado LERF contribution is 3.80E-8. Therefore, for this analysis the CDF is 5.27E-5, and the LERF is 1.67E-6.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists that can only be detected by performing a Type A ILRT. The probability of leakage detectable by a Type A ILRT is calculated to determine the impact of extending the testing interval. The Class 3 calculation is divided into two classes: Class 3a is defined as a small liner breach (L_a < leakage < $10L_a$), and Class 3b is defined as a large liner breach ($10L_a$ < leakage < $100L_a$).

Data reported in EPRI 1009325, Revision 2-A [Reference 24] states that two events could have been detected only during the performance of an ILRT and thus impact risk due to change in ILRT frequency. There were a total of 217 successful ILRTs during this data collection period. Therefore, the probability of leakage is determined for Class 3a as shown in the following equation:

$$P_{class3a} = \frac{2}{217} = 0.0092$$

Multiplying the CDF by the probability of a Class 3a leak yields the Class 3a frequency contribution in accordance with guidance provided in Reference 24. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Therefore, LERF contributions from CDF are removed. The frequency of a Class 3a failure is calculated by the following equation:

$$Freq_{class3a} = P_{class3a} * (CDF - LERF)$$
$$= \frac{2}{217} * (5.27\text{E-}5 - 1.67\text{E-}6) = 4.70\text{E-}7$$

In the database of 217 ILRTs, there are zero containment leakage events that could result in a large early release. Therefore, the Jeffreys non-informative prior is used to estimate a failure rate and is illustrated in the following equations:

$$\text{Jeffreys Failure Probability} = \frac{Number\ of\ Failures + 1/2}{Number\ of\ Tests + 1}$$

$$P_{class3b} = \frac{0 + 1/2}{217 + 1} = 0.0023$$

The frequency of a Class 3b failure is calculated by the following equation:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF)$$
$$= \frac{.5}{218} * (5.27\text{E-}5 - 1.67\text{E-}6) = 1.17\text{E-}7$$

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is

100L_a. These assignments are consistent with the guidance provided in Reference 24.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). Since the PRA model does not contain a Level 2 model, Class 1 is calculated as CDF – LERF. This overestimates the Intact frequency, which is conservative for this analysis because it leads to a higher calculated change in risk due to extending the ILRT frequency. The frequency per year is initially determined from the EPRI Accident Class 1 frequency listed in Table 5-7 and then subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$$Freq_{class1} = Freq_{class1} - (Freq_{class3a} - Freq_{class3b})$$

Class 2 Sequences. This group consists of core damage accident progression bins with large containment isolation failures. This is calculated by ANDing the ZL gate (Large Containment Isolation Failure) with a flag to calculate the contribution of large containment isolation failure to LERF. Since this flag is in cutsets that contribute 0.214% of LERF, which is 1.67E-6, the Class 2 contribution is 3.58E-9. The frequency per year for these sequences is obtained from the EPRI Accident Class 2 frequency listed in Table 5-7.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. 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 the analysis, consistent with approved methodology.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components occurs. Because the 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, consistent with approved methodology.

Class 6 Sequences. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. All other failure modes are bounded by the Class 2 assumptions. This accident class is also not evaluated further.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). This frequency is calculated by subtracting the Class 1, 2, and 8 frequencies from the total CDF. For this analysis, the frequency is determined from the EPRI Accident Class 7 frequency listed in Table 5-7.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass or SGTR occur, which contribute 25.8% and 23.3%, respectively, of LERF; LERF frequencies are shown in Table 5-6. For this analysis, the total frequency is listed in Table 5-7.

LERF quantification is distributed into EPRI categories based on release categories. Table 5-5 shows this distribution.

Table 5-6 – Release Category Frequencies

Containment End State	EPRI Category	Frequency (/yr)
Intact Containment	1	5.10E-05
Large Isolation Failure	2	3.58E-09
Failures Induced by Phenomena	7	8.49E-07
Other Containment Bypass	8	4.32E-07
SGTR	8	3.90E-07

Table 5-7 – Accident Class Frequencies

EPRI Category	Frequency (/yr)
Class 1	5.10E-05
Class 2	3.58E-09
Class 6	N/A – Included in Class 2
Class 7	8.49E-07
Class 8	8.22E-07
Total (CDF)	5.27E-05

Table 5-8 – Baseline Risk Profile

Class	Description	Frequency (/yr)
1	No containment failure	5.04E-05[2]
2	Large containment isolation failures	3.58E-09
3a	Small isolation failures (liner breach)	4.70E-07
3b	Large isolation failures (liner breach)	1.17E-07
4	Small isolation failures - failure to seal (type B)	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]
7	Severe accident phenomena induced failure (early and late)	8.49E-07
8	Containment bypass	8.22E-07
	Total	5.27E-05

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 3a and 3b frequencies are subtracted from Class 1 to preserve total CDF.

5.2.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose)

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on CNS-specific dose calculations summarized in Table 5-3. Table 5-3 provides a correlation of CNS population dose to EPRI Accident Class. Table 5-10 provides population dose for each EPRI accident class.

The population dose for EPRI Accident Classes 3a and 3b were calculated based on the guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24] as follows:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 1.72E{+}3 = 1.72E{+}4$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 1.72E{+}3 = 1.72E{+}5$$

Table 5-9 – Mapping of Population Dose to EPRI Accident Class

EPRI Category	Frequency (/yr)	Dose (person-rem)
Class 1	5.04E-05	1.72E+03
Class 2	3.58E-09	9.41E+04
Class 6	N/A – Included in Class 2	
Class 7	8.49E-07	7.69E+05
Class 8	8.22E-07	8.08E+06[1]

1. The Class 8 dose value differs from the value presented in Reference 30 because the dose is weighted based on frequency of the two Class 8 contributors: ISLOCA and SGTR.

Table 5-10 – Baseline Population Doses

Class	Description	Population Dose (person-rem)
1	No containment failure	1.72E+03
2	Large containment isolation failures	9.41E+04
3a	Small isolation failures (liner breach)	1.72E+04[1]
3b	Large isolation failures (liner breach)	1.72E+05[2]
4	Small isolation failures - failure to seal (type B)	N/A
5	Small isolation failures - failure to seal (type C)	N/A
6	Containment isolation failures (dependent failure, personnel errors)	N/A
7	Severe accident phenomena induced failure (early and late)	7.69E+05
8	Containment bypass	8.08E+06[3]

1. $10*L_a$
2. $100*L_a$
3. The Class 8 dose value differs from the value presented in Reference 30 because the dose is weighted based on frequency of the two Class 8 contributors: ISLOCA and SGTR.

5.2.3 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 10-year interval to a 15-year interval. To do this, an evaluation must first be made of the risk associated with the 10-year interval, since the base case applies to 3-year interval (i.e., a simplified representation of a 3-to-10 interval).

Risk Impact Due to 10-Year Test Interval

As previously stated, 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). Thus, only the frequency of Class 3a and Class 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 5.1.3 by a factor of 10/3 compared to the base case values. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 5.10\text{E-}5 = 1.57\text{E-}6$$

$$Freq_{Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 5.10\text{E-}5 = 3.90\text{E-}7$$

The results of the calculation for a 10-year interval are presented in Table 5-11.

Table 5-11 – Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	4.90E-05	93.11%	1.72E+03	8.43E-02
2	Large containment isolation failures	3.58E-09	0.01%	9.41E+04	3.37E-04
3a	Small isolation failures (liner breach)	1.57E-06	2.97%	1.72E+04	2.69E-02
3b	Large isolation failures (liner breach)	3.90E-07	0.74%	1.72E+05	6.70E-02
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	8.49E-07	1.61%	7.69E+05	6.52E-01
8	Containment bypass	8.22E-07	1.56%	8.08E+06	6.64E+00
	Total	5.27E-05			7.48E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

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 leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5 compared to the 3-year interval value, as described in Section 5.1.3. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 5.10\text{E-}5 = 2.35\text{E-}6$$

$$Freq_{Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 5.10\text{E-}5 = 5.85\text{E-}7$$

The results of the calculation for a 15-year interval are presented in Table 5-12.

Table 5-12 – Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	4.80E-05	91.25%	1.72E+03	8.26E-02
2	Large containment isolation failures	3.58E-09	0.01%	9.41E+04	3.37E-04
3a	Small isolation failures (liner breach)	2.35E-06	4.46%	1.72E+04	4.04E-02
3b	Large isolation failures (liner breach)	5.85E-07	1.11%	1.72E+05	1.01E-01
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	8.49E-07	1.61%	7.69E+05	6.52E-01
8	Containment bypass	8.22E-07	1.56%	8.08E+06	6.64E+00
	Total	5.27E-05			7.52E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

5.2.4 Step 4 – Determine the Change in Risk in Terms of LERF

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could, in fact, result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 [Reference 4] defines very small changes in risk as resulting in increases of CDF less than 10^{-6}/year and increases in LERF less than 10^{-7}/year, and small changes in LERF as less than 10^{-6}/year. Since containment overpressure is not required in support of ECCS performance to mitigate design basis accidents at CNS, the ILRT extension does not impact CDF. Therefore, the relevant risk-impact metric is LERF.

For CNS, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a 10-year test interval from Table 5-11, the Class 3b frequency is 3.90E-07/year; based on a 15-year test interval from Table 5-12, the Class 3b frequency is 5.85E-07/year. 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 is 4.68E-07/year. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.95E-07/year. As can be seen, even with the conservatisms included in the

evaluation (per the EPRI methodology), the estimated change in LERF is within the criteria for a small change when comparing the 15-year results to the current 10-year requirement and the original 3-year requirement. Table 5-13 summarizes these results.

Table 5-13 – Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	1.17E-07	3.90E-07	5.85E-07
ΔLERF (3 year baseline)		2.73E-07	4.68E-07
ΔLERF (10 year baseline)			1.95E-07

The increase in the overall probability of LERF due to Class 3b sequences is greater than 10^{-7}. As stated in RG 1.174 [Reference 4], "When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year." Baseline LERF (excluding external events) is 1.67E-6/year (1.75E-6/year if fire and tornado are included). Therefore, there is significant margin for both the ΔLERF and baseline LERF to the upper limits of Region II in RG 1.174 [Reference 4].

5.2.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability

Another parameter that the NRC guidance in RG 1.174 [Reference 4] states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The CCFP is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \frac{f(ncf)}{CDF}$$

where *f*(ncf) is the frequency of those sequences that do not result in containment failure; this frequency is determined by summing the Class 1 and Class 3a results [Reference 24]. Table 5-14 shows the steps and results of this calculation. The difference in CCFP between the 3-year test interval and 15-year test interval is 0.888%.

Table 5-14 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	5.09E-05	5.06E-05	5.04E-05
f(ncf)/CDF	0.966	0.961	0.957
CCFP	0.0340	0.0392	0.0429
ΔCCFP (3 year baseline)		0.518%	0.888%
ΔCCFP (10 year baseline)			0.370%

As stated in Section 2.0, a change in the CCFP of up to 1.5% is assumed to be small. The increase in the CCFP from the 3 in 10 year interval to 1 in 15 year interval is 0.888%. Therefore, this increase is judged to be very small.

5.2.6 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using a methodology similar to the Calvert Cliffs liner corrosion analysis [Reference 5]. The Calvert

Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. 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 steel liner 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

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-15, Step 1).
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs previous analysis are assumed to still be applicable.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis) (See Table 5-4, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel liner 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 liner ages (See Table 5-15, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere, given that a liner flaw exists, was estimated as 1.1% for the cylinder and dome, 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. For CNS, the ILRT is performed at or slightly below the design pressure of 15 psig [Reference 27]. Probabilities of 1% for the cylinder and dome, and 0.1% for the basemat are used in this analysis, and sensitivity studies are included in Section 5.3.2 (See Table 5-15, Step 4).
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region (See Table 5-15, 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. To date, all liner corrosion events have been detected through visual inspection (See Table 5-15, Step 5).
- 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 5-15 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (85%)		Containment Basemat (15%)	
1	Historical liner flaw likelihood Failure data: containment location specific Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces	Events: 2 (Brunswick 2 and North Anna 2) 2 / (70 x 5.5) = 5.19E-03		Events: 0 Assume a half failure 0.5 / (70 x 5.5) = 1.30E-03	
2	Aged adjusted liner flaw likelihood During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.	Year	Failure rate	Year	Failure rate
		1	2.05E-03	1	5.13E-04
		average 5-10	5.19E-03	average 5-10	1.30E-03
		15	1.43E-02	15	3.57E-03
		15 year average = 6.44E-03		15 year average = 1.61E-03	
3	Increase in flaw likelihood between 3 and 15 years Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.	0.73% (1 to 3 years) 4.18% (1 to 10 years) 9.66% (1 to 15 years)		0.18% (1 to 3 years) 1.04% (1 to 10 years) 2.41% (1 to 15 years)	
4	Likelihood of breach in containment given liner flaw	1%		0.1%	
5	Visual inspection detection failure likelihood	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 x 4 x 5)	0.00073% (3 years) 0.73% x 1% x 10% 0.00418% (10 years) 4.18% x 1% x 10% 0.00966% (15 years) 9.66% x 1% x 10%		0.000180% (3 years) 0.18% x 0.1% x 100% 0.00104% (10 years) 1.04% x 0.1% x 100% 0.00241% (15 years) 2.41% x 0.1% x 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, as summarized below for CNS.

Table 5-16 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for CNS

Description
At 3 years: 0.00073% + 0.000180% = 0.00091%
At 10 years: 0.00418% + 0.00104% = 0.00522%
At 15 years: 0.00966% + 0.00241% = 0.01207%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF.

The two corrosion events that were initiated from the non-visible (backside) portion of the containment liner used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this containment analysis. These events, one at North Anna Unit 2 (September 1999) caused by timber embedded in the concrete immediately behind the containment liner, and one at Brunswick Unit 2 (April 1999) caused by a cloth work glove embedded in the concrete next to the liner, were initiated from the nonvisible (backside) portion of the containment liner. A search of the NRC website LER database identified two additional events have occurred since the Calvert Cliffs analysis was performed. In January 2000, a 3/16-inch circular through-liner hole was found at Cook Nuclear Plant Unit 2 caused by a wooden brush handle embedded immediately behind the containment liner. The other event occurred in April 2009, where a through-liner hole approximately 3/8-inch by 1-inch in size was identified in the Beaver Valley Power Station Unit 1 (BVPS-1) containment liner caused by pitting originating from the concrete side due to a piece of wood that was left behind during the original - construction that came in contact with the steel liner. Two other containment liner through-wall hole events occurred at Turkey Point Units 3 and 4 in October 2010 and November 2006, respectively. However, these events originated from the visible side caused by the failure of the coating system, which was not designed for periodic immersion service, and are not considered to be applicable to this analysis. More recently, in October 2013, some through-wall containment liner holes were identified at BVPS-1, with a combined total area of approximately 0.395 square inches. The cause of these through-wall liner holes was attributed to corrosion originating from the outside concrete surface due to the presence of rayon fiber foreign material that was left behind during the original construction and was contacting the steel liner [Reference 28]. For risk evaluation purposes, these five total corrosion events occurring in 66 operating plants with steel containment liners over a 17.1 year period from September 1996 to October 4, 2013 (i.e., 5/(66*17.1) = 4.43E-03) are bounded by the estimated historical flaw probability based on the two events in the 5.5 year period of the Calvert Cliffs analysis (i.e., 2/(70*5.5) = 5.19E-03) incorporated in the EPRI guidance.

5.3 Sensitivities

5.3.1 Potential Impact from External Events Contribution

An assessment of the impact of external events is performed. The primary purpose for this investigation is the determination of the total LERF following an increase in the ILRT testing interval from 3 in 10 years to 1 in 15 years.

Catawba is transitioning to NFPA 805 licensing basis for fire protection and submitted a License Amendment Request (LAR) [Reference 33]. This transition includes performing a Fire PRA and installing modifications to reduce the fire-induced CDF and LERF. It is anticipated that many, but not all, of the Fire PRA related modifications will be completed by the next scheduled ILRT. The next scheduled ILRTs for the two units are fall 2016 and spring 2018. Compensatory measures have been implemented to reduce the fire risk until the modifications that reduce the Fire PRA CDF and LERF are implemented. These measures may be actions to reduce fire initiating event probabilities, actions to improve suppression probability, and/or actions to recover or protect systems that mitigate core damage and large early release accident sequences. Therefore, the Fire PRA model is deemed applicable for this calculation.

The Fire PRA model fire_cr3a_r3v14 was used to obtain the fire CDF and LERF values [Reference 18]. To reduce conservatism in the model, the methodology of subtracting existing

LERF from CDF is also applied to the Fire PRA model. The following shows the calculation for Class 3b:

$$Freq_{U1class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (3.57E\text{-}5 - 3.41E\text{-}6) = 7.41E\text{-}8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (3.64E\text{-}5 - 3.48E\text{-}6) = 7.55E\text{-}8$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (3.57E\text{-}5 - 3.41E\text{-}6) = 2.47\text{E-}7$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (3.64E\text{-}5 - 3.48E\text{-}6) = 2.52\text{E-}7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (3.57E\text{-}5 - 3.41E\text{-}6) = 3.70\text{E-}7$$

$$Freq_{U2class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (3.64E\text{-}5 - 3.48E\text{-}6) = 3.78\text{E-}7$$

The Seismic PRA results from the IPEEE Seismic PRA estimate a CDF of 1.6E-5/year [Reference 32]. The cr3b model contains a Seismic PRA model; when only the seismic portion of this model is quantified (cr3b_seismic.cut), the CDF is 1.15E-5. Applying the internal event LERF/CDF ratio to the seismic CDF yields an estimated seismic LERF of 3.62E-7, as shown by the equation below.

$$\text{LERF}_{\text{Seismic}} \approx \text{CDF}_{\text{Seismic}} * \text{LERF}_{\text{IE}} / \text{CDF}_{\text{IE}} = 1.15\text{E-}5 * 1.67\text{E-}6 / 5.27\text{E-}5 = 3.62\text{E-}7$$

Subtracting seismic LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.15\text{E-}5 - 3.62\text{E-}7) = 2.55\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.15\text{E-}5 - 3.62\text{E-}7) = 8.52\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = \frac{15}{3} * \frac{0.5}{218} * (1.15\text{E-}5 - 3.62\text{E-}7) = 1.28\text{E-}7$$

As stated in Section 7.6 of the high wind (HW) PRA [Reference 34], the CDF and LERF are 7.02E-6 and 6.48E-7, respectively. At the time of the original HW PRA development, many of the siding panels on the Turbine Building were not sufficiently fastened to provide full protection against high winds, resulting in relatively high fragilities. This had a considerable effect on the CDF and LERF. Analyses were performed on the fragilities of equipment within the Turbine Building and the CNS Main Transformers, which were shown to be vulnerable to siding impact. As evidenced by the completed work orders to replace the wall fasteners on the Turbine Building, Catawba has since sufficiently fastened the siding panels [References 47-49]. Therefore, the model with the fasteners fixed is used for this application.

Subtracting HW LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (7.02\text{E-}6 - 6.48\text{E-}7) = 1.46\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (7.02\text{E-}6 - 6.48\text{E-}7) = 4.87\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = \frac{15}{3} * \frac{0.5}{218} * (7.02\text{E-}6 - 6.48\text{E-}7) = 7.31\text{E-}8$$

The fire, seismic, and high wind contributions to Class 3b frequencies are then combined to obtain the total external event contribution to Class 3b frequencies. The change in LERF is calculated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Table 5-17.

Table 5-17 – Unit 1 CNS External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	1.14E-07	3.81E-07	5.71E-07	4.57E-07
Internal Events	1.17E-07	3.90E-07	5.85E-07	4.68E-07
Combined	2.31E-07	7.70E-07	1.16E-06	9.25E-07

Table 5-18 – Unit 2 CNS External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	1.16E-07	3.86E-07	5.78E-07	4.63E-07
Internal Events	1.17E-07	3.90E-07	5.85E-07	4.68E-07
Combined	2.33E-07	7.75E-07	1.16E-06	9.30E-07

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined, the total change in LERF of 9.25E-7 for Unit 1 and 9.30E-7 for Unit 2 meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than 1.0E-6 change in LERF. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5. The total LERF value is calculated below:

$$\mathrm{LERF_{U1}} = \mathrm{LERF_{internal}} + \mathrm{LERF_{seismic}} + \mathrm{LERF_{U1fire}} + \mathrm{LERF_{HW}} + \mathrm{LERF_{class3Bincrease}}$$
$$= 1.67\text{E-}6/\mathrm{yr} + 3.66\text{E-}7/\mathrm{yr} + 3.41\text{E-}6/\mathrm{yr} + 6.48\text{E-}7/\mathrm{yr} + 9.25\text{E-}7/\mathrm{yr} = 7.02\text{E-}6/\mathrm{yr}$$

$$\mathrm{LERF_{U2}} = \mathrm{LERF_{internal}} + \mathrm{LERF_{seismic}} + \mathrm{LERF_{U2fire}} + \mathrm{LERF_{HW}} + \mathrm{LERF_{class3Bincrease}}$$
$$= 1.67\text{E-}6/\mathrm{yr} + 3.66\text{E-}7/\mathrm{yr} + 3.48\text{E-}6/\mathrm{yr} + 6.48\text{E-}7/\mathrm{yr} + 9.30\text{E-}7/\mathrm{yr} = 7.10\text{E-}6/\mathrm{yr}$$

Although the total change in LERF is somewhat close to the Regulatory Guide 1.174 limit [Reference 4] when external event risk is included, several conservative assumptions were made in this ILRT analysis, as discussed in Sections 4.0, 5.1.3, 5.2.1, and 5.2.4; therefore the total change in LERF is considered conservative for this application. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.3.2 Potential Impact from Steel Liner Corrosion Likelihood

A quantitative assessment of the contribution of steel liner corrosion likelihood impact was performed for the risk impact assessment for extended ILRT intervals. As a sensitivity run, the internal event CDF was used to calculate the Class 3b frequency. The impact on the Class 3b frequency due to increases in the ILRT surveillance interval was calculated for steel liner corrosion likelihood using the relationships described in Section 5.2.6. The EPRI Category 3b frequencies for the 3 per 10-year, 10-year, and 15-year ILRT intervals were quantified using the

internal events CDF. The change in the LERF, change in CCFP, and change in Annual Dose Rate due to extending the ILRT interval from 3 in 10 years to 1 in 10 years, or to 1 in 15 years are provided in Table 5-19 – Table 5-21. Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without containment spray refinement is used to calculate the CCFP. The Annual Dose Rate calculations are performed using the containment spray adjustments. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, the corrosion likelihood is relatively insensitive to the results.

Table 5-19 – Steel Liner Corrosion Sensitivity Case: 3B Contribution

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
Internal Event 3B Contribution	1.17E-07	3.90E-07	5.85E-07	2.73E-07	4.68E-07	1.95E-07
Corrosion Likelihood X 1000	2.71E-10	9.41E-10	1.50E-09	6.70E-10	1.23E-09	5.62E-10
Corrosion Likelihood X 10000	2.93E-10	1.36E-09	2.96E-09	1.07E-09	2.67E-09	1.60E-09
Corrosion Likelihood X 100000	5.12E-10	5.56E-09	1.75E-08	5.05E-09	1.70E-08	1.20E-08

Table 5-20 – Steel Liner Corrosion Sensitivity: CCFP

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	3.40E-02	3.92E-02	4.29E-02	5.18E-03	8.88E-03	3.70E-03
Corrosion Likelihood X 1000	3.40E-02	3.93E-02	4.30E-02	5.23E-03	8.96E-03	3.73E-03
Corrosion Likelihood X 10000	3.42E-02	3.99E-02	4.39E-02	5.65E-03	9.69E-03	4.04E-03
Corrosion Likelihood X 100000	3.60E-02	4.59E-02	5.30E-02	9.90E-03	1.70E-02	7.07E-03

Table 5-21 – Steel Liner Corrosion Sensitivity: Dose Rate

	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Dose Rate	2.01E-02	6.70E-02	1.01E-01	4.69E-02	8.05E-02	3.35E-02
Corrosion Likelihood X 1000	2.03E-02	7.05E-02	1.13E-01	5.02E-02	9.24E-02	4.22E-02
Corrosion Likelihood X 10000	2.19E-02	1.02E-01	2.22E-01	8.01E-02	2.00E-01	1.20E-01
Corrosion Likelihood X 100000	3.84E-02	4.17E-01	1.31E+00	3.79E-01	1.28E+00	8.97E-01

5.3.3 Expert Elicitation Sensitivity

Another sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in Reference 24. In this sensitivity case, an expert elicitation was conducted to develop probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data.

Using the expert knowledge, this information was extrapolated into a probability-versus-magnitude relationship for pre-existing containment defects [Reference 24]. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in Reference 24. 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 expert elicitation results are used to develop sensitivity cases for the risk impact assessment. Employing the results requires the application of the ILRT interval methodology using the expert elicitation to change the probability of pre-existing leakage in the containment.

The baseline assessment uses the Jeffreys non-informative prior and the expert elicitation sensitivity study uses the results of 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 basic methodology (i.e., 10 L_a for small and 100 L_a for large) are used here. Table 5-22 presents the magnitudes and probabilities associated with the Jeffreys non-informative prior and the expert elicitation used in the base methodology and this sensitivity case.

Table 5-22 – CNS Summary of ILRT Extension Using Expert Elicitation Values (from Reference 24)

Leakage Size (L_a)	Jeffreys Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.70E-02	3.88E-03	86%
100	2.70E-03	9.86E-04	64%

Taking the baseline analysis and using the values provided in Table 5-10 – Table 5-12 for the expert elicitation sensitivity yields the results in Table 5-23.

Table 5-23 – CNS Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	5.10E-05	5.07E-05	1.72E+03	8.73E-02	5.02E-05	8.63E-02	4.97E-05	8.56E-02
2	3.58E-09	3.58E-09	9.41E+04	3.37E-04	3.58E-09	3.37E-04	3.58E-09	3.37E-04
3a	N/A	1.98E-07	1.72E+04	3.40E-03	6.59E-07	1.13E-02	9.89E-07	1.70E-02
3b	N/A	5.03E-08	1.72E+05	8.65E-03	1.68E-07	2.88E-02	2.51E-07	4.32E-02
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	8.49E-07	8.49E-07	7.69E+05	6.52E-01	8.49E-07	6.52E-01	8.49E-07	6.52E-01
8	8.22E-07	8.22E-07	8.08E+06	6.64E+00	8.22E-07	6.64E+00	8.22E-07	6.64E+00
Totals	5.27E-05	5.27E-05	9.14E+06	7.40E+00	5.27E-05	7.42E+00	5.27E-05	7.44E+00
ΔLERF (3 per 10 yrs base)	N/A				1.17E-07		2.01E-07	
ΔLERF (1 per 10 yrs base)	N/A				N/A		8.38E-08	
CCFP	3.27%				3.50%		3.66%	

The results illustrate how the expert elicitation reduces the overall change in LERF and the overall results are more favorable with regard to the change in risk.

5.3.4 Large Leak Probability Sensitivity Study

The large leak probability is a vital portion of determining the Class 3b frequency. CNS had previously calculated the large leak probability using the WCAP method. Table 5-24 presents the large leak probabilities for the baseline test, 10 year test interval, and 15 year test interval. Table 5-24 was developed using the same process as to calculate Class 3b.

Table 5-24 – CNS Large Leak Probabilities Using the WCAP Method

Test Interval	WCAP Large Leak Probability	EPRI Accident Class 3b Frequency
3 per 10 years	2.47E-4	1.26E-08
10 years	7.41E-4	3.78E-08
15 years	1.11E-3	5.66E-08

Using the same EPRI approach, but with an updated Class 3b frequency calculated from the WCAP large leak probability data, Table 5-25 contains the final results.

Table 5-25 – Impact on LERF due to Extended Type A Testing Intervals with WCAP CDF

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	1.26E-08	3.78E-08	5.66E-08
ΔLERF (3 year baseline)		2.52E-08	4.40E-08
ΔLERF (10 year baseline)			1.88E-08

These results demonstrate that the EPRI methodology is conservative when used to calculate a large leak probability as compared to the WCAP method.

5.3.5 SGTR Success Criteria

In response to F&O AS-04, a sensitivity was performed to conservatively change the PRA model logic to approximate the SGTR success criteria changes necessary to reconstruct the SGTR portion of the model. Changes in Success Criteria modeling are based on guidance provided in Reference 43. Since no top logic was deleted from the model (failure logic was added), the risk can only increase as a result of this sensitivity. Under the SGTR Containment Bypass logic, OR gate G086 was added to add new logic; under gate G086, three OR gates (G067, G087, G099) were added to represent three sets of accident scenarios that lead to core damage following a SGTR initiating event. Under gate G067 is AND gates G063, G068, and G073; under gate G087 is AND gates G088 and G089; and under gate G099 is AND gates G100 and G101 with scenario failure logic. Table 5-26 describes the scenario failure logic (and model gates) for these seven gates.

Table 5-26 – Added SGTR Failure Scenarios

Gate	SGTR Scenario Description
G063	SSHR Success; Failure of SG Isolation (YOLARGE), High Pressure Injection (YU), SG Depressurization (YD3SG)
G068	SSHR Success; Failure of SG Isolation (YOLARGE), High Pressure Injection (YU), RHR via Shutdown Cooling (SC-LX[1])
G073	SSHR Success; Failure of SG Isolation (YOLARGE), FWST Refill (ND0RWSTDHE), Primary System Depressurization using Sprays (YD1) or Pressurizer PORV (YD2)
G088	SSHR and SG Isolation Success; Failure of High Pressure Injection (YU), Primary System Depressurization using Sprays (YD1) or Pressurizer PORV (YD2), SG Depressurization (YD3SG)
G089	SSHR and SG Isolation Success; Failure of FWST Refill (ND0RWSTDHE), Primary System Depressurization using Sprays (YD1) or Pressurizer PORV (YD2), SG Depressurization (YD3SG)
G100	Failure of SSHR (F1), Feed and Bleed (SBPU10, YU)
G101	Failure of SSHR (F1), High Pressure Sump Recirculation (SX03)

1. Gate SC-LX is replicated from LX; the only difference is the gates for RHR suction from the sump (L107R, L207R) are replaced with suction from the corresponding hot leg (loop C hot leg for ND pump B, loop B hot leg for ND pump A).

Total CDF increases by less than 1%; therefore, there is only a negligible change in Class 3b frequency. Total LERF increases significantly. Therefore, the only significant effect this sensitivity has on the ILRT extension application is to the overall LERF criteria for Region II in RG 1.174 [Reference 4]. This sensitivity results in a LERF increase of 3.0E-7/year. Therefore, baseline LERF (excluding external events) is 1.97E-6/year (2.05E-6/year if fire and tornado risk from the original quantification is included [Reference 17]), and there is still significant margin for both the ΔLERF and baseline LERF to the upper limits of Region II in RG 1.174 [Reference 4].

6.0 RESULTS

The results from this ILRT extension risk assessment for CNS are summarized in Table 6-1.

Table 6-1 – ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	1.72E+03	5.04E-05	8.67E-02	4.90E-05	8.43E-02	4.80E-05	8.26E-02
2	9.41E+04	3.58E-09	3.37E-04	3.58E-09	3.37E-04	3.58E-09	3.37E-04
3a	1.72E+04	4.70E-07	8.08E-03	1.57E-06	2.69E-02	2.35E-06	4.04E-02
3b	1.72E+05	1.17E-07	2.01E-02	3.90E-07	6.70E-02	5.85E-07	1.01E-01
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	7.69E+05	8.49E-07	6.52E-01	8.49E-07	6.52E-01	8.49E-07	6.52E-01
8	8.08E+06	8.22E-07	6.64E+00	8.22E-07	6.64E+00	8.22E-07	6.64E+00
Total		**5.27E-05**	**7.41E+00**	**5.27E-05**	**7.48E+00**	**5.27E-05**	**7.52E+00**

ILRT Dose Rate from 3a and 3b				
ΔTotal Dose Rate	From 3 Years	N/A	6.34E-02	1.09E-01
	From 10 Years	N/A	N/A	4.53E-02
%ΔDose Rate	From 3 Years	N/A	0.86%	1.47%
	From 10 Years	N/A	N/A	0.61%
3b Frequency (LERF)				
ΔLERF	From 3 Years	N/A	2.73E-07	4.68E-07
	From 10 Years	N/A	N/A	1.95E-07
CCFP %				
ΔCCFP%	From 3 Years	N/A	0.518%	0.888%
	From 10 Years	N/A	N/A	0.370%

7.0 CONCLUSIONS AND RECOMMENDATIONS

Based on the results from Section 5.2 and the sensitivity calculations presented in Section 5.3, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to 15 years:

- Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF less than 1.0E-06/year and increases in LERF less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 4.68E-07/year using the EPRI guidance (this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included), and baseline LERF is 1.67E-6. As such, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4].
- The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.109 person-rem/year. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that a very small population dose is defined 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. The results of this calculation meet these criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure probability from the 3 in 10 year interval to 1 in 15 year interval is 0.888%. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that increases in CCFP of ≤ 1.5% is very small. Therefore, this increase is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the CNS risk profile.

Previous Assessments

The NRC in NUREG-1493 [Reference 6] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 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 or Type 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 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for CNS confirm these general findings on a plant-specific basis considering the severe accidents evaluated for CNS, the CNS containment failure modes, and the local population surrounding CNS.

A. ATTACHMENT 1

A.1. Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension

The CNS internal events PRA model (Revision 3b) is used to calculate CDF and LERF for the permanent 15-year ILRT extension. Any elements of the supporting requirements detailed in ASME/ANS RA-Sa-2009 that could be significantly affected by the application are required to meet Capability Category II requirements.

The internal events PRA provides an adequate base model for the development of the permanent 15-year ILRT extension. In accordance with RG 1.200, the most recent full scope CNS Internal Events PRA Peer Review was performed in March 2002 using the peer review process described in NEI 00-02 (Attachment U of Reference 33). More recently, focused scope peer reviews have been conducted on the CNS LERF PRA model and the CNS Internal Flooding PRA model. The results from these focus scope peer reviews are discussed in section A.1.1 for LERF and A.1.2 for Internal Flooding.

In March 2002, the CNS internal events PRA model received a peer review to certify the acceptability of PRAs before a consensus PRA Standard was available. The industry-developed process and methodology outlined in NEI 00-02 was used for the peer review. The review process was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the NEI Risk Applications Task Force.

Revision 2b of the CNS internal events PRA was the model of record at the time of the peer review. The Revision 3 model was used as the basis for the Fire PRA model which supports the NFPA 805 transition.

The NEI 00-02 Peer Review process used grades to assess the relative technical merits and capabilities of each sub-element reviewed. The grades provide guidance on appropriate use of the information covered by the sub-element for risk-informed applications. Per NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard", in general, the following approximate correspondence exists between the NEI 00-02 grading system and the ASME/ ANS PRA Standard RA-Sa-2009:

NEI 00-02	ASME PRA Standard
Grade 1	No equivalent "grade"
Grade 2	Capability Category I
Grade 3	Capability Category II
Grade 4	Capability Category III

Approximately 73% of the graded sub-elements received grades of 3 or higher. None of the sub-elements received a grade of 1 (or contingent 2), and 27 % of the sub-elements received a grade of 2 or contingent 3 (roughly 75% of this group was contingent grade 3).

F&Os from the 2002 peer review were assigned a significance level of A, B, C, D, or S based on guidance in NEI 00-02. Significance level A and B are equivalent to "Findings" in NEI 05-04 Revision 2. There were no level A F&Os; there were 32 level B F&Os, and 1 superior notation. In the time since the NEI 00-02 peer review, focused peer reviews have been performed for the internal flood and LERF models, which supersede one of the 32 F&Os.

In 2008, Duke Energy performed a self-assessment that evaluated the differences between the original peer review against NEI 00-02 and RA-S-2008 of the ASME/ANS PRA Standard, as endorsed by Regulatory Guide 1.200, Revision 1.

In 2013, Duke Energy performed a self-assessment against the ASME/ANS PRA Standard RA-Sa-2009 supporting requirements, as endorsed by Reg. Guide 1.200 Revision 2.

Table A-1 presents an assessment of all ASME/ANS PRA Standard RA-Sa-2009 supporting requirements that were assessed to be "Not Met" at the equivalent of Capability Category II in the 2002 peer review, were not assessed in the 2002 peer review (no equivalent NEI 00-02 sub-elements), or were assessed to be "Met" but had related Findings. Regulatory Guide 1.200, Appendix B was used to correlate NEI 00-02 sub-elements to ASME/ANS PRA Standard RA-Sa-2009 supporting requirements for the assessments. F&Os from the 2002 peer review are dispositioned for the applicable ASME/ANS PRA Standard RA-Sa-2009 SRs.

All changes to the CNS internal events PRA model since the last full-scope peer review have been reviewed and, with the exception of the LERF and Internal Flood PRA models for which focused-scope peer reviews were performed, there are no changes that are considered PRA upgrades as defined in ASME/ANS-RA-Sa-2009, as endorsed by Regulatory Guide 1.200 Revision 2. The CNS Internal Events PRA was judged to meet Capability Category II consistent with RG 1.200 guidance.

A.1.1 LERF PRA Quality Statement

In December 2012, a focused scope peer review was performed of the CNS LERF PRA against selected requirements of the ASME/ANS PRA Standard RA-Sa-2009, and any Clarifications and Qualifications provided in the NRC endorsement of the Standard contained in Revision 2 to RG 1.200. The peer review was performed using the process defined in NEI 05-04. The scope of the review was limited to the High Level Requirements and SRs in Part 2, Requirements for Internal Events At-Power PRA, Tables 2-2.8-1 and 2-2.8-2 through 2-2.8-8, of the ASME/ANS PRA Standard. The model reviewed was the LERF portion of CNS Internal Events PRA Model.

The ASME/ANS PRA Standard contains a total of 41 numbered SRs for the LERF portion of the internal events standard requirements. Two of the LERF SRs were determined to be not applicable to the CNS LERF PRA. Of the 39 applicable SRs, 26 SRs, or 67%, were rated as SR Met, Capability Category I/II, or greater. Only two SRs were not met. However, 11, or 28%, of the SRs were assessed at Capability Category I. CNS uses a LERF model based on the simplified LERF model in NUREG/CR-6595. While a NUREG/CR-6595 model is classified as Capability Category I, the NRC has determined this to be of sufficient capability to support risk-informed applications.

In the course of this review, 9 new F&Os were prepared, including 6 suggestions and 3 findings. Table A-2 lists the 13 SRs that were assessed at Capability Category I or Not Met and the related findings, including the peer review assessment comments, the disposition and status for each of the findings, and an assessment of the impact on the Fire PRA and NFPA 805 application.

A.1.2 Internal Flood PRA Quality Statement

In September 2012, a focused scope peer review was performed of the CNS Internal Flood PRA using the NEI 05-04 process and the ASME PRA Standard ASME/ANS RA-Sa- 2009, along with the NRC clarifications provided in Regulatory Guide 1.200, Revision 2. The peer review concluded that 56 of the total 62 numbered SRs outlined within the 2009 ASME PRA Standard for At-Power Internal Flood met Capability Category II or greater. Five of the SRs were rated as Not Met and 1 was rated as CC I.

The independent peer review identified 17 new F&Os which are comprised of 9 findings, 7 suggestions, and 1 best practice. Table A-3 presents the SRs and related F&O findings,

including the peer review assessment comments, the disposition and status for each of the findings, and an assessment of the impact on the Fire PRA and NFPA 805 application.

A.2. Fire PRA Quality Statement for Permanent 15-Year ILRT Extension

In accordance with RG 1.205 position 4.3:
"The licensee should submit the documentation described in Section 4.2 of Regulatory Guide 1.200 to address the baseline PRA and application-specific analyses. For PRA Standard "supporting requirements" important to the NFPA 805 risk assessments, the NRC position is that Capability Category II is generally acceptable."

The Catawba Internal Events model was also updated to support the Catawba Fire PRA. The CNS Fire PRA Peer Review was performed on July 12-16, 2010 using RG 1.200, Revision 2, the combined PRA standard, ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2, and the NEI 07-12 Fire PRA peer review process (Attachment V of Reference 33). The purpose of this review was to provide a method for establishing the technical quality and adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The peer review findings were addressed and the dispositions reviewed to validate that no changes were made which meet the definition of a PRA model upgrade per RG 1.200. Therefore, no additional peer reviews, partial scope or focused scope, were required to be conducted for the CNS Fire PRA.

The CNS Fire PRA was judged to meet Capability Category II consistent with RG 1.205 guidance. A total of twenty (20) F&O findings and twenty-nine (29) F&O suggestions (plus 1 best practice F&O) were generated. The capability categories are defined in ASME/ANS RA-Sa-2009, Part 4, "Requirements for Fires At-Power PRA." The peer review report noted that there were 13 SRs where the standard was not met. Sixteen F&Os were issued against SRs which met Capability Category I (some classified as "findings" and some addressed via "suggestions"). The findings have been resolved with the dispositions summarized in Table A-4. The impact of those areas where only the Capability Category I requirement was met is summarized in Table A-5. All F&Os that were defined as suggestions have been dispositioned. No changes were made in the resolution of the findings that meet the definition of a model upgrade as defined by RG 1.200; therefore, a follow-up peer review is not required. The Fire PRA is judged to be adequate to support the ILRT extension.

A.3. High Wind PRA Quality Statement for Permanent 15-Year ILRT Extension

The HWPRA was assessed by a peer team against ASME/ANS PRA standard with RG 1.200 Revision 2 clarifications in August of 2013. The peer team documented the Facts and Observations (F&Os) that pertain to the CNS HWPRA in LTR-RAM-II-13-077 [Reference 44]. Each of these F&Os are resolved or dispositioned in order to ensure the capability category of each individual Standard Requirement is met so that the CNS HWPRA can be used to support risk-informed applications. Table A-6 shows the findings and resolutions.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
IE-A1	IDENTIFY those initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage using a structured, systematic process for identifying initiating events that accounts for plant-specific features. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA). Existing lists of known initiators are also commonly employed as a starting point.	Dispositioned	F&O IE-03: Although SAAG 691 states that a review of plant systems was performed to search for support initiators, documentation of the review was not located. Each system notebook includes a section indicating whether or not it was determined that loss of that system leads to an initiating event. However, there was no discussion in SAAG 691 or the system notebooks to indicate that the process followed was sufficiently structured to capture potential initiators across various system alignments and support system alignments, and to consider initiating event precursors. This finding was made against NEI SR IE-10 with grade 3 being contingent on its resolution.	The NEI SRs applicable to this ASME SR are IE-7, IE-8, IE-9, and IE-10, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated IE-7 and IE-9 as "3" and IE-8 and IE-10 as "3 with contingencies." IE-10 has one level "B" F&O: IE-03. F&O IE-03: Support systems were reviewed to identify plant specific initiating events and documentation of the review and approach has been added to CNC-1535.00-00-0114 Rev 0. Since the Peer Review rated all of the applicable NEI SRs as "3" and there are no remaining open level "B" F&Os, this ASME SR is now Met Cat II.	There were no F&Os with "A" level of significance at CNS and there are no remaining open F&Os with "B" level of significance related to this SR. There is no impact to the ILRT extension.
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	Dispositioned	F&O IE-06: The Loss of HVAC initiator was removed, because operators may shut down the plant from remote locations (the Auxiliary Shutdown Panel and the SSF) if the Control Room is incapable of maintaining inventory control. This is an inadequate reason to omit an IE. If loss of HVAC causes a plant trip and requires SSD from the ASP, that sequence should be identified and modeled. Note that the switchgear room may also be affected by failed HVAC. A particular example is the possibility that the switchgear chiller is working, in which case the operators may not diagnose the situation in time.	The NEI SRs applicable to this ASME SR are IE-5, IE-7, IE-9, and IE-10, and there are no NRC objections. There is an industry action to confirm that the appropriate initiators were included. The original Peer Review rated IE-7 and IE-9 as "3" and IE-5 and IE-10 as "3 with contingencies." IE-5 has one level "B" F&O: IE-06; IE-10 has one level "B" F&O: IE-03. F&O IE-03 is more applicable to SRs IE-A1, IE-A5 and IE-A6, and is dispositioned under those SRs. F&O IE-06: This SR is not met because the loss of switchgear HVAC initiating event is not included in the PRA. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	LOCAs (LOCAs that cannot be 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	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.	Dispositioned	F&O IE-03: Although SAAG 691 states that a review of plant systems was performed to search for support initiators, documentation of the review was not located. Each system notebook includes a section indicating whether or not it was determined that loss of that system leads to an initiating event. However, there was no discussion in SAAG 691 or the system notebooks to indicate that the process followed was sufficiently structured to capture potential initiators across various system alignments and support system alignments, and to consider initiating event precursors. This finding was made against NEI SR IE-10 with grade 3 being contingent on its resolution.	The NEI SRs applicable to this ASME SR are IE-5, IE-7, IE-9, and IE-10, and there are no NRC objections. There is an industry action to check for initiating events that can be caused by a train failure or a system failure. The original Peer Review rated IE-7 and IE-9 as "3" and IE-5 and IE-10 as "3 with contingencies." IE-5 has one level "B" F&O: IE-06; IE-10 has one level "B" F&O: IE-03. F&O IE-06 is more applicable to SR IE-A2 and is dispositioned under that SR. F&O IE-03: Support systems were reviewed to identify plant specific initiating events and documentation of the review and approach has been added to CNC-1535.00-00-0114 Rev 0. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.
IE-A6	When performing the systematic evaluation required in IE-A5, INCLUDE initiating events resulting	Dispositioned	F&O IE-03: Although SAAG 691 states that a review of plant systems was performed to search for support initiators, documentation of the review was	The NEI SRs applicable to this ASME SR are IE-5, IE-7, IE-9, and IE-10, and there are no NRC objections. There is an industry action to check for initiating events that can be caused by multiple failures, if	Based on the disposition, the requirements of Cat II are considered met. There is no

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	from multiple failures, if the equipment failures result from common cause, and from routine system alignments.		not located. Each system notebook includes a section indicating whether or not it was determined that loss of that system leads to an initiating event. However, there was no discussion in SAAG 691 or the system notebooks to indicate that the process followed was sufficiently structured to capture potential initiators across various system alignments and support system alignments, and to consider initiating event precursors. This finding was made against NEI SR IE-10 with grade 3 being contingent on its resolution.	the equipment failures result from a common cause or from routine system alignments. The original Peer Review rated IE-7 and IE-9 as "3" and IE-5 and IE-10 as "3 with contingencies." IE-5 has one level "B" F&O: IE-06; IE-10 has one level "B" F&O: IE-03. F&O IE-06 is more applicable to SR IE-A2 and is dispositioned under that SR. F&O IE-03: Support systems were reviewed to identify plant specific initiating events and documentation of the review and approach has been added to CNC-1535.00-00-0114 Rev 0. CNC-1535.00-00-0114 Rev 0 documents the reviews of the common cause failure events and review of maintenance rule function for consideration of initiating events from multiple failures. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR.	impact to the ILRT extension.
IE-A8	INTERVIEW plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	Open	None	There are no NEI SRs applicable to this ASME SR. An extensive search for initiating events has been performed in CNC-1535.00-00-0114 Rev 0, so it is unlikely that interviews with plant personnel would result in the addition of any new initiators to the internal events model. However, the interviews need to be performed and documented. Self-assessment DPC-1535.00-00-0013, Rev. 3 indicates that this requirement has not been met for CNS. Specifically, no interviews with plant personnel have been performed or documented.	Based on the disposition, the requirements for this SR are considered not met, but it is unlikely that interviews with plant personnel would change the model in a way that would affect the ILRT extension. There is no impact to the ILRT extension.
IE-A9	REVIEW plant-specific operating experience for initiating event precursors, for identifying additional initiating events. For example, plant-specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	Dispositioned	F&O IE-03: Although SAAG 691 states that a review of plant systems was performed to search for support initiators, documentation of the review was not located. Each system notebook includes a section indicating whether or not it was determined that loss of that system leads to an initiating event. However, there was no discussion in SAAG 691 or the system notebooks to indicate that the process followed was sufficiently structured to capture potential initiators across various system alignments and support system alignments, and to consider initiating event precursors. This finding was made against NEI SR IE-10 with grade 3 being contingent on its resolution.	The NEI SRs applicable to this ASME SR are IE-10 and IE-16, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated IE-16 as "3" and IE-10 as "3 with contingencies." IE-10 has one level "B" F&O: IE-03. F&O IE-03: Plant-specific operating experience has been reviewed for initiating event precursors. This review is documented in CNC-1535.00-00-0114 Rev 0. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR.	There were no F&Os with "A" level of significance at CNS and there are no remaining open F&Os with "B" level of significance related to this SR. No impact on the ILRT extension.
IE-B2	USE a structured, systematic process for grouping initiating events. For example, such a	Dispositioned	None	The NEI SRs applicable to this ASME SR are IE-4 and IE-7, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated IE-7 as "3" and IE-4 as "3 with	There were no F&Os with "A" level of significance at CNS and there are no level

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	systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA).			contingencies." There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os associated with either of these NEI SRs. Initiating events were combined into groups and a systematic approach was used as documented in CNC-1535.00-00-0114 Rev 0 and CNC-1535.00-00-0031 Rev 0 (SAAG 691). Self-assessment DPC-1535.00-00-0013, Rev. 3 indicates that this requirement has not been met for CNS. Specifically, documentation of a structured, systematic approach to grouping initiating events was found to need enhancement.	"B" F&Os related to this SR. Documentation issue has no impact on the ILRT extension.
IE-C5	CALCULATE initiating event frequencies on a reactor year basis [Note (1)]. INCLUDE in the initiating event analysis the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power.	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. Initiating event frequencies are calculated on a reactor year basis. The plant availability factor is included in the calculations. This is considered to meet CAT II of the ASME/ANS PRA Standard.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.
IE-C6	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events or groups from further evaluation: (a) the frequency of the event is less than 1E-7 per reactor year (/yr), and the event does not involve either an ISLOCA, containment bypass, or reactor pressure vessel rupture (b) the frequency of the event is less than 1E-6/yr, and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator, or (c) the resulting reactor shutdown is	Dispositioned	F&O IE-06: The Loss of HVAC initiator was removed, because operators may shut down the plant from remote locations (the Auxiliary Shutdown Panel and the SSF) if the Control Room is incapable of maintaining inventory control. This is an inadequate reason to omit an IE. If loss of HVAC causes a plant trip and requires SSD from the ASP, that sequence should be identified and modeled. Note that the switchgear room may also be affected by failed HVAC. A particular example is the possibility that the switchgear chiller is working, in which case the operators may not diagnose the situation in time.	There are no NEI SRs applicable to this ASME SR. Although IE-C6 does not correlate directly to an NEI SR assessed by the peer review team, F&O IE-06 is judged to be applicable to this SR. The loss of Switchgear and Control Room HVAC systems are not modeled as initiating events in the Catawba PRA, and are excluded based on judgment, but the criteria in IE-C6 should be used to justify the exclusion. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on the ILRT extension application.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.

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	not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically). If either criterion (a) or (b) above is used, then CONFIRM that the value specified in the criterion meets the applicable requirements in Data Analysis (2-2.6) and Level 1 Quantification (2-2.7).				
IE-C9	If fault tree modeling is used for initiating events, QUANTIFY the initiating event frequency [as opposed to the probability of an initiating event over a specific time frame, which is the usual fault tree quantification model described in Systems Analysis (2-2.4)]. MODIFY, as necessary, the fault tree computational methods that are used so that the top event quantification produces a failure frequency rather than a top event probability as normally computed. USE the applicable requirements in Data Analysis (2-2.6) for the data used in the fault-tree quantification.	Open	F&O IE-08: The estimation of the frequency of the loss service water (RN) is incorrect in the application of common cause factors. A "mission time" of 72 hours is used to describe the failure of all four pumps in the calculation of a yearly frequency. The equation used is basically: Lambda*72 hours * Beta * Gamma * Delta. Note that Lambda*72 hours is the frequency of a pump failing to run for 72 hours. The CCF factors are dimensionless and represent the failure of the other three pumps. The equation above calculates the frequency of failure in a 72 hour period. The "mission time" must be consistent with the frequency being calculated. That is, one would expect the frequency for an 18 month period (a refueling cycle) to be 1.5 times the frequency for a year. The current equation would provide the same frequency for a year, a refueling cycle, or the life of the plant. Ignoring a plant availability factor, the annual frequency is given by: Lambda*8760 hours * Beta * Gamma * Delta. Given the set of MGL parameters, the current equation underestimates the frequency by a factor	There are no NEI SRs applicable to this ASME SR. Although IE-C8 does not correlate directly to an NEI SR assessed by the peer review team, F&O IE-08 is judged to be applicable to this SR. The F&O remains open (PRATracker C-03-0049) to implement a widely-accepted approach for CCF treatment of components in initiator analyses once developed.	At the time this F&O was issued, there was no industry consensus on how to model CCFs across running and standby trains in IE fault trees. Since then, EPRI has published guidance in technical report 1016741 [Reference 46]. For Catawba, the exposure time frame for potential common cause run failure is based on a consideration of mean time to repair (MTTR) and Tech. Spec. Completion Time rather than application of a full year, consistent with the guidance provided in Reference 46. The EPRI document recommends using a 24-hour MTTR in the IE fault trees for model simplification and to yield

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			of 365/3 ~ 122. One upper bound is provided by NUREG/CR-5750, which estimates the frequency at about 1E-3 per critical operating year. This value is based on individual unit critical years, and may not be appropriate for cases where the failure is a station failure, not a single unit failure. An alternative approach is to develop, via NUREG/CR-4780 techniques, more realistic MGL parameters that deal with loss of a system as an initiating event not as a design basis function. Note the discussion does not question the MGL parameters. The point being made is the use of the parameters in calculating the frequency.		results that are not excessively conservative. Also, use of a 24-hour MTTR mission period allows use of the same value in every model. The EPRI guidance notes that use of Tech. Spec. Completion Time periods is also a viable option that typically yields somewhat more conservative results. Since the industry consensus modeling approach is used in the Catawba PRA, the CNS loss of service water frequency is not underestimated and is appropriate. Therefore, there is no impact on the overall CDF and LERF. Therefore, there is no impact to the ILRT extension.
IE-C10	If fault-tree modeling is used for initiating events, CAPTURE within the initiating event fault tree models all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components.	Open	F&O IE-08: The estimation of the frequency of the loss service water (RN) is incorrect in the application of common cause factors. A "mission time" of 72 hours is used to describe the failure of all four pumps in the calculation of a yearly frequency. The equation used is basically: Lambda*72 hours * Beta * Gamma * Delta. Note that Lambda*72 hours is the frequency of a pump failing to run for 72 hours. The CCF factors are dimensionless and represent the failure of the other three pumps. The equation above calculates the frequency of failure in a 72 hour period. The "mission time" must be consistent with the frequency being calculated. That is, one would expect the frequency for an 18 month period (a refueling cycle) to be 1.5 times the frequency for a year. The current equation would provide the same frequency for a year, a refueling	There are no NEI SRs applicable to this ASME SR. Although IE-C8 does not correlate directly to an NEI SR assessed by the peer review team, F&O IE-08 is judged to be applicable to this SR. The F&O remains open (PRATracker C-03-0049) to implement a widely-accepted approach for CCF treatment of components in initiator analyses once developed.	At the time this F&O was issued, there was no industry consensus on how to model CCFs across running and standby trains in IE fault trees. Since then, EPRI has published guidance in technical report 1016741 [Reference 46]. For Catawba, the exposure time frame for potential common cause run failure is based on a consideration of mean time to repair (MTTR) and Tech. Spec. Completion Time rather than application of a full year, consistent with the guidance provided in

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			cycle, or the life of the plant. Ignoring a plant availability factor, the annual frequency is given by: Lambda*8760 hours * Beta * Gamma * Delta. Given the set of MGL parameters, the current equation underestimates the frequency by a factor of 365/3 ~ 122. One upper bound is provided by NUREG/CR-5750, which estimates the frequency at about 1E-3 per critical operating year. This value is based on individual unit critical years, and may not be appropriate for cases where the failure is a station failure, not a single unit failure. An alternative approach is to develop, via NUREG/CR-4780 techniques, more realistic MGL parameters that deal with loss of a system as an initiating event not as a design basis function. Note the discussion does not question the MGL parameters. The point being made is the use of the parameters in calculating the frequency.		Reference 46. The EPRI document recommends using a 24-hour MTTR in the IE fault trees for model simplification and to yield results that are not excessively conservative. Also, use of a 24-hour MTTR mission period allows use of the same value in every model. The EPRI guidance notes that use of Tech. Spec. Completion Time periods is also a viable option that typically yields somewhat more conservative results. Since the industry consensus modeling approach is used in the Catawba PRA, the CNS loss of service water frequency is not underestimated and is appropriate. Therefore, there is no impact on the overall CDF and LERF. Therefore, there is no impact to the ILRT extension.
IE-C12	COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonableness check of the results.	Dispositioned	F&O IE-04: The initiating event frequency for a stuck open PORV or safety valve is taken from NUREG/CR-5750 but is conservative for the following reasons. The NUREG assigned a value to these events based on a non-informative prior updated with 0 events and the total number of critical reactor years in the study. In the case of a spurious opening of a primary safety valve, the model should address the potential for the valve to close as the pressure decreased, effectively terminating the loss of coolant. The evaluation of the subsequent reclosure of the PORV is not as straightforward. The cause of the opening PORV	The NEI SR applicable to this ASME SR is IE-13, and there are no industry self-assessment actions and no NRC objections. IE-13 was given a grade of "2" with F&O IE-04. F&O IE-04 appears to be an observation of conservatism in usage of generic industry data for stuck open SRV and PORV initiating events. However, this treatment is judged to be appropriate, and so this is considered to meet CAT II of the ASME/ANS PRA Standard.	There were no F&Os with "A" level of significance at CNS and there are no remaining open F&Os with "B" level of significance related to this SR. In addition, use of generic data for stuck open PORV or safety valve initiating event frequency is conservative and judged to be appropriate. There is

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			would need to be addressed. However, either the PORV could be closed or the block valve could be closed.		negligible impact to the ILRT extension.
IE-C14	In the ISLOCA frequency analysis, INCLUDE the following features of plant and procedures that influence the ISLOCA frequency: (a) configuration of potential pathways including numbers and types of valves and their relevant failure modes and the existence, size, and positioning of relief valves (b) provision of protective interlocks (c) relevant surveillance test procedures (d) the capability of secondary system piping (e) isolation capabilities given high flow/differential pressure conditions that might exist following breach of the secondary system	Open	None	The NEI SR applicable to this ASME SR is IE-14, and there are no NRC objections. There is an industry action to confirm that secondary pipe system capability and isolation capability under high flow or differential pressures are included. The original Peer Review rated this NEI SR as "3". There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os associated with this NEI SR. Even though the NEI equivalent to this SR was assessed to be Grade 3 by the peer review team in 2002, the industry action relates to PRA Tracker Open Item C-02-0001, which indicates this item is still open. Credit may have been given to MOVs that will not function under the differential pressure conditions that result from ruptured check valves. This item could result in an increase in the probability of certain ISLOCAs and may impact the base model CDF and LERF.	If credit for the MOVs is removed, the base internal events CDF and LERF will increase. Impact on the internal events PRA LERF is expected to be greater than impact on CDF. The CDF/LERF and delta CDF/LERF values are in the middle of Region II of the RG 1.174 acceptance criteria, and the expected increase in CDF/LERF with credit for the MOVs removed is small enough that the risk metrics would remain in Region II. The ILRT extension impact is expected to be insignificant.
IE-C15	CHARACTERIZE the uncertainty in the initiating event frequencies and PROVIDE mean values for use in the quantification of the PRA results.	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. There is no equivalent NEI SR, however, the CNS self-assessment evaluated this SR as being met, and there is documentation of the mean value and error factor for the initiating event frequencies, so this is considered to meet CAT II of the ASME/ANS PRA Standard.	Based on the disposition, Cat II of the PRA Standard is met. There is no impact on the ILRT extension
AS-A1	USE a method for accident sequence analysis that (a) explicitly models the appropriate combinations of system responses and operator actions that affect the key safety functions for each modeled initiating event;	Open	AS-04: There were several observations on the modeling of event D3 in the SGTR tree: Event D3 is generally defined as the event to cooldown to RHR conditions using 2/3 SG for depressurization. D3 includes the HEP YAGRCOLDHE, which is directed by ECA 3.1 and 3.2. 1. D3 is defined as "primary system cooldown via secondary system depressurization". Primary	The NEI SRs applicable to this ASME SR are AS-4 and AS-8, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-4 as "3" and AS-8 as "3 with contingencies." AS-8 has one level "B" F&O: AS-04. F&O AS-04 is only applicable to SGTR events. The modeling of SGTR events was changed to be consistent with industry standards using the guidance in WCAP-15955. Success criteria runs were performed for the MNS PRA and are applicable to CNS.	There were no F&Os with "A" level of significance at CNS. Open level "B" F&O AS-04 is only applicable to SGTR events. A sensitivity was done in Section 5.3.5 to approximate the necessary SGTR success criteria modeling changes. Changes

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	(b) includes a graphical representation of the accident sequences in an "event tree structure" or equivalent such that the accident sequence progression is displayed; and (c) provides a framework to support sequence quantification.		system depressurization must be accomplished in some sequences (YD1D2D3, YOD3, YUOD3), by either PORV, aux spray, or main spray. These functions are not included in D3. 2. Sequence YUOD3 needs a T/H justification that D3 can actually prevent core damage in this circumstance. This sequence has no injection and no SG isolation. This is "core cooling recovery" with an unisolated SGTR. ECA3 specifies cool down at less than 100F/hr. The core cannot be maintained covered for the amount of time it takes to cooldown to RHR conditions at 100F/hr. Suggested resolution is to use a separate function for this heading, using an operator action directed by FRC.1 and without RCP operating. 3. Sequence YUD1QD3. comment #2 applies to this sequence as well. This is a stuck open relief PORV with no injection.	Reconstruction of the CNS SGTR success criteria is needed to close this F&O.	in Success Criteria modeling are based on guidance provided in Reference 43. Other than a small change to overall risk, there is no impact on the ILRT extension.
AS-A2	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage. [See Note (1).]	Open	AS-04: There were several observations on the modeling of event D3 in the SGTR tree: Event D3 is generally defined as the event to cooldown to RHR conditions using 2/3 SG for depressurization. D3 includes the HEP YAGRCOLDHE, which is directed by ECA 3.1 and 3.2. 1. D3 is defined as "primary system cooldown via secondary system depressurization". Primary system depressurization must be accomplished in some sequences (YD1D2D3, YOD3, YUOD3), by either PORV, aux spray, or main spray. These functions are not included in D3. 2. Sequence YUOD3 needs a T/H justification that D3 can actually prevent core damage in this circumstance. This sequence has no injection and no SG isolation. This is "core cooling recovery" with an unisolated SGTR. ECA3 specifies cool down at less than 100F/hr. The core cannot be maintained covered for the amount of time it takes to cooldown to RHR conditions at 100F/hr. Suggested resolution is to use a separate function for this heading, using an operator action directed by FRC.1 and without	The NEI SRs applicable to this ASME SR are AS-6, AS-7, AS-8, AS-9, and AS-17, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-6 and AS-17 as "3" and AS-7, AS-8 and AS-9 as "3 with contingencies." AS-8 has one level "B" F&O: AS-04, and AS-9 has one level "B" F&O: AS-07. F&O AS-04 is only applicable to SGTR events. The modeling of SGTR events was changed to be consistent with industry standards using the guidance in WCAP-15955. Success criteria runs were performed for the MNS PRA and are applicable to CNS. Reconstruction of the CNS SGTR success criteria is needed to close this F&O. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved.	There were no F&Os with "A" level of significance at CNS. Open level "B" F&O AS-04 is only applicable to SGTR events. A sensitivity was done in Section 5.3.5 to approximate the necessary SGTR success criteria modeling changes. Changes in Success Criteria modeling are based on guidance provided in Reference 43. Other than a small change to overall risk, there is no impact on the ILRT extension.

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			RCP operating. 3. Sequence YUD1QD3. comment #2 applies to this sequence as well. This is a stuck open relief PORV with no injection. AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.)		
AS-A3	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).]	Dispositioned	TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. SY-03: System success criteria are specified in the	The NEI SRs applicable to this ASME SR are AS-17, AS-7, and SY-17, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-17 as "3" and AS-8 and SY-17 as "3 with contingencies." SY-17 has two level "B" F&Os: TH-03 and SY-03. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O SY-03 - Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to these system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the PRA results is expected to be negligible and not significantly affect the applicability to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis.		
AS-A5	DEFINE the accident sequence model in a manner that is consistent with the plant-specific: system design, EOPs, abnormal procedures, and plant transient response.	Dispositioned	TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs.	The NEI SRs applicable to this ASME SR are AS-5, AS-18, AS-19, and SY-5, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-5, AS-19 and SY-5 as "3" and AS-18 as "3 with contingencies." AS-18 has one level "B" F&O: TH-03. The success criteria for all LOCA events were revisited since the Peer Review. For MLOCA and SLOCA events, thermal/hydraulic calculations were performed at the upper and lower ends of the spectrum, as well as at several midpoints where changes in thermal/hydraulic behavior occur to determine the success criteria for those events. Availability of accumulators is also addressed. The design basis requirements for mitigation are used are used as the primary basis the success criteria for LLOCA events. Thus F&O TH-03 has been resolved. Since the Peer Review rated all of the applicable NEI SRs as "3" and there are no remaining open level "A" or "B" F&Os, this ASME SR is Met Cat II.	There were no F&Os with "A" level of significance at CNS and there are no remaining open F&Os with "B" level of significance related to this SR. No impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
AS-A7	DELINEATE the possible accident sequences for each modeled initiating event, unless the sequences can be shown to be a non-contribution using qualitative arguments.	Open	AS-04: There were several observations on the modeling of event D3 in the SGTR tree: Event D3 is generally defined as the event to cooldown to RHR conditions using 2/3 SG for depressurization. D3 includes the HEP YAGRCOLDHE, which is directed by ECA 3.1 and 3.2. 1. D3 is defined as "primary system cooldown via secondary system depressurization". Primary system depressurization must be accomplished in some sequences (YD1D2D3, YOD3, YUOD3), by either PORV, aux spray, or main spray. These functions are not included in D3. 2. Sequence YUOD3 needs a T/H justification that D3 can actually prevent core damage in this circumstance. This sequence has no injection and no SG isolation. This is "core cooling recovery" with an unisolated SGTR. ECA3 specifies cool down at less than 100F/hr. The core cannot be maintained covered for the amount of time it takes to cooldown to RHR conditions at 100F/hr. Suggested resolution is to use a separate function for this heading, using an operator action directed by FRC.1 and without RCP operating. 3. Sequence YUD1QD3. comment #2 applies to this sequence as well. This is a stuck open relief PORV with no injection. AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.)	The NEI SRs applicable to this ASME SR are AS-4, AS-5, AS-6, AS-7, AS-8, and AS-9, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-4, AS-5 and AS-6 as "3" and AS-7, AS-8, and AS-9 as "3 with contingencies." AS-8 has one level "B" F&O: AS-04, and AS-9 has one level "B" F&O: AS-07. F&O AS-04 is only applicable to SGTR events. The modeling of SGTR events was changed to be consistent with industry standards using the guidance in WCAP-15955. Success criteria runs were performed for the MNS PRA and are applicable to CNS. Reconstruction of the CNS SGTR success criteria is needed to close this F&O. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved.	There were no F&Os with "A" level of significance at CNS. Open level "B" F&O AS-04 is only applicable to SGTR events. A sensitivity was done in Section 5.3.5 to approximate the necessary SGTR success criteria modeling changes. Changes in Success Criteria modeling are based on guidance provided in Reference 43. Other than a small change to overall risk, there is no impact on the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
AS-A8	DEFINE the end state of the accident progression as occurring when either a core damage state or a steady-state condition has been reached.	Dispositioned	TH-02: The original definition of core damage used in the Catawba PRA was the eutectic melting point of the fuel (4040 degF). This has been informally revised (i.e., not in a Workplace Procedure but known to the Duke PRA analysts associated with performing success criteria) based on a McGuire PRA Peer Review observation, to "success criteria is defined as the hottest core node remained below 2000 degF" as predicted by MAAP or other T/H code. The reference used by Duke for this definition is EPRI document NP-6328, "Release of Volatile Fission Products From Irradiated LWR Fuel: Mass Spectrometry Studies", Final Report, April 1989. The revised criterion is more in line with industry practice. In specific instances, it is possible that the 2000 degF criterion could be pushing the limit of acceptability for the code used, and investigation of the sensitivity of the results to a lower temperature value might be warranted (e.g., the ASME PRA Standard suggests 1800 degF for a code like MAAP, or even 1200 degF if there is prolonged core uncovery).	The NEI SRs applicable to this ASME SR are AS-20, AS-21, AS-22, and AS-23. There are no NRC objections, but since the explicit requirement for steady-state conditions for end state was not contained in NEI 00-02, this should be demonstrated. The original Peer Review rated all of these NEI SRs as "3". AS-22 has one level "C" F&O: TH-02, but a similar F&O for McGuire was level "B" so it is retained here. An evaluation was performed in DPC-1535.00-00-0010 which provides the definition of core damage for PRA applications using the MAAP analysis code, determined to be 2500 F. This criterion is used in the development of success criteria and timing of operator actions. This evaluation meets the requirements of Section 1-2.2 of the Standard, and thus F&O TH-02 is resolved. The ASME SR is considered Met as reported in Duke self-assessments CNC-1535.00-00-0155 and DPC-1535.00-00-0013. More recently, McGuire success criteria calculations have been revised to define success criteria as core temperature remains below 2000 Deg F. As noted in MCC-1535.00-00-0172, the difference in time to core damage is not significant when using either 2000 Deg F or 4000 Deg F because the exothermic nature of the zircaloy-water reaction rapidly increases the fuel temperature. Therefore, the revised success criterion does not have an impact on the time available for human recoveries or other non-recovery events such as loss of offsite power recoveries. There is also no impact on the equipment required for mitigation of any accident sequence. Even though the Catawba success criteria have not been revised, the conclusions from McGuire are considered applicable to Catawba due to the similarities between the plants.	Changing the success criteria for core damage to a temperature provided in ASME/ANS PRA Standard SR SC-A2 examples for Cat II/III is expected to have negligible impact on system time windows used in the human reliability analysis, and should not impact the amount of equipment required for successful mitigation of sequences. The impact on the ILRT extension is negligible.
AS-A9	USE realistic, applicable (i.e., from similar plants) thermal hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems.	Dispositioned	TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range	The NEI SRs applicable to this ASME SR are AS-18 and TH-4. There are no NRC objections, but the focus should be on the environmental conditions challenging the equipment during the accident sequence. The original Peer Review rated both of these NEI SRs as "3 with contingencies". AS-18 and TH-4 have the same level "B" F&O: TH-03. The success criteria for all LOCA events were revisited since the Peer Review. For MLOCA and SLOCA events, thermal/hydraulic calculations were performed at the upper and lower ends of the	There were no F&Os with "A" level of significance at CNS and there are no remaining open F&Os with "B" level of significance related to this SR. No impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs.	spectrum, as well as at several midpoints where changes in thermal/hydraulic behavior occur to determine the success criteria for those events. Availability of accumulators is also addressed. The design basis requirements for mitigation are used are used as the primary basis the success criteria for LLOCA events. Thus F&O TH-03 has been resolved. Since the Peer Review rated all of the applicable NEI SRs as "3" and there are no remaining open level "A" or "B" F&Os, this ASME SR is Met Cat II.	
AS-A10	In constructing the accident sequence models, INCLUDE, for each modeled initiating event, sufficient detail that differences in requirements on systems and required operator interactions (e.g., systems initiations or valve alignment) are captured. Where diverse systems and/or operator actions provide a similar function, if choosing one over another changes the requirements for operator intervention or the need for other systems, MODEL each separately.	Open	AS-04: There were several observations on the modeling of event D3 in the SGTR tree: Event D3 is generally defined as the event to cooldown to RHR conditions using 2/3 SG for depressurization. D3 includes the HEP YAGRCOLDHE, which is directed by ECA 3.1 and 3.2. 1. D3 is defined as "primary system cooldown via secondary system depressurization". Primary system depressurization must be accomplished in some sequences (YD1D2D3, YOD3, YUOD3), by either PORV, aux spray, or main spray. These functions are not included in D3. 2. Sequence YUOD3 needs a T/H justification that D3 can actually prevent core damage in this circumstance. This sequence has no injection and no SG isolation. This is "core cooling recovery" with an unisolated SGTR. ECA3 specifies cool down at less than 100F/hr. The core cannot be maintained covered for the amount of time it takes to cooldown to RHR conditions at 100F/hr. Suggested resolution is to use a separate function for this heading, using an operator action directed by FRC.1 and without RCP operating. 3. Sequence YUD1QD3. comment #2 applies to this sequence as well. This is a stuck open relief PORV with no injection. AS-07: The success criteria for AFW for SGTR is 1	The NEI SRs applicable to this ASME SR are AS-4, AS-5, AS-6, AS-7, AS-8, AS-9, AS-19, SY-5, SY-8, and HR-23, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-4, AS-5, AS-19, SY-5 and SY-18 as "3" and AS-7, AS-8, AS-9, and HR-23 as "3 with contingencies." AS-8 has one level "B" F&O: AS-04; AS-9 has one level "B" F&O: AS-07; and HR-23 has one level "B" F&O: HR-05. F&O AS-04 is only applicable to SGTR events. The modeling of SGTR events was changed to be consistent with industry standards using the guidance in WCAP-15955. Success criteria runs were performed for the MNS PRA and are applicable to CNS. Reconstruction of the CNS SGTR success criteria is needed to close this F&O. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved. Elements of F&O HR-05 related to this F&O are considered resolved. Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets.	There were no F&Os with "A" level of significance at CNS. Open level "B" F&O AS-04 is only applicable to SGTR events. A sensitivity was done in Section 5.3.5 to approximate the necessary SGTR success criteria modeling changes. Changes in Success Criteria modeling are based on guidance provided in Reference 43. Other than a small change to overall risk, there is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.) F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR TH-5 with grade 3 being contingent on its resolution.		
AS-B1	For each modeled initiating event, IDENTIFY mitigating systems impacted by the occurrence of the initiator and the extent of the impact. INCLUDE the impact of	Open	AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however,	The NEI SRs applicable to this ASME SR are IE-4, IE-5, IE-10, AS-4, AS-5, AS-6, AS-7, AS-8, AS-9, AS-10, AS-11, and DE-5, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-4, AS-5, AS-6 and AS-11 as "3" and all of the other NEI SRs as "3 with contingencies." IE-5 has one	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42].

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	initiating events on mitigating systems in the accident progression either in the accident sequence models or in the system models.		the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.) IE-06: The Loss of HVAC initiator was removed, because operators may shut down the plant from remote locations (the Auxiliary Shutdown Panel and the SSF) if the Control Room is incapable of maintaining inventory control. This is an inadequate reason to omit an IE. If loss of HVAC causes a plant trip and requires SSD from the ASP, that sequence should be identified and modeled. Note that the switchgear room may also be affected by failed HVAC. A particular example is the possibility that the switchgear chiller is working, in which case the operators may not diagnose the situation in time. DE-04: HVAC cooling of the essential switchgear rooms is stated as being required. The IPE quantitative analysis does not provide adequate success criteria. For example, the following are not specified: temperature limits of equipment, minimum number of Air Handling Units, or minimum number of chillers. The evaluation also states there is no concern within 24 hours provided that only those loads needed to provide core cooling are operated. There is no discussion of electrical load shedding for those loads not required, and of the human interface to execute load shedding. The human interface can be complex, involving both a discovery process (control room annunciators, or in the case of a local AHU failure, discovery through operator walkaround), and procedures and training to direct operation actions.	level "B" F&O: IE-06; IE-10 has one level "B" F&O: IE-03; AS-8 has one level "B" F&O: AS-04; AS-9 has one level "B" F&O: AS-07; AS-10 has one level "B" F&O: DE-04; and DE-5 has two level "B" F&Os: AS-07 and QU-02. Of the F&Os, AS-07, IE-06, and DE-04 appear to be related to this ASME SR, i.e., mitigating systems impacted by the occurrence of the initiator. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved. F&Os IE-06 and DE-04 are not resolved because the loss of switchgear HVAC initiating event is not included in the PRA. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on the ILRT extension application.	The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.
AS-B3	For each accident sequence, IDENTIFY the phenomenological	Open	DE-04: HVAC cooling of the essential switchgear rooms is stated as being required. The IPE	The NEI SRs applicable to this ASME SR are AS-10, SY-11, DE-10, and TH-8, and there are no industry self-assessment actions and no	Room heat-up analyses were performed for the

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	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.		quantitative analysis does not provide adequate success criteria. For example, the following are not specified: temperature limits of equipment, minimum number of Air Handling Units, or minimum number of chillers. The evaluation also states there is no concern within 24 hours provided that only those loads needed to provide core cooling are operated. There is no discussion of electrical load shedding for those loads not required, and of the human interface to execute load shedding. The human interface can be complex, involving both a discovery process (control room annunciators, or in the case of a local AHU failure, discovery through operator walkaround), and procedures and training to direct operation actions. TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. (Duke is already aware of this issue.) SY-06: For Catawba, there was no evaluation of the ability of non-qualified (non-EQ) equipment to survive in a degraded environment following an accident such as a steam line of feedwater line break outside of containment.	NRC objections. The original Peer Review rated AS-10 and DE-10 as "3 with contingencies" and SY-11 as "2". TH-8 was unrated. AS-10 has one level "B" F&O: DE-04; SY-11 has one level "B" F&O: SY-06; DE-10 has one level "B" F&O: TH-06; TH-8 has one level "B" F&O: TH-06. Of the F&Os, DE-04, TH-06, and SY-06 appear to be related to this ASME SR, i.e., phenomenological conditions created by the accident progression. F&O DE-06 is also associated with DE-10 but has been superseded by the more recent focus-scope peer review for the Flooding PRA model. F&Os DE-04 and TH-06 are not resolved because the loss of switchgear HVAC initiating event is not included in the PRA, and room heatup calculations for loss of ventilation are not performed for that and other locations. Room heatup calculations should be performed in all locations in which HVAC can be lost to justify not modeling those systems and/or determine timing of operator coping actions and equipment damage. If no room heatup calculation is performed, it should be assumed that the HVAC system is required in those locations. The appropriate dependencies should be included in the PRA model, including possible initiating events. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on the results for the ILRT extension application. F&O SY-06 is resolved because high-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). Since Peer Review F&Os DE-04, TH-06, and SY-06 are still open, this ASME SR is Not Met.	switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. High-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). This is considered resolved. There is no impact on the ILRT extension.
AS-B5	DEVELOP the accident sequence models to a level of detail sufficient to identify intersystem dependencies and train level interfaces, either in the event trees or through a combination of event tree and fault tree models and associated logic.	Dispositioned	QU-02: The IE's for certain support system failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus	The NEI SRs applicable to this ASME SR are AS-10, AS-11, DE-4, DE-5, DE-6, and QU-25, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-11 and DE-6 as "3" and AS-10, DE-4, and DE-5 as "3 with contingencies." QU-25 was found not applicable. AS-10 has one level "B" F&O: DE-04; DE-4 has one level "B" F&O: DE-04; DE-5 has two level "B" F&Os: QU-02 and AS-07. Of the F&Os, QU-02 and AS-07 appear to be related to this ASME SR, i.e., intersystems dependencies.	Review for dependencies takes place in the cut set file. This F&O will have no effect on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE. AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.)	F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved.	
SC-A1	USE the definition of "core damage" provided in Section 1-2 of this Standard. If core damage has been defined differently than in Section 1-2, (a) IDENTIFY any substantial differences from the Section 1-2 definition (b) PROVIDE the bases for the selected definition	Dispositioned	TH-02: The original definition of core damage used in the Catawba PRA was the eutectic melting point of the fuel (4040 degF). This has been informally revised (i.e., not in a Workplace Procedure but known to the Duke PRA analysts associated with performing success criteria) based on a McGuire PRA Peer Review observation, to "success criteria is defined as the hottest core node remained below 2000 degF" as predicted by MAAP or other T/H code. The reference used by Duke for this definition is EPRI document NP-6328, "Release of Volatile Fission Products From Irradiated LWR Fuel: Mass Spectrometry Studies", Final Report, April 1989. The revised criterion is more in line with industry practice. In specific instances, it is possible that the 2000 degF criterion could be pushing the limit of acceptability for the code used, and investigation of the sensitivity of the results to a lower temperature value might be warranted (e.g., the ASME PRA Standard suggests 1800 degF for a code like MAAP, or even 1200 degF if there is prolonged core uncovery).	The NEI SRs applicable to this ASME SR are AS-20 and AS-22, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-20 as "3" and AS-22 as "3 with contingencies." AS-22 has one level "C" F&O: TH-02, but a similar F&O for McGuire was level "B" so it is retained here. An evaluation was performed in DPC-1535.00-00-0010 which provides the definition of core damage for PRA applications using the MAAP analysis code, determined to be 2500 F. This criterion is used in the development of success criteria and timing of operator actions. This evaluation meets the requirements of Section 1-2.2 of the Standard, and thus F&O TH-02 is resolved. The ASME SR is considered Met as reported in Duke self-assessments CNC-1535.00-00-0155 and DPC-1535.00-00-0013. More recently, McGuire success criteria calculations have been revised to define success criteria as core temperature remains below 2000 Deg F. As noted in MCC-1535.00-00-0172, the difference in time to core damage is not significant when using either 2000 Deg F or 4000 Deg F because the exothermic nature of the zircaloy-water reaction rapidly increases the fuel temperature. Therefore, the revised success criterion does not have an impact on the time available for	Changing the success criteria for core damage to a temperature provided in examples in ASME/ANS PRA Standard SR SC-A2 for Cat II/III is expected to have negligible impact on system time windows used in the human reliability analysis, and should not impact the amount of equipment required for successful mitigation of sequences. The impact on the ILRT extension is negligible.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				human recoveries or other non-recovery events such as loss of offsite power recoveries. There is also no impact on the equipment required for mitigation of any accident sequence. Even though the Catawba success criteria have not been revised, the conclusions from McGuire are considered applicable to Catawba due to the similarities between the plants.	
SC-A2	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. SELECT these parameters such that determination of core damage is as realistic as practical, in a manner consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, in a manner consistent with the requirements specified under HLR-SC-B. Examples of measures for core damage suitable for Capability Category II/III, that have been used in PRAs, include (a) collapsed liquid level less than 1/3 core height or code-predicted peak core temperature >2,500°F (BWR) (b) collapsed liquid level below top of active fuel for a prolonged period, or code-predicted core peak node temperature >2,200°F using a code	Dispositioned	F&O TH-02 - The original definition of core damage used in the Catawba PRA was the eutectic melting point of the fuel (4040 °F). This has been informally revised (i.e., not in a Workplace Procedure but known to the Duke PRA analysts associated with performing success criteria) based on a McGuire PRA Peer Review observation, to "success criteria is defined as the hottest core node remained below 2000 °F" as predicted by MAAP or other T/H code. The reference used by Duke for this definition is EPRI document NP-6328, "Release of Volatile Fission Products From Irradiated LWR Fuel: Mass Spectrometry Studies", Final Report, April 1989. The revised criterion is more in line with industry practice. In specific instances, it is possible that the 2000 °F criterion could be pushing the limit of acceptability for the code used, and investigation of the sensitivity of the results to a lower temperature value might be warranted (e.g., the ASME PRA Standard suggests 1800 °F for a code like MAAP, or even 1200 °F if there is prolonged core uncovery). F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft^2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes	The NEI SRs applicable to this ASME SR are TH-4, TH-5, TH-7, and AS-22. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3 with contingencies", except TH-5, which is rated a "2". TH-4 has one level "B" F&O: TH-03; TH-5 has two level "B" F&Os: HR-05 and TH-05; TH-7 has one level "B" F&O: TH-01; and AS-22 has one level "C" F&O: TH-02, but a similar F&O for McGuire was level "B" so it is retained here. An evaluation was performed in DPC-1535.00-00-0010 which provides the definition of core damage for PRA applications using the MAAP analysis code, determined to be 2500 F. This criterion is used in the development of success criteria and timing of operator actions. This evaluation meets the requirements of Section 1-2.2 of the Standard, and thus F&O TH-02 is resolved. The ASME SR is considered Met as reported in Duke self-assessments CNC-1535.00-00-0155 and DPC-1535.00-00-0013. More recently, McGuire success criteria calculations have been revised to define success criteria as core temperature remains below 2000 Deg F. As noted in MCC-1535.00-00-0172, the difference in time to core damage is not significant when using either 2000 Deg F or 4000 Deg F because the exothermic nature of the zircaloy-water reaction rapidly increases the fuel temperature. Therefore, the revised success criterion does not have an impact on the time available for human recoveries or other non-recovery events such as loss of offsite power recoveries. There is also no impact on the equipment required for mitigation of any accident sequence. Even though the Catawba success criteria have not been revised, the conclusions from McGuire are considered applicable to Catawba due to the similarities between the plants. TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria	Changing the success criteria for core damage to a temperature provided in examples in ASME/ANS PRA Standard SR SC-A2 for Cat II/III is expected to have negligible impact on system time windows used in the human reliability analysis, and should not impact the amount of equipment required for successful mitigation of sequences. The impact on the ILRT extension is negligible. Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria, so the impact on ILRT extension is expected to be negligible. Peer Review F&O TH-01 is still open. While the success criteria has been updated, it has not been incorporated into the PRA model. However, there are no

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	with detailed core modeling; or code-predicted core peak node temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped parameter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR).		from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. F&O HR-05 - In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. F&O TH-01 - Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple	sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. HR-05 - Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. This is considered to resolve the elements of this F&O related to this SR, and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met.	significant changes to the success criteria, so the impact on the ILRT extension is expected to be negligible.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. F&O TH-05 -The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided.		
SC-A3	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18 with grade 3 and SY-17 with a grade 3 being contingent on its resolution. F&O QU-02 - The IE's for certain support system	The NEI SRs applicable to this ASME SR are AS-7, AS-17, AS-18, SY-17, TH-9, IE-6, SY-8 and DE-5. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-17, IE-6, and SY-8 as "3" and all of the other NEI SRs as "3 with contingencies." AS-18 has one level "B" F&O: TH-03; SY-17 has two level "B" F&Os: SY-03 and TH-03; TH-9 has two level "B" F&Os: TH-05 and TH-06; and DE-5 has two level "B" F&Os: AS-07 and QU-02. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria, so the impact on the ILRT

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE. F&O AS-07 - The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.) This finding was made against NEI SR DE-5 with grade 3 being contingent on its resolution. F&O SY-03 - System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN	F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved. F&O TH-04 - Feed and bleed success criteria changes were implement in Rev. 3 of the Transient Analysis Notebook and supported with MAAP analyses. The MAAP-code acceptance criteria applied are identified in Appendix H, Section 3.4 of the Catawba PRA Rev. 3 Transient Analysis Notebook. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR. F&O SY-03 - Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to these system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met. F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN F&O TH-06 is not resolved because the loss of switchgear HVAC initiating event is not included in the PRA, and room heatup calculations for loss of ventilation are not performed for that and other locations. Room heatup calculations should be performed in all locations in which HVAC can be lost to justify not modeling those systems and/or determine timing of operator coping actions and equipment damage. If no room heatup calculation is performed, it should be assumed that the HVAC system is required in those locations. The appropriate dependencies should be included in the PRA model, including possible initiating events. A recent evaluation	extension is expected to be negligible. Peer Review F&O SY-03 is still open due to documentation. While the success criteria has been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. This finding was made against NEI SR SY-17 with grade 3 being contingent on its resolution. F&O TH-05 -The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided. This finding was made against NEI SR TH-5 with grade 2. F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution.	was performed (PIP C-13-05664) to determine the impact on the Fire PRA of not including switchgear room and battery HVAC modeling. The evaluation concluded that any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on the Fire PRA results or results for the NFPA-805 application.	
SC-A4	IDENTIFY mitigating systems that are shared between units, and the manner in which the sharing is performed should both units experience a common initiating event (e.g., LOOP).	Dispositioned	F&O AS-07 - The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is	The NEI SRs applicable to this ASME SR are IE-6 and DE-5. There are no NRC objections, but since the explicit requirement for shared mitigating systems between units was not contained in NEI 00-02, this should be demonstrated. The original Peer Review rated IE-6 as "3" and DE-5 as "3 with contingencies." DE-5 has two level "B" F&Os: AS-07 and QU-02. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss	Review for dependencies takes place in the cut set file. This F&O will have no effect on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.) This finding was made against NEI SR DE-5 with grade 3 being contingent on its resolution. F&O QU-02 - The IE's for certain support system failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE. This finding was made against NEI SR DE-5 with grade 3 being contingent on its resolution.	of AFW pump, so AS-07 is considered resolved. F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance.	
SC-A6	CONFIRM that the bases for the success criteria are consistent with the features, procedures, and operating philosophy of the plant.	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP	The NEI SRs applicable to this ASME SR are AS-5, AS-18, AS-19, TH-4, TH-5, TH-6, TH-8, ST-4, ST-5, ST-7, ST-9 and SY-5. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-5, AS-19, ST-4 and SY-5 as "3" and AS-18, TH-4, TH-6 and ST-9 as "3 with contingencies." TH-8 and St-5 were unrated. TH-5 is rated a "2." AS-18 has one level "B" F&O: TH-03; TH-4 has one level "B" F&O: TH-03; TH-5 has two level "B" F&Os: HR-05 and TH-05; and TH-8 has one level "B" F&O: TH-06. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18 and TH-4 with grade 3 being contingent on its resolution. F&O TH-05 -The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided. This finding was made against NEI SR TH-5 with grade 2. F&O HR-05 - In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR HR-5 with grade 2.	F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O HR-05 - Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. This is considered to resolve the elements of this F&O related to this SR, and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible.

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			F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution.		
SC-B1	USE appropriate realistic generic analyses/evaluations that are applicable to the plant for thermal/hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. (See SC-B4.) Realistic models or analyses may be supplemented with plant-specific/generic FSAR or other conservative analysis applicable to the plant, but only if such supplemental analyses do not affect the determination of which combinations of systems and trains of systems are required to respond to an initiating event.	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18, SY-17 and TH-4 with grade 3 being contingent on its resolution. F&O SY-03 - System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The	The NEI SRs applicable to this ASME SR are AS-18, SY-17, TH-4, TH-6 and TH-7. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3 with contingencies". AS-18 has one level "B" F&O: TH-03; SY-17 has two level "B" F&Os: SY-03 and TH-03; TH-4 has one level "B" F&O: TH-03; and TH-7 has one level "B" F&O: TH-01. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O SY-03 - Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to these system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met. F&O TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current	Peer Review F&Os SY-03 and TH-01 are still open. While the success criteria has been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire (sister plant for Catawba) has been updated and no change in mitigation equipment was identified.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. This finding was made against NEI SR SY-17 with grade 3 being contingent on its resolution. F&O TH-01 - Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. This finding was made against NEI SR TH-7 with grade 3 being contingent on its resolution.	model of record does not reflect the updated information and as a result the ASME SR is considered Not Met.	
SC-B2	DO NOT USE expert judgment except in those situations in which there is lack of available information regarding the condition or response of a modeled SSC, or a lack of analytical methods upon which to base a prediction of SSC condition or response. USE the requirements	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to	The NEI SRs applicable to this ASME SR are TH-4 and TH-8. There are no NRC objections, but the requirement to assess the availability of documentation was not contained in NEI 00-02, this should be demonstrated. The original Peer Review rated TH-4 as "3 with contingencies". TH-8 is unrated. TH-4 has one level "B" F&O: TH-03; and TH-8 has one level "B" F&O: TH-06. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely

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	in 1.-4.3 when implementing an expert judgment process.		large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18 and TH-4 with grade 3 being contingent on its resolution. F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution.	computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.
SC-B3	When defining success criteria, USE thermal/hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping (HLR-IE-B) and accident sequence modeling (HLR-AS-A and HLR-AS-B).	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the	The NEI SRs applicable to this ASME SR are AS-18, TH-4, TH-6 and TH-7. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3 with contingencies", except TH-5, which is rated a "2". AS-18 has one level "B" F&O: TH-03; TH-4 has one level "B" F&O: TH-03; TH-5 has two level "B" F&Os: HR-05 and TH-05; and TH-7 has one level "B" F&O: TH-01. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using	Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria, so the impact on the ILRT extension is expected to be negligible.

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			accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18 and TH-4 with grade 3 being contingent on its resolution. F&O TH-05 -The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided. This finding was made against NEI SR TH-5 with grade 2. F&O HR-05 - In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA	alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O HR-05 - Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. This is considered to resolve the elements of this F&O related to this SR, and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met.	Peer Review F&O TH-01 is still open. While the success criteria has been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible.

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			limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR TH-5 with grade 2. F&O TH-01 - Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. This finding was made against NEI SR TH-7 with grade 3 being contingent on its resolution.		
SC-B4	USE analysis models and computer codes that have sufficient capability to model the conditions of interest in the determination of success criteria for CDF, and that provide results representative of the plant. A qualitative evaluation of a relevant application of codes, models, or analyses that has been used for a similar class of plant (e.g., Owner's Group generic studies) may be used. USE computer codes and models only within known limits of applicability.	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made	The NEI SRs applicable to this ASME SR are AS-18, TH-4, TH-6, and TH-7. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3 with contingencies". AS-18 has one level "B" F&O: TH-03; TH-4 has one level "B" F&O: TH-03; and TH-7 has one level "B" F&O: TH-01. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current	Peer Review F&O TH-01 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			against NEI SR AS-18 and TH-4 with grade 3 being contingent on its resolution. F&O TH-01 - Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. This finding was made against NEI SR TH-7 with grade 3 being contingent on its resolution.	model of record does not reflect the updated information and as a result the ASME SR is considered Not Met.	
SC-B5	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include: (a) comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features (b) comparison with results of similar analyses performed with other plant-specific codes (c) check by other means appropriate to the particular analysis	Open	F&O TH-01 - Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. This finding was made against NEI SR TH-7 with grade 3 being contingent on its resolution. F&O TH-05 -The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided. This finding was made against NEI SR	The NEI SRs applicable to this ASME SR are TH-7 and TH-9. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated both of these NEI SRs as "3 with contingencies". TH-7 has one level "B" F&O: TH-01; and TH-9 has two level "B" F&Os: TH-05 and TH-06. F&O TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			TH-5 with grade 2. F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution.	current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Peer Review F&O TH-01 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria, so the impact on the ILRT extension is expected to be negligible.
SC-C1	DOCUMENT the success criteria in a manner that facilitates PRA applications, upgrades, and peer review.	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. This finding was made against NEI SR AS-18 and SY-17 with grade 3 being contingent on its resolution.	The NEI SRs applicable to this ASME SR are ST-13, SY-10, SY-17, SY-27, TH-8, TH-9, TH-10, AS-17, AS-18, AS-24 and HR-30. There are no industry self-assessment actions and no NRC objections. The original Peer Review rated ST-13, TH-10, AS-17, and AS-24 as "3" and SY-17, ST-27, TH-9, and AS-18 as "3 with contingencies." TH-8 and HR-30 were unrated. SY-10 is rated a "2." SY-10 has one level "B" F&O: TH-06; SY-17 has two level "B" F&Os: SY-03 and TH-03; SY-27 has one level "B" F&O: SY-03; TH-8 has one level "B" F&O: TH-06; TH-9 has two level "B" F&Os: TH-05 and TH-06; AS-18 has one level "B" F&O: TH-03; and HR-30 has one level "B" F&O: HR-05. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O HR-05 - Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the

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			F&O HR-05 - In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR HR-30 with grade 2. F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution. F&O SY-03 - System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no	References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. This is considered to resolve the elements of this F&O related to this SR, and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application. F&O SY-03 - Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to these system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.

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			basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. This finding was made against NEI SR SY-17 with grade 3 being contingent on its resolution.		
SC-C2	DOCUMENT the processes used to develop overall PRA success criteria and the supporting engineering bases, including the inputs, methods, and results. For example, this documentation typically includes: (a) the definition of core damage used in the PRA including the bases for any selected parameter value used in the definition (e.g., peak cladding temperature or reactor vessel level) (b) calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used (c) identification of computer codes	Open	F&O TH-03 - Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs.	The NEI SRs applicable to this ASME SR are ST-13, SY-10, SY-17, SY-27, TH-8, TH-9, TH-10, AS-17, AS-18, AS-24 and HR-30. There are no NRC objections, but the requirement to assess the availability of documentation was not explicitly contained in NEI 00-02, this should be demonstrated. The original Peer Review rated ST-13, TH-10, AS-17, and AS-24 as "3" and SY-17, ST-27, TH-9, and AS-18 as "3 with contingencies." TH-8 and HR-30 were unrated. SY-10 is rated a "2." SY-10 has one level "B" F&O: TH-06; SY-17 has two level "B" F&Os: SY-03 and TH-03; SY-27 has one level "B" F&O: SY-03; TH-8 has one level "B" F&O: TH-06; TH-9 has two level "B" F&Os: TH-05 and TH-06; AS-18 has one level "B" F&O: TH-03; and HR-30 has one level "B" F&O: HR-05. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	or other methods used to establish plant-specific success criteria (d) a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) of the calculations or codes (e) the uses of expert judgment within the PRA, and rationale for such uses (f) a summary of success criteria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA (g) the basis for establishing the time available for human actions (h) descriptions of processes used to define success criteria for grouped initiating events or accident sequences		This finding was made against NEI SR AS-18 and SY-17 with grade 3 being contingent on its resolution. F&O HR-05 - In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR HR-30 with grade 2. F&O TH-06 - There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. This finding was made against NEI SR TH-9 with grade 3 being contingent on its resolution.	F&O HR-05 - Success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. This is considered to resolve the elements of this F&O related to this SR, and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application. F&O SY-03 - Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to these system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.

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			F&O SY-03 - System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. This finding was made against NEI SR SY-17 with grade 3 being contingent on its resolution.		
SY-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.	Dispositioned	None	NEI 00-02 does not have a precise equivalent to SY-A4. System analysis subelements DE-11, SY-10, and SY footnote 5 provide partial coverage. DE-11 requires walkdowns but specifies walkdowns to examine spatial dependencies, not, as SY-A4 requires, to confirm that the systems analysis correctly reflects the as-built, as-operated plant. Likewise, SY-10 including Footnote 5 and F&O TH-06 specifically concern spatial or environmental dependencies, which are assessed as part of SRs SY-A21 and SY-A22. F&O TH-06 does not apply to SR SY-A4. Therefore the Grade 3 given to DE-11 and the Grade 2 given to SY-10 by the peer review team do not apply to SR SY-A4. Compliance with SR-A4 is assessed by reference to the peer review teams notes on SY-10 and DE-11. The 2002 peer review's notes show that the peer review team specifically reviewed walkdown notes oriented to confirming that the systems analysis correctly reflects the as-built, as-operated plant (see notes R4 and R19 below). Therefore, SY-A4 is considered met. The proposed resolution in Table C of the self-assessment suggests an enhancement to system documentation (see below). Completion of the enhancement will solidify the assessment that SR SY-A4 is met.	Self-assessment Table E assesses the impact as follows: To support system model development, walkdowns and plant personnel interviews were performed. However, documentation of an up-to date system walkdown is not included with each system notebook. This documentation issue does not impact the ILRT extension.

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				Self-assessment Table C indicates the following resolution under SY-A4: Enhance the system documentation to include an up-to-date system walkdown checklist and system engineer review for each system. Consider revising workplace procedure XSAA-106 to require that such documentation be revisited with each major PRA revision.	
SY-A5	INCLUDE the effects of both normal and alternate system alignments, to the extent needed for CDF and LERF determination.	Dispositioned	None	SY-A5 corresponds to NEI 00-02 subelements SY-8, SY-11, QU-12, and QU-13, which provide partial coverage. Because subelement SY-11 and associated F&O SY-06 are limited to consideration of system performance in degraded environments, they are not closely related to this SR, and are assessed as part of SRs SY-A21, SY-A22, and SY-B14. QU-12 and especially QU-13 refer to model asymmetry, which does not cover the effects of normal and alternate alignments. QU-12 and QU-13 were given a Grade 3 by the peer review team.SY-8 is also related to modeling normal and alternate alignments to the extent that it mentions test and maintenance availabilities, but does not fully cover the requirement to model normal and alternate alignments. SY-8 was given a Grade 3 by the peer review team. By self-assessment, it was determined that normal and alternate system alignments are included to the extent necessary for CDF and LERF determination. Fault tree top events are included for CDF and LERF. This is considered sufficient to meet CAT II of the ASME/ANS PRA Standard.	Based on the disposition, the requirements of Cat II are considered met. There is no impact on the ILRT extension.
SY-A10	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. Example causes of variable system success criteria are: (a) different accident scenarios. Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled	Open	F&O DE-04: HVAC cooling of the essential switchgear rooms is stated as being required. The IPE quantitative analysis does not provide adequate success criteria. For example, the following are not specified: temperature limits of equipment, minimum number of Air Handling Units, or minimum number of chillers. The evaluation also states there is no concern within 24 hours provided that only those loads needed to provide core cooling are operated. There is no discussion of electrical load shedding for those loads not required, and of the human interface to execute load shedding. The human interface can be complex, involving both a discovery process (control room annunciators, or in the case of a local AHU failure, discovery through operator walkaround), and procedures and training to direct	SY-A10 corresponds to NEI 00-02 system analysis subelements SY-12, SY-13, SY-17, and SY-23 and accident sequence subelements AS-10, AS-13, AS-16, AS-17, and AS-18. The 2002 peer review team assigned unconditional Grade 3 to AS-13, AS-17, SY-12, SY-13, and SY-13, and a contingent Grade 3 to AS-10, AS-16, AS-18 and SY-17 with level "B" F&Os: DE-04, AS-01, SY-03 and TH-03. F&O DE-04 (PRATracker Item C-03-0052) and TH-03 (PRATracker Item C-03-0050) are both open. F&O DE-04 is not resolved because the loss of switchgear HVAC initiating event is not included in the PRA. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.

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	initiating event); (b) dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if non-critical loads are not isolated); (c) time dependence. Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident); (d) sharing of a system between units. Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).		operation actions. F&O AS-01: SAAG 427 describes the ATWS event tree analysis. Section 4, event B, describes how main feedwater is recovered after an ATWS. The probabilities used for main feedwater recovery are .05, following a T2 (Loss of Load) and .2 following a T4 (Loss of MFW). In the non-ATWS analysis, the following non-recoveries (From SAAG 427) are: T1 non-rec = .05 T4 - non-rec = .1 Considering that the critical time for FW to come on line in an ATWS event involving a loss of main feedwater is very short, even for conditions of favorable MTC, the use of non-recovery probabilities of this magnitude does not appear to be justified without supporting analyses. F&O TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs.	F&O AS-01 - Credit for Main Feedwater has been removed from the ATWS model, which resolves this F&O. Recovery for MFW in ATWS events initiated by a loss of feedwater has no impact on Fire PRA. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.

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			F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis.		
SY-A11	INCLUDE in the system model those failures of the equipment and components that would affect system operability (as identified in the system success criteria), except when excluded using the criteria in SY-A15. This equipment includes both active components (e.g., pumps, valves, and air compressors) and passive components (e.g., piping, heat exchangers, and tanks) required for system operation.	Open	None	The peer review found that the NEI 00-02 subelements SY-6, SY-7, SY-8, SY-9, SY-12, SY-13, and SY-14 were met at Grade 3. However, RG 1.200, Rev 2 indicates that the subelements listed provide only partial coverage. As evidenced by the note below from the peer review report, the peer review team considered the modeling of passive failures. Because the requirement for including passive failures is different in subelement SY-7 and SY-A11, this SR is considered not met. With respect to passive failure, the peer review report notes that: (R11) Passive failures were found in the model for check valves leaks and ruptures, heat exchanger leaks and fouling, orifice plugging, and valves transferring closed. An event for Service Water (RN) failure due to clams was also noted. Per Duke self-assessment, the system models include multiple failure modes for most components, and all modeled components and associated failure modes are documented in the system notebooks. Assumptions regarding components or failure modes excluded from the model are documented in the assumptions section of the system	Passive equipment typically has high reliability. Therefore, passive equipment is not expected to contribute significantly to risk. Multiple spurious operations are considered in the PRA. There is no impact on the ILRT extension.

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				write-ups. Passive failures such as tanks and heat exchangers are modeled, although passive piping failures are generally not modeled since they are probabilistically insignificant. (A few exceptions: basic event BWSTPTHDEX, "Pipe Rupture Fails Flow From the BWST;" SUMPRECDEX, "Pipe Rupture Fails Flow From the RE Emergency Sump;" WSSPIPEDEX, "Random Pipe Break in SSW System.")	
SY-A12	DO NOT INCLUDE in a system model component failures that would be beneficial to system operation, unless omission would distort the results. Example of a beneficial failure: A failure of an instrument in such a fashion as to generate a required actuation signal.	Dispositioned	None	According to RG 1.200, Rev. 2, this SR is not fully covered by NEI 00-02. Partial coverage is provided by subelements SY-6, SY-7, SY-8, SY-9, SY-12, SY-13, SY-14 which were all met at Grade 3. The peer review report does not provide evidence that beneficial failures were excluded. Therefore, this SR was not fully evaluated in the 2003 peer review. The two Catawba PRA Quality Self-Assessments (DPC-1535.00-00-0013 and CNC-1535.00-00-0155) found that this SR is met, and noted that beneficial failures are not included in the system models. By the nature of the system analysis methodology (fault trees), such failures are not easily included, because lower level failures result in failure of higher level functions.	Based on the disposition, the requirements of Cat II are considered met. There is no impact on the ILRT extension.
SY-A13	INCLUDE those failures that can cause flow diversion pathways that result in failure to meet the system success criteria.	Dispositioned	F&O: SY-04: In the KC System Notebook (SAAG File No. 294), there is no basis provided in Section 11.3 for excluding the failure to isolate the Non-Essential Reactor Building Header from the fault trees. In discussion with the PRA engineer responsible for the notebook update, it was determined that three valves need to fail to close for flow diversion to take place, but there could be a common cause failure of these valves that was not justified to be excluded. F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN	SY-A13 corresponds to NEI 00-02 subelements SY-15 and SY-17. F&Os SY-04 and TH-03 are also associated with this SR. Although neither subelement SY-15 nor SY-17 specifically requires diversion pathways to be considered, F&O SY-04 makes it clear that the peer reviewers evaluated the sufficiency of the modeling of flow diversions. The F&O has been addressed by justifying excluding the potential diversion flowpath in the system notebook. In order for an open flow path to the non-essential header to starve flow to required components there would have to be failures of multiple components (pumps and valves). The following assumption has been added to the KC system notebook: "The reactor building non-essential headers are not included in the fault tree as potential diversion pathways. In addition to failure of the reactor non-essential headers valves (KC3, -18, -228, and -230), valves KC338, -424, and -425, which receive an Sp signal to close, would have to fail. Common cause failure of all of the involved valves is considered probabilistically insignificant." CNC-1535.00-00-0038, Section K.11.3 "Assumptions" provides some qualitative consideration based upon the number of valves that would be required to fail. In addition CNC-1535.00-00-0011, Section 3.1, provides discussion regarding consideration of the loss of KC due to flow diversion pathway.	Flow diversion is considered in the PRA. There is no impact on the ILRT extension. Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.

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			notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. F&O TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs.	F&O SY-04 is closed by the clarification in the system notebook. The F&O itself is evidence that the peer review assessed to some extent whether diversion flow paths were adequately modeled. However, because NEI 00-02 does not explicitly require flow diversion pathways to be considered, the SR is considered not met at Category II. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	
SY-A15	In meeting SY-A11 and SY-A14, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met (a) A component may be excluded	Dispositioned	None	This SR is not covered in NEI 00-02. Therefore, this SR was not evaluated in the 2002 peer review. The Catawba PRA Quality Self-Assessment (CNC-1535.00-00-0155) found that this SR is met. The earlier self-assessment noted that some failure modes are excluded in a qualitative fashion rather than by using quantitative criteria. It was noted that it was an issue of not documenting the quantitative evaluations for screening.	Based on the disposition, the requirements of Cat II are considered met. There is no impact on the ILRT extension.

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	from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. (b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same				
SY-A18	INCLUDE in either the system model or accident sequence modeling those conditions that cause the system to isolate or trip, or those conditions that once exceeded cause the system to fail, or SHOW that their exclusion does not impact the results. For example, conditions that isolate or trip a system include: (a) system-related parameters such as a high temperature within the system (b) external parameters used to protect the system from other failures [e.g., the high reactor pressure vessel (RPV) water level isolation signal used to prevent water intrusion into the turbines of the RCIC and HPCI pumps of a	Open	F&O SY-06: For Catawba, there was no evaluation of the ability of non-qualified (non-EQ) equipment to survive in a degraded environment following an accident such as a steam line of feedwater line break outside of containment. F&O TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not	SY-A18 corresponds to NEI 00-02 AS-13, SY-10, SY-11, SY-13, and SY-17. The peer review gave Grade 2 to SY-10 and SY-11, which have to do with the proper consideration of spatial dependencies and adverse environmental conditions (notes R19 and R20 below are given by the peer review team in support of Grade 2.) SY-17, which treats the bases for success criteria, received a contingent Grade 3 due to level "B" F&Os TH-03 and SY-03. This SR is considered not met at Category II because the corresponding NEI 00-02 subelements are not given Grade 3 or better. R19. Evidence of plant walkdowns being performed was found in the design-basis calculation performed for the flooding analysis. The only spatial information in the system notebooks is a basic description of equipment locations. No discussion of room cooling dependence for systems was found in the system notebooks and heatup calculations were not retrievable (see F&O TH-06). R20. The PRA staff confirmed that there was no search performed for non-qualified equipment that was credited in the PRA to perform in degraded environments. See F&O SY-06.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. High-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). This is considered resolved. There is no impact on the

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	BWR] (c) adverse environmental conditions (see SY-A22)		appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. F&O TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. (Duke is already aware of this issue.)	F&O SY-06 is resolved because high-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	ILRT extension.
SY-A20	INCLUDE events representing the simultaneous unavailability of	Dispositioned	None	This SR is not covered in NEI 00-02. Therefore, this SR was not evaluated in the 2003 peer review. The Catawba PRA Quality Self-	Based on the disposition, the requirements of Cat II of the

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	redundant equipment when this is a result of planned activity (see DA-C14).			Assessments (DPC-1535.00-00-0013 and CNC-1535.00-00-0155) found that this SR is met noting that maintenance events are generally treated as independent within the PRA model, however, after the model is solved, cut sets involving coincident maintenance are deleted where such combinations are prohibited by the technical specifications, as documented in the model integration notebook. Cut sets involving coincident maintenance combinations prohibited by the online risk assessment tool are retained, but have their probability reduced.	ASME/ANS Standard are considered to be met. There is no impact on the ILRT extension.
SY-A21	IDENTIFY system conditions that cause a loss of desired system function, (e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.).	Dispositioned	F&O TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. F&O TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms	RG 1.200 Rev. 2 maps several NEI 00-02 subelements to this SR: AS-18, DE-10, SY-11, SY-13, SY-17, and TH-8. The 2002 peer review report associates the following level "B" F&Os to one or more of these subelements: TH-03, TH-06, SY-03, and SY-06. F&O DE-06 is also associated with DE-10 but has been superseded by the more recent focus-scope peer review for the Flooding PRA model. Based on the 2002 peer review report's assignment of Grade 2 to subelement SY-11, and contingent Grade 3 to subelements SY-17, AS-18, and DE-10 the SR is considered not met at SR Category II. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been	High-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). This is considered resolved. There is no impact on the ILRT extension. Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria [Reference

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			and small rooms housing critical pumps. (Duke is already aware of this issue.) F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. F&O SY-06: For Catawba, there was no evaluation of the ability of non-qualified (non-EQ) equipment to survive in a degraded environment following an accident such as a steam line of feedwater line break outside of containment.	added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met. F&O SY-06 is resolved because high-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38).	45], so the impact on the ILRT extension is expected to be negligible. Additionally the success criteria for McGuire, (sister plant for Catawba), has been updated and no change in mitigation equipment was identified.
SY-A22	TAKE CREDIT for system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	Dispositioned	F&O: SY-04: In the KC System Notebook (SAAG File No. 294), there is no basis provided in Section 11.3 for excluding the failure to isolate the Non-Essential Reactor Building Header from the fault trees. In discussion with the PRA engineer responsible for the notebook update, it was determined that three valves need to fail to close for flow diversion to take place, but there could be a common cause failure of these valves that was not justified to be excluded. F&O SY-06: For Catawba, there was no evaluation	RG 1.200 Rev. 2 maps several NEI 00-02 subelements to this SR: AS-19, SY-5, SY-11, SY-13, SY-22, and TH-8. The 2002 peer review report assigns Grade 2 to subelement SY-11 and contingent Grade 3 to subelement SY-22. In addition NEI 00-02 only provides partial coverage of this SR. Therefore, the SR is considered not met at SR Category II. The 2002 peer review report associates the following level "B" F&Os to one or more of these subelements: TH-06, SY-04, and SY-06. F&O SY-04 has been addressed by justifying excluding the potential diversion flowpath in the system notebook. The following assumption	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no

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			of the ability of non-qualified (non-EQ) equipment to survive in a degraded environment following an accident such as a steam line of feedwater line break outside of containment. F&O TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. (Duke is already aware of this issue.)	has been added to the KC system notebook: "The reactor building non-essential headers are not included in the fault tree as potential diversion pathways. In addition to failure of the reactor non-essential headers valves (KC3, -18, -228, and -230), valves KC338, -424, and -425, which receive an Sp signal to close, would have to fail. Common cause failure of all of the involved valves is considered probabilistically insignificant." F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN The loss of HVAC was screened as an initiating event in the CNP model based on judgment that sufficient time would allow for diagnosis of the loss of HVAC and recovery of standby equipment and/or alternate means of cooling. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application. F&O SY-06 is not resolved because an evaluation of potential adverse effects on equipment operation due to degraded environmental conditions resulting from accidents in the PRA model has not been performed for events like steam line breaks and feed line breaks (Ref: PRATracker C-03-0055). The FPRA considers the impact of fire on the environment in the HGL analysis. High energy line breaks are not relevant to the FPRA.	impact to the ILRT extension. High-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). This is considered resolved. There is no impact on the ILRT extension.
SY-B5	ACCOUNT explicitly for the modeled system's dependency on support systems or interfacing systems in the modeling process. This may be accomplished in one of the following ways: (a) for the fault tree linking approach by modeling the dependencies as a link to an appropriate event or gate in the support system fault tree; (b) for the linked event tree approach, by using event tree logic rules, or calculating a probability for	Open	F&O DE-04: HVAC cooling of the essential switchgear rooms is stated as being required. The IPE quantitative analysis does not provide adequate success criteria. For example, the following are not specified: temperature limits of equipment, minimum number of Air Handling Units, or minimum number of chillers. The evaluation also states there is no concern within 24 hours provided that only those loads needed to provide core cooling are operated. There is no discussion of electrical load shedding for those loads not required, and of the human interface to execute load shedding. The human interface can be complex, involving both a discovery process (control room annunciators, or in the case of a local AHU failure, discovery through operator walkaround), and procedures and training to direct operation actions.	This SR is covered by NEI 00-02 subelements DE-4, DE-5, DE-6, and SY-12. The 2002 peer review report assigns Grade 3 to subelements DE-6 and SY-12 and contingent Grade 3 to subelements DE-4 and DE-5. Level "B" F&Os associated with subelements DE-4 and DE-5 are DE-04, QU-02, and AS-07. F&O DE-04 is not resolved because the loss of switchgear HVAC initiating event is not included in the PRA. A recent evaluation was performed (PIP C-13-05664) to determine the impact on the Fire PRA of not including switchgear room and battery HVAC modeling. The evaluation concluded that any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on the Fire PRA results or results for the NFPA-805 application. F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension. System level initiators modeled directly as fault

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	each split fraction conditional on the scenario definition.		F&O QU-02: The IE's for certain support system failures (RN, KC) are not input in the top event logic as a boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE. F&O AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.)	it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved.	trees will have little effect on the ILRT extension.
SY-B7	BASE support system modeling on realistic success criteria and timing, unless a conservative approach can be justified (i.e., if their use does not impact risk significant contributors.)	Dispositioned	F&O: SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN	This SR is covered by NEI 00-02 subelements AS-18, SY-13, SY-17, TH-7, and TH-8. The 2002 peer review gave contingent Grade 3 to subelements SY-17 and TH-7. Therefore, this SR is not met at Category II. Associated level "B" F&Os are: TH-01, TH-03, TH-06 and SY-03. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	Peer Review F&O SY-03 is still open. While the success criteria have been updated, it has not been incorporated into the PRA model. However, there are no significant changes to the success criteria, so the impact on the ILRT extension is negligible. Additionally, the success criteria for McGuire, (sister

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			notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. F&O TH-01: Success Criteria (Level 1 and Level 2) for some systems and sequences are supported by MAAP runs with MAAP 3b, Version 16. This version of MAAP has been found to have limitations which can impact conclusions and results. In particular for the Catawba PRA, the simple pressurizer model likely impacts the analyses that involve RCS cooldown and depressurization using SG heat removal by permitting RCS depressurization to match RCS cooldown for transients, without the possible need for pressurizer PORVs, spray or aux spray. F&O TH-03: Success Criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, MLOCA success criteria analyses are done for a 3.5 inch break (file SAAG 96), while the MLOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large LOCA (8.25 ft2 break analyzed in SAAG 97) vs a break range down to 6 inches, and small LOCA (1 inch break analyzed, SAAG 95) vs. break sizes from 3/8 to 2 inches. Further, it was not clear that the MLOCA MAAP runs adequately match the accident sequence being modeled in the PRA. Cases in SAAG 96 do not appear to disable accumulators when defining the minimum ECC requirements, but accumulators are not required by the resulting MLOCA success	TH-01 - An updated success criteria calculation was completed using MAAP 4.0.7 (Section 2.2) and is documented into the updated CNS Success Criteria Notebook. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. F&O TH-03 - As part of establishing success criteria, a series of analyses were performed over a range of applications to ensure that computer codes employed provided realistic results. Success criteria sensitivities included analyses for a range of possible conditions, including the LOCA break sizes and availability of accumulators. In addition, a review of other industry design-basis calculations using alternate methods was employed to consider code limitations. This is considered to resolve the finding and achieve grade 3 of the NEI SR/ meet cat II of the ASME SR. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN F&O TH-06 is not resolved because the loss of switchgear HVAC initiating event is not included in the PRA, and room heatup calculations for loss of ventilation are not performed for that and other locations. Room heatup calculations should be performed in all locations in which HVAC can be lost to justify not modeling those systems and/or determine timing of operator coping actions and equipment damage. If no room heatup calculation is performed, it should be assumed that the HVAC system is required in those locations. The appropriate dependencies should be included in the PRA model, including possible initiating events. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	plant for Catawba), has been updated and no change in mitigation equipment was identified. Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.

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			criteria. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. F&O TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. (Duke is already aware of this issue.)		
SY-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. Example: Use results of plant walkdowns as a source of information regarding spatial/environmental hazards, for resolution of spatial/environmental issues, or evaluation of the impacts of such hazards.	Open	F&O TH-06: There is no room heatup analysis notebook / evaluation of loss of HVAC to equipment rooms for the Catawba PRA, and apparently no retrievable room heatup calculations or documentation to support the assumption that room cooling need not be modeled in the PRA. Other PRAs have found that room cooling is required for some rooms such as electrical equipment rooms and small rooms housing critical pumps. (Duke is already aware of this issue.)	This SR is covered by NEI 00-02 subelements DE-11 and SY-10. This SR covers the same requirement as NEI 00-02 subelement SY-10, but is more specific. The 2002 peer review gave Grade 2 to SY-10. Therefore, this SR is not met at Category II. F&O TH-06 - CNP PRA Tracker ID C-03-0052 for TH-06 - OPEN F&O TH-06 is not resolved because the loss of switchgear HVAC initiating event is not included in the PRA, and room heatup calculations for loss of ventilation are not performed for that and other locations. Room heatup calculations should be performed in all locations in which HVAC can be lost to justify not modeling those systems and/or determine timing of operator coping actions and equipment damage. If no room heatup calculation is performed, it should be assumed that the HVAC system is required in those locations. The appropriate dependencies should be included in the PRA model, including possible initiating events. Any additional risk added by including the VC/YC systems in the PRA model would be small and would not have a significant impact on results for the ILRT extension application.	Room heat-up analyses were performed for the switchgear rooms, battery rooms, and the control room [References 40, 41, and 42]. The results of these analyses show that equipment in these rooms will not be adversely impacted by the loss of HVAC over the 24-hour mission time. There is no impact to the ILRT extension.
SY-B10	MODEL those systems that are required for initiation and actuation of a system. In the model quantification, INCLUDE the presence of the conditions needed for automatic actuation (e.g., low vessel water level). INCLUDE permissive and lockout signals that	Dispositioned	None	This SR is covered by NEI 00-02 subelements SY-8, SY-12 and SY-13. Even though the 2002 peer review gave unconditional Grade 3 to all of these NEI 00-02 subelements, NEI 00-02 does not explicitly address permissives and control logic. The reviewers' notes in the peer review report do not show that they assessed the model with respect to permissives and control logic. Therefore, the peer review cannot be used to fully assess compliance with this requirement. The Catawba PRA Quality Self-Assessments (DPC-1535.00-00-0013 and	Based on the disposition, the requirements of Cat II of the ASME/ANS Standard are considered to be met. There is no impact on the ILRT extension.

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	are required to complete actuation logic.			CNC-1535.00-00-0155) determined that this SR is met, noting that systems required for initiation and actuation of other systems (e.g., ESFAS) are explicitly modeled, and the presence of conditions needed for automatic actuation and permissive and lockout signals required to complete actuation logic are included.	
SY-B12	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system such as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.	Dispositioned	None	This SR is not covered in NEI 00-02. The Catawba PRA Quality Self-Assessments (DPC-1535.00-00-0013 and CNC-1535.00-00-0155) found that this SR is met, noting that no systems are excluded based on proceduralized recovery actions. In addition, proceduralized recovery actions are modeled for some support systems (e.g., manually actuate systems after ESFAS failure).	Based on the disposition, the requirements of Cat II of the ASME/ANS Standard are considered to be met. There is no impact on the ILRT extension.
SY-B13	Some systems use components and equipment that are required for operation of other systems. INCLUDE components that, using the criteria in SY-A15, may be screened from each system model individually, if their failure affects more than one system (e.g., a common suction pipe feeding two separate systems).	Dispositioned	None	NEI 00-02 does not fully address this SR; subelements DE-6 and AS-6 partially address it. Therefore, compliance with this SR was not completely evaluated in the 2002 peer review. However, the peer review gave unconditional Grade 3 to both of these NEI 00-02 subelements, The Catawba PRA Quality Self-Assessments (DPC-1535.00-00-0013 and CNC-1535.00-00-0155) found that this SR is met, noting that the system notebooks include assumptions regarding components or failure modes excluded from the model. Piping and other passive failures are not modeled if they are probabilistically insignificant. However, some pipe breaks and passive failure of tanks and heat exchangers are modeled.	Based on the disposition, the requirements of Cat II of the ASME/ANS Standard are considered to be met. There is no impact on the ILRT extension.
SY-B14	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include (a) LOCA inside containment with failure of containment heat removal (b) safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) (c) steam line breaks outside	Open.	F&O: SY-06: For Catawba, there was no evaluation of the ability of non-qualified (non-EQ) equipment to survive in a degraded environment following an accident such as a steam line of feedwater line break outside of containment.	This SR is covered by NEI 00-02 subelement SY-11. Subelement SY-11 received a Grade 3 contingent on resolution of F&O SY-06. F&O SY-06 is not resolved because an evaluation of potential adverse effects on equipment operation due to degraded environmental conditions resulting from accidents in the PRA model has not been performed for events like steam line breaks and feed line breaks (Ref: PRATracker C-03-0055). The FPRA considers the impact of fire on the environment in the HGL analysis. High energy line breaks are not relevant to the FPRA. The SR is not met at Category II because the peer review gave Grade 2 to subelement SY-11.	The FPRA considers the impact of fire on the environment in the HGL analysis. High-energy line breaks (e.g., steam line breaks and feed line breaks) are addressed in the Internal Flood PRA (Reference 38). This is considered resolved. There is no impact on the ILRT extension.

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	containment (d) debris that could plug screens/filters (both internal and external to the plant) (e) heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability (f) loss of NPSH for pumps (g) steam binding of pumps				
SY-C1	DOCUMENT the systems analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Dispositioned	F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis.	SY-C1 corresponds to NEI-00-02 subelements SY-5, SY-6, SY-9, SY-18, SY-23, SY-25, SY-26, SY-27. The 2002 peer review report gives Grade 3 to these subelements except SY-27 which is contingent on resolution of F&O SY-03. Based on the 2002 peer review report's contingent Grade 3 for subelement SY-27, the SR is considered not met at SR Category II. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met.	No impact from documentation changes.
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and results. For example, this documentation typically includes:	Dispositioned	F&O SY-03: System success criteria are specified in the system notebooks in sufficient detail to describe the overall fault tree top events, but no basis is provided in the system notebooks for the number of pumps or flow rate requirements. The Reference section 18.1 does not contain a link to an appropriate success criteria calculation. For example, in the KC notebook, it is stated without a source reference that both pumps and the associated heat exchanger in a train are required for success when the ND (RHR) heat exchanger is	The peer review found that the NEI-00-02 subelements corresponding to SR SY-C2 (SY-5, SY-6, SY-9, SY-18, SY-23, SY-25, SY-26, and SY-27), according to RG 1.200, Rev. 2, were met at Grade 3 (with a contingent grade for SY-27 corresponding to F&O SY-03). In addition, RG 1.200 Rev. 2 indicates that the corresponding NEI-00-02 subelements only partially cover the current requirements in the SR. Therefore, the SR is considered not met. F&O SY-03 – Although XSAA-115 (PRA Modeling Guidelines) has been revised to require success criteria reference to be provided, references to the appropriate system success criteria have not been	No impact from documentation changes.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	(a) system function and operation under normal and emergency operations (b) system model boundary (c) system schematic illustrating all equipment and components necessary for system operation (d) information and calculations to support equipment operability considerations and assumptions (e) actual operational history indicating any past problems in the system operation (f) system success criteria and relationship to accident sequence models (g) human actions necessary for operation of system (h) reference to system-related test and maintenance procedures (i) system dependencies and shared component interface (j) component spatial information (k) assumptions or simplifications made in development of the system models (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes (m) a description of the modularization process (if used) (n) records of resolution of logic loops developed during fault tree linking (if used) (o) results of the system model evaluations (p) results of sensitivity studies (if used) (q) the sources of the above information (e.g., completed		required. Similarly, in Section 12 of the RN notebook, it is stated that the top events simply represent "failure to provide sufficient flow" to components requiring cooling without defining a flow rate or number of pumps (in Section 13 of the notebook it does state that failure to provide flow requires failure of all four pump trains). The CA notebook has a similar statement without a tie to a specific basis. F&O DE-01: No specific guidance is given regarding modeling of system dependencies in the system notebooks; however, a highly knowledgeable analyst could reproduce the given results. A dependency matrix is provided but contains little detailed explanation of how dependencies were determined. The Internal Flood Analysis does not seem to provide the detail required to reproduce the results except by a highly knowledgeable analyst.	added to the system notebooks. As a result, this F&O remains open due to incomplete documentation. This F&O remains open with grade 3 of NEI SR / meet CAT II of the ASME SR being not met. F&O DE-01: PRA Modeling Guidelines XSAA-115 was revised to provide guidance regarding modeling of system dependencies.	

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	checklist from walkdowns, notes from discussions with plant personnel) (r) basic events in the system fault trees so that they are traceable to modules and to cutsets. (s) the nomenclature used in the system models.				
HR-B1	If screening is performed, ESTABLISH rules for screening individual activities from further consideration. Example: Screen maintenance and test activities from further consideration only if (a) equipment is automatically re-aligned on system demand, or (b) following maintenance activities, a post-maintenance functional test is performed that reveals misalignment, or (c) equipment position is indicated in the control room, status is routinely checked, and realignment can be affected from the control room, or (d) equipment status is required to be checked frequently (i.e., at least once a shift)	Dispositioned	F&O HR-02: A screening value of 3E-3 was initially used for all pre-initiator HEPs. There were 7 HEPs quantified in more detail, because the HEP importance was too high. However, there were 7 Latent Human Error events with a 3E-3 probability in the top 100 importance events in the CR2b quantification. This observation does not necessarily have a large impact on the PRA results. However, per the HR subtier criteria, screening HEPs should not be used for actions appearing in important contributors.	The NEI SRs applicable to this ASME SR are HR-5 and HR-6, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-5 as "3" but HR-6 as "2" with associated level "B" F&Os HR-02 and TH-05. TH-05 does not seem relevant to this SR because pre-initiator HEPs are not based on thermal-hydraulics timing. Both Self-assessments identified this element as N/A on the basis that "Screening is not performed." Section 2.1 of the revised HRA Calc CNC-1535.00-00-0030 states that "The screening values permitted those pre-initiator actions that could be important with respect to the frequencies of core-damage sequences to be highlighted during the quantification process. Interactions that were not important to any of the core-damage sequences based on use of the screening values were not modeled or quantified further. Those interactions that surfaced as potentially important during the sequence quantification process were then evaluated in more detail in the second stage." As a result, Table 2. Summary of Pre-Initiator (Type A) Human Interactions shows that 23 of the 56 pre-initiators were quantified with detailed analysis. F&O HR-02: F&O HR-02 remains open (Ref: PRATracker C-03-0058) to provide detailed quantification of the dominant pre-initiator HEPs. Detailed evaluations have been performed for 24 of 58 (41%) of the pre-initiator human error events (LHEs). Different LHEs may be more significant for fire than for internal event sequences since a fire can fail multiple components. However, cut sets that contain the screening value LHEs would be expected to decrease in importance since detailed evaluations tend to lower the probabilities assigned to the LHEs. Review of the cutsets data verified incorporation of mean LHE values into the database.	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
HR-B2	DO NOT screen activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems (HR-A3).	Dispositioned	F&O HR-02: A screening value of 3E-3 was initially used for all pre-initiator HEPs. There were 7 HEPs quantified in more detail, because the HEP importance was too high. However, there were 7 Latent Human Error events with a 3E-3 probability in the top 100 importance events in the CR2b quantification. This observation does not necessarily have a large impact on the PRA results. However, per the HR subtier criteria, screening HEPs should not be used for actions appearing in important contributors.	The NEI SRs applicable to this ASME SR are DA-5, DA-6, HR-5, HR-6, HR-7, and HR-26, and there are no NRC objections. There is an industry action to ensure single actions with multiple train consequences are evaluated in pre-initiators, since the screening rules in HR-6 do not preclude screening of activities that can affect multiple trains of a system. The original Peer Review rated DA-5, HR-5, HR-7, and HR-26 as "3", but HR-6 as "2" with associated level "B" F&Os HR-02 and TH-05, and DA-6 was "N/A". TH-05 does not seem relevant to this SR because pre-initiator HEPs are not based on thermal-hydraulics timing. DA-5 also has one level "B" F&Os: DA-01, but this F&O is not related to this SR, since the F&O is on component boundaries. Both Self-assessments identified this element as N/A on the basis that "Screening is not performed." CNC-1535.00-00-0030, Appendix F Catawba Nuclear Station Miscalibration Human Reliability Analysis discusses HR-B2 in Section 3, Screening. It states that "According to the ASME PRA Standard supporting requirement HR-B2, activities that could simultaneously impact multiple trains of redundant or diverse equipment are not to be screened out. The simultaneous impact does not mean that an activity simultaneously impacts redundant trains while the activity is being performed, but that the activity or activities performed in a procedure can render redundant or diverse trains unavailable simultaneously. For example, a calibration procedure would sequentially step through the calibrations of redundant channels measuring the same parameter. Although only one channel is calibrated at a time, more than one channel may be miscalibrated – impacting redundant channels simultaneously. In general, calibration activities performed on redundant channels should therefore not be screened out." F&O HR-02: F&O HR-02 remains open (Ref: PRATracker C-03-0058) to provide detailed quantification of the dominant pre-initiator HEPs. Detailed evaluations have been performed for 24 of 58 (41%) of the pre-initiator human error events (LHEs). Different LHEs may be more significant for fire than for internal event sequences since a fire can fail multiple components. However, cut sets that contain the screening value LHEs would be expected to decrease in importance since detailed evaluations tend to lower the probabilities assigned to	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				the LHEs. Review of the cutsets data verified incorporation of mean LHE values into the database.	
HR-D1	ESTIMATE the probabilities of human failure events using a systematic process. Acceptable methods include THERP [2-5] and ASEP [2-6].	Dispositioned	F&O HR-02: A screening value of 3E-3 was initially used for all pre-initiator HEPs. There were 7 HEPs quantified in more detail, because the HEP importance was too high. However, there were 7 Latent Human Error events with a 3E-3 probability in the top 100 importance events in the CR2b quantification. This observation does not necessarily have a large impact on the PRA results. However, per the HR subtier criteria, screening HEPs should not be used for actions appearing in important contributors.	The NEI SR applicable to this ASME SR is HR-6, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-6 as "2" with associated level "B" F&Os HR-02 and TH-05. TH-05 does not seem relevant to this SR because pre-initiator HEPs are not based on thermal-hydraulics timing. Section 2.1 of the revised HRA Calc CNC-1535.00-00-0030 states that "The screening values permitted those pre-initiator actions that could be important with respect to the frequencies of core-damage sequences to be highlighted during the quantification process. Interactions that were not important to any of the core-damage sequences based on use of the screening values were not modeled or quantified further. Those interactions that surfaced as potentially important during the sequence quantification process were then evaluated in more detail in the second stage." As a result, Table 2. Summary of Pre-Initiator (Type A) Human Interactions shows that 23 of the 56 pre-initiators were quantified with detailed analysis. F&O HR-02: F&O HR-02 remains open (Ref: PRATracker C-03-0058) to provide detailed quantification of the dominant pre-initiator HEPs. Detailed evaluations have been performed for 24 of 58 (41%) of the pre-initiator human error events (LHEs). Different LHEs may be more significant for fire than for internal event sequences since a fire can fail multiple components. However, cut sets that contain the screening value LHEs would be expected to decrease in importance since detailed evaluations tend to lower the probabilities assigned to the LHEs. Review of the cutsets data verified incorporation of mean LHE values into the database.	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.
HR-D2	For significant HFEs, USE detailed assessments in the quantification of pre-initiator HEPs. USE screening values based on a simple model, such as ASEP in the quantification of the pre-initiator HEPs for non-significant human failure basic events. When bounding values are used, ENSURE they are based on	Dispositioned	F&O HR-02: A screening value of 3E-3 was initially used for all pre-initiator HEPs. There were 7 HEPs quantified in more detail, because the HEP importance was too high. However, there were 7 Latent Human Error events with a 3E-3 probability in the top 100 importance events in the CR2b quantification. This observation does not necessarily have a large impact on the PRA results. However, per the HR	The NEI SR applicable to this ASME SR is HR-6, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-6 as "2" with associated level "B" F&Os HR-02 and TH-05. TH-05 does not seem relevant to this SR because pre-initiator HEPs are not based on thermal-hydraulics timing. Section 2.1 of the revised HRA Calc CNC-1535.00-00-0030 states that "The screening values permitted those pre-initiator actions that	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	limiting cases from models such as ASEP [2-6].		subtier criteria, screening HEPs should not be used for actions appearing in important contributors.	could be important with respect to the frequencies of core-damage sequences to be highlighted during the quantification process. Interactions that were not important to any of the core-damage sequences based on use of the screening values were not modeled or quantified further. Those interactions that surfaced as potentially important during the sequence quantification process were then evaluated in more detail in the second stage." As a result, Table 2. Summary of Pre-Initiator (Type A) Human Interactions shows that 23 of the 56 pre-initiators were quantified with detailed analysis. F&O HR-02: F&O HR-02 remains open (Ref: PRATracker C-03-0058) to provide detailed quantification of the dominant pre-initiator HEPs. Detailed evaluations have been performed for 24 of 58 (41%) of the pre-initiator human error events (LHEs). Different LHEs may be more significant for fire than for internal event sequences since a fire can fail multiple components. However, cut sets that contain the screening value LHEs would be expected to decrease in importance since detailed evaluations tend to lower the probabilities assigned to the LHEs. Review of the cutsets data verified incorporation of mean LHE values into the database.	
HR-D3	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	Dispositioned	None	NEI 00-02 does not explicitly address this SR and states "This item is implicitly included in the peer review of HRA by virtue of the assessment of the crew's ability to implement the procedure in an effective and controlled manner. The pre-initiator HRA adequacy is determined reasonable and representative considering the procedure quality." CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc, section 3.1 Quantification of Type A Interactions states that "Once each pre-initiator human interaction was further defined in terms of the specific failures of interest, the conditions that would affect their probabilities of occurrence were identified. These conditions, which were drawn from Table 5-2 of the ASEP methodology, include the following [Ref. 6]: (1) Whether status of the unavailable component would be indicated by a compelling signal in the control room. (2) Whether component status would be positively verified by a post-maintenance or post-calibration test. (3) Whether there would be a requirement for an independent verification of the status of the component after test or maintenance	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				activities. (4) Whether there would be a check of the component status each shift or each day, using a written checklist. An event tree was constructed to provide a framework for applying these conditions in evaluating individual pre-initiator interactions."	
HR-D4	When taking into account self-recovery or recovery from other crew members in estimating HEPs for specific HFEs, USE pre-initiator recovery factors in a manner consistent with selected methodology. If recovery of pre-initiator errors is credited (a) ESTABLISH the maximum credit that can be given for multiple recovery opportunities (b) USE the following information to assess the potential for recovery of pre-initiator: (1) post-maintenance or post-calibration tests required and performed by procedure (2) independent verification, using a written check-off list, that verifies component status following maintenance/testing (3) a separate check of component status made at a later time, using a written check-off list, by the original performer (4) work shift or daily checks of component status, using a written check-off list.	Dispositioned	F&O HR-02: A screening value of 3E-3 was initially used for all pre-initiator HEPs. There were 7 HEPs quantified in more detail, because the HEP importance was too high. However, there were 7 Latent Human Error events with a 3E-3 probability in the top 100 importance events in the CR2b quantification. This observation does not necessarily have a large impact on the PRA results. However, per the HR subtier criteria, screening HEPs should not be used for actions appearing in important contributors.	The NEI SR applicable to this ASME SR is HR-6, and there are no NRC objections. There is an industry action to use the ASME/ANS PRA Standard for requirements, since NEI 00-02 does not explicitly cite the treatment of recovery actions for pre-initiators. The original Peer Review rated HR-6 as "2" with associated level "B" F&Os HR-02 and TH-05. TH-05 does not seem relevant to this SR because pre-initiator HEPs are not based on thermal-hydraulics timing. The Type A operator action quantification spreadsheets addressed post maintenance testing, independent verification and separate checks using an event tree approach. F&O HR-02: F&O HR-02 remains open (Ref: PRATracker C-03-0058) to provide detailed quantification of the dominant pre-initiator HEPs. Detailed evaluations have been performed for 24 of 58 (41%) of the pre-initiator human error events (LHEs). Different LHEs may be more significant for fire than for internal event sequences since a fire can fail multiple components. However, cut sets that contain the screening value LHEs would be expected to decrease in importance since detailed evaluations tend to lower the probabilities assigned to the LHEs. Review of the cutsets data verified incorporation of mean LHE values into the database.	There is no impact to the ILRT extension since pre-initiator (Type A) human actions are not modified.
HR-D6	PROVIDE an assessment of the uncertainty in the HEPs in a manner	Dispositioned	None	NEI 00-02 does not address this supporting requirement.	There is no impact to the ILRT extension since pre-

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	consistent with the quantification approach. USE mean values when providing point estimates of HEPs.			DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment identified this in Table 3 as Not Met and in Table C as an Open Item. DPC-1535.00-00-013, Rev. 2., 2008 Self-assessment Table 1 states that "The Type A HEPs are not identified to be mean values and error factors are not provided in the summary table of the HR notebook (Table 2)." This is not uncommon in HRA. CNC-1535.00-00-0155, Rev. 0, 2013 Self-assessment states that this is Met and cites the Catawba Rev. 3a PRA Database as a reference.	initiator (Type A) human actions are not modified.
HR-E1	When identifying the key human response actions REVIEW: (a) the plant-specific emergency operating procedures, and other relevant procedures (e.g., AOPs, annunciator response procedures) in the context of the accident scenarios (b) system operation such that an understanding of how the system(s) functions and the human interfaces with the system is obtained	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number.	The NEI SRs applicable to this ASME SR are HR-9, HR-10, HR-16, AS-19, and SY-5, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-9, AS-19, and SY-5 as "3", but HR-10 and HR-16 as "2" with associated level "B" F&Os HR-05 and HR-04, respectively. F&Os HR-04 and HR-05: While these F&Os remain open (PRATracker items C-03-0059 and C-03-0060); CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030. DPC-1535.00-00-013, Rev. 2., 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation." Section 2.2 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that, "To delineate system response to particular types of upset events, it can be as important to understand the intended response of the operating crew in using the system as it is to understand the design of the system itself. Thus, in defining the sequence delineation for particular initiating events, it was necessary to review carefully the operating procedures, including the emergency procedures and the various abnormal procedures. This review was aimed at identifying any operator-driven considerations that would affect the modeling process, such as the priorities that might come into play when multiple options were available for maintaining core cooling, or the cues that might indicate the need to change operating modes. These procedure	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in the HRA Calc. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	reviews were augmented by obtaining input from operators. This was done by having current and former operators review the sequence logic and system fault trees; through extensive discussions with operators regarding specific scenarios; and, to the extent possible, by observing simulator exercises."	
HR-E2	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.	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	The NEI SRs applicable to this ASME SR are HR-8, HR-9, HR-10, HR-21, HR-22, HR-23, and HR-25, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-8, HR-9, HR-21 and HR-25 as "3" and HR-22 and HR-23 as "3 with contingencies." HR-10 was rated "2" with associated level "B" F&O HR-05. Also, NEI SRs HR-22 and HR-23 have F&Os HR-04 and HR-05, respectively. F&Os HR-04 and HR-05: While these F&Os remain open (PRATracker items C-03-0059 and C-03-0060); CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030. DPC-1535.00-00-013, Rev. 2., 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "The identification of human response actions included those actions required to initiate, operate, control, isolate, or terminate those systems and components modeled by the PRA, as well as those actions performed by the control room staff either in response to procedural direction or as skill-of-the-craft to recover a failed function, system or component." Section 2.2 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that " To delineate system response to particular types of upset events, it can be as important to understand the intended response of the operating crew in using the system as it is to understand the design of the system itself. Thus, in defining the sequence delineation for particular initiating events, it was necessary to review carefully the operating procedures, including the emergency procedures and the various abnormal procedures. This review was aimed at identifying any operator-driven considerations that would affect the modeling process, such as the priorities that might come into play when multiple options were available for maintaining core cooling, or the cues that might indicate the need to change operating modes. These procedure reviews were augmented by obtaining input from operators. This was done by having current and former operators review the sequence	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in the HRA Calc. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				logic and system fault trees; through extensive discussions with operators regarding specific scenarios; and, to the extent possible, by observing simulator exercises."	
HR-E3	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures.	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	The NEI SRs applicable to this ASME SR are HR-10, HR-14, and HR-20, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "2" with associated level "B" F&Os HR-04 and HR-05. F&Os HR-04 and HR-05: While these F&Os remain open (PRATracker items C-03-0059 and C-03-0060); CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030. DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "As documented in the HR notebook, talk-throughs with plant operations have been performed to confirm that interpretation of the procedures is consistent with plant observations and training procedures. This was done by having operators review the sequence logic and system fault trees, through extensive discussions with operators regarding specific scenarios." Section 4 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that: "The quantification of the human interactions required input from operations personnel, who often provided input on timing and qualitative insights that led to changes in the definition or application of specific events. The assessment for each event was reviewed in detail by at least one other PRA analyst. Review of the overall reasonableness of the events and their treatment was also gained during the final review of the sequence cut sets. This review process included both other members of the PRA project team and Catawba operations personnel."	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.
HR-E4	USE simulator observations or talk-throughs with operators to confirm	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not	The NEI SRs applicable to this ASME SR are HR-14 and HR-16, and there are no industry self-assessment actions and no NRC objections.	Based on the disposition, the requirements of Cat II are

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	the response models for scenarios modeled.		obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1.	The original Peer Review rated both of these NEI SRs as "2" with associated level "B" F&O HR-04. F&O HR-04: While this F&O remains open (PRATracker items C-03-0059), CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030. DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "As documented in the HR notebook, talk-throughs with plant operations have been performed to confirm the response models for scenarios modeled. This was done by having operators review the sequence logic and system fault trees, through extensive discussions with operators regarding specific scenarios, and, to the extent possible, by observing simulator exercises." Section 2.2 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that: "To delineate system response to particular types of upset events, it can be as important to understand the intended response of the operating crew in using the system as it is to understand the design of the system itself. Thus, in defining the sequence delineation for particular initiating events, it was necessary to review carefully the operating procedures, including the emergency procedures and the various abnormal procedures. This review was aimed at identifying any operator-driven considerations that would affect the modeling process, such as the priorities that might come into play when multiple options were available for maintaining core cooling, or the cues that might indicate the need to change operating modes. These procedure reviews were augmented by obtaining input from operators. This was done by having current and former operators review the sequence logic and system fault trees; through extensive discussions with operators regarding specific scenarios; and, to the extent possible, by observing simulator exercises."	considered met. There is no impact to the ILRT extension.
HR-F1	DEFINE human failure events (HFEs) that represent the impact of the human failures at the function, system, train, or component level as appropriate. Failures to correctly	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their	The NEI SRs applicable to this ASME SR are HR-16, AS-19, and SY-5, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-19 and SY-5 as "3", but HR-16 as "2" with associated level "B" F&O HR-04.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.

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	perform several responses may be grouped into one HFE if the impact of the failures is similar or can be conservatively bounded.		comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1.	F&O HR-04: While this F&O remains open (PRATracker items C-03-0059), CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment. DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "Based on a review of the PRA documentation, the PRA defines human failure events at the appropriate level: function, system, train, or component level." Section 2.2 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that: "Type CP interactions in the logic models were included at the highest level consistent with their effects. For example, the failure to initiate feed-and-bleed cooling following a total loss of feedwater is included in the supporting logic for the corresponding events in the event trees, rather than being broken down into individual faults associated with each piece of equipment in the system fault trees. This treatment helps to highlight the events, and focuses consideration on cognitive aspects of the response to upset conditions."	
HR-F2	COMPLETE THE DEFINITION of the HFEs by specifying (a) accident sequence specific timing of cues, and time window for successful completion (b) accident sequence specific procedural guidance (e.g., AOPs, and EOPs) (c) the availability of cues and other indications for detection and evaluation errors (d) the specific high level tasks (e.g., train level) required to achieve the goal of the response	Open	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the Catawba Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to	The NEI SRs applicable to this ASME SR are HR-11, HR-16, HR-17, HR-19, HR-20, AS-19, and SY-5, and there are no NRC objections. There is an industry action to determine whether the requirements of the ASME/ANS PRA Standard are met. The original Peer Review rated AS-19 and SY-5 as "3", but HR-16, HR-17, HR-19, and HR-20 as "2", with associated level "B" F&Os HR-04, HR-05, TH-05, and HR-04, respectively. HR-11 was assessed as "NA". F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. The date stamp on the HEP Excel Spreadsheets is still 2005 so it is not apparent that any updates have been made. No Thermal-	Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. F&O TH-05: The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided.	hydraulic analyses are referenced as the basis for the accident sequence specific timing for cues or overall time window, as required by the SR. F&O HR-05: While this F&O remains open (PRATracker item C-03-0060) for documentation issues, success criteria, plant parameters and associated acceptance criteria derived from the success criteria analyses are used to support the timing analysis used in the PRA HRA. References to MAAP analysis that support the timing actions are included in the HRA spreadsheets. F&O HR-04: While this F&O remains open (PRATracker item C-03-0059), CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the discussion from CNC-1535.00-00-0030. Section 2.2 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that: "To delineate system response to particular types of upset events, it can be as important to understand the intended response of the operating crew in using the system as it is to understand the design of the system itself. Thus, in defining the sequence delineation for particular initiating events, it was necessary to review carefully the operating procedures, including the emergency procedures and the various abnormal procedures. This review was aimed at identifying any operator-driven considerations that would affect the modeling process, such as the priorities that might come into play when multiple options were available for maintaining core cooling, or the cues that might indicate the need to change operating modes. These procedure reviews were augmented by obtaining input from operators. This was done by having current and former operators review the sequence logic and system fault trees; through extensive discussions with operators regarding specific scenarios; and, to the extent possible, by observing simulator exercises."	
HR-G1	PERFORM detailed analyses for the estimation of HEPs for significant HFEs. USE screening values for HEPs for non-significant human failure basic events.	Dispositioned	F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the Catawba Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental	The NEI SRs applicable to this ASME SR are HR-15, HR-17, and HR-18, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-15 as "3", but HR-17 as "2", with associated level "B" F&O HR-05. HR-18 was assessed as "N/A" (F&Os of HR-05, HR-09 and TH-05 were cited but they are not directly relevant to this SR).	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension. Detailed analysis has been performed for significant HFEs.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment Table 1 considers this SR to be met on the following basis: "The Type C HRA uses detailed analyses for the estimation of HEPs for significant HFEs. The human cognitive reliability model or the caused-based approach was used to quantify cognition errors, and an abbreviated version of THERP to quantify execution errors. Screening values have been used for HEPs for non-significant human failure basic events."	
HR-G3	When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of the required response	Dispositioned	F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the McGuire Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios. This finding was made against NEI SR HR-17 with an assignment of grade 2.	The NEI SRs applicable to this ASME SR are HR-17 and HR-18, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-17 as "2", with associated level "B" F&O HR-05. HR-18 was assessed as "N/A" (F&Os of HR-05, HR-09 and TH-05 were cited but they are not directly relevant to this SR). Reg Guide 1.200, Rev. 2, Table B-4 states that "NEI 00-02 does not explicitly enumerate the same level of detail that is included in the ASME standard. However, by invoking the standard HRA methodologies the performance shape factors are necessarily evaluated. The peer review team experience is relied upon to investigate the PRA given general guidance and criteria." CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc., Section 3.2 Quantification of Type Cp Interactions provides more detailed explanations for the HRA methods used and the PSFs that were considered using the HCR and Cause Based methods.	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in the HRA Calc. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	(h) environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc.				
HR-G4	BASE the time available to complete actions on appropriate realistic generic thermal-hydraulic analyses, or simulation from similar plants (e.g., plant of similar design and operation). SPECIFY the point in time at which operators are expected to receive relevant indications.	Open	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O TH-05: The HEP worksheets do not clearly refer to success criteria analyses to support timing for operator actions. Although most worksheets include an estimate of the time available for completion of an action, and some refer generally to information from MAAP analyses, specific references to MAAP (or other analysis) cases are not provided.	The NEI SRs applicable to this ASME SR are HR-18, HR-19, HR-20, and AS-13, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated AS-13 as "3", but HR-19 and HR-20 as "2", with associated level "B" F&Os TH-05 and HR-04, respectively. HR-18 was assessed as "N/A" (F&Os of HR-05, HR-09 and TH-05 were cited but they are not directly relevant to this SR). F&O TH-05 - Operator actions are considered as part of the CNP success criteria analyses with expected operator actions included for SLOCA (Section 3.3), SGTR (Section 3.4), and transient F&B (Section 3.6). Specific timing information from MAAP analyses can be found in Appendices A through F MAAP. This F&O is dispositioned based on the resolution of the finding and achieve grade 3 of the NEI SR. However, the CNS Assessment of Peer Review Open Items (May 2013) identifies this F&O as remaining open because the current model of record does not reflect the updated information and as a result the ASME SR is considered Not Met. The date stamp on the HEP Excel Spreadsheets is still 2005 so it is not apparent that any updates have been made. No Thermal-hydraulic analyses are referenced as the basis for the accident sequence specific timing for cues or overall time window, as required by the SR. F&O HR-04: While this F&O remains open (PRATracker items C-03-0059), CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030.	Peer Review F&O TH-05 is still open. While updated success criteria and timing data has been developed from MAAP 4.0.7 analyses, it has not been incorporated into the model of record. However, there are no significant changes to the success criteria [Reference 45], so the impact on the ILRT extension is expected to be negligible.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment considers this SR to be met. Section 4 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that" "The quantification of the human interactions required input from operations personnel, who often provided input on timing and qualitative insights that led to changes in the definition or application of specific events. The assessment for each event was reviewed in detail by at least one other PRA analyst. Review of the overall reasonableness of the events and their treatment was also gained during the final review of the sequence cut sets. This review process included both other members of the PRA project team and Catawba operations personnel." CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc., Section 3.2.1 The Human Cognitive Reliability Model states: "Ideally, the response and execution times would be collected from simulator exercises and actual plant events. In most cases, however, it was not practical to collect sufficient information, so the estimates of the SROs were used. The total time available was generally obtained from thermal-hydraulic calculations for the accidents of interest (e.g., from MAAP analyses, hand calculations, or other sources). Once the type of cognitive processing was determined and the time estimates were available, the correlation was quantified for failure to accomplish the action of interest within the available time window, TW, which represents the net time available to formulate the response to an event."	
HR-G5	When needed, BASE the required time to complete actions for significant HFEs on action time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.	Open	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1.	The NEI SRs applicable to this ASME SR are HR-16, HR-18, and HR-20, and there are no NRC objections. There is an industry action to evaluate proper inputs per the ASME/ANS PRA Standard or cite peer review documentation/conclusions or examples from your model. The original Peer Review rated HR-16 and HR-20 as "2", with associated level "B" F&O HR-04. HR-18 was assessed as "NA" (F&Os of HR-05, HR-09 and TH-05 were cited but they are not directly relevant to this SR). F&O HR-09: Addressed in Catawba Human Reliability Analysis CNC-1535.00-00-0030. F&O remains open (PRATracker C-03-0066) with action to define and document the four time parameters for all HEPs. Any changes to the HEPs are expected to be small. The internal events PRA human actions have been conservatively modified for	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			F&O HR-09: Define the four time parameters for all HEPs. Document the basis for all four times for each HEP. Make similar HEPs consistent with each other. Requantify HEP with new time data.	application in the FPRA. F&O HR-04: While this F&O remains open (PRATracker items C-03-0059), CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the self-assessment and the discussion from CNC-1535.00-00-0030. DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment considers this SR to be met. Section 4 of Rev. 2 of the HRA Calc CNC-1535.00-00-0030 states that" "The quantification of the human interactions required input from operations personnel, who often provided input on timing and qualitative insights that led to changes in the definition or application of specific events. The assessment for each event was reviewed in detail by at least one other PRA analyst. Review of the overall reasonableness of the events and their treatment was also gained during the final review of the sequence cut sets. This review process included both other members of the PRA project team and Catawba operations personnel." CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc., Section 3.2.1 The Human Cognitive Reliability Model states: "Ideally, the response and execution times would be collected from simulator exercises and actual plant events. In most cases, however, it was not practical to collect sufficient information, so the estimates of the SROs were used. The total time available was generally obtained from thermal-hydraulic calculations for the accidents of interest (e.g., from MAAP analyses, hand calculations, or other sources). Once the type of cognitive processing was determined and the time estimates were available, the correlation was quantified for failure to accomplish the action of interest within the available time window, TW, which represents the net time available to formulate the response to an event."	the HRA Calc. There is no impact on the ILRT extension.
HR-G6	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	Dispositioned	F&O QU-05: Event ND0RWSTDHE: This is a recovery action to terminate the NV and NI pumps in the event of failure of ND to provide recirculation after a SL. The event was quantified on the basis of tripping the pumps within 18 minutes. RWST refill was assumed to occur (from undescribed source) and pumps were restarted to continue injection. This recovery event is applied to	The NEI SRs applicable to this ASME SR is HR-12, and there are no NRC objections. There is an industry action to ensure they are met by citing peer review documentation/conclusions or examples from your model. The original Peer Review rated HR-12 as "3", with associated level "B" F&O QU-05. F&O QU-05: Event NDORWSTDHE has been redefined and failure	HFEs are reviewed by knowledgeable site personnel to assure high quality. Recent update of the Oconee PRA model demonstrated that the HRA methodology for operator actions used at the time of

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			a) loss of KC pumps b) SNSDRNVLHE - drain plug blockage c) CCF of ND pumps. The recovery event is intended to provide injection flow for the long term commensurate with the RWST make-up capability. The time of some of these failure is 20 minutes, when injection requirements are beyond the make-up capability of the RWST. Secondly, there are cutsets representing heat removal that cannot be recovered by continued injection of HHSI. The sequence needs continuous injection of HHSI and heat removal from containment.	probability recalculated for the Catawba Rev 3a PRA Model Integration Notebook. In DPC-1535.00-00-013, it is noted that the PRA notebooks do not document a review of the HFEs and their final HEPs relative to each other to check reasonableness given the scenario context, plant history, procedures, operational practices, and experience, and this SR is considered Not Met. Self-assessment, CNC-1535.00-00-0155, also lists this SR as Not Met. It is noted that, as part of model integration and results review, the probabilities associated with human error events and their reasonableness given the scenarios in which they occur are reviewed. To fully meet this SR, it is recommended that a meeting be held with the PRA model integrator, the HRA specialist and plant operators to perform a formal consistency check of the post-initiator human error probability quantifications.	the Catawba peer review produced conservative results, largely due to overestimation of the impact of dependencies. This issue is not expected to affect the overall conclusions of the ILRT extension LAR submittal. However, this review needs to be better documented. No impact on the ILRT extension is expected.
HR-G8	Characterize the uncertainty in the estimates of the HEPs in a manner consistent with the quantification approach, and PROVIDE mean values for use in the quantification of the PRA results.	Dispositioned	None	NEI 00-02 does not address this supporting requirement. DPC-1535.00-00-013, Rev. 2. 2008 Self-assessment identified this in Table 3 as Not Met and in Table C as an Open Item. CNC-1535.00-00-0155, Rev. 0, 2013 Self-assessment states that this is Met and cites the Catawba Rev. 3a PRA Database as a reference, which includes uncertainty parameters and mean values for use in quantification.	Uncertainties in the internal events HEPs are fed into the HRA. No impact to the ILRT extension.
HR-H1	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of significant accident sequences.	Dispositioned	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the Catawba Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the	The NEI SRs applicable to this ASME SR are HR-21, HR-22, and HR-23, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated HR-21 as "3" and HR-22 and HR-23 as "3 with contingencies." NEI SRs HR-22 and HR-23 have level "B" F&Os HR-04 and HR-05, respectively. F&Os HR-04 and HR-05: While these F&Os remain open (PRATracker items C-03-0059 and C-03-0060); CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the discussion from CNC-1535.00-00-0030. CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc, Section 3.3 Quantification for Non-Recovery (Type CR) Interactions says: "The consideration of actions that would constitute non-recovery events is outlined in Section 2.5. As noted, there, some of the non-recovery events assessed in this study represented failures to respond to the	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in the HRA Calc. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus, even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	loss of a system or function in a manner that was not explicitly directed by procedures. These events were added to the sequence-level minimal cut sets after the solution process. This process was originally accomplished by adding events to the cut sets on an individual basis. More recently, the addition of the events has been automated through the use of the QRECOVER program, which allows the analyst to define a set of rules which, if satisfied, cause the event to be added. The rules are formulated in terms of the combinations of events that must appear in a cut set (and, in some cases, the events that must not be present) for a particular recovery action to be valid."	
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis, the following occur: (a) a procedure is available and operator training has included the action as part of crew's training, or justification for the omission for one or both is provided (b) "cues" (e.g., alarms) that alert the operator to the recovery action provided procedure, training, or skill of the craft exist (c) attention is given to the relevant performance shaping factors provided in HR-G3 (d) there is sufficient manpower to perform the action.	Open	F&O HR-04: The operating staff at the plant had some input to the HRA in the beginning, but it is not obvious a thorough review of the dominant operator actions by the plant staff had been done, nor was it obvious there had been any feedback of their comments into the analysis. The level of detail and relation to the operating procedures is sparse. In some instances, the procedural steps are not mentioned. In some places, the reference to the procedure is incorrect, such as the emergency primary depressurization reference to ES 1.3, which actually occurs in FRC.1. F&O HR-05: In the Catawba HRA notebook for PRA Rev 2b (and similarly in the Catawba Rev 3 HRA notebook), the documentation of the bases for the HEPs is not sufficiently specified to assure that the analysis is reproducible. Specifically, the sequence context (e.g., previous failures in the event sequence, concurrent activities, environmental factors, etc.) and procedural steps applicable to each HEP are not consistently provided. Thus,	The NEI SRs applicable to this ASME SR are HR-22 and HR-23, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated both of these NEI SRs as "3 with contingencies". NEI SRs HR-22 and HR-23 have level "B" F&Os HR-04 and HR-05, respectively. F&Os HR-04 and HR-05: While these F&Os remain open (PRATracker items C-03-0059 and C-03-0060); CNC-1535.00-00-0030 contains the information needed to ascertain that the requirements for this SR are met, as noted below in the discussion from CNC-1535.00-00-0030. CNC-1535.00-00-0030, Rev. 2, July 2012, HRA Calc, Section 3.3 Quantification for Non-Recovery (Type CR) Interactions says: "The consideration of actions that would constitute non-recovery events is outlined in Section 2.5. As noted, there, some of the non-recovery events assessed in this study represented failures to respond to the loss of a system or function in a manner that was not explicitly directed by procedures. These events were added to the sequence-level minimal cut sets after the solution process. This process was originally accomplished by adding events to the cut sets on an individual basis. More recently, the addition of the events has been	Based on a review of the HR and SY notebooks, the identification of key human response actions employed reviews of the plant-specific operating procedures, including the emergency procedures and the various abnormal procedures, as well as human interfaces with systems operation. The issues raised by the peer review were addressed through added discussion in the HRA Calc. There is no impact on the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			even though there is evidence that the HEP worksheet information is being reviewed by plant Operations personnel, it is not clear that they would have sufficient supporting information with which to make an effective assessment of the HRA. Similarly, the timing, PSF, stress level, and all other contributing factors to the HEP were printed, but the basis was not provided. It would not have been possible for another analyst to determine the same factors and derive the same number. The lack of such information in the documentation of the HRA limits the ability to verify and reproduce the results, and to determine their applicability in specific scenarios.	automated through the use of the QRECOVER program, which allows the analyst to define a set of rules which, if satisfied, cause the event to be added. The rules are formulated in terms of the combinations of events that must appear in a cut set (and, in some cases, the events that must not be present) for a particular recovery action to be valid."	
DA-A1	IDENTIFY from the systems analysis the basic events for which probabilities are required. Examples of basic events include: (a) independent or common cause failure of a component or system to start or change state on demand (b) independent or common cause failure of a component or system to continue operating or provide a required function for a defined time period (c) equipment unavailable to perform its required function due to being out of service for maintenance (d) equipment unavailable to perform its required function due to being in test mode (e) failure to recover a function or system (e.g., failure to recover offsite-power)	Dispositioned	F&O DA-02: Some of the generic data from SAROS is quite dated, including WASH-1400, NUREG/CR-2815, Zion PRA, and NUREG/CR-4550. More recent generic data should be pursued. Component failures should be defined such that they encompass only those failures that would disable the component over the PRA mission time. It appears that this has not been considered. Specific examples of less than adequate reliability data characterization were identified through review of Tables 1 and 3 in SAAG-655, Catawba PRA Rev. 3 Failure Rate and Maintenance Unavailability Data. First, repeat events in a short duration, where there was insufficient component repair should be counted as one event. An example is PIP nos. 2-C97-2481 and 2-C97-2637 on 7/29/97 and 8/12/97 for incoming breaker 2CXI-5C. The first failure occurred "for no apparent reason", but the second failure was attributed to a failed relay. The first event should be omitted as a component failure as the component was left in the degraded condition. Second, component degradation that results in failure to meet normal criteria (e.g., to avoid component life degradation), may not impact the component mission for the PRA. For example, PIP no. 0-C98-2057 involved a 6/7/98 event for trouble	The CNS PRA model includes events of all of the types shown (other than component repair, which is not considered in the model). The 2009 ASME/ANS Cat II requirements for DA-A1 were evaluated in part under NEI technical elements DA-4, DA-5, DA-15, SY-8, and SY-15 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-5, SY-8, and SY-14. DA-4 and DA-15 were assigned a PSA grade of 3 contingent on resolution of F&Os DA-02, DA-05, and DA-06. F&O DA-02 is related to generic data sources; see SR DA-C1 for disposition. F&O DA-05 is related to specific component unavailabilities; see SR DA-C14 for disposition. F&O DA-06 concerns MOV rupture error factors; see SR DA-D3 for disposition.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	(f) failure to repair a component, system, or function in a defined time period		alarms for VI compressor F, and the compressor motor was found smoking. The evaluation addressed concern with overheating and insulation breakdown, but did not address whether run to failure would survive PRA mission. Similar pump failures due to routine vibration testing exceeding limits were found (LPR 2B & 1A, WO 93020502 & PIP 1-C93-1124). F&O DA-05: The unavailabilities computed for the basic events for PORV block valve closure, RNC031BDEX, 033ADEX, and 035BDEX, assume that each PORV is closed one week per quarter. However, there is no history of PORV closures for any extended period of time in the last few years. While this does use plant-specific data, the benefit derived from it is limited due to the highly conservative assumption regarding PORV out of service time. F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1...n$. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s.		

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-A2	ESTABLISH definitions of SSC boundaries, failure modes, and success criteria in a manner consistent with corresponding basic event definitions in Systems Analysis (SY-A5, SY-A7, SY-A8, SY-A9 through SY-A14 and SY-B4) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events in a manner consistent with corresponding definitions in Systems Analysis (SY-A19).	Dispositioned	None	NEI 00-02 did not address this supporting requirement. A review of the Catawba CAFTA Model of Record was completed to define existing failure modes (both in type-code and/or basic event file). The process was used to define a complete set of required data, which was used to define the failure modes. The boundaries are set by the data source and/or system modeling. The database development calculation (DPC-1535.00-00-0016) includes a listing of each of the specific component type/failure mode combinations that are considered, along with component boundaries definitions.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-A3	USE an appropriate probability model for each basic event. Examples include (a) binomial distributions for failure on demand (b) Poisson distributions for standby and operating failures and initiating events	Dispositioned	None	NEI 00-02 did not address this supporting requirement. A review of the Catawba CAFTA Model of Record was completed to define existing failure modes (both in type-code and/or basic event file). The process was used to define a complete set of required data, which includes failures per demand and time-dependent failures. Appropriate failure models are used for each event type.	Based on the disposition, the CNS PRA model meets the Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-A4	IDENTIFY the parameter to be estimated and the data required for	Dispositioned	F&O DA-02: Some of the generic data from SAROS is quite dated, including WASH-1400, NUREG/CR-	The appropriate parameters necessary for each type of basic event have been identified and the required data has been collected and	Update of the generic data addressed concerns of the

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	estimation. Examples are as follows: (a) For failures on demand, the parameter is the probability of failure, and the data required are the number of failures given a number of demands. (b) For standby failures, operating failures, and initiating events, the parameter is the failure rate, and the data required are the number of failures in the total (standby or operating) time. (c) For unavailability due to test or maintenance, the parameter is the unavailability on demand, and the alternatives for the data required include (1) the total time of unavailability OR a list of the maintenance events with their durations, together with the total time required to be available; OR (2) the number of maintenance or test acts, their average duration, and the total time required to be available.		2815, Zion PRA, and NUREG/CR-4550. More recent generic data should be pursued. Component failures should be defined such that they encompass only those failures that would disable the component over the PRA mission time. It appears that this has not been considered. Specific examples of less than adequate reliability data characterization were identified through review of Tables 1 and 3 in SAAG-655, Catawba PRA Rev. 3 Failure Rate and Maintenance Unavailability Data. First, repeat events in a short duration, where there was insufficient component repair should be counted as one event. An example is PIP nos. 2-C97-2481 and 2-C97-2637 on 7/29/97 and 8/12/97 for incoming breaker 2CXI-5C. The first failure occurred "for no apparent reason", but the second failure was attributed to a failed relay. The first event should be omitted as a component failure as the component was left in the degraded condition. Second, component degradation that results in failure to meet normal criteria (e.g., to avoid component life degradation), may not impact the component mission for the PRA. For example, PIP no. 0-C98-2057 involved a 6/7/98 event for trouble alarms for VI compressor F, and the compressor motor was found smoking. The evaluation addressed concern with overheating and insulation breakdown, but did not address whether run to failure would survive PRA mission. Similar pump failures due to routine vibration testing exceeding limits were found (LPR 2B & 1A, WO 93020502 & PIP 1-C93-1124). F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to	documented in calculation CNC-1535.00-00-0029 and its attached spreadsheets, and in the generic database, DPC-1535.00-00-0016. The 2009 ASME/ANS Cat II requirements for DA-A4 were evaluated in part under NEI technical elements DA-4, DA-5, DA-6, DA-7, and SY-8 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-5, DA-7 and SY-8. DA-6 was found to be not applicable to CNS. DA-4 was assigned a PSA grade of 3 contingent on resolution of F&Os DA-02, DA-04, and DA-06. F&O DA-02 is related to generic data sources; see SR DA-C1 for disposition. DA-04 is Level C and does not need to be addressed. F&O DA-06 concerns MOV rupture error factors; see SR DA-D3 for disposition.	peer review team. Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, i =1…n. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-B1	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage to the extent supported by data: (a) mission type (e.g., standby, operating) (b) service condition (e.g., clean vs. untreated water, air)	Dispositioned	F&O DA-01: Workplace Procedure XSAA-110 is the primary data gathering procedure. It is supplemented by SAAG-655, Catawba PRA Revision 3 Failure Rate And Maintenance Unavailability Data, and SAAG-670, the CCF analysis report. Also, noteworthy is attachment 3, which includes the CCF checklist. Additional details are provided by SAAG File 579 (Rev. 2b Summary Report) and the Rev 2 Summary Report. The data guidance is generally adequate; however it does not address component boundaries. Component boundaries are apparent from the data	The 2009 ASME/ANS Cat II requirements for DA-B1 were evaluated under NEI technical element DA-5 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-5, however, one Level B F&O was issued related to DA-5. F&O DA-01 was addressed in the referenced generic database development. Specifically, component boundaries are defined, time-dependent events for components such as motor-operated valves and check valves are developed, restrictions on the use of demand failures are provided, and data for standby vs. alternating and clean vs. water components are developed.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			as in the specific example in F&O DA-02, i.e., the incoming breaker and panelboard BLF. However, these should be defined in the guidance.		
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently)	Dispositioned	F&O DA-01: Workplace Procedure XSAA-110 is the primary data gathering procedure. It is supplemented by SAAG-655, Catawba PRA Revision 3 Failure Rate And Maintenance Unavailability Data, and SAAG-670, the CCF analysis report. Also, noteworthy is attachment 3, which includes the CCF checklist. Additional details are provided by SAAG File 579 (Rev. 2b Summary Report) and the Rev 2 Summary Report. The data guidance is generally adequate; however it does not address component boundaries. Component boundaries are apparent from the data as in the specific example in F&O DA-02, i.e., the incoming breaker and panelboard BLF. However, these should be defined in the guidance.	The 2009 ASME/ANS Cat II requirements for DA-B2 were evaluated under NEI technical elements DA-5 and DA-6 in the 2002 Catawba Peer Review. DA-6 was found to be not applicable to CNS. The peer review team assigned PSA grade of 3 to DA-5, however, one Level B F&O was issued related to DA-5. F&O DA-01 was addressed in the referenced generic database development as noted in DA-B1 disposition. No outlier components were inappropriately included in the established groupings. For unique failure modes (e.g. pressurizer safety valves and PORVs), unique failure probabilities are developed.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-C1	OBTAIN generic parameter estimates from recognized sources. ENSURE that the parameter 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],	Open	F&O DA-02: Some of the generic data from SAROS is quite dated, including WASH-1400, NUREG/CR-2815, Zion PRA, and NUREG/CR-4550. More recent generic data should be pursued. Component failures should be defined such that they encompass only those failures that would disable the component over the PRA mission time. It appears that this has not been considered. Specific examples of less than adequate reliability data characterization were identified through review of Tables 1 and 3 in SAAG-655, Catawba PRA Rev. 3 Failure Rate and Maintenance Unavailability Data. First, repeat events in a short duration, where there was insufficient component repair should be counted as one event. An example is PIP nos. 2-C97-2481 and 2-C97-2637 on 7/29/97 and 8/12/97 for incoming breaker 2CXI-5C. The first failure occurred "for no apparent reason", but the second failure was attributed to a failed relay. The first event should be omitted as a component failure as the component was left in the degraded condition. Second, component degradation that results in	NUREG/CR-6928 updated through 2010 is the primary data source. The 2009 ASME/ANS Cat II requirements for DA-C1 were evaluated under NEI technical elements DA-4, DA-7, DA-9, DA-19, and DA-20 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-7, DA-9, DA-19 and DA-20. DA-4 was assigned a PSA grade of 3 contingent on resolution of Level B F&Os DA-02 and DA-06. F&Os DA-02 was addressed by development and compilation of equipment failure rates for generic components as documented in DPC-1535.00-00-0016. The report, however, is limited to random independent failures for demand and time-dependent failures. F&O DA-02 remains open and is tracked as open item C-03-0057. F&O DA-06 concerns MOV rupture error factors; see SR DA-D3 for disposition.	Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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	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.		failure to meet normal criteria (e.g., to avoid component life degradation), may not impact the component mission for the PRA. For example, PIP no. 0-C98-2057 involved a 6/7/98 event for trouble alarms for VI compressor F, and the compressor motor was found smoking. The evaluation addressed concern with overheating and insulation breakdown, but did not address whether run to failure would survive PRA mission. Similar pump failures due to routine vibration testing exceeding limits were found (LPR 2B & 1A, WO 93020502 & PIP 1-C93-1124). F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1 \ldots n$. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is		

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			incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-C2	COLLECT plant-specific data for the basic event/parameter grouping corresponding to that defined by requirement DA-A1, DA-A3, DA-A4, DA-B1, and DA-B2.	Open	F&O DA-01: Workplace Procedure XSAA-110 is the primary data gathering procedure. It is supplemented by SAAG-655, Catawba PRA Revision 3 Failure Rate And Maintenance Unavailability Data, and SAAG-670, the CCF analysis report. Also, noteworthy is attachment 3, which includes the CCF checklist. Additional details are provided by SAAG File 579 (Rev. 2b Summary Report) and the Rev 2 Summary Report. The data guidance is generally adequate; however it does not address component boundaries. Component boundaries are apparent from the data as in the specific example in F&O DA-02, i.e., the incoming breaker and panelboard BLF. However, these should be defined in the guidance. F&O DA-02: Some of the generic data from SAROS is quite dated, including WASH-1400, NUREG/CR-2815, Zion PRA, and NUREG/CR-4550. More recent generic data should be pursued. Component failures should be defined such that they encompass only those failures that would disable the component over the PRA mission time. It appears that this has not been considered. Specific examples of less than adequate reliability data characterization were identified through review of Tables 1 and 3 in SAAG-655, Catawba PRA Rev. 3 Failure Rate and Maintenance Unavailability Data. First, repeat events in a short duration, where there was insufficient component repair should be counted as one event. An example is PIP nos. 2-C97-2481 and 2-C97-2637 on 7/29/97 and 8/12/97	The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). The events, failure modes, and parameters for which data are collected appear to be consistent with those used in the system models, and are collected for groups of components. The 2009 ASME/ANS Cat II requirements for DA-C2 were evaluated under NEI technical elements DA-4, DA-5, DA-6, DA-7, DA-14, DA-15, DA-19, and DA-20 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-5, DA-7, DA-9, DA-19 and DA-20. DA-4 was assigned a PSA grade of 3 contingent on resolution of Level B F&Os DA-02 and DA-06. DA-6 and DA-14 were found to be not applicable to CNS. F&O DA-01: F&O was issued related to component boundaries (see SR DA-B1 for disposition). F&O DA-02 is related to generic data sources; see SR DA-C1 for disposition. F&O DA-02 remains open and is tracked as open item C-03-0057. See SR DA-D3 for DA-06 disposition. F&O DA-05 is related to specific component unavailabilities; see SR DA-C14 for disposition. F&O DA-06 concerns MOV rupture error factors; see SR DA-D3 for disposition.	Minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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			for incoming breaker 2CXI-5C. The first failure occurred "for no apparent reason", but the second failure was attributed to a failed relay. The first event should be omitted as a component failure as the component was left in the degraded condition. Second, component degradation that results in failure to meet normal criteria (e.g., to avoid component life degradation), may not impact the component mission for the PRA. For example, PIP no. 0-C98-2057 involved a 6/7/98 event for trouble alarms for VI compressor F, and the compressor motor was found smoking. The evaluation addressed concern with overheating and insulation breakdown, but did not address whether run to failure would survive PRA mission. Similar pump failures due to routine vibration testing exceeding limits were found (LPR 2B & 1A, WO 93020502 & PIP 1-C93-1124). F&O DA-05: The unavailabilities computed for the basic events for PORV block valve closure, RNC031BDEX, 033ADEX, and 035BDEX, assume that each PORV is closed one week per quarter. However, there is no history of PORV closures for any extended period of time in the last few years. While this does use plant-specific data, the benefit derived from it is limited due to the highly conservative assumption regarding PORV out of service time. F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue.		

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, i =1…n. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for l is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-C3	COLLECT plant-specific data, in a manner consistent with uniformity in design, operational practices, and experience. JUSTIFY the rationale for screening or disregarding plant-specific data (e.g., plant design modifications, changes in operating practices).	Dispositioned	None.	The scope of NEI 00-02 only partially addresses this supporting requirement. For NEI technical elements that are partially related to this SR, the F&Os from the 2002 Catawba peer review are more closely associated other SRs and are addressed as part of SR DA-B1 and DA-C1. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). The events, failure modes, and parameters for which data are collected appear to be consistent with those used in the system models, and are collected for groups of components. The data is collected for groups of components. The Maintenance Rule Experience data tables identify those failures which apply to PRA components and failure modes, and those which are not PRA components (and therefore excluded).	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. In addition, any minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
DA-C4	When evaluating maintenance or other relevant records to extract plant-specific component failure event data, DEVELOP a clear basis for the identification of events as failures. DISTINGUISH between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would not have occurred (e.g., slow pick-up to rated speed). INCLUDE all failures that would have resulted in a failure to perform the mission as defined in the PRA.	Dispositioned	None	The scope of NEI 00-02 did not address this supporting requirement. The Workplace Procedure for Developing PRA Data (XSAA-110) provides specific guidelines for counting failures and demands for PRA purposes. In particular, a failure is counted only if the component would have failed to perform its function as defined in the PRA, under conditions applicable to the PRA. Numerous examples are provided. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). The Maintenance Rule Experience data tables identify those failures which apply to PRA components and those which are not PRA components, as well as the specific applicable failure mode.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. In addition, any minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-C5	COUNT repeated plant-specific component failures occurring within a short time interval as a single failure if there is a single, repetitive problem that causes the failures. In addition, COUNT only one demand.	Dispositioned	None	The scope of NEI 00-02 did not address this supporting requirement. The Workplace Procedure for Developing PRA Data (XSAA-110) specifies that repeated component failures occurring within a short period of time be counted as a single failure if there is a single, repetitive problem that causes the failures. In addition, only one demand is to be counted. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866).	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. In addition, any minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-C8	When required, USE plant-specific operational records to determine the time that components were configured in their standby status.	Open	None	The scope of NEI 00-02 did not address this supporting requirement. The denominators for calculation of plant-specific equipment failure data are determined in SAAG 492 by estimating the number of demands, run hours, or exposure hours for each component in the PRA. Each PRA system analyst reviewed each basic event in their system to determine the average annual number of demands, or the average number of operating hours or exposure hours for each component. However, other than some very brief analyst comments, there is no documented basis for the estimates provided and no determination of the time components are configured in standby. The documentation should be revised to clearly indicate how the time components are configured in their standby status is determined.	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the random failure rates of components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
DA-C9	ESTIMATE operational time from surveillance test practices for standby components, and from actual operational data.	Open	F&O DA-02: Some of the generic data from SAROS is quite dated, including WASH-1400, NUREG/CR-2815, Zion PRA, and NUREG/CR-4550. More recent generic data should be pursued. Component failures should be defined such that they encompass only those failures that would disable the component over the PRA mission time. It appears that this has not been considered. Specific examples of less than adequate reliability data characterization were identified through review of Tables 1 and 3 in SAAG-655, Catawba PRA Rev. 3 Failure Rate and Maintenance Unavailability Data. First, repeat events in a short duration, where there was insufficient component repair should be counted as one event. An example is PIP nos. 2-C97-2481 and 2-C97-2637 on 7/29/97 and 8/12/97 for incoming breaker 2CXI-5C. The first failure occurred "for no apparent reason", but the second failure was attributed to a failed relay. The first event should be omitted as a component failure as the component was left in the degraded condition. Second, component degradation that results in failure to meet normal criteria (e.g., to avoid component life degradation), may not impact the component mission for the PRA. For example, PIP no. 0-C98-2057 involved a 6/7/98 event for trouble alarms for VI compressor F, and the compressor motor was found smoking. The evaluation addressed concern with overheating and insulation breakdown, but did not address whether run to failure would survive PRA mission. Similar pump failures due to routine vibration testing exceeding limits were found (LPR 2B & 1A, WO 93020502 & PIP 1-C93-1124). F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is	The Workplace Procedure for Developing PRA Data (XSAA-110) specifies that equipment demands are counted based on actual operating experience, surveillance tests, preventive maintenance tests and unplanned demands. The denominators for calculation of plant-specific equipment failure data are determined in SAAG 492 by estimating the number of demands, run hours, or exposure hours for each component in the PRA. Each PRA system analyst reviewed each basic event in their system to determine the average annual number of demands, or the average number of operating hours or exposure hours for each component. Other than some very brief analyst comments, there is no documented basis for the estimates provided and no relationship shown between the surveillance test practices and operational data and the values in the denominator notebook. The documentation should be revised to clearly indicate the relationship between the surveillance test practices and operational data and the values in the denominator notebook. The 2009 ASME/ANS Cat II requirements for DA-C9 were evaluated under NEI technical elements DA-4, DA-6, and DA-7 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-7. DA-4 was assigned a PSA grade of 3 contingent on resolution of Level B F&Os DA-02 and DA-06. Element DA-6 was found to be not applicable to CNS. F&O DA-02 is related to generic data sources; see SR DA-C1 for disposition. F&O DA-02 remains open and is tracked as open item C-03-0057. F&O DA-06 concerns MOV rupture error factors; see SR DA-D3 for disposition.	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the random failure rates of components is not significant in the risk evaluations. Fire risk is dominated by fire impacts. There is negligible impact to the ILRT extension.

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			composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1 \ldots n$. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into sub-elements (or causes) that are fully	Dispositioned	None	The scope of NEI 00-02 did not address this supporting requirement. The Workplace Procedure for Developing PRA Data (XSAA-110) specifies that equipment demands are counted based on actual operating experience, surveillance tests, preventive maintenance tests and unplanned demands. The denominators for calculation of plant-specific equipment failure data are determined in SAAG 492 by estimating the number of demands, run hours, or exposure hours for each component in the PRA.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. In addition, any minor changes to the unavailability of the components is not significant in the risk evaluations There is negligible impact to the ILRT extension.

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	tested, then USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element sometimes has many more successes than another. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid tests would be significantly decreased.]				
DA-C11	When using data on maintenance and testing durations to estimate unavailabilities at the component, train, or system level, as required by the system model, only INCLUDE those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded.	Open	None	The scope of NEI 00-02 did not address this supporting requirement. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). Plant-specific unavailabilities are presented for about 25 component/trains. The unavailability data is based on that collected for performance reporting for INPO. Unavailabilities are listed in the system notebooks, however the basis for these unavailability values is not provided (only a list or summary description of applicable maintenance practices or procedures is provided). The documentation should be revised to provide a clearer basis for the unavailability values.	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the unavailability of components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-C12	When an unavailability of a front line system component is caused by an unavailability of a support system, COUNT the unavailability towards that of the support system and not the front line system, in order to avoid double counting and to capture the support system dependency properly.	Open	None	The scope of NEI 00-02 did not address this supporting requirement. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). Plant-specific unavailabilities are presented for about 25 component/trains. The unavailability data is based on that collected for performance reporting for INPO. Unavailabilities are listed in the system notebooks, the basis for these unavailability values is not provided (only a list or summary description of applicable maintenance practices or procedures is provided). The documentation should be revised to provide a clearer basis for the unavailability values.	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the unavailability of components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-C13	EVALUATE the duration of the actual time that the equipment was unavailable for each contributing activity. Since maintenance outages are a function of the plant status, INCLUDE only outages occurring during plant at power. Special attention should be paid to	Open	None	The scope of NEI 00-02 did not address this supporting requirement. The plant-specific equipment failure data collected is captured in Maintenance Rule Experience Documents thru 2005 (SAAG 866). Plant-specific unavailabilities are presented for about 25 component/trains. The unavailability data is based on that collected for performance reporting for INPO. There is no documentation of the duration due to each contributing activity or of the treatment for shared components. In addition, the unavailabilities for the remaining	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the unavailability of components is not significant in the risk evaluations. There is

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	the case of a multi-plant site with shared systems, when the Specifications (TS) requirements can be different depending on the status of both plants. Accurate modeling generally leads to a particular allocation of outage data among basic events to take this mode dependence into account. In the case that reliable estimates or the start and finish times are not available, INTERVIEW the knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events.			systems are either based on screening values or left to be calculated by the system analyst. The documentation should be revised to provide a clearer basis for the unavailability values.	negligible impact to the ILRT extension.
DA-C14	EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) that is a result of a planned, repetitive activity based on actual plant experience. CALCULATE coincident maintenance unavailabilities that are a result of a planned, repetitive activity that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares" (i.e., plant systems that have more redundancy than is addressed by tech specs). For example (intrasystem case), the charging system in some plants has a third train that may be out of service for extended periods of time coincident	Dispositioned	F&O DA-05: The unavailabilities computed for the basic events for PORV block valve closure, RNC031BDEX, 033ADEX, and 035BDEX, assume that each PORV is closed one week per quarter. However, there is no history of PORV closures for any extended period of time in the last few years. While this does use plant-specific data, the benefit derived from it is limited due to the highly conservative assumption regarding PORV out of service time.	The scope of NEI 00-02 did not address this supporting requirement. However, level "B" F&O DA-05 is considered to be most closely related to SR DA-C14. Maintenance restrictions imposed by the Tech. Specs. or the Maintenance Rule a(4) program are addressed by the model solution process as follows. The maintenance basic events are generally treated as independent within the PRA model. After the model is solved, cut sets involving coincident maintenance are deleted where such combinations are prohibited by the technical specifications, as documented in the model integration notebook. Cut sets involving coincident maintenance combinations allowed by the technical specifications but prohibited by the online risk assessment tool are retained, but have their probability reduced. Maintenance tasks that require a component to be out of service are performed under the same work window. For example, a pump lubrication PM could be bundled with the PM for its supply breaker. However, this type of maintenance coordination does not involve more than one train of equipment, and does not result in the plant taking on more risk.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. In addition, any minor changes to the unavailability of components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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	with one of the other trains and yet is in compliance with tech specs. Examples of intersystem unavailability include plants that routinely take out multiple components on a "train schedule" (such as AFW train A and HPI train A at a PWR, or RHR train A and LPCS train A at a BWR).			From SAAG 655, F&O DA-05 was addressed by revising unavailabilities of PORV block valves to more realistic values.	
DA-C16	Data on recovery from loss of offsite power, loss of service water, etc. are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service.	Dispositioned	F&O IE-04: The initiating event frequency for a stuck open PORV or safety valve is taken from NUREG/CR-5750 but is conservative for the following reasons. The NUREG assigned a value to these events based on a non-informative prior updated with 0 events and the total number of critical reactor years in the study. In the case of a spurious opening of a primary safety valve, the model should address the potential for the valve to close as the pressure decreased, effectively terminating the loss of coolant. The evaluation of the subsequent reclosure of the PORV is not as straightforward. The cause of the opening PORV would need to be addressed. However, either the PORV could be closed or the block valve could be closed. F&O AS-01: SAAG 427 describes the ATWS event tree analysis. Section 4, event B, describes how main feedwater is recovered after an ATWS. The probabilities used for main feedwater recovery are .05, following a T2 (Loss of Load) and .2 following a T4 (Loss of MFW). In the non-ATWS analysis, the following non-recoveries (From SAAG 427) are: T1 non-rec = .05, T4 - non-rec = .1 . Considering that the critical time for FW to come on line in an ATWS event involving a loss of main feedwater is very short, even for conditions of favorable MTC, the use of non-recovery probabilities of this magnitude does not appear to be justified without supporting analyses.	Catawba uses the EPRI report, Losses of Off-Site Power at U. S. Nuclear Power Plants Data, which includes the recovery time associated with each event. No plant specific recovery data is collected. The 2009 ASME/ANS Cat II requirements for DA-C16 were evaluated under NEI technical elements IE-13, IE-15, IE-16, AS-16, DA-15, SY-24, and QU-18 in the 2002 Catawba Peer Review. Level B F&Os associated with these elements are F&O IE-04, AS-01, DA-05, and QU-05. F&O IE-04 appears to be an observation of conservatism in usage of generic industry data for stuck open SRV and PORV initiating events and is judged to be applicable to IE-C12. However, this treatment is judged to be appropriate and that this F&O does not apply to DA-C16. F&O AS-01: Credit for Main Feedwater has been removed from the ATWS model, which resolves this F&O. Recovery for MFW in ATWS events initiated by a loss of feedwater has no impact on Fire PRA. F&O DA-05 is related to specific component unavailabilities; see SR DA-C14 for disposition. F&O QU-05: Event NDORWSTDHE has been redefined and failure probability recalculated for the Catawba Rev 3a PRA Model Integration Notebook.	There were no F&Os with "A" level of significance at CNS and there are no open Level B F&Os related to this SR. The CNS PRA model meets the requirements of Cat II for this SR. There is no impact to the ILRT extension.

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			F&O DA-05: The unavailabilities computed for the basic events for PORV block valve closure, RNC031BDEX, 033ADEX, and 035BDEX, assume that each PORV is closed one week per quarter. However, there is no history of PORV closures for any extended period of time in the last few years. While this does use plant-specific data, the benefit derived from it is limited due to the highly conservative assumption regarding PORV out of service time. F&O QU-05: Event ND0RWSTDHE: This is a recovery action to terminate the NV and NI pumps in the event of failure of ND to provide recirculation after a SL. The event was quantified on the basis of tripping the pumps within 18 minutes. RWST refill was assumed to occur (from undescribed source) and pumps were restarted to continue injection. This recovery event is applied to a) loss of KC pumps b) SNSDRNVLHE - drain plug blockage c) CCF of ND pumps. The recovery event is intended to provide injection flow for the long term commensurate with the RWST make-up capability. The time of some of these failure is 20 minutes, when injection requirements are beyond the make-up capability of the RWST. Secondly, there are cutsets representing heat removal that cannot be recovered by continued injection of HHSI. The sequence needs continuous injection of HHSI and heat removal from containment.		
DA-D1	CALCULATE realistic parameter estimates for significant basic events based on relevant generic and plant-specific evidence unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. When it is necessary to combine evidence from generic	Dispositioned	F&O DA-08: Another example of conservatism is the SBO following trip event, PACBOFTDEX. This event in the top 100 cutsets has a 1E-3 probability, and has not been updated since the IPE.	NEI 00-02 does not address this supporting requirement. However, level "B" F&O DA-08 is considered to be most closely related to SR DA-D1. Calculation CNC-1535.00-00-0029 documents a Bayesian update of generic component failure data with plant-specific experience. Where plant-specific data is not available, the generic data is used. Generic data has been updated as documented in DPC-1535.00-00-0016. Actual component unavailability data is derived from Maintenance Rule unavailability data.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Any minor changes to the random failure rate of the components is not significant in the risk evaluations. There is

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	and plant-specific data, USE a Bayes update process or equivalent statistical process that assigns appropriate weight to the statistical significance of the generic and plant-specific evidence and provides an appropriate characterization of uncertainty. CHOOSE prior distributions as either noninformative, or representative of variability in industry data. CALCULATE parameter estimates for the remaining events by using generic industry data.			F&O DA-08: The ac power notebook documents the development of the current value, which is within an acceptable range, e.g. another Westinghouse plant uses 2.4E-3 for LOOP following general transient. This F&O is considered resolved.	negligible impact to the ILRT extension.
DA-D2	If neither plant-specific data nor generic parameter estimates are available for the parameter associated with a specific basic event, USE data or estimates for the most similar equipment available, adjusting if necessary to account for differences. Alternatively, USE expert judgment and document the rationale behind the choice of parameter values.	Dispositioned	None	NEI 00-02 does not address this supporting requirement. Use of multiple data sources provides a means to define sources for all generic failure data. If exception is taken, the departure is defined and the basis provided. The SSF diesel generator data is an example of a departure. No specific instances were identified in which neither generic or plant-specific data is not available.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. Any minor changes to the random failure rate of the components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.
DA-D3	PROVIDE a mean value of, and a statistical representation of the uncertainty intervals for, the parameter estimates of significant basic events. Acceptable systematic methods include Bayesian updating, frequentist method, or expert judgment.	Open	F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1...n$. Such an operation often does not	Uncertainty distribution data has been calculated for all of the Bayesian-updated failure data. However, the data and CCF calcs (CNC-1535.00-00-0029 and CNC-1535.00-00-0028) do not document the error factors to be used for maintenance unavailability and CCF events. A review of the CAFTA database indicates that maintenance unavailability events have been assigned an error factor of 3 and CCFs have been assigned an error factor of 10. However, no basis for the use of these error factors is provided. Documentation should be revised to provide the basis for error factors. NEI 00-02 does not address this supporting requirement under the DA technical element; it is only partially addressed under QU-30, which was assigned a PSA grade of 3 by the 2002 CNS peer review team. However, F&O DA-06, issued at the 2002 Catawba peer review is most closely aligned with this SR.	This is a documentation issue that does not impact the PRA model. In addition, any minor changes to the uncertainty distributions of component failure rates is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for l is based on taking samples from the weighted sum of samples from each of the fi(λ)'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.	F&O DA-06: As noted in revision 1 to CNC-1535.00-00-0029, type code MVR error factor value was revised to 6.5, and Bayesian Mean was revised from 4.28E-08 to 4.08E-08, based on MVR generic ER = 6.	
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	Open	None	NEI 00-02 does not address this supporting requirement. There is no evidence in the documentation that the specific checks required by this SR have been performed on the Bayesian-updated data to ensure that the data is appropriate. However, a verification of the proper operation of the software within the expected data range (item d of the SR) was performed. A quick review of the current data did not reveal any unusual or unexpected results. Documentation should be revised to provide the basis for error factors.	This is a documentation issue that does not impact the PRA model. Any minor changes to the random failure rates of components is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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	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				
DA-D5	USE one of the following models for estimating CCF parameters for significant CCF basic events: (a) Alpha Factor Model (b) Basic Parameter Model (c) Multiple Greek Letter Model (d) Binomial Failure Rate Model JUSTIFY the use of alternative methods (i.e., provide evidence of peer review or verification of the method that demonstrates its acceptability).	Open	None	The 2009 ASME/ANS Cat II requirements for DA-D5 were partially evaluated under NEI technical elements DA-8 thru DA-14 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-8 thru DA-12. DA-13 and DA-14 were found to be not applicable. F&O DA-09 is related to element DA-12, which is Level "C" and is not addressed. The CNS PRA uses a "modified" MGL method as documented in CNC-1535.00-00-0028 (SAAG 670). In lieu of incorporating separate events for various combinations of 2 failures, 3 failures, etc., a set of combined CCF probability events are developed that include the relevant combinations of CCF failures that could impact system function. The approach appears reasonable. Generic estimates for error factors are used for the common cause events. However, the documentation for the selection of specific error factors used is not included in the CCF analysis.	The CNS PRA uses a "modified" MGL method. This is a documentation issue that does not impact the PRA model. There is negligible impact to the ILRT extension.
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	Open	None	The 2009 ASME/ANS Cat II requirements for DA-D6 were partially evaluated under NEI technical elements DA-8 thru DA-14 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-8 thru DA-12. DA-13 and DA-14 were found to be not applicable. F&O DA-09 is related to element DA-12, which is Level "C" and is not addressed Plant-specific CCF failure documentation (CNC-1535.00-00-0028)	Any minor changes to the CCF failure rates is not significant in the risk evaluations. There is negligible impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				was reviewed to ensure that the generic CCF estimates were consistent with plant operating experience. No evidence is provided to show that the component boundaries used in the CCF generic estimates are consistent with the component boundaries assumed for the PRA.	
DA-D8	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: (a) If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available for significant basic events; or (b) If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and assess the hypothetical effect on the historical data to determine to what extent the data can be used.	Dispositioned	None	NEI 00-02 does not address this supporting requirement. The referenced data analyses consider the applicability of the data. As noted in DPC-1535.00-00-0016, in most instances, a generic industry value and Catawba-specific experience are combined using a Bayesian update. In addition, Catawba has a living PRA database program (PRA Tracker) to provide the means for formal documentation, tracking and resolution of any potential changes to the PRA based on plant modifications, discovered errors or industry information. When an issue is identified that calls into question some aspect of the PRA model or related analysis, or if during the review of a site design change package some issue is identified, the issue is entered into the PRA Tracker program, and tracked to closure.	Based on disposition, the CNS PRA model meets the requirements of Cat II for this SR. There is no impact to the ILRT extension.
DA-E1	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Open	F&O DA-01: Workplace Procedure XSAA-110 is the primary data gathering procedure. It is supplemented by SAAG-655, Catawba PRA Revision 3 Failure Rate And Maintenance Unavailability Data, and SAAG-670, the CCF analysis report. Also, noteworthy is attachment 3, which includes the CCF checklist. Additional details are provided by SAAG File 579 (Rev. 2b Summary Report) and the Rev 2 Summary Report. The data guidance is generally adequate; however	The data analysis is appropriately documented in a manner that facilitates PRA applications, upgrades, and peer review, except as noted by the assessment comments and recommendations of other DA supporting requirements. The 2009 ASME/ANS Cat II requirements for DA-E1 were evaluated under NEI technical elements DA-1, DA-19 and DA-20 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-19 and 20. DA-1 was assigned a PSA grade of 3 contingent on resolution of F&Os DA-01 and DA-06.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this documentation SR; however, there is a documentation issue that does not impact the PRA model. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			it does not address component boundaries. Component boundaries are apparent from the data as in the specific example in F&O DA-02, i.e., the incoming breaker and panelboard BLF. However, these should be defined in the guidance. F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1...n$. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above	F&O DA-01 was addressed in the referenced generic database development. Specifically, component boundaries are defined, time-dependent events for components such as motor-operated valves and check valves are developed, restrictions on the use of demand failures are provided, and data for standby vs. alternating and clean vs. water components are developed. F&O DA-06: As noted in revision 1 to CNC-1535.00-00-0029, type code MVR error factor value was revised to 6.5, and Bayesian Mean was revised from 4.28E-08 to 4.08E-08, based on MVR generic ER = 6.	

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			equation and notes that it may produce a non-unimodal distribution.		
DA-E2	DOCUMENT the processes used for data parameter definition, grouping, and collection including parameter selection and estimation, including the inputs, methods, and results. For example, this documentation typically includes (a) system and component boundaries used to establish component failure probabilities (b) the model used to evaluate each basic event probability (c) sources for generic parameter estimates (d) the plant-specific sources of data (e) the time periods for which plant-specific data were gathered (f) justification for exclusion of any data (g) the basis for the estimates of common cause failure probabilities, including justification for screening or mapping of generic and plant-specific data (h) the rationale for any distributions used as priors for Bayesian updates, where applicable (i) parameter estimate including the	Open	F&O DA-01: Workplace Procedure XSAA-110 is the primary data gathering procedure. It is supplemented by SAAG-655, Catawba PRA Revision 3 Failure Rate And Maintenance Unavailability Data, and SAAG-670, the CCF analysis report. Also, noteworthy is attachment 3, which includes the CCF checklist. Additional details are provided by SAAG File 579 (Rev. 2b Summary Report) and the Rev 2 Summary Report. The data guidance is generally adequate; however it does not address component boundaries. Component boundaries are apparent from the data as in the specific example in F&O DA-02, i.e., the incoming breaker and panelboard BLF. However, these should be defined in the guidance. F&O DA-06: In SAAG 342, there is development of a failure probability for the rupture of an MOV. The type code for this event is MVR. This type code is used in the calculation of the ISLOCA frequency. In the SAROS database, this distribution is composed of three equally weighted distributions. The three distributions have error factors of close to 10.0. The error factor assigned to MVR is ~2.6. This is impossible – the error factor should be close to ten. The following provides additional explanation of this issue. Often it is useful to develop a distribution based on combining several distributions. That is $f(\lambda) = \Sigma i\ wi\ fi(\lambda)$, $i = 1...n$. Such an operation often does not possess a closed solution and Monte Carlo (MC) simulations are required. However, care must be taken in implementing the MC solution. People are often tempted to set up a MC process where one iteration for I is based on taking samples from the weighted sum of samples from each of the $fi(\lambda)$'s. This is incorrect. This, in effect, loses data and results in a unimodal function. In the case of two	The existing data documentation provided in CNC-1535.00-00-0028 and CNC-1535.00-00-0029 address most, but not all, of the specific items noted in this SR. More documentation needs to be added to discuss the exclusion of plant-specific data (e.g., pre-Maintenance Rule data), and the development of uncertainty estimates for Maintenance unavailability and CCF events. The 2009 ASME/ANS Cat II requirements for DA-E1 were evaluated under NEI technical elements DA-1, DA-19 and DA-20 in the 2002 Catawba Peer Review. The peer review team assigned PSA grade of 3 to DA-19 and 20. DA-1 was assigned a PSA grade of 3 contingent on resolution of F&Os DA-01 and DA-06. F&O DA-01 was addressed in the referenced generic database development. Specifically, component boundaries are defined, time-dependent events for components such as motor-operated valves and check valves are developed, restrictions on the use of demand failures are provided, and data for standby vs. alternating and clean vs. water components are developed. F&O DA-06: As noted in revision 1 to CNC-1535.00-00-0029, type code MVR error factor value was revised to 6.5, and Bayesian Mean was revised from 4.28E-08 to 4.08E-08, based on MVR generic ER = 6.	Based on the disposition, the CNS PRA model meets the requirements of Cat II for this SR. There is no impact to the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	characterization of uncertainty, as appropriate.		equally weighted functions A and B where every point on A is less than any point on B, the lower points of A and the higher points of B would not be in the resulting distribution. While the mean is preserved, the variance is understated and is incorrect. The proper method is to obtain samples for A, weight them, and put them in a pool. Then obtain samples for B, weight them, and put them in the pool. The points in the pool are MC distribution and, in this case, would be bi-modal. Note that page 5-38 of NUREG/CR-2300 uses the above equation and notes that it may produce a non-unimodal distribution.		
DA-E3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the data analysis.	Open	None	NEI 00-02 does not address this supporting requirement under the DA technical element; it is only partially addressed under QU technical element. The methodology, search and rationale are included in the documentation in order to support the prior supporting requirements. The data selection meets the intent by not deviating from the accepted consensus values for failure rates which is consistent with guidance document. However, The data analysis calculations, do not explicitly include an "Assumptions" section.	The disposition identifies a documentation issue that does not impact the PRA model. There is no impact to the ILRT extension.
QU-A2	PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF to identify significant accident sequences/cutsets and confirm the logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.	Open	F&O QU-12: The Conditional core damage Probability of several Initiators from the CR2b results were evaluated. The results are: 8.30E-03 Loss Of RN 8.38E-03 Loss Of KC 5.04E-03 Small LOCA 2.30E-04 Secondary Line Break Inside Containment 5.47E-05 LOOP 1.24E-05 Inadvertent SS Actuation 1.24E-05 Loss Of Instrument Air 1.04E-05 Steamline Break Outside Containment 9.54E-06 FDW Line Break Outside Containment 2.26E-06 Loss Of Main Feedwater 2.89E-06 SGTR 7.75E-07 Loss Of Load 5.01E-07 Reactor Trip These results show a discrepancy between Small LOCA and SGTR that is not consistent with what is normally seen in PRAs in the industry. The CCDP for small LOCA and SGTR are usually in the same	The NEI SR applicable to this ASME SR is QU-8, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated this NEI SR as "3 with contingencies", with associated level "B" F&O QU-12. F&O QU-12: The Catawba PRA has updated the small LOCA (SL) initiator to be redefined to only include small pipe breaks. The SL and SGTR initiating event frequencies are found in the CNS U1&2 internal initiator events frequency data notebook. This is considered to resolve the finding. The Catawba PRA model consists of a top logic fault tree that links the fault tree models for the frontline and support systems, and is solved to produce an overall CDF and LERF. Results for individual accident sequences are not calculated, although individual cutsets are provided in order to review the overall model logic. This ASME SR is considered still open.	There is no impact to the ILRT extension. The Catawba PRA model consists of a top logic fault tree that links the fault tree models for the frontline and support systems, and is solved to produce an overall CDF and LERF. Results for individual accident sequences are not calculated, although individual cutsets are provided in order to review the overall model logic. This ASME SR is considered still open. However, this has no impact on the results of the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			order of magnitude because the initiators have similar mitigation functions such as safety injection, secondary side heat removal, primary cooldown and depressurization, and long term injection if cooldown and depressurization are not successful. A difference of 3 orders of magnitude is unusual. Also, the CCDP value for the Loss of Instrument Air probability is identical to the Inadvertent SS Actuation probability (to 3 significant figures), which seemed surprising.		
QU-A3	ESTIMATE the mean CDF accounting for the "state-of-knowledge" correlation between event probabilities when significant [Note (1)].	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. An uncertainty analysis is performed for both CDF and LERF to estimate the mean values from internal and external (excluding seismic) events. The analysis is described in the Catawba Rev 3a PRA Model Integration Notebook. A correlation factor has been developed and is used to apply a multiplier to those ISLOCA cut sets having two MVR or CVR type code events in the same cut set.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.
QU-A4	SELECT a method that is capable of discriminating the contributors to the CDF commensurate with the level of detail in the model.	Dispositioned	F&O QU-12: The Conditional core damage Probability of several Initiators from the CR2b results were evaluated. The results are: 8.30E-03 Loss Of RN 8.38E-03 Loss Of KC 5.04E-03 Small LOCA 2.30E-04 Secondary Line Break Inside Containment 5.47E-05 LOOP 1.24E-05 Inadvertent SS Actuation 1.24E-05 Loss Of Instrument Air 1.04E-05 Steamline Break Outside Containment 9.54E-06 FDW Line Break Outside Containment 2.26E-06 Loss Of Main Feedwater 2.89E-06 SGTR 7.75E-07 Loss Of Load 5.01E-07 Reactor Trip These results show a discrepancy between Small LOCA and SGTR that is not consistent with what is normally seen in PRAs in the industry. The CCDP for small LOCA and SGTR are usually in the same order of magnitude because the initiators have similar mitigation functions such as safety injection, secondary side heat removal, primary cooldown and	The NEI SRs applicable to this ASME SR are QU-4, QU-8, QU-9, QU-10, QU-11, QU-12, and QU-13, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated QU-4, QU-9, QU-10 and QU-12 as "3" and QU-8 and QU-11 as "3 with contingencies." QU-8 has one level "B" F&O: QU-12; and QU-11 has one level "B" F&O: QU-02. F&O QU-12: The Catawba PRA has updated the small LOCA (SL) initiator to be redefined to only include small pipe breaks. The SL and SGTR initiating event frequencies are found in the CNS U1&2 internal initiator events frequency data notebook. This is considered to resolve the finding F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance. The Catawba PRA model consists of a top logic fault tree that links the fault tree models for the frontline and support systems, and is solved to produce an overall CDF and LERF. The results produced	Based on the disposition, this SR is considered met. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			depressurization, and long term injection if cooldown and depressurization are not successful. A difference of 3 orders of magnitude is unusual. Also, the CCDP value for the Loss of Instrument Air probability is identical to the Inadvertent SS Actuation probability (to 3 significant figures), which seemed surprising. QU-02: The IE's for certain support system failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE.	include individual cutsets (consisting of basic event combinations) that are provided in order to determine the significant and non-significant contributors to CDF. The CNS self-assessment considered this SR met.	
QU-A5	INCLUDE recovery actions in the quantification process in applicable sequences and cut sets (see HR-H1, HR-H2, and HR-H3).	Dispositioned	F&O QU-05: Event ND0RWSTDHE: This is a recovery action to terminate the NV and NI pumps in the event of failure of ND to provide recirculation after a SL. The event was quantified on the basis of tripping the pumps within 18 minutes. RWST refill was assumed to occur (from undescribed source) and pumps were restarted to continue injection. This recovery event is applied to a) loss of KC pumps b) SNSDRNVLHE - drain plug blockage c) CCF of ND pumps. The recovery event is intended to provide injection flow for the long term commensurate with the RWST make-up capability. The time of some of these failure is 20 minutes, when injection requirements are beyond the make-up capability of the RWST. Secondly, there are cutsets representing heat removal that cannot be recovered by continued	The NEI SRs applicable to this ASME SR are QU-18 and QU-19, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated QU-18 as "3" and QU-19 as "3 with contingencies," with associated level "B" F&Os QU-05 and QU-08, respectively. Catawba Rev 3a PRA Model Integration Notebook describes the general recovery rules development and describes development of the recovery rules to address dependencies among HEP combinations. Some comments are included in the general rule files to describe the basis for the included recovery rules and the HRA documentation includes a spreadsheet which determines HEP dependencies and the associated recoveries needed. F&O QU-05: Event NDORWSTDHE has been redefined and failure probability recalculated for the Catawba Rev 3a PRA Model Integration Notebook.	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			injection of HHSI. The sequence needs continuous injection of HHSI and heat removal from containment. F&O QU-08: Documentation of mutually exclusive events is limited to the text file, cr2b_rul.txt. The rule recovery file allows different numbers of max recoveries depending on the combinations in question. There is no documentation regarding how the max recoveries were established for each set of events. Examples of the content of the file are DBLMAINT, DELSEQ, and NSHEATEX, which function as recoveries set to 0.0 Note that this applies to recovery events as well as deleted combinations.	F&O QU-08 is tied to the corresponding NEI SR QU-19.The F&O applies to SR QU-B8 and is not evaluated against SR QU-A4.	
QU-B2	TRUNCATE accident sequences and associated system models at a sufficiently low cutoff value that dependencies associated with significant cutsets or accident sequences are not eliminated. NOTE: Truncation should be carefully assessed in cases where cutsets are merged to create a solution (e.g., where system level cutsets are merged to create sequence level cutsets).	Dispositioned	F&O QU-01: The truncation limit of the baseline CDF at 1E-9 is not low enough to defend convergence toward a stable result. This is shown on page 12 of SAAG-579. Use of the 1E-9 truncation limit yields 4485 cutsets, while the 1E-10 truncation limit yields 31512 cutsets. Thus, although the PRA runs using 1E-9 are capturing about 85% of the CDF predicted with a cutoff of 1E-10, they are capturing only 13% of the cutsets using the 1E-10 truncation limit.	The NEI SRs applicable to this ASME SR are QU-21, QU-22, QU-23, and QU-24, and there are no NRC objections. There is an industry action to confirm that this requirement is met. The original Peer Review rated QU-21, QU-22 and QU-23 as "3" and QU-24 as "3 with contingencies." QU-24 has one level "B" F&O: QU-01. F&O QU-01: Catawba Rev 3a PRA Model Integration Notebook documents that a truncation study was performed to calculate the truncation limit that would meet the criteria of being four orders of magnitude below the calculated baseline CDF and captures 90% of the bounding CDF risk and the percent change in increase in calculated CDF should be less than 5% from the previous decade. A truncation limit of 5.0E-10 for CDF (and 5.0E-11 for LERF) was calculated to meet the criteria. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact on the ILRT extension.
QU-B3	ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no significant accident sequences are inadvertently eliminated. For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in	Dispositioned	F&O QU-01: The truncation limit of the baseline CDF at 1E-9 is not low enough to defend convergence toward a stable result. This is shown on page 12 of SAAG-579. Use of the 1E-9 truncation limit yields 4485 cutsets, while the 1E-10 truncation limit yields 31512 cutsets. Thus, although the PRA runs using 1E-9 are capturing about 85% of the CDF predicted with a cutoff of 1E-10, they are capturing only 13% of the cutsets using the 1E-10 truncation limit.	The NEI SRs applicable to this ASME SR are QU-21, QU-22, QU-23, and QU-24, and there are no NRC objections. There is an industry action to confirm that the final truncation limit is such that convergence toward a stable CDF is achieved. The original Peer Review rated QU-21, QU-22 and QU-23 as "3" and QU-24 as "3 with contingencies." QU-24 ha one level "B" F&O: QU-01. F&O QU-01: Catawba Rev 3a PRA Model Integration Notebook documents that a truncation study was performed to calculate the truncation limit that would meet the criteria of being four orders of	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact on the ILRT extension.

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SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	decreasing changes in CDF or LERF, and the final change is less than 5%.			magnitude below the calculated baseline CDF and captures 90% of the bounding CDF risk and the percent change in increase in calculated CDF should be less than 5% from the previous decade. A truncation limit of 5.0E-10 for CDF (and 5.0E-11 for LERF) was calculated to meet the criteria. This is considered to resolve the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR	
QU-B6	ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.	Open	F&O AS-04: There were several observations on the modeling of event D3 in the SGTR tree: Event D3 is generally defined as the event to cooldown to RHR conditions using 2/3 SG for depressurization. D3 includes the HEP YAGRCOLDHE, which is directed by ECA 3.1 and 3.2. 1. D3 is defined as "primary system cooldown via secondary system depressurization". Primary system depressurization must be accomplished in some sequences (YD1D2D3, YOD3, YUOD3), by either PORV, aux spray, or main spray. These functions are not included in D3. 2. Sequence YUOD3 needs a T/H justification that D3 can actually prevent core damage in this circumstance. This sequence has no injection and no SG isolation. This is "core cooling recovery" with an unisolated SGTR. ECA3 specifies cool down at less than 100F/hr. The core cannot be maintained covered for the amount of time it takes to cooldown to RHR conditions at 100F/hr. Suggested resolution is to use a separate function for this heading, using an operator action directed by FRC.1 and without RCP operating. 3. Sequence YUD1QD3. comment #2 applies to this sequence as well. This is a stuck open relief PORV with no injection. F&O AS-07: The success criteria for AFW for SGTR is 1 CA pump to 2 steam generators. The ruptured SG is assumed to be one of the two steam generators that supply steam to the turbine-driven AFW pump. In the Catawba Rev. 2b fault tree model, however, the dependency of the TDP on the SGTR initiator is not modeled. Thus, the TDP	The NEI SRs applicable to this ASME SR are AS-8, AS-9, QU-4, QU-20, and QU-25, and there are no NRC objections. There is an industry action to check for proper accounting of success terms. The original Peer Review rated QU-4 and QU-20 as "3" and AS-8 and AS-9 as "3 with contingencies." QU-25 is rated as "NA". AS-8 has one level "B" F&O: AS-04; and AS-9 has one level "B" F&O: AS-07. In the Catawba Rev 3a PRA Model Integration Notebook system successes are credited by post-processing recovery rules. F&O AS-04 is only applicable to SGTR events. The modeling of SGTR events was changed to be consistent with industry standards using the guidance in WCAP-15955. Success criteria runs were performed for the MNS PRA and are applicable to CNS. Reconstruction of the CNS SGTR success criteria is needed to close this F&O. F&O AS-07 is only applicable to SGTR events. The CA notebook was updated to reflect the correct success criteria due to SGTR loss of AFW pump, so AS-07 is considered resolved. This is considered to resolve the findings and achieve grade 3 of NEI SR / meet CAT II of the ASME SR.	There were no F&Os with "A" level of significance at CNS. Open level "B" F&O AS-04 is only applicable to SGTR events. A sensitivity was done in Section 5.3.5 to approximate the necessary SGTR success criteria modeling changes. Changes in Success Criteria modeling are based on guidance provided in Reference 43. Other than a small change to overall risk, there is no impact on the ILRT extension.

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			supply is not degraded by the initiating event in the model logic, so the model is incorrect. (This item is already on the list of corrective actions for the Catawba PRA, and Duke has indicated that it will be implemented in the Rev. 3 PRA.)		
QU-B7	IDENTIFY cutsets (or sequences) containing mutually exclusive events in the results.	Dispositioned	F&O QU-08: Documentation of mutually exclusive events is limited to the text file, cr2b_rul.txt. The rule recovery file allows different numbers of max recoveries depending on the combinations in question. There is no documentation regarding how the max recoveries were established for each set of events. Examples of the content of the file are DBLMAINT, DELSEQ, and NSHEATEX, which function as recoveries set to 0.0 Note that this applies to recovery events as well as deleted combinations.	The NEI SR applicable to this ASME SR is QU-26, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated this NEI SRs as "3", with associated level "B" F&O QU-08. F&O QU-08: Mutually exclusive event combinations (e.g., double initiating events, double maintenance, and other invalid combinations of events) are included in the general recovery rule file for the purpose of eliminating cutsets with those combinations from the quantification results which is shown in Catawba Rev 3a PRA Model Integration Notebook. The NEI grade of 3 was assigned to the correlated element. CDF and LERF Model Integration notebook 1535.00-00-0061 and Results and Insights for Catawba PRA 1535.00-00-0075 were revised and this F&O is considered resolved.	There were no F&Os with "A" level of significance at CNS, and there are no remaining open level "B" F&Os related to this SR. There is no impact on the ILRT extension.
QU-B8	CORRECT cutsets containing mutually exclusive events by either (a) developing logic to eliminate mutually exclusive situations, or (b) deleting cutsets containing mutually exclusive events	Dispositioned	F&O QU-08: Documentation of mutually exclusive events is limited to the text file, cr2b_rul.txt. The rule recovery file allows different numbers of max recoveries depending on the combinations in question. There is no documentation regarding how the max recoveries were established for each set of events. Examples of the content of the file are DBLMAINT, DELSEQ, and NSHEATEX, which function as recoveries set to 0.0 Note that this applies to recovery events as well as deleted combinations.	The NEI SR applicable to this ASME SR is QU-26, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated this NEI SR as "3", with associated level "B" F&O QU-08. F&O QU-08: Mutually exclusive event combinations (e.g., double initiating events, double maintenance, and other invalid combinations of events) are included in the general recovery rule file for the purpose of eliminating cutsets with those combinations from the quantification results which is shown in Catawba Rev 3a PRA Model Integration Notebook. The NEI grade of 3 was assigned to the correlated element. CDF and LERF Model Integration notebook 1535.00-00-0061 and Results and Insights for Catawba PRA 1535.00-00-0075 were revised and this F&O is considered resolved.	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact on the ILRT extension.
QU-C1	IDENTIFY cutsets with multiple HFEs that potentially impact significant accident sequences/cutsets by requantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. The final quantification of these	Dispositioned	QU-02: The IE's for certain support system failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the	The NEI SRs applicable to this ASME SR are QU-10, QU-17, HR-26, and HR-27, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated QU-10, HR-26 and HR-27 as "3" and QU-17 as "3 with contingencies." QU-17 has one level "B" F&O: QU-02. Catawba Rev 3a PRA Model Integration Notebook describes the steps taken to solve the tree at 5.0E-11 (an order of magnitude lower	Based on the disposition, this SR is considered met. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	post-initiator HFEs may be done at the cutset level or saved sequence level.		available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE.	than the typical truncation) with a special database where HFEs with a nominal value of less than 0.1 have been increased to 0.1 to ensure that cutsets involving multiple human events are not truncated and can be evaluated for dependencies. The identified combinations are evaluated and quantified by an HRA analyst. Multiple human error events within a cut set are replaced with a single human error event that considers the sequence of the operator actions and their interdependence. F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance.	
QU-C2	ASSESS the degree of dependency between the HFEs in the cutset or sequence in accordance with HR-D5 and HR-G7.	Dispositioned	QU-02: The IE's for certain support system failures (RN, KC) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree. However, failures that cause the IE may also affect the mitigating system, such that there is a dependency between the initiating event and the available mitigation. Examples are an electrical bus that failed one train of KC and could fail one train of mitigating equipment. Another example is the operator error in the loss of KC to start the standby train of KC (KKCSTNBDHE). The HRA notebook states this event has dependencies with HYDBACKDHE.	The NEI SRs applicable to this ASME SR are QU-10 and QU-17, and there are no NRC objections. There is an industry action to verify dependencies in cutsets/sequences are assessed. The original Peer Review rated QU-10 as "3" and QU-17 as "3 with contingencies." QU-17 has one level "B" F&O: QU-02. Catawba Rev 3a PRA Model Integration Notebook describes the steps taken to solve the tree at 5.0E-11 (an order of magnitude lower than the typical truncation) with a special database where HFEs with a nominal value of less than 0.1 have been increased to 0.1 to ensure that cutsets involving multiple human events are not truncated and can be evaluated for dependencies. The identified combinations are evaluated and quantified by an HRA analyst. Multiple human error events within a cut set are replaced with a single human error event that considers the sequence of the operator actions and their interdependence. F&O QU-02: System level initiators represented as fully developed sub-tree structures are not in the Rev 3 model. Duke Energy feels that it is acceptable to not develop system level initiators as long as a review for dependencies takes place in the cut set file. This process has been proceduralized and is contained in Section 4 of Workplace Guideline XSAA-103, Guidelines For Determining Risk Significance.	Based on the disposition, this SR is considered met. There is no impact to the ILRT extension.
QU-D4	COMPARE results to those from similar plants and IDENTIFY causes for significant differences.	Open	F&O QU-12: The Conditional core damage Probability of several Initiators from the CR2b results were evaluated. The results are:	The NEI SRs applicable to this ASME SR are QU-8, QU-11, and QU-31, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated QU-31 as "3" and QU-8	There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	For example: Why is LOCA a large contributor for one plant and not another?		8.30E-03 Loss Of RN 8.38E-03 Loss Of KC 5.04E-03 Small LOCA 2.30E-04 Secondary Line Break Inside Containment 5.47E-05 LOOP 1.24E-05 Inadvertent SS Actuation 1.24E-05 Loss Of Instrument Air 1.04E-05 Steamline Break Outside Containment 9.54E-06 FDW Line Break Outside Containment 2.26E-06 Loss Of Main Feedwater 2.89E-06 SGTR 7.75E-07 Loss Of Load 5.01E-07 Reactor Trip These results show a discrepancy between Small LOCA and SGTR that is not consistent with what is normally seen in PRAs in the industry. The CCDP for small LOCA and SGTR are usually in the same order of magnitude because the initiators have similar mitigation functions such as safety injection, secondary side heat removal, primary cooldown and depressurization, and long term injection if cooldown and depressurization are not successful. A difference of 3 orders of magnitude is unusual. Also, the CCDP value for the Loss of Instrument Air probability is identical to the Inadvertent SS Actuation probability (to 3 significant figures), which seemed surprising.	and QU-11 as "3 with contingencies." QU-8 has one level "B" F&O: QU-02; and QU-11 has one level "B" F&O: QU-12. Only F&O QU-12 is related to this ASME SR. F&O QU-12: The Catawba PRA has updated the small LOCA (SL) initiator to be redefined to only include small pipe breaks. The SL and SGTR initiating event frequencies are found in the CNS U1&2 internal initiator events frequency data notebook. This is considered to resolve the finding. Catawba needs to perform and document a comparison of results between the CNS PRA and other similar plants to be incorporated into the Catawba PRA model integration notebook.	
QU-D6	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. INCLUDE SSCs and operator actions that contribute to initiating event frequencies and event mitigation.	Dispositioned	F&O QU-12: The Conditional core damage Probability of several Initiators from the CR2b results were evaluated. The results are: 8.30E-03 Loss Of RN 8.38E-03 Loss Of KC 5.04E-03 Small LOCA 2.30E-04 Secondary Line Break Inside Containment 5.47E-05 LOOP 1.24E-05 Inadvertent SS Actuation 1.24E-05 Loss Of Instrument Air 1.04E-05 Steamline Break Outside Containment 9.54E-06 FDW Line Break Outside Containment 2.26E-06 Loss Of Main Feedwater 2.89E-06 SGTR	The NEI SRs applicable to this ASME SR are QU-8 and QU-31, and there are no NRC objections. There is an industry action to confirm that this requirement is met. The original Peer Review rated QU-31 as "3" and QU-8 as "3 with contingencies." QU-8 has one level "B" F&O: QU-02. The Results and Insights from Catawba PRA Notebook provides a summary of the CDF results by IE, the most important operator actions and top SSCs. F&O QU-12: The Catawba PRA has updated the small LOCA (SL) initiator to be redefined to only include small pipe breaks. The SL and SGTR initiating event frequencies are found in the CNS U1&2 internal initiator events frequency data notebook. This is considered to resolve	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			7.75E-07 Loss Of Load 5.01E-07 Reactor Trip These results show a discrepancy between Small LOCA and SGTR that is not consistent with what is normally seen in PRAs in the industry. The CCDP for small LOCA and SGTR are usually in the same order of magnitude because the initiators have similar mitigation functions such as safety injection, secondary side heat removal, primary cooldown and depressurization, and long term injection if cooldown and depressurization are not successful. A difference of 3 orders of magnitude is unusual. Also, the CCDP value for the Loss of Instrument Air probability is identical to the Inadvertent SS Actuation probability (to 3 significant figures), which seemed surprising.	the finding and achieve grade 3 of NEI SR / meet CAT II of the ASME SR	
QU-D7	REVIEW the importance of components and basic events to determine that they make logical sense.	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. The Results and Insights from Catawba PRA Notebook provides the importances and top SSCs.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.
QU-E3	ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the "state-of-knowledge" correlation.	Dispositioned	None	The NEI SR applicable to this ASME SR is QU-30, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated this NEI SR as "3". There are no level "A" or "B" F&Os associated with this NEI SR. NEI 00-02 only partially addresses this supporting requirement under QU-30. An uncertainty analysis is performed for both CDF and LERF to estimate the mean values from internal and external (excluding seismic) events. The analysis is described in the Catawba Rev 3a PRA Model Integration Notebook. A correlation factor has been developed and is used to apply a multiplier to those ISLOCA cut sets having two MVR or CVR type code events in the same cut set.	There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os related to this SR. There is no impact to the ILRT extension.
QU-E4	For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success	Dispositioned	None	The NEI SRs applicable to this ASME SR are QU-28, QU-29, and QU-30, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3". There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os associated with any of these NEI SRs. NEI 00-02 only partially addresses this supporting requirement under QU-28, QU-29 and QU-30.	With the sensitivity of the model and characterization of uncertainties unknown there is potential to impact the ILRT extension. However the impact is expected to be minimal.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	criterion, introduction of a new initiating event) [Note (1)].			Although general modeling assumptions are provided in the PRA Modeling Guidelines (XSAA-115) and specific assumptions related to system design, operation, and modeling are documented in the various PRA notebooks, the sensitivity of the results to model uncertainties and assumptions has not been thoroughly documented.	
QU-F2	DOCUMENT the model integration process including any recovery analysis, and the results of the quantification including uncertainty and sensitivity analyses. For example, documentation typically includes (a) records of the process/results when adding non-recovery terms as part of the final quantification (b) records of the cutset review process (c) a general description of the quantification process including accounting for systems successes, the truncation values used, how recovery and post-initiator HFEs are applied (d) the process and results for establishing the truncation screening values for final quantification demonstrating that convergence towards a stable result was achieved (e) the total plant CDF and contributions from the different initiating events and accident classes (f) the accident sequences and their contributing cutsets (g) equipment or human actions that are the key factors in causing the accidents to be non-dominant (h) the results of all sensitivity studies (i) the uncertainty distribution for the	Dispositioned	F&O QU-04: More guidance or creation of a procedure is needed to address the quantification steps. For example, there is no desktop guide or procedure as there is for developing system fault trees. There is no discussion in any of the documentation on what codes are used for the quantification process, or what files are needed to establish the CAFTA run parameters. Develop quantification guidance for PSA analysts. This should include information on the quantification codes and the run parameters. Standards for quantification commensurate with the application type should be included.	The NEI SRs applicable to this ASME SR are QU-4, QU-12, QU-13, QU-27, QU-28, QU-31, QU-32, and MU-7, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3". There were no F&Os with "A" level of significance at CNS. Level "B" F&O QU-04 was written against NEI subelement QU-3 which is not mapped to any of the SRs in the current PRA Standard, however, it is associated with this SR. F&O QU-04: SAAG 791, Catawba Rev 3 PRA Integration Notebook (1535.00-00-0061), has been greatly expanded with respect to providing quantification guidance for PSA analysts. The model integration process and basic quantification results are documented in the Catawba Rev 3a PRA Model Integration Notebook. However the documentation of the PRA model needs to be expanded to address all required items. This is documented in the Level C F&O QU-10. The NEI grade of 3 was assigned to each correlated element.	There were no F&Os with "A" level of significance at CNS and there are no remaining open level "B" F&Os related to this SR. There is no impact to the ILRT extension.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	total CDF (j) importance measure results (k) a list of mutually exclusive events eliminated from the resulting cutsets and their bases for elimination (l) asymmetries in quantitative modeling to provide application users the necessary understanding of the reasons such asymmetries are present in the model (m) the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process				
QU-F3	DOCUMENT the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary. PROVIDE a detailed description of significant accident sequences or functional failure groups.	Dispositioned	None	The NEI SR applicable to this ASME SR is QU-31, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated this NEI SR as "3". There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os associated with this NEI SR. NEI 00-02 only partially addresses this supporting requirement under QU-31. The Results and Insights from Catawba PRA Notebook provides a summary of the CDF results by IE, the most important operator actions and top SSCs.	There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os related to this SR. There is no impact to the ILRT extension.
QU-F4	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).	Dispositioned	None	The NEI SRs applicable to this ASME SR are QU-27, QU-28, and QU-32, and there are no industry self-assessment actions and no NRC objections. The original Peer Review rated all of these NEI SRs as "3". There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os associated with any of these NEI SRs. NEI 00-02 only partially addresses this supporting requirement under QU-27, QU-28, and QU-32. General modeling assumptions are provided in the PRA Modeling Guidelines (XSAA-115) and specific assumptions related to system design, operation, and modeling are documented in the various PRA notebooks.	There were no F&Os with "A" level of significance at CNS and there are no level "B" F&Os related to this SR. There is no impact to the ILRT extension.
QU-F5	DOCUMENT limitations in the quantification process that would impact applications.	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. The PRA Modeling Guidelines (XSAA-115) describes some basic MAAP limitations. For applications, workplace procedure XSAA103 describes the need to resolve the integrated model if the failure	Based on the disposition, the requirements of Cat II are considered met. There is no

Table A-1 Internal Events PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
				probability associated with a modeled SSC increases. The procedure also notes that the initiator model is resolved prior to resolving the integrated model if an SSC of interest is included in an initiator fault tree.	impact to the ILRT extension.
QU-F6	DOCUMENT the quantitative definition used for significant basic event, significant cutset, and significant accident sequence. If it is other than the definition used in Part 2, JUSTIFY the alternative.	Dispositioned	None	There are no NEI SRs applicable to this ASME SR. The Results and Insights from Catawba PRA notebook identifies the risk-significant accident sequences, systems, components and operator actions. However there is no discussion of a specific quantitative definition for significant basic events, cutsets, accident sequences or functional failures.	Based on the disposition, the requirements of Cat II are considered met. There is no impact to the ILRT extension.

Table A-2 LERF PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
LE-B2	DETERMINE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in LE-B1 using applicable generic or plant-specific analyses for significant containment challenges. USE conservative treatment or a combination of conservative and realistic treatment for non-significant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-C1	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in LE-B1 and analyzed in LE-B2. Compare the containment challenges analyzed in LE-B with the containment structural capability analyzed in LE-D and identify accident progressions that have the potential for a large early release. JUSTIFY any generic or plant-specific calculations or references used to categorize releases as non-LERF contributors based on release magnitude or timing. NUREG/CR-6595, App. A [2-16] provides an acceptable definition of LERF source terms.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-C3	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair (i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability [see SY-A24, DA-C15, and DA-D8]). AC power recovery based on generic data applicable to the plant is acceptable.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-C4	INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, effect of fission product	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be

Table A-2 LERF PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features.				conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-C9	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-C11	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-D2	EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows and INCLUDE as potential containment challenges, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-D3	When containment failure location [Note (1)] affects the event classification of the accident progression as a large early release, DEFINE failure location based on a realistic containment assessment that accounts for	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be

Table A-2 LERF PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	plant-specific features. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.				conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-D6	PERFORM an analysis of thermally induced SG tube rupture that includes plant specific procedures and design features and conditions that could impact tube failure. An acceptable approach is one that arrives at plant-specific split fractions by selecting the SG tube conditional failure probabilities based on NUREG-1570 [2-17] or similar evaluation for induced SG failure of a similarly designed SGs and loop piping. SELECT failure probabilities based on (a) RCS and SG post-accident conditions to sufficient to describe the important risk outcomes (b) secondary side conditions including plant-specific treatment of MSSV and ADV failures JUSTIFY assumptions and selection of key inputs. An acceptable justification can be obtained by the extrapolation of the information in NUREG-1570 [17] to obtain plant-specific models, use of reasonably bounding assumptions, or performance of sensitivity studies indicating low sensitivity to changes in the range in question.	Dispositioned	None	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-E2	USE realistic parameter estimates to characterize accident progression phenomena for significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic estimates for non-significant accident progression sequences resulting in a large early release.	Open	LE-E2-01 (F): Catawba basically used the conservative parameter estimates from NUREG/CR-6595 to characterize the accident progression phenomena. This approach would satisfy CC-I. However, Duke is using the Conditional Containment Failure Probabilities (CCFPs) from Rev. 0 of NUREG/CR-6595 rather than the more restrictive values from Revision 1. To meet	The Westinghouse focused peer review concluded that the Catawba LERF model was adequate to meet Cat I; however, the LERF report needed to include the Duke white paper addressing the use of CCFPs from Rev 0 of the NUREG as opposed to Rev 1.	When the Westinghouse documentation recommendation is implemented, the SR will meet Cat I. Then, per Reference 36, the LERF model is sufficient to support risk-informed applications. Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be

Table A-2 LERF PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			this requirement would require using the NUREG/CR-6595, Rev. 1 CCFP values or providing an engineering analysis to defend use of the older values. At the time of the peer review, Duke did have a white paper, "Conditional Containment Failure Probabilities for the McGuire and Catawba Large Early Release Frequency Models", November 2012, (Reference 10) that discusses the basis for the use of the CCFPs from Revision 0 of NUREG/CR-6595. However, this white paper was not provided as part of the official documentation for the review and as such, was not directly reviewed. A later review of this white paper indicates that Duke appears to have a reasonable basis for using the revision 0 CCFP values based on plant-specific analysis. Duke should include this information in their LERF analysis reports.	The Duke white paper reviewed the supporting analyses for the conditional containment failure probabilities provided in the various revisions to NUREG/CR-6595. Based on the plant specific analyses performed, the CCFPs utilized in the current CNS LERF analyses (based on NUREG/CR-6595 original issue) are judged to be better estimates than the estimates available from NUREG/CR-6427 or NUREG/CR-6595 revision 1 and are appropriate for a LERF model at CC I. The CNS peer review noted that the position paper appeared to be a reasonable basis for using the NUREG/CR-6595 revision 0 results. Duke also believes that the NUREG/CR-6595 revision 0 results are better estimates than the revision 1 results.	conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-F1	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 2-2.8-3.	Dispositioned	LE-G3-01 (F): In CNC-1535.00-00-061, Catawba documents the significant contributors to LERF in terms of contribution by initiating events. However, they did not document the relative contribution of contributors such as plant damage states, accident progression sequences, phenomena, containment challenges and containment failure modes. To move from CC-I to CC-II/III, Catawba needs to evaluate the relative contributions to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes.	Per Reference 36, LERF model is sufficient to support risk-informed applications.	Limitations with the NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states,	Dispositioned	LE-G3-01 (F): In CNC-1535.00-00-061, Catawba documents the significant	The Westinghouse report suggested that improving this SR	Per Reference 36, LERF model is sufficient to support risk-informed applications. Limitations with the

Table A-2 LERF PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.		contributors to LERF in terms of contribution by initiating events. However, they did not document the relative contribution of contributors such as plant damage states, accident progression sequences, phenomena, containment challenges and containment failure modes. To move from CC-I to CC-II/III, Catawba needs to evaluate the relative contributions to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes.	to a Cat II would require evaluating the relative contributions to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges and containment failure modes. Meeting Cat I already satisfies the application and improving to Cat II would only result in improved documentation.	NUREG/CR-6595 LERF approach used for Catawba include consideration of whether the estimated LERF is significantly below (about an order of magnitude or more) the R.G. 1.174 acceptance guideline. Duke believes the LERF model to be conservative and believes the NUREG/CR-6595 method is acceptable for fire since it produces adequate risk insights. Overall, the impacts to the ILRT extension are expected to be minimal.
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Open	LE-G6-01 (F): Catawba did not document the quantitative definition of significant accident progression sequence. Catawba needs to add a definition for significant accident progression sequence to CNC-1535.00-00-0143, Rev. 0 or CNC-1535.00-00-061. This can be accomplished by adding a specific definition of referencing the appropriate definition in Section 1-2 of RA-Sa-2009.	The Westinghouse report suggested that Catawba add a definition for significant accident progression sequence to the documentation.	Per Reference 36, LERF model is sufficient to support risk-informed applications. This finding is documentation only and does not impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
IFPP-A2	DEFINE flood areas at the level of individual rooms or combined rooms/halls for which plant design features exist to restrict flooding.	Dispositioned	IFPP-A2-01 (F): Discrepancies exist regarding the defining of flood areas at the level of individual rooms. The CNS PRA clearly meets Capability Category I which is based on defining flood areas at the building level. There are some areas within the buildings which are not clearly part of flood areas, where boundaries are vaguely defined, or for where the flood area boundaries are defined inconsistent with the requirements of the Standard. Pipe trenches on AB522 are not included in any of the Flood Areas as defined in Att. A to the -022 calculation. The flood Zone boundary between 594A01 and 594A05 goes through the middle of a hallway south of the HVAC room (similarly for Unit 2 side). There is no discussion on why this is an appropriate boundary for a flood zone. It appears this flood zone should have been defined by the walls and doors to the immediate south of the Main Control Room. Floods could clearly propagate without being impeded as this flood boundary was defined. Given the number of rooms with enclosed doors and flood sources within the plant, the Flood Zones would have been better defined with greater level of detail. As done, this results in many flooding barriers within a Zone. The AB522 flood area, for example, could have been subdivided since the Containment Spray (NS) and Residual Heat Removal (ND) pumps are in separate rooms, with the ND pumps behind doors. Also, while liquid will spray down into all of these rooms, propagation within the room complex could differ depending on flood sources at higher elevations. Figure A-3 in CN-RAM-11-022 shows an incorrect location of the NI pump room s and NV pump rooms. Room 233 is not defined as part of any	Per LTR-RAM-II-13-008, flood zone drawings were evaluated and updated to provide required level of detail to meet Cat II.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Flood Area in the Appendix A Figures. Details of flood zone boundaries south of zone 560Z09 are unclear.		
IFPP-A5	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify: (a) spatial information needed for the development of flood areas (b) plant design features credited in defining flood areas. Note: Walkdown(s) may be done in conjunction with the requirements of IFSO-A6, IFSN-A17, and IFQU-A11.	Dispositioned	IFSO-A6-01 (F): Walkdowns were completed however, there were some discrepancies in walkdown notes. See IFSO-A6.	Per LTR-RAM-II-13-008, walkdown documentation was revised to resolve discrepancies noted by the peer review team.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
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.	Dispositioned	IFSO-B3-01 (F): No discussion of impact of sources of uncertainty with respect to partitioning was documented. No discussion of impact of sources of uncertainty with respect to flooding sources was documented.	Per LTR-RAM-II-13-008, the existing documentation was reviewed and two additional assumptions were added to cover any associated uncertainties related to source or plant partitioning.	The supporting requirement concentrates on internal flooding events. Documentation of uncertainty does not impact the ILRT extension.
IFSO-A1	For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) (b) plant internal sources of flooding (e.g., tanks or pools) located in the flood area (c) plant external sources of flooding (e.g., reservoirs or rivers) that are	Dispositioned	IFSO-A1-01 (F): For each flood area, the potential sources of flooding are to be identified, including equipment (e.g., piping, valves, pumps, tanks) located in the area. Section 5.5 (Table 5-4) documents potential flood sources for each flood area; however some flood areas are missing potential sources of flooding: Turbine Building (577T) does not include Conventional Low Pressure Service Water (RL) or Recirculating Cooling Water System (KR). Sections 5.3.11 and 5.3.12 indicate these piping systems are in the Turbine Building but failure of these systems is not addressed. Likewise, steam (HELB) systems are not included (e.g., Main Steam, Extraction Steam, Reheat Steam, etc). No piping less than or equal to 2-inch diameter is included for flooding and this pipe is not considered as spray source in most areas.	Per LTR-RAM-II-13-008, missing flood sources were added and relevant scenarios were carried forward into the other calculation notes and adequately documented.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
	connected to the area through some system or structure (d) in-leakage from other flood areas (e.g., backflow through drains, doorways, etc.)		Drinking Water (YD) piping was not included as a flood or spray source. Drinking Water system is in the Internal Events PRA model (HYDBACKTRM, YD System is Unavailable) as backup cooling to the 1A NV pump. Failure os this piping may fail that function and flood spray other SSCs. Many tanks (e.g., Liquid Waste Tanks) were identified on the Peer Review walkdowns that are not documented as potential flood sources.		
IFSO-A2	For multi-unit sites with shared systems or structures, INCLUDE any potential sources with multi-unit or cross-unit impacts.	Dispositioned	IFSO-A2-01 (F): One important case where the cross-unit impact is not considered is the consideration of a cross-unit flood from one turbine building affecting the other turbine building. Flooding in one turbine building can propagate to the other turbine building, per Figure 5-1 of CN-RAM-11-022 and per General Arrangement drawings. This cross-unit source should have been considered but was not considered. Discussion of how flooding of Unit 2 offsite power transformers in zone 577T02 would or would not impact Unit 1 is not documented.	Per LTR-RAM-II-13-008, after review of plant characteristics, there was found to be no cross unit impact on the offsite power transformers. Documentation was revised to include this review.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
IFSO-A5	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	Dispositioned	IFSO-A5-01 (F): The internal flooding PRA documentation does not identify system capacities for many systems, nor does it identify flow rates for most releases, not does it identify the pressure and temperature of the source. While flood breach size is characterized in Assumption 9 of CN-RAM-11-023, maximum flow rates for failures in individual systems are not provided. Capacity of sources are discussed for some systems in CN-RAM-11-022 (e.g., Nuclear Service Water infinite capacity, volume of RWST for ECCS breaks), system capacities that would be released are not documented or identified for most systems (e.g., KC, for which a surge tank volume is identified but that does not correspond to the entire system/train volume subject to release).	Per LTR-RAM-II-13-008, documentation was revised to include tables showing system capacities, flow, pressure and temperature to ensure that the information needed for flood sources is documented. References for all values are included in the tables.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			(refer also to discussion relating to documentation under SR IFSN-B1 and its Finding)		
IFSO-A6	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to determine or verify the location of flood sources and in-leakage pathways. Note: Walkdown(s) may be done in conjunction with the requirements of IFPP-A5, IFSN-A17, and IFQU-A11.	Dispositioned	IFSO-A6-01 (F): Discrepancies exist in Walkdown Notes. Walkdown notes identify a number of rooms which are part of the 554A01 and 554A02 Flood Areas as at elevation 560'. Walkdown notes for 577A01 general area does not list CCW HX's, which are very large components, as flood sources. Various radwaste tanks located in general areas in teh auxiliary building were apparently not identified in teh walkdown notes. Note no equipment elevation information is provided for the 577A02 or 577A03 (rooms 419 and 427) mechanical penetration area PRA equipment in the walkdown notes. Main control room walkdown notes indicate two double watertight doors to the main control room. Drawing CN-1040-04 appears to show at least three double doors and three other doors that are not identified in the walkdown notes. Note Table B-1 of CN-RAM-11-023 lists an 18 inch critical height for CA pumps, whereas Table B-2 uses a 16 inch critical height. Note the header on many of the walkdown notes is incorrect. The 560A01 general area has a header of "Aux. Bldg. 577' General Area." (p,212 ff.) The 577 General area 577A01 has a heading referencing to a stairwell. This introduces some confusion in checking the information in teh walkdown notes, including checks for validity. No walkdown sheets for Rooms 205A, 215, and 215B. It is inefficient to have the walkdown notes indexed based on room descriptions rather than by Flood	Per LTR-RAM-II-13-008, walkdown documentation was revised to resolve discrepancies noted by the peer review team.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Areas or Room Numbers with no map of room numbers provided.		
IFSO-B1	DOCUMENT the internal flood sources in a manner that facilitates PRA applications, upgrades, and peer review.	Dispositioned	IFSN-B1-01 (F): The documentation was found lacking either supporting calculations that produced applied values (e.g., flood rates) or references to calculations outside the IFPRA documentation. References to calculations such as break flow were not identified.	Flood rates were based on EPRI methodology to support the IEF generation for given scenarios of spray, flood, major flood or HELB. For flood rates this resulted in evaluating the upper bound of a system's capacity and assigning the appropriate failure mechanisms (e.g. for a system with a 1,000 gpm max flow rate spray and floods were deemed appropriate failure mechanisms). All calculations used and references were adequately explained and documented throughout the analysis. For example the drain flow rate calculations were performed and documented in a manner which would allow for them to be reproduced independently. The documents were re-examined and no other calculations were found to need additional documentation or clarification.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
IFSO-B3	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood sources.	Dispositioned	IFSO-B3-01 (F): No discussion of impact of sources of uncertainty with respect to partitioning was documented. No discussion of impact of sources of uncertainty with respect to flooding sources was documented.	Per LTR-RAM-II-13-008, the existing documentation was reviewed and two additional assumptions were added to cover any associated uncertainties related to source or plant partitioning.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
IFSN-A7	In applying SR IFSN-A6 to determine susceptibility of SSCs to flood-induced failure mechanisms, TAKE CREDIT for the operability of SSCs identified in IFSN-A5 with respect to internal flooding impacts only if supported by an appropriate combination of (a) test or operational data (b) engineering analysis (c) expert judgment	Dispositioned	IFSN-A7-01 (F): Catawba is assuming that floor drains are capable of responding to Spray (100GPM) events so that such events do not need to be analyzed or further evaluated. Many Internal Flooding PRA's do not take any credit for drains, even for 100GPM Spray events, due to generally poor maintenance practices and availability for Sump Pumps, as well as due to the possibility that whatever flood event is going on will cause any debris in the room (self-generated or left after maintenance) to clog the drains and/or damage sump pumps. Generally, Preventive Maintenance Tasks or surveillance requirements for drains should exist prior to crediting the drains for flood	Per LTR-RAM-II-13-008, the finding was addressed by re-evaluating the credit of drains in the source of the spray. The drain system at CNS was only credited for spray scenarios in which the flow rate from the associated break is 100gpm or less. This was substantiated by documented calculations in which one drain was shown to be able to accommodate over 100gpm with a minimal amount of standing water. This credit was considered conservative and was not taken for flood and major flood mitigation or timing. Additionally all PRA-related components are considered failed within the originating flood	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			mitigation. Also, the geometry of the drain system should be investigated to ensure that it is not prone to blockages (e.g., check valve failures). This assumption would specifically impact not evaluating a 100GPM spray scenario in the Diesel Generator buildings, since if undetected this would flood the diesel room to four feet and then start flooding the switchgear areas. It would be legitimate to credit the drains in the switchgear areas –walkdown notes indicate five 6 inch diameter drains in the switchgear rooms (zone 560A05 and 560A06). Water height for a 100 GPM release would be minimal, less than one inch, so the equipment in the switchgear rooms would not be damaged if the flood progressed to that point. While in general drains outside the immediate area would suffice to prevent flooding, this needs to be confirmed given the small number of drains that are reported for the large Aux. Bldg. general areas.	area. Documentation was revised to reflect the re-evaluation of credit for drains.	
IFSN-A10	DEVELOP flood scenarios (i.e., the set of information regarding the flood area, source, flood rate and source capacity, operator actions, and SSC damage that together form the boundary conditions for the interface with the internal events PRA) by examining the equipment and relevant plant features in the flood area and areas in potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs.	Dispositioned	IFSN-A10-01 (F): CN-RAM-12-005, Identification and Estimation of Initiating Event Frequencies, Table 5-4, CNS Passive System Failure Frequency by Flood Area: Table 5-4 (page 33) lists KF as a potential flood source in flood area 543A01. There is no flood or major flood initiator for KF in this flood area in Table 5-5, CNS Passive System Failure Frequency by Initiator. There is no scenario developed for failure of this piping (see CN-RAM-11-023, Characterization of Flood Scenarios, Section 5.4.2). Table 5-4 (page 33) lists CS and KR as potential flood sources in flood area 543A02. There are no flood initiators for CS or KR in this flood area in Table 5-5. There are no scenarios developed for failure of these piping systems in this area (see CN-RAM-11-023, Section 5.4.3).	IFSN-A10-01: Per LTR-RAM-II-13-008, the initiating event frequency documentation and scenario documentation was re-examined to determine whether all potential flood sources are identified and evaluated. Identified discrepancies, including the KF piping in flood area 543A01, were added to the analysis and documentation is being revised. IFSO-A1-01: Per LTR-RAM-II-13-008, missing flood sources were added and relevant scenarios were carried forward into the other calculation notes and adequately documented.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			IFSO-A1-01 (F): IFSO-A1-01 (F): For each flood area, the potential sources of flooding are to be identified, including equipment (e.g., piping, valves, pumps, tanks) located in the area. Section 5.5 (Table 5-4) documents potential flood sources for each flood area; however some flood areas are missing potential sources of flooding: Turbine Building (577T) does not include Conventional Low Pressure Service Water (RL) or Recirculating Cooling Water System (KR). Sections 5.3.11 and 5.3.12 indicate these piping systems are in the Turbine Building but failure of these systems is not addressed. Likewise, steam (HELB) systems are not included (e.g., Main Steam, Extraction Steam, Reheat Steam, etc). No piping less than or equal to 2-inch diameter is included for flooding and this pipe is not considered as spray source in most areas. Drinking Water (YD) piping was not included as a flood or spray source. Drinking Water system is in the Internal Events PRA model (HYDBACKTRM, YD System is Unavailable) as backup cooling to the 1A NV pump. Failure of this piping may fail that function and flood spray other SSCs. Many tanks (e.g., Liquid Waste Tanks) were identified on the Peer Review walkdowns that are not documented as potential flood sources.		
IFSN-A11	For multi-unit sites with shared systems or structures, INCLUDE multi-unit scenarios.	Dispositioned	IFSO-A2-01 (F): One important case where the cross-unit impact is not considered is the consideration of a cross-unit flood from one turbine building affecting the other turbine building. Flooding in one turbine building can propagate to the other turbine building, per Figure 5-1 of CN-RAM-11-022 and per General Arrangement drawings. This cross-unit source should have been considered but was not considered.	Per LTR-RAM-II-13-008, after review of plant characteristics, there was found to be no cross unit impact on the offsite power transformers. Documentation was revised to include this review.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Discussion of how flooding of Unit 2 offsite power transformers in zone 577T02 would or would not impact Unit 1 is not documented.		
IFSN-B1	DOCUMENT the internal flood scenarios in a manner that facilitates PRA applications, upgrades, and peer review.	Dispositioned	IFSN-B1-01 (F): The documentation was found lacking either supporting calculations that produced applied values (e.g., flood rates) or references to calculations outside the IFPRA documentation. References to calculations such as break flow were not identified.	Flood rates were based on EPRI methodology to support the IEF generation for given scenarios of spray, flood, major flood or HELB. For flood rates this resulted in evaluating the upper bound of a system's capacity and assigning the appropriate failure mechanisms (e.g. for a system with a 1,000 gpm max flow rate spray and floods were deemed appropriate failure mechanisms). All calculations used and references were adequately explained and documented throughout the analysis. For example the drain flow rate calculations were performed and documented in a manner which would allow for them to be reproduced independently. The documents were re-examined and no other calculations were found to need additional documentation or clarification.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
IFEV-A4	For multi-unit sites with shared systems or structures, INCLUDE multi-unit impacts on SSCs and plant-initiating events caused by internal flood scenario groups.	Dispositioned	IFSO-A2-01 (F): One important case where the cross-unit impact is not considered is the consideration of a cross-unit flood from one turbine building affecting the other turbine building. Flooding in one turbine building can propagate to the other turbine building, per Figure 5-1 of CN-RAM-11-022 and per General Arrangement drawings. This cross-unit source should have been considered but was not considered. Discussion of how flooding of Unit 2 offsite power transformers in zone 577T02 would or would not impact Unit 1 is not documented.	Per LTR-RAM-II-13-008, after review of plant characteristics, there was found to be no cross unit impact on the offsite power transformers. Documentation was revised to include this review.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.
IFEV-A5	DETERMINE the flood-initiating event frequency for each flood scenario group by using the applicable requirements in 2-2.1.	Dispositioned	IFSO-A1-01: For each flood area, the potential sources of flooding are to be identified, including equipment (e.g., piping, valves, pumps, tanks) located in the area. Section 5.5 (Table 5-4) documents potential flood sources for each flood area; however some flood areas are missing potential sources of flooding: Turbine Building (577T) does not include	Per LTR-RAM-II-13-008, missing flood sources were added and relevant scenarios were carried forward into the other calculation notes and adequately documented.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-3 Internal Flood PRA Peer Review – Facts and Observations

SR	2009 ASME/ANS Cat II Requirement	Status	Finding/Observation	Disposition	Impact on ILRT Extension
			Conventional Low Pressure Service Water (RL) or Recirculating Cooling Water System (KR). Sections 5.3.11 and 5.3.12 indicate these piping systems are in the Turbine Building but failure of these systems is not addressed. Likewise, steam (HELB) systems are not included (e.g., Main Steam, Extraction Steam, Reheat Steam, etc). No piping less than or equal to 2-inch diameter is included for flooding and this pipe is not considered as spray source in most areas. Drinking Water (YD) piping was not included as a flood or spray source. Drinking Water system is in the Internal Events PRA model (HYDBACKTRM, YD System is Unavailable) as backup cooling to the 1A NV pump. Failure of this piping may fail that function and flood spray other SSCs. Many tanks (e.g., Liquid Waste Tanks) were identified on the Peer Review walkdowns that are not documented as potential flood sources.		
IFEV-B1	DOCUMENT the internal flood-induced initiating events in a manner that facilitates PRA applications, upgrades, and peer review.	Dispositioned	IFSN-B1-01 (F): The documentation was found lacking either supporting calculations that produced applied values (e.g., flood rates) or references to calculations outside the IFPRA documentation. References to calculations such as break flow were not identified.	Flood rates were based on EPRI methodology to support the IEF generation for given scenarios of spray, flood, major flood or HELB. For flood rates this resulted in evaluating the upper bound of a system's capacity and assigning the appropriate failure mechanisms (e.g. for a system with a 1,000 gpm max flow rate spray and floods were deemed appropriate failure mechanisms). All calculations used and references were adequately explained and documented throughout the analysis. For example the drain flow rate calculations were performed and documented in a manner which would allow for them to be reproduced independently. The documents were re-examined and no other calculations were found to need additional documentation or clarification.	The supporting requirement concentrates on internal flooding events. Since internal flooding represents such a small portion of internal events risk, resolutions to address internal flood peer review findings do not significantly impact the ILRT extension.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A11-01	IDENTIFY instances where cable routing is assumed.	Closed / Met	The "Y3" assessment in Appendix B of CNC-1535.00-00-0109 excludes cables for a small number of components that are not in the ARTRAK (e.g., 7 components for Air). While the routing of the cables from the electrical panel to the compressor may be sufficient to determine that power is available, the compressor itself has instrumentation and controls, that could cause spurious trips or spurious starts that do not appear to be included in the review of Y3 components and may not be limited to the routing areas in the assumed routing. For instance, the compressor control cable will likely go to the control room for switches, alarms and other controls. Similar information would be needed for other systems credited in the Y3 list as well. This SR was judged to be not met.	An assumption was added to the FPRA Summary Report to indicate that the Y3 components are based on assumed routing. The Y3 list of basic events was developed considering both power and control cables in which each Y3 component could be credited. Sensitivity analysis performed in the FPRA Summary Report show that the impact of the Y3 components on quantification is relatively minimal. Credit by exclusion was used as a reasonable alternative to cable routing of Fire PRA components of lesser importance. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
CS-81-01	ANALYZE all electrical distribution buses credited in the FPRA Plant Response Model for proper overcurrent coordination and protection.	Closed/CC-I	CNS performed a review of their existing electrical overcurrent coordination and protection analysis. As a result of this review, CNS identified a number of deficiencies in terms of scope and level of detail. CNS is currently in the process of completely redoing their electrical overcurrent coordination and protection analysis. The new analysis will increase the level of detail and increase the scope to include all Appendix R equipment, the PRA equipment and the NPO equipment. As part of this re-analysis, CNS is making plant modifications as needed. However, at this time, this analysis is not complete. SR considered met at CC-I.	The update of the breaker coordination and protection analysis was completed subsequent to the peer review and has since been incorporated into the Fire PRA. Breaker coordination related interlocks from pseudo components modeled in DATATRAK that were tabulated in Section 6.0 of the CNS Appendix R Coordination Study (Document No. 32-9043224) have been included in the Fire PRA as described in the Cable Selection Report. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.
CS-C3 (no F&O)	If the provision of SR CS-A11 is used, DOCUMENT the assumed cable routing and the basis for concluding that the routing is reasonable in a manner that facilitates FPRA applications, upgrades, and peer review.	Closed / Met	The review of the components selected for Y3 in Appendix B do not provide justification that the components and routings for Y3 are a complete list and that the systems can be credited in all of the fire areas and scenarios where they have been excluded from the UNL list. This SR was judged to be not met.	Refer to disposition for cross-referenced F&O CS-A11-01.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
ES-C1-01	IDENTIFY instrumentation that is relevant to the quantification of HEPs for operator actions that are to be addressed in the FPRA and quantified per SR HRAC 1.	Closed / Met	HRA events are reviewed for instrumentation in attachment B of CNC-1535.00-00-0108 revision 0. The documentation for HRA events that do not have instrumentation in the internal events model is not clear. Instrumentation is described in general terms without information on the number of trains or the number of instruments available. There is not enough documentation to justify the diverse and redundant argument. This SR was judged to be not met.	Additional details were added to Appendix B of the Component Selection calculation to support the redundant (multiple trains) and diverse (multiple parameters such as level and pressure) argument. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
ES-C2-01	IDENTIFY instrumentation associated with each operator action to be addressed, based on fire-induced failure of any single instrument whereby one of the modes of failure to be considered is spurious operation of the instrument.	Closed / CC-II	The Equipment Selection calculation CNC-1535.00-00-0108 revision 0, addresses spurious instrumentation under "Errors of Commission." This section states "No specific instruments were identified that would cause an undesired operator action without first taking one or more confirmatory actions." The results of the assessment are provided, but no details are provided on who performed the review, what method was used, and what procedures were reviewed. This SR was judged to be not met.	The Component Selection calculation was updated to include additional details of the instrument review including the names of the reviewers. Using the guidance provided in Section 9.7 of the calculation and their firsthand knowledge of CNS, the reviewers evaluated the applicable EP(s), OP(s), & AP(s) in order to determine the important parameters that would be relied on for successful execution of each modeled operator action. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
FQ-A2-01	For each fire scenario selected per the FSS requirements that will be quantified as a contributor to fire-induced plant CDF and/or LERF, IDENTIFY the specific initiating event or events (e.g., general transient. LOOP) that will be used to quantify CDF and LERF.	Closed / Met	Loss of Offsite Power (LOOP) events are not adequately represented in the Fire PRA model. Scenarios resulting in a LOOP are modeled by setting % T1 to TRUE along with the basic events for 6900V Switchgear 1TA/1TD and transformers 1ATC/1ATD. However, this does not satisfy all the LOOP logic, such as the PORV and SRV response following a LOOP, impact on Instrument Air and the ability to recover Main Feedwater. SR judged to be met.	The Fire PRA model was updated to collect offsite power cables under 1/2SYS-OSP which have been linked to basic event PACBOFTDEX under new gate TQ76A to address LOOP logic. This assures that the LOOP affects are reflected in the PORV, IA, MFW, and SRV logic structure. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
FQ-F1-01	DOCUMENT the CDF and LERF analyses in accordance with the HLRQU-F and HLR-LE-G high level requirements and their supporting	Closed / Met	There are asymmetries in the model results between train A and B; this is due to the assumption that A train components are normally running and B train components are in standby (and thus all maintenance is assigned to that train). This results in asymmetrical	A discussion of the model asymmetries and the potential impact on the Fire PRA results has been added to the Fire PRA Model Development report. Additionally, a comparison of risk results and importance measures for A-train versus B-train fire areas and selected equipment demonstrated the

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
	requirements in the ASME PRA Standard and DEVELOP a basis supporting the claim of nonapplicability of any of the requirements under HLR-QU-A Part 2.		results and is not discussed in the document. This SR was judged to be not met.	impact from model asymmetry to be insignificant with respect to FRE conclusions based on 1.174 acceptance thresholds (refer to Section 6.2.5). Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
FQ-F1-02	DOCUMENT the CDF and LERF analyses in accordance with the HLRQU-F and HLR-LE-G high level requirements and their supporting requirements in the ASME PRA Standard and DEVELOP a basis supporting the claim of nonapplicability of any of the requirements under HLR-QU-A Part 2.	Closed / Met	Many specific details from HLR-QU-F and HLR-LE-G are not documented. Specifically: - QU-F2: Review process, identification of key equipment and operator actions, bases for mutually exclusive events, and the process used to illustrate computer code correctness. - QU-F5 and LE-G5: Limitations in the quantification process that would impact applications. - QU-F6 and LE-G6: Quantitative definition for 'significant'. - LE-G2: Containment failure analysis and failure probability estimate for containment implosion due to spurious NS or VX activation.	An appendix to the CNS Fire PRA Summary Report has been added to include an importance measure report from the integrated cutset results to address QU-F2 (the key equipment/actions). Sections 3.1 and 3.2 of the Model Development Report have had additional discussion provided to address LE-G2 (Spurious NS, VX). Section 4.0 of the Model Development Report addresses the quantitative definition of significant, QU-F6 and LE-G6. Section 7.0 of the Application Calculation and Sections 2.2 of the model development report have been updated to addresses QUF2 and LE-G5 (computer code correctness and limitations). Section 6.2 of the Model Development report was updated to provide the basis for mutually exclusive event recovery rules. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
FSS-A1-01	IDENTIFY all risk-relevant ignition sources, both fixed and transient, in each unscreened physical analysis unit within the global analysis boundary.	Closed / Met	Documentation of the potential sources of fire in each compartment has not been completed. SR judged to be met.	The Fire Scenario report documentation was updated to list the ignition sources that were screened from quantification for each fire compartment. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
FSS-A2-01	GROUP all risk-relevant damage targets in each unscreened physical analysis unit within the global analysis boundary into one or more damage target sets and for each	Addressed with no impacts to the NFPA 805 application / Met	Target Sets and related Failure Modes are not listed in a comprehensive and organized fashion, and then linked to ignition sources. Example: For FA1, targets are (1) trays in pump room, (2) NS Pump 1A motor, (3) NS Pump 1A itself, (4) NS Pump 1A motor junction box, etc. Tray identification may be needed in some fire areas. Then postulated ignition sources are linked to	As described in the CNS Fire Scenario Report, only those targets within the zone of influence of an ignition source (which may also be a target as in the case of an NS pump) are identified for a given scenario. It was considered not practical to group target sets and then locate ignition sources. However, a list of scenarios where a specific

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
	target set, SPECIFY the equipment and cable failures, including specification of the failure modes.		each target or group of targets (i.e., oil leak to pump and motor, etc.). SR judged to be met.	target was impacted (either a tray target or a specific component) can be derived using the FRANC database. Note that all of the targets in the NS and ND pump rooms in FA 1 were assumed to be failed by the pump fire. Since this F&O has been met, there is negligible impact to the ILRT extension.
FSS-H10-01	DOCUMENT the walkdown process and results.	Closed / Met	Fire Area walkdown notes were input to a computer database, but no output has been created for documentation purposes. In addition, plant drawings identifying the fire areas and the ratings of boundaries to these fire areas have not been found. SR judged to be met.	The fire scenario walkdown information has been appended to the Fire Scenario Report (Appendix F) for documentation purposes. CN-1209 series drawings identify the fire areas and boundaries. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
HRA-A2-01	For each fire scenario, identify any new fire-specific safe shutdown actions called out in the plant fire response procedures in a manner consistent with the scope of selected equipment from the ES and PRM elements of the RA-S-2009 standard and in accordance with HLR-HR-E and its SRs in Part 2.	N/A	This requirement states that HRAs are identified in a manner similar to HLR-HR-E from part 2 of the standard with emphasis on fire scenarios. SR HR-E1 discusses a systematic review of the applicable procedures for operator actions of interest. However, the Fire Modeling documentation does not discuss the review of Plant Fire procedure or other applicable procedures to identify fire specific actions. If this review was performed, then some evidence of the actions considered should be provided. The SR was judged to be not met.	The goal is to have a post transition plant with as few fire specific actions as possible. Consequently, no fire specific actions are added to the Fire PRA model. Any important actions identified as necessary to reduce risk can be added to the procedures and model at a later time. No operator actions have been identified at this time; the requirement is N/A at this time. Since this F&O is N/A, there is negligible impact to the ILRT extension.
HRA-A4-01	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures relevant to actions identified in SRs HRA-A1, HRA-A2, and HRA-A3 is consistent with plant operational and training practices.	Addressed with no impacts to the NFPA 805 application / CC-I	Information on operator walk-throughs or talk-throughs for the impact of fires on the operator actions is not present in CNC-1535.00-00-0111. There is information in the HRA calculator sheets for the new operator actions developed but it has no information concerning when these actions where discussed or with whom. This information should be maintained as backup information or included in the applicable document. Also, if the talk-throughs have not been updated since the IPE or IPEEE days, the procedural changes may require updating for the FPRA. SR considered met at CC-I.	The Fire PRA uses a set of multipliers as described in the model development report to account for fire impacts on human reliability. This process is intended to implicitly account for (in a conservative manner) factors influencing operator performance such as fire effects on instrumentation, operator stress, and possible impact on timing. This conservative approach is judged to be consistent with a CC-I approach as indicated in SR HRA-C1 of the standard. With the HRA at CC-I, the Fire PRA results possess a conservative bias from this aspect of the analysis.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
				With overall risk metric results of the Fire PRA acceptable, the conservatism does not impede the use of the Fire PRA for the transition to NFPA 805. No actions have been taken to bring this HRA element to CC-II. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
HRA-83-01	COMPLETE the definitions of HFEs identified previously in HRA-81 and HRA-82 of this Standard and, within the context in the fire scenarios in the Fire PRA, specify the following: accident sequence specific timing of cues, time window for completion and procedure guidance. Also specify the availability of indications for detection and high-level tasks needed to achieve the goal of the response.	Addressed with no impacts to the NFPA 805 application / CC-I	The methodology for HRA adjustments does not explicitly address instrumentation, timing and procedural impacts other than simple vs. complex actions, which per HRA-81-01 were noted as not defined in the documentation. The SR was judged to be not met.	The HEP multiplier process is intended to implicitly address timing, procedure use, and instrument availability (when considered along with the instrument review documented in the component selection calculation). No changes have been made to bring the HRA to CC-II. Any effects on the ILRT extension are small and would not have a significant impact on results for the ILRT extension application.
HRA-C1-02	For each selected fire scenario, quantify the HEPs for all HFEs, accident sequences that survive initial quantification and account for relevant fire-related effects using conservative estimates, in accordance with the SRs for HLR-HR-G in Part 2 set forth under CC-I.	Closed/ CC-II	A finding from the FPIE evaluation stated that HEPs are not converted from medians to means. This issue was said to be addressed with a sensitivity case. However, this issue should be addressed in the Fire PRA. SR considered met at CC-I.	The HEP values have been converted from median to mean in the Fire PRA model. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
HRA-D2-01	[HR-H2] CREDIT operator recovery actions only if a procedure is available and	Closed / Met	The one recovery action developed for the Fire PRA (TSSPZRLRHE) is not proceduralized nor is it trained on. There is no discussion of why this action can be	The operator action referenced in the finding is not a "fire recovery" in the context of NFPA 805. This is an action added to the Fire PRA model to

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
	operator training has included the action as part of crew's training, or justification for the omission for one or both is provided.		credited, which is contrary to the requirements of HR-H2 so this SR is Not Met.	address a specific accident sequence (not fire specific) that was not yet included in the internal events model. Section 5.1 of the Model Development Report has been updated to better describe the basis for crediting this operator action. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
PRM-B2-01	Verify the peer review exceptions and deficiencies for the Internal Events PRA are dispositioned, and the disposition does not adversely affect the development of the Fire PRA plant response model.	Addressed with no significant impacts to the NFPA 805 application / Met	Section 4 of CNC-1535.00-00-0111 addresses PRA model quality for fire PRA use. Two potentially significant items not addressed are HRA pre-initiators (HR-A3 and HR-06) and failure probability data (DA-81) from DPC-1535.00-00-0013 revision 2. Section 4 of the FPRA Model Development should address these two items. This SR was judged to be not met.	The CNS internal events peer review was conducted under NEI 00-02. There is no internal event PRA finding which corresponds to element HR-06 in the ASME/ANS PRA Standard; however, the HEPs have been quantified using mean values in the Fire PRA. There were no internal events peer review findings against HR-A3. No changes have been made to the Fire PRA. Compared to post-initiator HEPs and fire induced failures, latent human error probabilities, equipment random failure rates and maintenance unavailability, calibration HEPs and misalignment of multiple trains of equipment are not expected to contribute significantly to overall equipment unavailability. Thus there is no material impact on the Fire PRA and no changes to the pre-initiator human error modeling have been made for the Fire PRA. The internal events peer review identified a finding against DA-81 (F&O DA-01) which noted that the data development workplace procedure did not identify component boundaries. The finding went on to note that component boundaries are apparent from the data. The change to the workplace procedure does not impact the Fire PRA quantification and no examples where the data was found to be incorrect were identified. Modest changes to the random failure rates have little impact on the results as fire-induced failures are far more significant in the Fire PRA results. Since this F&O has been met, there is negligible impact to the ILRT extension.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-B5-01	For those fire-induced initiating events included in the Internal Events PRA model, REVIEW the corresponding accident sequence models and IDENTIFY any existing accident sequences that will require modification based on unique aspects of the plant fire response procedures in accordance with HLR-AS-A and HLRAS-B of the ASME PRA Standard and their support requirements and IDENTIFY any new accident sequences that might result from a fire event that were not included in the Internal Events PRA in accordance with HLRAS-A and HLR-AS-B of the ASME PRA Standard and their supporting requirements.	Closed / Met	Reactor trip was used for fire initiating events in the model, although feedwater is failed due to lack of routing information. The plant response model is not the same for the plant trip and loss of feedwater initiating events, for example the probability of lifting a PORV or SRV is 1E-2 for loss of feedwater and 1E-3 for plant trip. The SR was judged to be met.	The Fire PRA model was modified to add gate IEFIRES which enables the fire initiating events to inherit the plant response logic for any transient event. The transient logic in the IEPRA and consequently the Fire PRA includes transfers to all of the necessary support systems logic. Updated Section 6.3 of the Fire PRA Model Development report. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
PRM-B6-01	MODEL accident sequences for any new initiating events identified per PRM-82 and any accident sequences identified per PRM-84 reflective of the possible plant responses to the fire-induced initiating events in accordance with HLR-AS-A and HR-AS-B and their SRs in the ASME PRA Standard and DEVELOP a defined basis to support the claim	Closed / Met	CNS added several new accident sequences to address some fire-specific issues that were not part of the base PRA. The model was reviewed and generally found to follow the process from the internal events PRA. The one issue was identified in that one of the new sequences included a new operator action, TSSPZRLRHE, but did not provide a basis for the assumed timing. In the HRA quantification section, CNS indicated that this was an ex-control room action with more than an hour available to perform the action. However, CNS did not provide the basis for saying that more than an hour was available.	Section 5.1 of the Fire PRA Model Development report has been updated to provide additional basis for the action and the assumed HEP value. Note that this HEP is not an NFPA 805 fire-specific recovery event. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
	of nonapplicability of any of these requirements in the ASME PRA Standard.			
PRM-B7-01	IDENTIFY any cases where new or modified success criteria will be needed to support the FPRA consistent with the HLR-SC-A and HLR-SC-8 of the ASME PRA Standard and their SRs.	Closed / Met	The self-assessment indicated that success criteria issues were considered in the Model Development Report. However, no evidence could be found that success criteria had been discussed in the Model Development report. The SR was judged to be not met.	A discussion addressing success criteria has been added to the Fire PRA Model Development report (section 3.4). Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
PRM-B11 (no F&O)	MODEL all operator actions and operator influences in accordance with the HRA element of this standard.	Closed / Met	This SR is judged to be not met because of a number of issues associated with the identification and incorporation of fire related HFEs. See HRA F&Os.	Refer to disposition for cross-referenced F&Os PRM-86-01, HRA-A4-01, & HRA-81-01. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
SF-A3-01	ASSESS the potential for common-cause failure of multiple fire suppression systems due to the seismically-induced failure of supporting systems such as fire pumps, fire water storage tanks, yard mains, gaseous suppression storage tanks, or building stand-pipes.	Closed / Met	The seismic/fire interaction evaluation is discussed in Section 3.13 of CNC-1535.00-00-0112. In general, CNS relies upon the assessments performed for the IPEEE analyses, in particular, the walkdowns. The IPEEE walkdown is documented in CNC-1435.00-00-0007 and the overall IPEEE is documented in the IPEEE Submittal Report. There is no indication in the documents provided that both seismic-induced fire as well as seismic-induced failure of fire mitigation systems has been considered. The SR was judged to be not met.	Section 3.13 of the Fire PRA Summary report has been updated to indicate that both seismic-induced fire and seismic-induced failure of fire mitigation systems were considered in the seismic/fire interaction evaluation. No impact on quantification. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.
SF-A5-01	Review (a} plant fire brigade training procedures and assess the extent to which training has prepared firefighting personnel to respond to potential fire alarms and fires in the wake of an earthquake and (b) the storage and placement of firefighting support equipment and fire	Closed / Met	This SR basically requires that CNS qualitatively assess their existing fire brigade training procedures to determine if the training has prepared the brigade to respond to fire alarms after an earthquake, to review their staging of fire mitigation equipment and to assess whether or not the occurrence of a seismically induced fire and any associated damage might compromise either of these elements. The CNS seismic/fire interaction evaluation is discussed in Section 3.13 of CNC-1535.00-00-0112. In general, CNS relies upon the assessments performed for the IPEEE analyses, in	Section 3.13 of the Fire PRA Summary Report has been updated to include an evaluation of seismically induced fire and the potential impacts on brigade response and equipment staging. No impact on quantification. Since this F&O has been closed and met, there is negligible impact to the ILRT extension.

Table A-4 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
	brigade access routes, and (c) assess the potential that an earthquake might compromise one or more of these features.		particular, the walkdowns. The IPEEE walkdown is documented in CNC-1435.00-00-0007 and the overall IPEEE is documented in the IPEEE Submittal Report. A review of these documents does not show any evaluation of a seismically induced fire and the potential impacts on brigade response and equipment staging.	

Table A-5 Fire PRA – Category I Summary

SR	Topic	Status
PP-B3	Spatial separation not relied upon for compartment assignments. This SR, PP-B3, is judged to be met at CC-I since no spatial separation is credited and CC-II/III requires crediting of spatial separation as credited in the regulatory fire protection program. Basis for Significance: As performed, adequate compartmentalization is used for the Catawba fire PRA. Possible Resolution: For CC-II/III, refinement of the compartmentalization to smaller compartment is required. (F&O PRM-B3-01)	As indicated by the reviewer, adequate compartmentalization is used for the CNS Fire PRA. Not crediting spatial separation as a partitioning feature is conservative; therefore, CC-I for this SR is acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
PP-B5	No active fire barrier elements are credited for Catawba Fire PRA compartmentalization. The credited passive fire barriers correspond to barriers credited in the regulatory fire protection program. Basis for Significance: The Catawba fire PRA credits only fire rated passive barriers. In order to meet Capability Category II/III, crediting of active fire barrier elements in fire compartment boundaries, with appropriate justification, is necessary. Possible Resolution: If CC-II/III is desired, the PRA compartmentalization must be revised with credits for active fire barriers, with appropriate justification. (F&O PRM-B5-01)	The Catawba Fire PRA analysis did not credit active barriers for partitioning, which is a conservative treatment. CC-I for this SR is acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
CS-B1	CNS performed a review of their existing electrical over-current coordination and protection analysis. As a result of this review, CNS identified a number of deficiencies in terms of scope and level of detail. CNS is currently in the process of completely redoing their electrical over-current coordination and protection analysis. The new analysis will increase the level of detail and to increase the scope to include all Appendix R equipment, the PRA equipment and the NPO equipment. As part of this re-analysis, CNS is making plant modifications as needed. However, at this time, this analysis is not complete. Basis for Significance: A review of the existing electrical over-current coordination and protection analysis is required to meet the SR even at the CC-I level. Possible Resolution: To move from CC-I up to CC-II/III complete the ongoing update of the electrical over-current coordination and protection analysis and formally issue the report. (F&O CS-B1-01)	The update of the breaker coordination and protection analysis was completed subsequent to the peer review and has since been incorporated into the Fire PRA. In some cases, the Fire PRA results were updated to reflect a larger cable footprint for affected power sources by including cables for uncoordinated loads (via related pseudo components). In other cases, the Fire PRA credited modifications to minimize or eliminate the coordination issues. This SR is now considered met at CC-II/III. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.
FSS-B2	Section 3.1.3 of CNC-1535.00-00-010 and Appendix E of that document identifying fire-driven parameters necessitating abandonment discuss the conditions that are assumed for fire scenarios W1 and W2 addressed in the document. A bounding type analysis for the control room was performed. To achieve Capability Category II requires a realistic characterization. The scenario analyzed are bounding in nature but could be tweaked for more realism and analysis with additional detail in order to achieve a Capability Category II rating. Basis for Significance: Analysis presented satisfies Capability Category I requirements. Possible Resolution: If Capability Category II is desired, perform	The MCR abandonment evaluation employed acceptable fire modeling methods and the calculated CCDP is based on the proceduralized success path provided by the SSF. CC-I for this SR is bounding; therefore CC-I is considered acceptable for the NFPA 805 application. However, the contribution of MCR abandonment is not inordinate to the overall fire CDF/LERF. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.

Table A-5 Fire PRA – Category I Summary

SR	Topic	Status
	additional control room analysis with more realistically modeled scenarios, crediting panel design and other specific features of the Catawba control room design. (F&O FSS-B2-01)	
FSS-C1	A two-point treatment was used for isolated selected scenarios such as low energy panels but not for "each selected" scenario. Basis for Significance: Analysis performed addresses Capability Category I requirements and more but not to the extent to qualify for a Capability Category II rating. Possible Resolution: If Capability Category II rating is desired, a preponderance of evaluated scenarios should be evaluated using two-point methodology. (F&O FSS-C1-01)	The Fire PRA analysis was updated to increase the number of scenario refinements using a 2-point treatment. Analysis has since been updated; SR now considered met at CC-II. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.
FSS-C2	Peak heat release rates reflected in NUREG 6850 were utilized. Time-dependent growth heat release rate curves were not discussed. Basis for Significance: Analysis performed meets industry practice. Possible Resolution: If Capability Category II rating is desired, then additional analysis utilizing time-dependent heat release rate information is required. (F&O FSS-C2-01)	Time-dependent HRR profiles have since been incorporated into numerous high risk scenarios. Analysis has since been updated; SR now considered met at CC-II. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.
FSS-C3	Burn out was considered in analysis for hot gas layer impact but did not seem to be based on fuel exhaustion but rather taking the room condition to total involvement. Additional discussion and detail addressing fuel exhaustion is required for improved rating. Basis for Significance: Analysis performed appears to satisfy requirement but does not address detail for higher than Capability Category I rating. Possible Resolution: If Capability Category II/III is desired, additional analysis considering the impact of fuel exhaustion in each compartment is required. (F&O FSS-C3-01)	The treatment for the hot gas layer is a conservative screening evaluation; therefore CC-I is considered acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
FSS-F2	Structural collapse is not deemed likely or addressed further. This meets Capability Category I which does not have any requirements identified. The discussion of structural collapse is qualitative in nature which does not meet the requirements for Capability Category II/III structural collapse analyses. Basis for Significance: Capability Category I has no requirements identified, so that SR CC-I is met. Capability Category II/III required more in-depth scenario development, identifying the criteria for structural collapse. Possible Resolution: If Capability Category II/III is desired, then more detailed structural analysis is required to be incorporated into the model. However, this may not always be cost effective. (F&O FSS-F2-01)	The Fire PRA locations were reviewed and determined to not meet the definition in FSS-F1. Therefore, this SR is N/A. Therefore, this F&O does not adversely affect the ILRT extension.
FSS-F3	No quantitative discussion is provided. A qualitative discussion of structural collapse is provided in Section 3.2 of CNC-1535.00-00-011.	The Fire PRA locations were reviewed and determined to not meet the definition in FSS-F1. Therefore, this SR is N/A.

Table A-5 Fire PRA – Category I Summary

SR	Topic	Status
	Basis for Significance: Qualitative discussion meets criterion for Capability Category I. Capability Category II/III requires specific risk determination for structural collapse. Possible Resolution: If Capability Category II/III grading is required, then update of the Catawba Fire PRA is required that specifically determines fire risk resulting in structural collapse as per the SR. (F&O FSS-F3-01)	Therefore, this F&O does not adversely affect the ILRT extension.
FSS-G4	Plans indicate that some three-hour boundaries are constructed with two-hour block with grout filled cells. No justification for this arrangement and its adequacy was provided. This is also a plant partitioning issue. Basis for Significance: Used three-hour fire rated fire area boundaries and allowed for barrier failure in screening analysis, Attachment 4 of the Fire Scenario Report [CNC-1535.00-00-0110]. Possible resolution: To achieve Capability Category II, provide original plant construction documents and/or industry test information and building code acceptance information to justify the validity of two-hour block with grout filled cells being equivalent to a three-hour barrier. (F&O FSS-G4-01)	While conditions within the plant may still be impacted by the fire event, the major actions associated with fire mitigation are assumed to be complete within a 1 to 2 hour time frame. From NUREG/CR-6850, fire barriers with a minimum fire protection endurance rating of one hour can be credited to prevent the spread of fire. Therefore, the difference between a 2 or 3 hour barrier rating is inconsequential to the Fire PRA. Therefore, this F&O is inconsequential the ILRT extension.
FSS-H2	Duke testing was not used. Hughes report was the default report for damage mechanisms resulting in zone of influence damage criteria. Basis for Significance: Used zone of influence scoping and documented in Generic Fire Modeling Treatments Report for project 1SPH.02902.030 and CNC-1535.00-00-0110. Thresholds for target damage were based on industry criteria for damage with zone of influence assessment for Catawba. Catawba specific damage criteria were not used. Possible Resolution: In order to meet Capability Category II/III classification, the use of Catawba plant-specific damage criteria is required. Determination of plant-specific damage criteria is required with a well document technical basis. Revise and update the Fire PRA as noted above. (F&O FSS-H2-01)	The damage criteria applied in the Generic Fire Modeling Treatments are taken from NUREG/CR-6850. No plant-specific data is available for use in lieu of NUREG/CR-6850. Since the plant-specific ignition sources are comparable to those in the Generic Fire Modeling Treatments, use of ZOI information based on the generic configurations is considered acceptable for the NFPA 805 application. Therefore, this F&O does not adversely affect the ILRT extension.
HRA-A3	The Equipment Selection Calculation CNC-1535.00-00-0108 revision 0, addresses spurious instrumentation under "Errors of Commission". This section states "No specific instruments were identified that would cause an undesired operator action without first taking one or more confirmatory actions". The results of the assessment are provided, but no details are provided on who performed the review, what method was used, and what procedures were reviewed. Basis for Significance: There is not sufficient documentation to determine the SR is met. Possible Resolution: Add documentation describing what procedures were reviewed, what method was applied during the review, and what the qualification of the individual performing the review was. (F&O ES-C2-01)	Analysis has since been updated; SR now considered met at CC-II. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.

Table A-5 Fire PRA – Category I Summary

SR	Topic	Status
HRA-A4	Information on operator walk-throughs or talk-throughs for the impact of fires on the operator actions is not presented in CNC-1535.00-00-0111. There is information in the HRA Calculator sheets for the new operator actions developed, but it has no information concerning when these actions were discussed or with whom. This information should be maintained as backup information or included in the applicable document. Also, if the talk-throughs have not been updated since the IPE or IPEEE days, the procedural changes may require updating for the FPRA. Basis for Significance: A review of procedural impacts for the fire is required to determine correct impacts on the HEPs due to events such as fire. Talk-throughs will also help verify that any additional actions are not missed. Possible Resolution: If talk-throughs were performed for this FPRA, the information should be maintained as backup information or included in the applicable document. If the talk-throughs have not been performed or adequately documented since the IPEEE, then the talk-throughs should be performed and documented in a manner that will help future updates. (F&O HRA-A4-01)	The Fire PRA uses a set of multipliers as described in the model development report to account for fire impacts on human reliability. This process is intended to implicitly account for (in a conservative manner) factors influencing operator performance such as fire effects on instrumentation, operator stress, and possible impact on timing. This conservative approach is judged to be consistent with a CC-I approach as indicated in SR HRA-C1 of the standard. With the HRA at CC-I, the Fire PRA results possess a conservative bias from this aspect of the analysis. With overall risk metric results of the Fire PRA acceptable, the conservatism does not impede the use of the Fire PRA for the transition to NFPA 805. CC-I is considered acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
HRA-B4	HRA events are reviewed for instrumentation in Attachment B of CNC-1535.00-00-0108, Rev. 0. The documentation for HRA events that do not have instrumentation in the internal events model is not clear. Instrumentation is described in general terms without any information on the number of trains or the number of instruments available. There is not enough documentation to justify the diverse and redundant argument. Basis for Significance: Based on the available documentation, reviewers were unable to determine if the instrumentation supporting credited HRA events was diverse and redundant enough to credit associated events. Possible Resolution: Provided additional details on the number, type, and trains of instrumentation being credited. (F&O ES-C1-01)	Analysis has since been updated; SR now considered met at CC-II. The model used for the ILRT extension application included this update; therefore, there is no impact to the ILRT extension.
HRA-C1	A finding from the FPIE evaluation stated that HEPs are not converted from medians to means. This issue was said to be addressed with a sensitivity case. However, this issue should be addressed in the Fire PRA. Basis for Significance: This finding will have a minor impact on post-accident HEP, but will cause a 2-3 times increase in pre-accident HEPs. Possible Resolution: Ensure that the HEPs are completely based on means rather than medians. (F&O HRA-C1-02)	The Fire PRA uses a set of multipliers as described in the model development report to account for fire impacts on human reliability. This process is intended to implicitly account for (in a conservative manner) factors influencing operator performance such as fire effects on instrumentation, operator stress, and possible impact on timing. This conservative approach is judged to be consistent with a CC-I approach as indicated in SR HRA-C1 of the standard. With the HRA at CC-I, the Fire PRA results possess a conservative bias from this aspect of the analysis. With overall risk metric results of the Fire PRA acceptable, the conservatism does not impede the use of the Fire PRA for the

Table A-5 Fire PRA – Category I Summary

SR	Topic	Status
		transition to NFPA 805. CC-I is considered acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.
HRA-D1	CNS added several new accident sequences to address some fire-specific issues that were not part of the base PRA. The model was reviewed and generally found to follow the process from the internal events PRA. One issue was identified: One of the new sequences included a new operator action, TSSPZRLRHE, but the documentation did not provide a basis for the assumed timing. In the HRA quantification section, CNS indicated that this was an ex-control room action with more an hour was available to perform the action. However, CNS did not provide the basis for saying that more than an hour was available. Basis for Significance: This important information needs to be documented in relation to inclusion of a new operator action in the PRA. Possible Resolution: CNS needs to explicitly define the basis for stating that more than an hour is available to perform an ex-control room fire-specific action. Also, CNS should review all ex-control room actions to confirm that they have reasonable bases for the assumed time available. (F&O PRM-86-01)	The Fire PRA uses a set of multipliers as described in the model development report to account for fire impacts on human reliability. This process is intended to implicitly account for (in a conservative manner) factors influencing operator performance such as fire effects on instrumentation, operator stress, and possible impact on timing. This conservative approach is judged to be consistent with a CC-I approach as indicated in SR HRA-C1 of the standard. With the HRA at CC-I, the Fire PRA results possess a conservative bias from this aspect of the analysis. With overall risk metric results of the Fire P RA acceptable, the conservatism does not impede the use of the Fire PRA for the transition to NFPA 805. CC-I is considered acceptable for the NFPA 805 application. The only potential effects on the ILRT extension are conservative; therefore, this F&O does not adversely affect the ILRT extension.

Table A-6 – High Wind PRA F&Os and Resolutions

F&O #	Review Element	Level	Issue
WPR-A1-02	WPR-A1	Finding	SR CC I/II/III Ensure that wind-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the wind PRA system model using a systematic process. **Discussion:** Assumption 5 in Section 6.1 of CNC-1535.00-00-0154 does not reflect the as-built as-operated plant and impacts the cutsets in the model. **Basis for Significance:** Assumption 5 in Section 6.1 of the CNC-1535.00-00-0154 states: "A high wind initiating event is assumed to have the operators tripping the reactor if there is also high wind failure of a SSC modeled in the fault tree." This assumption is not correct based on the directions provided in procedure RP/0/A/5000/007. Refer to WPR-A1-01 for discussion on consideration of RP/0/A/5000/007 procedures. This does not reflect the as-built as-operated plant and impacts the cutsets in the model. **Possible Resolution:** Proposed solution: Delete this assumption and associated model logic. **Actual Resolution:** The assumption has been deleted. The model logic has been revised as discussed in the resolution to WPR A1-01 to induce a reactor trip for each applicable High Wind initiating event per the procedure. **ILRT Extension:** Since this Finding has been resolved, there is not effect on the ILRT extension.
WPR-A1-03	WPR-A1	Finding	SR CC I/II/III Ensure that wind-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the wind PRA system model using a systematic process. **Discussion:** No discussion was provided for screening out failure modes that could result in the loss of ultimate heat sink due to a high wind event other than those related to the class 1 structures housing the service water system. **Basis for Significance:** No discussion was provided for screening out failure modes that could result in the loss of ultimate heat sink due to a high wind event other than those related to the class 1 structures housing the service water system. This initiator is important since it can affect multiple units. For example, no discussion was provided to evaluate the consequences of wind borne debris being deposited in the lake supplying safety related cooling water and choking off the intake. The basis for significance is that a potential major initiator that can affect both units is not evaluated. **Possible Resolution:** Evaluate the potential for losing ultimate heat sink due to debris blocking the intake. **Actual Resolution:** The potential for losing the ultimate heat sink due to debris blocking the intake has been evaluated and judged to be insignificant in the CNS HWPRA. The CNS unit's intake suction through piping located near the bottom of the lake. It is considered unlikely that sufficient debris would be in the lake and that the debris would sink to the low level intake and plug the intake. Documentation of this judgment has been added to section 7.3.2.16 of CNC-1535.00-00-0154 Revision 1.

Table A-6 – High Wind PRA F&Os and Resolutions

F&O #	Review Element	Level	Issue
			ILRT Extension: Since this Finding has been resolved, there is not effect on the ILRT extension.
WPR-A4-01	WPR-A4	Finding	SR CC I/II/III In each of the following aspects of the high wind PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. DEVELOP a defined basis to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are: (a) initiating-event analysis, (b) accident-sequence analysis, (c) success-criteria analysis, (d) systems analysis, (e) data analysis, (f) human-reliability analysis, (g) use of expert judgment. When Part 2 requirements are used FOLLOW the capability category designations in Part 2, and for consistency use the same Capability Category in this analysis. **Discussion:** There was no documented evidence in *CNC-1535.00-00-0154,* the CNS HWPRA report, to show that the high wind PRA systems-analysis work SATISFIES the corresponding requirements in Part 2 (of the PRA Standard). A defined basis to support non-applicability of any exceptions was not included. Peer Review Team did not have the time to perform a detailed review of the assessment of Part 2 i.e., P2A (calc DPC-1535.00-00-0013, PRA Quality Self-Assessment, received during the review). It is not within the scope that the Peer Review Team scope to perform the assessment of Part 2 SRs as part of this Peer Review. Without being provided with a compliance review of Part 2 SRs, the Peer Review also cannot judge that the technical elements as specified in the applicable Part 2 SRs are satisfied or not. So this F&O asks for more evidence and systematic assessment of the applicable SRs in Part 2 to meet this SR WPR-A4 and document it accordingly. **Basis for Significance:** No evidence of satisfying the requirements of Part 2, or basis for exceptions to the requirements, was provided or cross referenced in *CNC-1535.00-00-0154,* the CNS HWPRA report. **Possible Resolution:** Document that the requirements of Part 2 are satisfied. Whenever an exception is taken, the PRA team needs to be cognizant of the underlying rationale for the specific Part 2 requirement so as to ensure that this rationale is considered when the exception is taken.

Table A-6 – High Wind PRA F&Os and Resolutions

F&O #	Review Element	Level	Issue
			Actual Resolution: Documentation of compliance to the part 2 SRs is provided in appendix H of CNC-1535.00-00-0154. The CNS model is undergoing update to R.G. 1.200 Rev. 2 and all deficiencies noted in the assessment will be corrected. None of the deficiencies has an impact on the HWPRA results. **ILRT Extension:** Since this Finding has been resolved, there is not effect on the ILRT extension.
WPR-A4-02	WPR-A4	Finding	SR CC I/II/III In each of the following aspects of the high wind PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. DEVELOP a defined basis to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are: (a) initiating-event analysis, (b) accident-sequence analysis, (c) success-criteria analysis, (d) systems analysis, (e) data analysis, (f) human-reliability analysis, (g) use of expert judgment. When Part 2 requirements are used FOLLOW the capability category designations in Part 2, and for consistency use the same Capability Category in this analysis. **Discussion:** Peer Review Team disagrees with some of the assessment results as stated in the P2A (DPC-1535.00-00-0013, PRA Quality Self-Assessment) report. **Basis for Significance:** For example, SY-B7 is a CCI because the base system modeling used conservative success criteria verses realistic as required to meet CCII - the peer review team would need to review the high wind analysis in detail to understand if the risk importance of low speed straight winds is justified. Also, given its significant importance the impact of the siding on the AC system should be documented in a system notebook to address compliance with SY-B9. We see no evidence that the AC notebook was modified or that a new notebook addressing structures was developed in compliance with SY-B9; e.g. siding integrity is essential for maintaining the integrity of the AC system during "low-speed" straight winds - these issues need to be documented in accordance with the SY requirements described in SY-C2. **Possible Resolution:** Review the P2A assessment in detail, correct any errors and enhance the documentation. If a materially important mistake is discovered, its impact shall be analyzed and appropriate action taken.

Table A-6 – High Wind PRA F&Os and Resolutions

F&O #	Review Element	Level	Issue
			Actual Resolution: Documentation of compliance to the part 2 SRs is provided in appendix H of CNC-1535.00-00-0154. The CNS model is undergoing update to R.G. 1.200 Rev. 2 and all deficiencies noted in the assessment will be corrected. None of the deficiencies has an impact on the HWPRA results. **ILRT Extension:** Since none of the deficiencies have an impact on the HWPRA results, there is not effect on the ILRT extension.
WPR-C3-01	WPR-C3 (also affects WPR-A10)	Finding	SR CC I/II/III DOCUMENT the sources of model uncertainty and related assumptions associated with the high wind plant response model development. **Discussion:** Several assumptions in CNC-1535.00-00-0154, Section 6.1 need clarification: Assumption 1 states that one tornado missile hit to a PRA SSC results in functional failure except for the main transformers in the Yard, which require two missile hits. No basis is provided for this assumption. Was fragility of the component considered or was any missile at any speed assumed to result in a functional failure? Why are two missiles needed for a functional failure of the main transformers and how were the two missiles modeled? Assumption 4 states: F2 and greater peak gusty winds at CNS will automatically induce a LOOP. Explain basis why an F2 or greater peak gust wind automatically induces LOOP verses >F1 or >F3, for example. Assumption 5 states: A high wind initiating event is assumed to have the operators tripping the reactor if there is also high wind failure of a SSC modeled in the fault tree. Procedure RP/0/A/5000/007 requires operators take both units to hot shutdown for winds 73mph or higher - without a concurrent high wind failure. As such this assumption does not reflect the way the operators will respond. Assumption 6 states: Conservatism is introduced when initiating event %T3, LOOP, and the High Wind-Induced LOOP events are OR'd under the same parent gate. Some High Wind- Induced LOOP events may be double counted due to inclusion in the %T3 model frequency. A characterization (e.g. sensitivity analysis) of the impact of this assumption on the model results is needed. Assumption 7 states: Some components were modeled by high wind analysis. These components had no representation in the fault tree. Only one of the four MSSVs and one of the four MSIVs are modeled due to the Internal Event model assumption of a SGTR occurs on SG "B". Thus high wind fragilities on the other three MSSV/MSIVs are not in the fault tree. The example did not clarify if this is a conservative assumption – please explain why this assumption is conservative and if a sensitivity analysis is needed. Assumption 8 states: System YD is the Drinking Water System is assumed failed for all high wind events. A basis is needed for this assumption including the impact on model results.

Table A-6 – High Wind PRA F&Os and Resolutions

F&O #	Review Element	Level	Issue
			Assumption 11 states: This analysis is for Unit 1 with shared Unit 2 SSCs. Applicability of Unit 2 with shared Unit 1 SSCs is assumed for this analysis. Explain why was Unit 1 selected and why a Unit 2 model is not needed. Is the internal events CDF and LERF for Unit 1 significantly different from Unit 2? Assumption 1 in CNC-1535.00-00-0154 Appendix A Section B.1. CREDITING RECOVERY OF SEAL INJECTION AFTER FIRST HOUR states: Straight Line or Tornado Wind conditions will not prevent access to the SSF after one hour has elapsed from the Wind-Induced LOOP Initiating Event. What Is the basis for this assumption? It is previously stated that reaching the SSF requires travelling 100 feet outside. Debris and structural integrity issues may preclude using this path, this would not only lead to path and door blockage but personnel safety. Access may require obtaining debris removal equipment and performing structural reviews which may not be possible within one hour. In addition, the Calculation Section Addressed reference for SR WPR-C3 in Table 4-1 should be Section 6.0, not Section 7.3.3. **Basis for Significance:** Basis for key assumptions must be clear to facilitate review, applications and future updates. **Possible Resolution:** Improve the quality of the assumptions in the High Wind PRA report. **Actual Resolution:** Each of the assumptions has been reviewed and several have been clarified and/or enhanced. Assumptions 1, 5, 6, and 7 have been deleted as they were determined to be inapplicable. Assumptions 4, 8, 11 and Assumption 1 in Appendix G have been revised and enhanced to provide a clearer basis for each. The only model change that resulted from this review is the assumption that a failure of a PRA SSC is required to induce the wind initiating event in the model. This assumption has been removed as a reactor trip was modeled in accordance with RP/0/A/5000/007. **ILRT Extension:** Since this Finding has been resolved, there is not effect on the ILRT extension.