Duke Energy®
Carolinas

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January 20, 2009

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

Subject: Duke Energy Carolinas, LLC (Duke)
Catawba Nuclear Station, Unit 2
Docket Number 50-414
Proposed Technical Specification (TS) Amendment
TS 3.3.1, Reactor Trip System (RTS) Instrumentation

Pursuant to 10 CFR 50.4, 10 CFR 50.90, and 10 CFR 50.91(a)(6), Duke proposes a one-time exigent limited duration extension of TS Surveillance Requirement (SR) 3.3.1.4 Frequency. SR 3.3.1.4 is a Trip Actuating Device Operational Test (TADOT) of the reactor trip breakers (RTBs) and reactor trip bypass breakers. The SR Frequency was recently revised by Catawba Amendments 247/240 to 62 days on a staggered test basis. Amendments 247/240 were issued by the NRC on December 30, 2008.

On January 8, 2009, during testing of the Unit 2 RTBs, personnel discovered a problem while attempting to test Train 2A RTB. Investigation to date indicates that the problem most likely lies in the contacts associated with Train 2A reactor trip bypass breaker cubicle cell switch. As a result of this issue, SR 3.3.1.4 cannot be performed for Train 2A reactor trip breaker. Both trains of the RTS are currently fully operable.

Duke is requesting, on a one-time basis, that the SR 3.3.1.4 Frequency for RTBs be extended until March 10, 2009 at 0500 hours. The planned date and time for Mode 3 is March 7, 2009 at 0500 hours. This represents a one-time extension of this SR Frequency by 20 days, as the SR is currently set to expire (including applicable grace period) on February 19, 2009. The details regarding the reason that the extension request is being made for both trains is explained fully in Attachment 1 to this letter. Duke is requesting that this TS SR Frequency extension be approved via a license condition to Facility Operating License (FOL) NPF-52. A similar precedent has been established for granting a temporary TS Completion Time extension at Catawba via a license condition. This extension will allow the repair of Train 2A reactor trip bypass breaker cubicle cell switch to be performed during the refueling outage and will prevent an unnecessary transient shutdown cycle of Unit 2.

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Unit 2 is currently at 100% power. Therefore, in order to avoid the unnecessary shutdown of Unit 2, Duke requests approval of this proposed license amendment on a one-time exigent basis by February 15, 2009.

Attachment 1 provides the technical information necessary to support this amendment request. Attachment 2 contains the existing FOL pages marked up to show the proposed change. Attachment 3 contains the retyped (clean) FOL pages. Attachment 4 contains a Catawba Probabilistic Risk Analysis (PRA) technical adequacy discussion. This request is considered a risk-informed license amendment request in accordance with NRC Regulatory Guides 1.174, 1.177, and 1.200.

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

Implementation of this amendment request will not require changes to the Catawba Updated Final Safety Analysis Report (UFSAR).

There are no regulatory commitments associated with this amendment request.

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

By copy of this letter, Duke is also notifying the NRC of an administrative error introduced during the issuance of Amendments 248/241 on January 9, 2009. Amendments 248/241 were followup amendments to Amendments 247/240. On page 4 of the FOL pages for the units, the amendment numbers were inadvertently reversed. Unit 1 (FOL NPF-35) should actually be Amendment 248 and Unit 2 (FOL NPF-52) should actually be Amendment 241. Duke is requesting that the NRC correct this error at the earliest opportunity.

Should you have any questions concerning this information, please call R. D. Hart at (803) 701-3622 or L. J. Rudy at (803) 701-3084.

Very truly yours,

James R. Morris

James R. Morris

LJR/s

Attachments

James R. Morris affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

James R. Morris, Vice President

Subscribed and sworn to me: 1/20/09
Date

Notary Public

My commission expires: 7/2/2014
Date

SEAL

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RGC File
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ATTACHMENT 1

SUMMARY DESCRIPTION
DETAILED DESCRIPTION
TECHNICAL JUSTIFICATION
REGULATORY EVALUTION
ENVIRONMENTAL ASSESSMENT

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.4, 10 CFR 50.90, and 10 CFR 50.91(a)(6), Duke proposes a one-time exigent limited duration extension of TS SR 3.3.1.4 Frequency. TS SR 3.3.1.4 is a TADOT of the RTBs and bypass breakers. The SR Frequency is 62 days on a staggered test basis. The requested extension would allow continued operation of Unit 2 for 20 days. The current SR is set to expire (including applicable grace period) on February 19, 2009.

On January 08, 2009, at approximately 0900 hours, while performing alignment of Train 2A reactor trip bypass breaker, unexpected conditions were experienced in preparation for Train 2A RTB/SSPS testing. Investigation to date indicates that the problem is likely related to contacts associated with the bypass breaker cubicle cell switch. It has been determined that maintenance on the cubicle cell switch cannot be conducted with Unit 2 at power. Catawba plans to perform this maintenance during the upcoming Unit 2 End of Cycle 16 Refueling Outage scheduled to begin on March 7, 2009. In order to avoid an unnecessary shutdown of Unit 2, Duke requests approval of this one-time exigent license amendment request by February 15, 2009.

1.1 Background

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and reactor coolant system pressure boundary during anticipated operational occurrences and to assist the engineered safety features systems in mitigating accidents.

Reactor trip switchgear, including the RTBs and reactor trip bypass breakers, provides the means to interrupt power to the control rod drive mechanisms and allows the control rods to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs while the unit is at power.

The RTB TS required Functions must be operable in Mode 1 or 2 when the reactor is critical. In Mode 3, 4, or 5, the RTB TS required Functions must be operable when the RTBs or associated bypass breakers are closed, and the control rod drive system is capable of rod withdrawal.

TS SR 3.3.1.4 is a TADOT of the RTBs and bypass breakers. This SR must be performed on each bypass breaker prior to placing the breaker in service. The SR Frequency is 62 days on a staggered test basis.

2.0 DETAILED DESCRIPTION

On January 8, 2009, at approximately 0900 hours, Unit 2 RTB/SSPS testing was being conducted. When the testing personnel attempted to rack in Train 2A reactor trip bypass breaker, they noted that the breaker was harder to rack in than in the past. They stopped work and notified supervision and the control room. The breaker was removed from the cubicle and was evaluated.

Due to the problem with the breaker, Unit 1 Train 1B reactor trip bypass breaker was moved to Train 2A reactor trip bypass breaker's cubicle. Unit 2 RTB and SSPS testing was again initiated. Train 2A reactor trip bypass breaker was racked in and closed. After closing the breaker, personnel obtained unexpected testing results. They again stopped and notified supervision and the control room.

After evaluation, Train 2A reactor trip bypass breaker was opened and racked out per the restoration portion of the applicable procedure.

This issue was documented in Catawba Problem Investigation Process (PIP) C-09-00135. Investigation to date indicates that the problem most likely lies in the contacts associated with the breaker cubicle cell switch. The cell switch in question provides Train 2A P-4 (reactor trip) signal to the safety functions of turbine trip and main feedwater isolation when Train 2A RTB is open and the bypass breaker is not racked in and closed. It also interfaces with non-safety related systems such as the condenser steam dump system and the main feedwater pump speed control system. The cell switch believed to be the problem does not affect the operation (i.e., open/close function) of any RTB itself. The SSPS General Warning feature is also supported by this cell switch. This feature trips the reactor if both trains of SSPS/RTBs are placed in the test/bypass mode simultaneously.

The postulated failure mechanism was evaluated for transportability to other Unit 2 trains/components and to Unit 1. The failure mechanism was determined to be limited to the identified cell switch. Other failure mechanisms ruled out by the transportability evaluation included human error, problems with the testing equipment, and problems with the breaker itself.

Despite the fact that the failure mechanism itself is limited to Train 2A reactor trip bypass breaker cubicle cell switch, regulatory relief is also being requested for the SR 3.3.1.4 Frequency as it applies to Train 2B RTB/SSPS testing. Testing of the RTB as described in the UFSAR requires the testing of the trip function of the opposite train's bypass breaker. Racking Train 2A reactor trip bypass breaker to the TEST position may disturb the suspect cell switch and generate a General Warning Alarm. Since the other train is also bypassed at this time, this would generate a reactor trip. Based on the last successful performance of SR 3.3.1.4 on December 4, 2008 (for Train 2B), the latest date for which this SR can next be performed (including applicable grace period) is February 19, 2009, which is prior to the start of the Unit 2 End of Cycle 16 Refueling Outage (March 7, 2009). Therefore, it is necessary to request relief from the SR 3.3.1.4 Frequency for both trains. Note that relief is being requested until March 10, 2009 at 0500 hours. Although Mode 3 of the refueling outage is currently targeted to be achieved on March 7, 2009 at 0500 hours, an extra 3 days has been incorporated into this request to account for any unexpected delays in achieving Mode 3.

Relevant Maintenance and Testing History Concerning RTBs and Bypass Breakers

Following are prior examples of RTB/bypass breaker cell switch problems at Catawba:

- On May 26, 1991, with Unit 1 in Mode 5, a main feedwater isolation occurred during racking out activities of the Train A reactor trip bypass breaker. Operations personnel had previously started the Manual Reactor Trip Functional Test, but stopped to pursue higher priority work. Later, after shift change, Operations personnel resumed the procedure to align the breakers for Engineered Safety Features (ESF) Train A testing. During this alignment, the reactor trip bypass breaker was racked from the connect to the disconnect position. During breaker transit, the Operator heard relays activating and the apparent closure of main feedwater isolation valves. The main feedwater isolation was immediately reset and the valves were realigned for ESF testing. No problems were found with the breaker.
- On September 23, 2004, the cell switch lever for RTB 2A was angled to the left rather than aligning at a right angle to the switch shaft. This had not caused any problem during operation, but it was misaligned enough that it needed to be corrected. The breaker contact plate must strike the cell switch lever in the proper position when the breaker is racked into the cubicle. A work order (WO) was written to correct this deficiency. When Maintenance personnel checked a cell switch kit out of the warehouse, the same problem was found with the new parts. Initial examination indicated the lever was defective and did not orient squarely on the switch shaft. Engineering personnel inspected the three other cell switch kits in the warehouse and none of the others had the alignment problem. One of the good kits was checked out to use for the repair and the defective one was held for followup action. It appears that the lever installed in the RTB 2A cubicle and the first one checked out of the warehouse had the same problem. The levers in the other Unit 2 RTB cubicles were aligned properly.
- On September 7, 2006, during RTB/SSPS testing, relay X4B failed to energize when Train 1B reactor trip bypass breaker was racked out. Per the applicable procedure, Maintenance personnel notified the control room that the runback circuit for both feedwater pumps to minimum speed was defeated for a reactor trip. Also, the main feedwater pump recirculation valve (1CF6 and 1CF13) automatic opening on a reactor trip would not have occurred due to the relay malfunction defeating the circuit. A review of the circuit concluded that the most

likely failed components were the cell switch (33) or the auxiliary relay (X4B). Other than conductors (wiring), a cell switch, and the X4B relay, there were no other components in the circuit. The cell switch is a cam operated mechanism in the breaker cubicle that operates during the racking in or out of the breaker. It provides a closed contact to the relay X4B to energize it when the breaker is racked out. To determine the likely cause, a voltage measurement was to be performed to determine if the cell switch had actually provided the relay a signal to energize. When Maintenance personnel returned to the switchgear on the following day to perform the voltage measurement, the relay was found already energized (i.e., in its normal configuration). A monitoring frequency was established to provide additional assurance that the circuit would perform.

- On November 16, 2006, a new cell switch to be installed in a RTB cubicle was found with cracked mounting threads. RTB cell switches are being replaced according to a "5R" frequency (every fifth refueling outage) switch replacement program. Each switch is mounted to a bracket in the breaker cubicle with two cap screws. The screws are inserted into threaded holes in the switch base. From a pre-installation inspection of one of the new switches, a Maintenance technician found a crack in the threads of one hole in the switch base. This switch was tagged not to be used and was turned over to Engineering personnel. No other new cell switches were found with this problem.

For the cell switch issue identified in this amendment request (Train 2A reactor trip bypass breaker cell switch), the planned method of repair is to remove and repair or replace the cell switch (either the complete assembly or the affected components).

As part of routine scheduled preventive maintenance, this cell switch was replaced in September 2007 during the Unit 2 End of Cycle 15 Refueling Outage and in March 2000 during the Unit 2 End of Cycle 10 Refueling Outage. Review of Train 2B and the equivalent Unit 1 trains identified the following preventive maintenance history (no corrective maintenance history was identified). The preventive maintenance replacement frequency is every fifth refueling outage (approximately every 7.5 years).

1A RTB Cell Switch	Model WO 00882624	1EOC16	WO 01125193
		1EOC11	WO 00945449
		1EOC05	WO 00883354
1B RTB Cell Switch	Model WO 00882625	1EOC16	WO 01125194
		1EOC11	WO 00945450
		1EOC05	WO 00883355
2A RTB Cell Switch	Model WO 00880400	2EOC15	WO 01726214
		2EOC10	WO 00960668
2B RTB Cell Switch	Model WO 00880402	2EOC15	WO 01726216
		2EOC10	WO 00960670

1A Bypass Cell Switch	Model WO 00882623	1EOC16	WO 01125192
1B Bypass Cell Switch	Model WO 00882626	1EOC16	WO 01125195
		1EOC11	WO 00945451
		1EOC05	WO 00883356
2B Bypass Cell Switch	Model WO 00880403	2EOC15	WO 01726217
		2EOC10	WO 00960671

Since all Catawba RTBs and bypass breakers are identical and interchangeable, the last three years of test results for all breakers was reviewed. A total of 80 tests of RTBs and bypass breakers were successfully completed and all breakers properly opened in response to all diverse trip signals.

Summary of RTB tests from January 1, 2006 through December 31, 2008:

Trip Breaker	Number of Tests	Number of Failures
1A	19	0
1B	20	0
2A	20	0
2B	21	0
Total	80	0

The system health associated with RTBs has been "green" since the second trimester of 2006. The system health was "yellow" in the first trimester of 2006 due to a problem with a RTB occurring during Unit 1 RTB/SSPS testing on March 23, 2006. Train 1B RTB spuriously opened; no cause could be determined. The breaker was replaced with a spare. The breaker was examined by the vendor and the cause could not be determined. The breaker was returned to the Unit 1 RTB B cubicle on July 13, 2006 and has passed all surveillance tests since being returned to service.

2.1 Intended Resolution of Proposed Amendment

10 CFR 50.91(a)(6) states that where the NRC finds that exigent circumstances exist, in that a licensee and the NRC must act quickly and that time does not permit the publishing of a Federal Register notice allowing 30 days for prior public comment, and it also determines that the amendment involves no significant hazards considerations, the NRC will either issue a Federal Register notice providing for a limited period of opportunity for public comment or will utilize alternate means of communication as necessary to allow for public comment. The NRC will also require the licensee to explain the exigency and why the licensee cannot avoid it.

2.2 Reason Exigent Situation Has Occurred

Previous performances of SR 3.3.1.4 have been successful until this situation unexpectedly occurred on January 8, 2009. Upon discovery of this issue, Duke contacted the NRC to verbally provide all relevant available information. Catawba management took appropriate action in support of a resolution to this issue. This license amendment request was developed and submitted after this situation occurred. However, due to the impending expiration of the SR 3.3.1.4 Frequency on February 19, 2009, insufficient time exists for processing this amendment request through normal channels. Sufficient time exists for processing this amendment request through exigent channels as delineated in 10 CFR 50.91(a)(6).

2.3 Description of Proposed Changes

Duke proposes the following license condition in Appendix B, Additional Conditions, for FOL NPF-52:

Amendment Number
TBD

Additional Condition
The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling Outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this license condition shall expire.

Implementation Date
February 19, 2009

3.0 TECHNICAL JUSTIFICATION

Duke has qualitatively and quantitatively evaluated the risk impact for extending TS SR 3.3.1.4 a total of 20 days for both Train 2A and Train 2B reactor trip bypass breakers from February 19, 2009 until March 10, 2009.

Qualitative Assessment

It was qualitatively determined that the overall risk impact on the requested SR Frequency extension was expected to be minimal since the Anticipated Transient Without Scram (ATWS) contribution to the Core Damage Frequency (CDF) is less than 1% (reference

Duke LAR submittal to NRC dated December 11, 2007 to relax completion times and surveillance test intervals for the RTS and ESFAS) and the contribution to the Large Early Release Frequency (LERF) is less than 4%.

Additionally, the current PRA model reflects failure probabilities that support a 31-day staggered test basis Frequency for SR 3.3.1.4. When the 62-day staggered test basis surveillance test interval is implemented, the failure probabilities for reactor trip breakers (including common cause failure) will be based on the revised failure probabilities from NUREG/CR-5500, Vol. 2 (INEEL/EXT-97-00740), "Reliability Study: Westinghouse Reactor Protection System, 1984-1995", December 1998 as utilized in WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times", March 2003, which are about one order of magnitude lower than the current failure probabilities.

The net effect of this change will be to further reduce the importance of these components and their impact on this application. From the above, it is qualitatively assessed that the individual reactor trip breakers are of very low risk significance and that a surveillance test interval extension of 20 days would not affect this qualitative assessment.

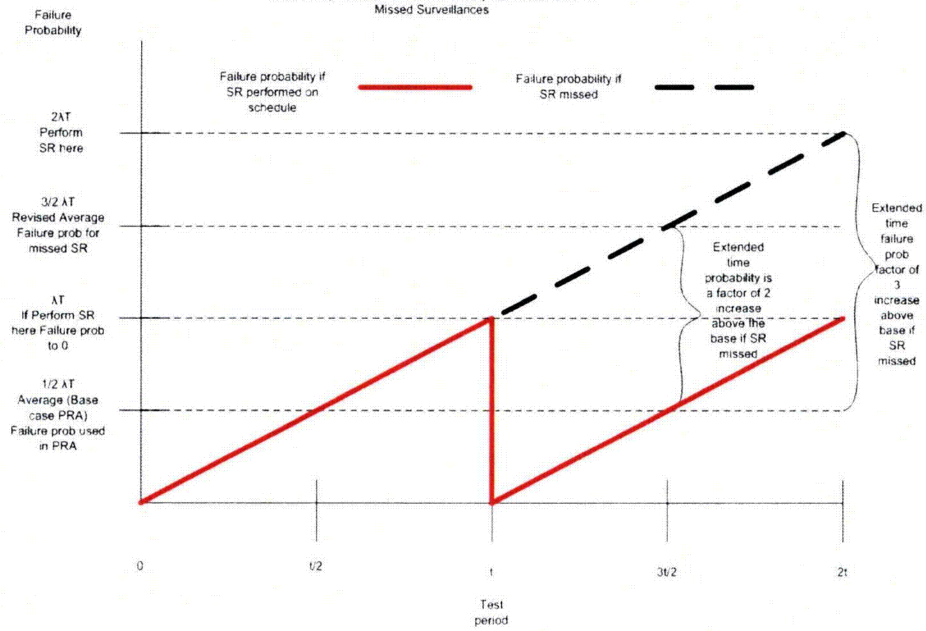
Quantitative Assessment

The subject SR verifies proper operation of the bypass breaker prior to it being placed in service. Upon failure of the cell switch, a turbine trip or a feedwater isolation signal would be generated. This initiates a reactor trip from full power. For the purposes of the PRA, extension of the surveillance test interval leads directly to an increased failure probability for the basic event, reactor trip breakers fail to open. The PRA directly models this condition.

Since the proposed extension is being requested for both Train 2A and Train 2B RTBs, the common cause event, QRPBKRSCOM, CCF of Reactor Trip Breakers To Open, is used for the analysis. The nominal base case probability of this occurrence is 1.6E-05. The analysis consists of using an elevated failure probability for this event corresponding to the requested extension duration, and calculating the increase in CDF and LERF and comparing it to the base case CDF and LERF to obtain the dCDF and the dLERF.

The choice of the revised failure probability is made using the methodology discussed in the Industry Implementation Guidance for TSTF-358, Revision 6, "Missed Surveillance Requirements", TSTF-IG-06-01. The resultant conclusion that if a surveillance test interval is doubled, then for the second half of the interval (i.e., the extended part) the average failure probability of the event in question is three times the average failure probability of the first half of the interval is used. This assumes the increase in failure probability is linear. This can be graphically seen below:

Figure 2
Relationship Between Failure Probability and Test Period for Missed Surveillances



Using the above to ratio the failure probability to the extension, a revised failure probability for QRPBKRSCOM for a 20-day extension is calculated as:

$$(20/62) \times (4.8E\text{-}05 - 1.6E\text{-}05) + 1.6E\text{-}05 = 2.6E\text{-}05$$

This revised value when substituted into the PRA model provided the following results.

CDF/rx-yr	CDFbase/rx-yr	dCDF/rx-yr
1.850E-05	1.844E-05	6.0E-08

LERF/rx-yr	LERFbase/rx-yr	dLERF/rx-yr
1.279E-06	1.247E-06	3.2E-08

The values obtained represent a negligible increase in risk and meet Regulatory Guide 1.174 guidelines.

A sensitivity study is performed by multiplying the original failure probability for QRPBKRSCOM by a factor of 3 to obtain a revised failure probability of 4.8E-05. This value represents a surveillance test interval extension of 62 days. When substituted into the PRA model the results are:

Sensitivity Case:

CDF/rx-yr	CDFbase/rx-yr	dCDF/rx-yr
1.862E-05	1.844E-05	1.8E-07

LERF/rx-yr	LERFbase/rx-yr	dLERF/rx-yr
1.348E-06	1.247E-06	1.0E-07

As before, the values obtained represent a very small increase in risk when compared to the guidelines in Regulatory Guide 1.174.

The sequences are all ATWS (initiated by loss of load or loss of feedwater) as expected.

Conclusion

The calculated dCDF and dLERF values represent negligible increases in risk when compared to the Regulatory Guide 1.174 guidelines. As noted earlier, these results are conservative since they do not use the lower failure probabilities that are supportive of the 62-day staggered test basis Frequency for SR 3.3.1.4. The qualitative and quantitative assessments are in agreement and support a one-time 20-day extension for SR 3.3.1.4 for both Train 2A and Train 2B RTBs.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Applicable regulatory requirements are contained in 10 CFR 50, Appendix A, General Design Criteria 20, 21, 22, and 23. These are stated below.

Criterion 20--Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21--Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22--Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23--Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

These four criteria will continue to be complied with for the duration of the operating cycle upon NRC granting approval of this amendment request. Both RTS trains and all RTS protection functions (including P-4) remain fully operable and all required redundancy continues to be maintained. The RTS continues to remain fully testable (apart from the identified issue) and the postulated failure mechanism is not transportable to other trains/components of Unit 2 or to Unit 1. In addition, even if the suspect reactor trip bypass breaker cell switch were to change state, this does not impact the operability of the automatic or manual reactor trip functions.

4.2 Precedent

Amendments 247/240, issued by the NRC on December 30, 2008, extended the SR 3.3.1.4 Frequency to 62 days on a staggered test basis. The amendment request submitted herein, following approval by the NRC, will provide for a one-time extension of the current 62-day surveillance test interval.

4.3 Evaluation of Significant Hazards Considerations

Duke has concluded that operation of Catawba Unit 2 in accordance with the proposed license condition does not involve a significant hazards consideration. Duke's conclusion is based upon its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

Response: No.

The Reactor Trip System (RTS) serves as accident mitigation equipment and is not required to function unless an accident occurs. The reactor trip bypass breakers are utilized to support testing of the reactor trip breakers (RTBs) while at power. This equipment does not affect any accident initiators or precursors. The proposed extension of the Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.4 Frequency for RTBs does not affect its interaction with any system whose failure or malfunction could initiate an accident. Therefore, the probability of an accident previously evaluated is not significantly increased.

The risk evaluation performed in support of this amendment request demonstrates that the consequences of an accident are not significantly increased. As such, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of the NRC granting of this proposed change. No changes are being made to the plant which will introduce any new or different accident causal mechanisms.

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

Response: No.

Based on the availability of the RTS equipment and the low probability of an accident, Catawba concludes that the proposed extension of the surveillance test interval does not result in a significant reduction in the margin of safety.

The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be significantly impacted by the proposed change. The risk implications of this request were evaluated and found to be acceptable.

4.4 Conclusions

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 NRC'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.

5.0 ENVIRONMENTAL ASSESSMENT

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for the categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ATTACHMENT 2

MARKED-UP CATAWBA FOL PAGES

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

(6) Mitigation Strategies

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 233 are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below and pursuant to evaluations contained in the referenced SER and SSERs. These include, (a) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of the SER, and SSERs # 3 and #4), (b) exemption from the requirement of paragraph III.A.(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #3), and (c) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #3). These exemptions are authorized by law, will not present an undue risk to the public health and safety, are consistent

Amendment Number	Additional Condition		Implementation Date
165	The schedule for the performance of new and revised surveillance requirements shall be as follows: For surveillance requirements (SRs) that are new in Amendment No. 165 the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment No. 165. For SRs that existing prior to Amendment No. 165, including SRs with modified acceptance criteria and SRs who intervals of performance are being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of amendment No. 165. For SRs that existed prior to Amendment No. 165, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of Amendment No. 165		By January 31, 1999
172	The maximum rod average burnup for any rod shall be limited to 60 GWd/mtU until the completion of an NRC environmental assessment supporting an increased limit.		Within 30 days of date of amendment.
233	This amendment requires the licensee to use administrative controls, as described in the licensee's letter of April 30, 2007, and evaluated in the Staff's Safety Evaluation dated October 31, 2007, to restrict the primary to secondary leakage through any one steam generator to 75 gallons per day and through all steam generators to 300 gallons per day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 16 operation.		Prior to any entry into Mode 4 during Cycle 16 operation

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Amendment Number	Additional Condition		Implementation Date
	The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling Outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this license condition shall expire.		February 19, 2009

ATTACHMENT 3

RETYPED (CLEAN) CATAWBA FOL PAGES

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

(6) Mitigation Strategies

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
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(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
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4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. , are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below, and pursuant to evaluations contained in the referenced SER and SSER. These include: (a) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of SSER #5), (b) exemption from the requirement of paragraph III.A.1(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #5), and (c) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, are consistent

Amendment Number	Additional Condition		Implementation Date
	The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling Outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this license condition shall expire.		February 19, 2009

ATTACHMENT 4

CATAWBA PRA TECHNICAL ADEQUACY DISCUSSION

Review of Outstanding PRA Model Change

Administrative controls exist to assure plant changes are reflected in the PRA model. Outstanding plant changes not yet reflected in the model, and whether those would impact this analysis have been reviewed. The following five items were noted as applying to this analysis but having negligible impact on this analysis.

PRA Change No.	Description	Affect on Current Calculation
C-09-0002	LAR 247/240 implemented a change to SR 3.3.1.4 surveillance test interval from 31 days staggered test basis to 62 days staggered test basis. The analysis by West. utilized NUREG/CR-5500 data that are smaller than that used in the current model. Model needs to be revised to reflect this change.	Risk decrease. Current model is bounding.
C-07-0016	Add alternate feedwater makeup line to each S/G. (Reference Letter to NRC 2/26/07)	Many of the cut sets involve a failure of SSHR. This would be a risk reduction because of adding a new way to get feedwater to the steam generators. Therefore the current model is bounding.
C-07-0007	Consider not applying operator recovery event XHM1A1BDHE, "Operators Fail to Supply Power to Hydrogen Igniters," to ATWS sequences in which the time to core damage is relatively short.	A review of the LERF cut sets indicates that there are only three fast-acting ATWS sequences that include operator action XHM1A1BDHE. The maximum cut set value is 1.8E-09. This has an insignificant impact on the results. (This recovery is not included in the CDF model.)
C-06-0003	Hydrogen igniter logic in the LERF model uses the wrong power logic.	The LERF is overestimated by less than 1E-09/year. Insignificant impact on present analysis.
C-05-0022	Missing failure modes for certain air-operated valves (AOVs) in the KC system fault tree and the CA system fault tree.	The omitted failure modes would be relatively small contributors to CA system unavailability and therefore

		do not have a significant impact on the present analysis.

PRA Technical Adequacy Discussion

Regulatory Guide 1.200 Assessment

In accordance with American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", ASME-RA-Sc-2007 and with Regulatory Guide 1.200, Duke has made an assessment of all the ASME Supporting Requirements (SRs). This assessment is documented in approved procedures.

The Catawba PRA fully meets 224 of the 306 ASME PRA Standard Supporting Requirements (SRs), as modified by Regulatory Guide 1.200. In addition, 24 of the SRs are not applicable to the Catawba PRA, either because the referenced techniques are not utilized in the PRA or because the SR is not required for Capability Category II.

Of the 58 open SRs, 15 are of a technical nature. The remaining open SRs require enhanced documentation. However, none of the open items are expected to have a significant impact on the PRA results or insights, as discussed in Table A-1 in the Supporting Documents section.

PRA Model

The Catawba PRA is a full scope PRA including both internal and external events. The model includes the necessary initiating events (e.g., LOCAs, transients) to evaluate the frequency of accidents. The previous reviews of the Catawba PRA, NRC and peer reviews have not identified deficiencies related to the scope of initiating events considered.

The Catawba PRA includes models for those systems needed to estimate core damage frequency. These include all of the major support systems (e.g., ac power, service water, component cooling, and instrument air) as well as the mitigating systems (e.g., emergency core cooling). These systems are generally modeled down to the component level, pumps, valves, and heat exchangers. This level of detail is sufficient for this application.

Truncation Limit

Truncation limit is not an issue with this risk calculation. The analysis for the current configuration was performed at the same truncation level as the base case (5.0E-10 for CDF and 5.0E-11 for LERF). The event of interest was set to 1.0 initially in the analysis to ensure cut sets were not inappropriately being truncated when the final value was used. There is adequate representation of the expected failure in the results (appears in every cut set) and all appearances of the event in cut sets do not appear within a factor of 10 of the truncation limit. A sensitivity study determined that the Regulatory Guide 1.174 risk guidelines were met for this analysis. Additionally, an explicit truncation limit analysis was performed for Revision 3a of the PRA consistent with ASME standard and Regulatory Guide 1.200 requirements to ensure truncation limit would not be an issue for most applications.

Uncertainty and Sensitivity
Duke agrees with the Regulatory Guide 1.177 statement that risk analyses of completion time extensions are relatively insensitive to uncertainties and that similar results are expected for surveillance test interval changes. The PRA did not credit equipment repair so there are no uncertainties to be evaluated for that issue. The sensitivity analysis presented in the technical justification addresses the important assumptions made in the submittal and shows that the risk resulting from the proposed surveillance test interval extension is relatively insensitive to uncertainties.

Results of Reviews with Respect to this LAR
A review of the analyses (cut sets and pertinent accident sequences) post processing was made for accuracy and completeness. The process applied in post processing the cut sets for this analysis is identical to that utilized in the base case PRA. No changes to the post processing of cut sets are made for this analysis. This process is documented in Duke procedures.

Consistent with the work place procedures governing PRA analysis, this calculation has undergone independent checking by a qualified reviewer. Additionally the Catawba Plant Operations Review Committee (PORC) and Duke Nuclear Safety Review Board (NSRB) reviewed and approved the original license amendment request package.

<u>Tier 2 and Tier 3 Discussion</u>

Tier 2 Assessment: Avoidance of Risk-significant Plant Equipment Outage Configurations
Tier 2 provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change.

Duke has several Work Process Manual procedures and Nuclear System Directives that are in place at Catawba Nuclear Station to ensure that risk-significant plant configurations are avoided. The key documents are as follows:

- Nuclear System Directive 415, "Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a.4)".
- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)".
- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL".
- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL".

The proposed changes are not expected to result in any significant changes to the current configuration risk management program. The existing program uses a blended approach of quantitative and qualitative evaluation of each configuration assessed. The Catawba on-line computerized risk tool, ORAM-Sentinel, considers both internal and external initiating events with the exception of seismic events. Thus, the overall change in plant risk during maintenance activities is expected to be addressed adequately in accordance with Regulatory Guide 1.174 and 1.177 considering the proposed surveillance test interval extension period.

Tier 3 Assessment: Maintenance Rule Configuration Control
10 CFR 50.65(a)(4), Regulatory Guide 1.182, and NUMARC 93-01 require that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increase in risk that may result from proposed maintenance activities. These requirements are applicable for all plant modes. NUMARC 91-06 requires utilities to assess and manage the risks that occur during the performance of outages.

As stated above, Duke has approved procedures and directives in place at Catawba to ensure the requirements of the Maintenance Rule are implemented. These documents are used to address the Maintenance Rule requirements, including the on-line (and off-line) Maintenance Policy requirement to control the safety impact of combinations of equipment removed from service.

More specifically, the Nuclear System Directives address the process; define the program, and state individual group responsibilities to ensure compliance with the Maintenance Rule. The Work Process Manual procedures provide a consistent process for utilizing the computerized software assessment tool, ORAM-SENTINEL, which manages the risk associated with equipment inoperability.

ORAM-SENTINEL is a Windows-based computer program designed by the Electric Power Research Institute as a tool for plant personnel to use to analyze and manage the risk associated with all risk significant work activities including assessment of combinations of equipment removed from service. It is independent of the requirements of Technical Specifications and Selected Licensee Commitments.

The ORAM-SENTINEL models for Catawba are based on a "blended" approach of probabilistic and traditional deterministic approaches. The results of the risk assessment include a prioritized listing of equipment to return to service, a prioritized listing of equipment to remain in service, and potential contingency considerations.

Additionally, prior to the release of work for execution, Operations personnel must consider the effects of severe weather and grid instabilities on plant operations. This qualitative evaluation is inherent of the duties of the Work Control Center Senior Reactor Operator (SRO). Responses to actual plant risk due to severe weather or grid instabilities are programmatically incorporated into applicable plant emergency or response procedures.

External Events Discussion

The Catawba PRA is a full scope model that includes both internal and external events. The following is a list of the reviews conducted on the PRA modeling which assures the technical adequacy of the existing PRA model with respect to external events:

- A peer review sponsored by the Electric Power Research Institute (EPRI) was conducted on the original Catawba PRA.
- An SER has been received on the IPE and IPEEE for Catawba.
- In March 2002, a peer review of the Catawba PRA was conducted as part of the WOG PRA Certification Program.
- In August 2008, a PRA Technical Adequacy Self-Assessment was conducted against the Supporting Requirements (SRs) in the ASME standard (American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME-RA-Sc-2007) and Regulatory Guide 1.200 for Catawba.

These previous reviews did not identify deficiencies related to the scope of external initiating events considered. No fundamental plant weaknesses or vulnerabilities with regard to external events were identified during the IPEEE examination for Catawba. There were no plant changes identified from the IPEEE that would significantly reduce the risk from external events. The seismic, fire, and tornado modeling that exists in the Catawba PRA is at the level of detail used to support the IPE and IPEEE submittals and is consistent with the ASME standard and Regulatory Guide 1.200 supporting requirements.

In general it has been noted previously (reference Duke LAR submittal to NRC dated December 11, 2007 to relax completion times and surveillance test intervals for the RTS and ESFAS) that RTS actuation signal failures or unavailabilities are very small contributors to the CDF for external events. For this specific application, the generated cut sets from this analysis that went into the dCDF and dLERF values were all ATWS sequences. Cut sets involving external events (i.e., fire, high winds, external flooding, and other external events) that were generated as a result of the increased failure probability of the basic event used to justify the proposed extension were also part of the base case CDF, and as such were eliminated from the final results in the calculation of the dCDF and the dLERF.

Seismic has a negligible effect on this analysis since the seismic frequency is low and the frequency of a seismic event combined with an ATWS is even lower and therefore very unlikely. Additionally, other key plant equipment and supporting systems are more susceptible to the impact of a seismic event than the reactor vessel internals.

Catawba PRA – Open ASME PRA Standard Supporting Requirements

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
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. (See SC-B4.)	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Perform analyses with the most up-to-date version of MAAP.	Technical	No impact is expected since success criteria are consistent with peer plants per the PWROG PSA Database.
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)	Partial	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to segregate standby and operating component data. Segregate components by service condition to the extent supported by the data.	Technical	This is a refinement to the equipment failure rates. However, since most components are grouped appropriately, the overall impact should be small.
DA-C13	EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) based on actual plant experience. CALCULATE coincident maintenance unavailabilities that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares," i.e., plant systems that have more redundancy than is addressed by tech specs. For example, the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with tech specs.	Partial	Developing PRA Data, Workplace Procedure XSAA-110, Rev. 4, July 2007; Catawba Component Failure Rate Denominator Estimates, SAAG 492, December 1997	Put in place a mechanism for identifying and quantifying coincident unavailabilities. Incorporate in the system models those maintenance events allowed by technical specifications where 2 or more components have maintenance events that are correlated with each other.	Technical	The on-line risk tool (ORAM-SENTINEL) and existing plant processes and procedures are sufficient to identify high risk configurations. No significant impact on this application.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-A2	IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Enhance the HRA to consider the potential for calibration errors.	Technical	Based on preliminary evaluations using the EPRI HRA calculator, calibration errors that result in failure of a single channel are expected to fall in the low 10^{-3} range. Calibration errors that result in failure of multiple channels are expected to fall in the low 10^{-5} range. Relative to post-initiator HEPs, equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-A3	IDENTIFY which of those work practices identified above (HR-A1, HR-A2) involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems [e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system (e.g., SLCS)].	No		Identify maintenance and calibration activities that could simultaneously affect equipment in either different trains of a redundant system or diverse systems.	Technical	Relative to post-initiator HEPs, equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability. See the Expected Impact on Applications for requirement HR-A2 above.
HR-D6	PROVIDE an assessment of the uncertainty in the HEPs. USE mean values when providing point estimates of HEPs.	No		Develop mean values for pre-initiator HEPs.	Technical	Pre-initiator HEPs are generally set to relatively high screening values. Thus the suggested data refinement is not expected to have a significant impact on this application.
HR-G9	Characterize the uncertainty in the estimates of the HEPs, and PROVIDE mean values for use in the quantification of the PRA results.	No		Develop mean values for post-initiator HEPs.	Technical	Use of mean values for HEPs is expected to result in an increase in post-initiator HEP values, in the base case model as well as for applications. However, a quantitative sensitivity study was performed for this application that showed that the resulting risk is very small compared to Regulatory Guide 1.174 guidelines.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A3	REVIEW the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience. See also IE-A7	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Perform a review of the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience.	Technical	Initiating events (other than ATWS) result in a plant trip and the generation of an LER. These events are reviewed as part of the initiating events analysis. Fire and flood events that don't result in a reactor trip could potentially impact the frequencies assigned to the fire and flood initiators. However, fire and flood sequences are not significant contributors to the delta CDF in the PRA analysis for the LAR. Thus this open SR does not have a significant impact.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-C2c	For each flood area not screened out using the requirements under IF-B1b, IDENTIFY the SSCs located in each defined flood area and IF-A2) along flood propagation paths that are modeled in the internal events PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibility per IF-C3, its spatial location in the area and any flooding mitigative features (e.g., shielding, flood or spray capability ratings).	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	For those flood areas addressed in the current flooding analysis, equipment important to accident mitigation and the associated critical flood heights are identified. However, given the expected increase in number of flood areas needed to satisfy requirement IF-A1, additional equipment will need to be identified and discussed in order to meet the requirements of the ASME Standard. The current flooding analysis does not discuss flood mitigative features and this will have to be corrected to satisfy the requirements of the ASME Standard.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
IF-C3	For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	The current flooding analysis identifies the submergence failure height of the equipment important to accident mitigation, but never addresses the impact of spray. Spray as a failure mechanism needs to be addressed in the analysis or a note made explaining why it was omitted.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-C3b	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads and the potential for barrier unavailability, including maintenance activities.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Provide more analysis of flood propagation flowpaths. Address potential structural failure of doors or walls due to flooding loads and the potential for barrier unavailability.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
IF-E6b	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and indirect effects such as submergence, jet impingement, and pipe whip, as applicable.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Address potential indirect effects.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
LE-C10	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Perform plant-specific T/H calculations for SGTR. Consider some credit for ISLOCA scrubbing; if no credit can be given, then this should be documented. It is not known whether or not the additional analysis will alter the LERF, but because these items dominate LERF, a more realistic analysis should be considered.	Technical	The conservative treatment will not mask the contribution of non-bypass events, because even if some credit were given to scrubbing, the unscrubbed bypasses would still dominate LERF over the non-bypass events. In addition, the limiting risk metric in the present analysis is CDF, not LERF.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
LE-C6	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of para. 4.5.5, as appropriate for the level of detail of the analysis.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	The only operator action expected to be important is RCS depressurization for small LOCAs. However, the current analysis lacks a formal dependency analysis for this action. The result is expected to be insensitive to this impact given that the SGTR so dominates the result.	Technical	This issue affects some small LOCAs. Because the small LOCA contribution to LERF is small, there is no significant impact associated with this open SR.
LE-D3	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for non-significant accident progression sequences resulting in a large early release. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	No	Catawba ISLOCA Analysis, CNC-1535.00-00-0053, Rev. 0, January 2006	For MNS/CNS, the ND heat exchanger is assumed to provide the largest break flow area. The ISLOCA is a dominant contributor and the evaluation is relatively conservative.	Technical	ISLOCA sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Cut set review during model integration and when supporting applications should address this. Suggest adding this guidance to workplace procedure XSAA-103.	Documentation	Phenomenological effects are already considered in the model.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
DA-A1a	ESTABLISH definitions of SSC boundaries, failure modes, and success criteria consistent with corresponding basic event definitions in Systems Analysis (SY-A5, SY-A7, SY-A8, SY-A10 through SY-A13 and SY-B4) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events consistent with corresponding definitions in Systems Analysis (SY-A18).	No	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to discuss component boundaries definitions.	Documentation	No impact is expected for documentation issues.
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)	Partial	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to include a specific discussion of outlier treatment (i.e., do any outliers exist? If so, how are these events considered and grouped?)	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
DA-D4	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Partial	Catawba PRA Common Cause Analysis, CNC-1535.00-00-0028, Rev. 0, December 2005	Enhance the documentation to include a discussion of the specific checks performed on the Bayesian-updated data, as required by this SR.	Documentation	No impact is expected for documentation issues.
DA-D6	USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities consistent with the component boundaries.	Partial	Catawba PRA Common Cause Analysis, CNC-1535.00-00-0028, Rev. 0, December 2005	Provide documentation in SAAG 637 of the comparison of the component boundaries assumed for the generic CCF estimates to those assumed in the Catawba PRA to ensure that these boundaries are consistent.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
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 meaning of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of detection, diagnosis and decision-making, and executing the required response (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.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Document in more detail the influence of performance shaping factors on execution human error probabilities.	Documentation	No impact is expected for documentation issues.
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) (See SC-B4.). SPECIFY the point in time at which operators are expected to receive relevant indications.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Enhance HRA documentation accordingly.	Documentation	No impact is expected for documentation issues.
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.	No		Document a review of the HFEs and their final HEPs relative to each other to confirm their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis: (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	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Develop more detailed documentation of operator cues, relevant performance shaping factors, and availability of sufficient manpower to perform the action.	Documentation	No impact is expected for documentation issues.
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.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006; Systems Analysis	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-A3a	REVIEW generic analyses of similar plants to assess whether the list of challenges included in the model accounts for industry experience.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Ensure the list of challenges included in the Catawba PRA accounts for industry experience using a more recent reference, such as the WOG PSA Model and Results Comparison Database - Revision 4.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A4	PERFORM a systematic evaluation of each system where necessary (e.g., down to the subsystem or train level), 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 an FMEA (failure modes and effects analysis), or other systematic process] to assess and document the possibility of an initiating event resulting from individual systems or train failures.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Provide documentation of a systematic evaluation of all plant systems, including support systems (including those not explicitly modeled in the PRA), to assess the possibility of an initiating event occurring due to a failure of the system.	Documentation	No impact is expected for documentation issues.
IE-A4a	When performing the systematic evaluation required in IE-A4, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from system alignments resulting from preventive and corrective maintenance.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-A5	In the identification of the initiating events, INCORPORATE (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation. (b) events resulting in a controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A6	INTERVIEW plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	No		Obtain plant personnel input (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-A7	REVIEW plant-specific operating experience for initiating event precursors, for the purpose of 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.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Include review of precursor events for their potential to be initiating events.	Documentation	No impact is expected for documentation issues.
IE-B1	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis element (para. 4.5.2) and to facilitate quantification in the Quantification element (para. 4.5.8).	No		Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-B2	USE a structured, systematic process for grouping initiating events. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA).	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Document a structured, systematic grouping of initiating events (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-B3	GROUP initiating events only when the following can be assured: (a) events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or (b) events can be subsumed into a group and bounded by the worst case impacts within the "new" group. DO NOT SUBSUME events into a group unless: (1) the impacts are comparable to or less than those of the remaining events in that group, AND (2) it is demonstrated that such grouping does not impact significant accident sequences.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance documentation of the grouping process (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-D3	DOCUMENT the assumptions and sources uncertainty with the initiating event analysis.	No		Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IF-B3	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) range of flow rates (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Enhance the Internal Flood analysis to address the potential for spray, jet impingement, and pipe whip failures. Additionally, document how these failures are included in the quantification.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-F2	DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification. For example, this documentation typically includes (a) flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined (b) flood areas used in the analysis and the reason for eliminating areas from further analysis (c) propagation pathways between flood areas and assumptions, calculations, or other bases for eliminating or justifying propagation pathways (d) accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification (e) assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability (f) screening criteria used in the analysis (g) flooding scenarios considered, screened, and retained (h) description of how the internal event analysis models were modified to model these remaining internal flooding scenarios (i) flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) (j) calculations or other analyses used to support or refine the flooding evaluation (k) results of the internal flooding analysis, consistent with the quantification requirements provided in HLR QU-D	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Need to document how the analysis addressed all of the items identified in this requirement.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
LE-F2	PROVIDE uncertainty analysis that identifies the sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004; Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	Documentation	No impact is expected for documentation issues.
LE-F3	IDENTIFY contributors to LERF and characterize LERF uncertainties consistent with the applicable requirements of Tables 4.5.8-2(d) and 4.5.8-2(e). NOTE: The supporting requirements in these tables are written in CDF language. Under this requirement, the applicable requirements of Table 4.5.8 should be interpreted based on LERF, including characterizing key modeling uncertainties associated with the applicable contributors from Table 4.5.9-3. For example, supporting requirement QU-D5 addresses the significant contributors to CDF. Under this requirement, the contributors would be identified based on their contribution to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Compare LERF results and uncertainties to similar plants and include in the LERF documentation.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Evaluate the relative contribution of the various contributors to the total LERF.	Documentation	No impact is expected for documentation issues.
LE-G4	DOCUMENT assumptions and sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	Documentation	No impact is expected for documentation issues.
LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	No		Include in the LERF documentation an assessment that identifies the limitations in the LERF analysis that could impact applications.	Documentation	No impact is expected for documentation issues.
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Provide a discussion of the significant cut sets and sequences.	Documentation	No impact is expected for documentation issues.
QU-D3	COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another?	No		Perform and document a comparison of results between the CNS PRA and other similar plants.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
QU-E4	EVALUATE the sensitivity of the results to model uncertainties and assumptions using sensitivity analyses [Note (1)].	No		Perform and document a set of sensitivity cases to determine the impact of the assumptions and sources of model uncertainty on the results.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation?	Expected Impact on Application
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 nonrecovery 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 accident sequences to be nonsignificant (h) the results of all sensitivity studies (i) the uncertainty distribution for the 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 regarding why 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.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Expand the documentation of CNS PRA model results to address all required items.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
QU-F6	DOCUMENT the quantitative definition used for significant basic event, significant cutset, significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Document the required definitions.	Documentation	No impact is expected for documentation issues.
SC-A4	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.
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	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Provide evidence that an acceptability review of the T/H analyses is performed.	Documentation	No impact is expected for documentation issues.
SC-C1	DOCUMENT the success criteria in a manner that facilitates PRA applications, upgrades, and peer review.	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
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 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	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-A14	In meeting SY-A12 and SY-A13, 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 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.	Partial	System analyses	Provide quantitative evaluations for screening.	Documentation	No impact is expected for documentation issues.
SY-A2	COLLECT pertinent information to ensure that the systems analysis appropriately reflects the as-built and as-operated systems. Examples of such information include system P&IDs, one-line diagrams, instrumentation and control drawings, spatial layout drawings, system operating procedures, abnormal operating procedures, emergency procedures, success criteria calculations, the final or updated SAR, Technical Specifications, training information, system descriptions and related design documents, actual system operating experience, and interviews with system engineers and operators.	Partial	System analyses	Need to update references per XSAA-115.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-A4	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	Partial	System analyses	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.	Documentation	No impact is expected for documentation issues.
SY-A8	ESTABLISH the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data. For example, a control circuit for a pump does not need to be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL as separate basic events of the model, those subcomponents (e.g., a valve limit switch that is associated with a permissive signal for another component) that are shared by another component or affect another component, in order to account for the dependent failure mechanism.	No		Enhance systems analysis documentation to discuss component boundaries.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-B15	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 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 (h) harsh environments induced by containment venting or failure that may occur prior to the onset of core damage	Partial	System analyses	Cut set review during applications should address this. Suggest adding this guidance to workplace procedure XSAA-103.	Documentation	No impact is expected for documentation issues.
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.	Partial	System analyses	Per Duke's PRA modeling guidelines, ensure that a walkdown/system engineer interview checklist is included in each system notebook. Based on the results of the system walkdown, summarize in the system write-up any possible spatial dependencies or environmental hazards that may impact system operation.	Documentation	No impact is expected for documentation issues.
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human	Partial	System analyses	Enhance system model documentation to comply with all ASME PRA	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
	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: (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 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.			Standard requirements.		