Kelvin Henderson
Vice President
Catawba Nuclear Station

Duke Energy
CNO1VP | 4800 Concord Road
York, SC 29745

o: 803.701.4251
f: 803.701.3221

CNS-15-049

May 11, 2015

10 CFR 50.59 and 10 CFR 50.4

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
2013/2014 10 CFR 50.59 Summary Report

Attached please find the 2013/2014 10 CFR 50.59 Summary Report. The report contains a brief description of changes, tests, and experiments, including a summary of the safety evaluations for Catawba Nuclear Station, Units 1 and 2. This report is submitted pursuant to the provisions of 10 CFR 50.59(d)(2) and 10 CFR 50.4.

If there are any questions regarding this report, please contact Cecil A. Fletcher II at (803) 701-3622.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 11, 2015.

Sincerely,

Kelvin Henderson
Vice President, Catawba Nuclear Station

Attachment

IE47
NRR

xc (with attachment):

V.M. McCree, Region II Administrator
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Avenue NE, Suite 1200
Atlanta, GA 30303-1257

G.E. Miller, NRC Project Manager (addressee only)
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Mail Stop 8 G9A
Rockville, MD 20852-2738

G.A. Hutto III, NRC Senior Resident Inspector
Catawba Nuclear Station

Attachment

Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
2013/2014 10 CFR 50.59 Summary Report

A/R Number: 00433539

Facility: CATAWBA NUCLEAR STATION

Unit(s): 1, 2

Activity Title: Revision to UFSAR Section 15.4.1 (Uncontrolled Bank Withdrawal at Zero Power (UCBWZP)) for a change in Core Thermal Power (3.469 E-6 MWth), NC Flow (384 K gpm), and Bypass Flow (9.2% for Peak Pressure and 8.5% for DNB)

Summary

The proposed activities are revisions to UFSAR Section 15.4.1 (Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical or Low Power Startup Condition Accident) departure from nucleate boiling analysis and peak primary pressure analysis. These activities arise from revisions to the associated calculation.

A summary of the 10CFR50.71(e) UFSAR changes include:

UFSAR Section 15.4.1 (Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition) was revised for text, input data, sequence of events, and figures for the peak primary pressure case and the departure from nucleate boiling analysis.

The change to the calculation producing these results is summarized as:

(a) an analysis change for both the departure from nucleate boiling and peak primary pressure case to appropriately apply initial neutron flux as 1.0E-9 RTP (rated thermal power) where RTP is 3,469 MWth for a Measurement Uncertainty Recapture (MUR) uprate. In the former analysis, initial neutron flux was applied as 1.0E-9 RTP of 3,411 MWth.

(b) an analysis change to use 384,000 gpm RCS flow.

(c) A change in core bypass flow. However, core bypass flow is not specifically described in the UFSAR description of the accident.

There is no increase in the frequency of occurrence of an accident previously evaluated in the UFSAR or the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR. There is also no increase in the consequences of an accident previously evaluated in the UFSAR or an increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. There is no accident of a different type or of a malfunction of an SSC with a different result. Design basis limits for fission products are not altered or exceeded. Finally, there is no departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. There is also no change to Technical Specifications. In conclusion, this activity does not require prior NRC approval.

A/R Number: 00436002

Facility: CATAWBA NUCLEAR STATION

Unit(s): 1, 2

Activity Title: Update to UFSAR 15.7.5 - Postulated Drop of a Fuel Cask at Catawba

Summary

A set of updates to the Catawba Nuclear Station UFSAR 15.7.5 and Table 15-14 have been completed. Previously, UFSAR 15.7.5 included a brief report of the analysis of a fuel cask drop (FCD) scenario in which a fuel cask was dropped or tipped towards a spent fuel pool. With the updates, the UFSAR includes a brief report of this and two additional FCD scenarios as follows:

- Dropping a loaded and sealed fuel cask onto the ground or fuel building floor (found to "screen out" from a full evaluation pursuant to 10 CFR 50.59).
- Dropping a loaded but unsealed fuel cask into a fuel cask pit.

The activity evaluated fully pursuant to 10 CFR 50.59 is the acknowledgment of the potential for a FCD into a fuel cask pit at Catawba. This FCD scenario was analyzed for three different effects as follows:

- Impact on subcritical margin of the dropped fuel cask and its contents (completed under the requirements of 10 CFR 72 and beyond the scope of this 10 CFR 50.59 evaluation)
- Impact on the spent fuel pool (UFSAR 9.1.2.3)
- Radiation doses (UFSAR 15.7.5)

It is acknowledged only that the equipment handling a fuel cask could fail in a position different from that associated with dropping or tipping a fuel cask towards a spent fuel pool. Acknowledging this implies no increase in the frequency of the FCD or the likelihood of failure of the equipment used to handle the fuel cask. The radiation doses calculated for this FCD scenario were under the threshold for "not more than a minimal increase in consequences" with significant margins. The scenario in this activity is another scenario for the FCD already evaluated in the UFSAR. Also, the FCD is another heavy load drop similar to a fuel handling accident and weir gate drop, the analyses of both of which are reported in the UFSAR. Acknowledgment of a FCD into a fuel cask pit does not imply the potential for a new accident or a malfunction of a structure, system, or component with a result different from any evaluated in the UFSAR. The analyses completed for the new FCD scenario confirm the adequacy of the design of the fuel pin cladding, the only fission product barrier relevant to this evaluation. Finally, the method of analysis as applied to the FCD represents no change to the methods in the current license basis of Catawba which are described in the UFSAR.

The potential for a FCD into a fuel cask pit may be acknowledged and UFSAR 15.7.5 and Table 15-14 updated to report the analysis of its radiological consequences without prior NRC approval.

A/R Number: 00442540

Facility: CATAWBA NUCLEAR STATION

Unit(s): 1, 2

Activity Title: Each unit at Catawba has two atmospheric steam dumps isolated due to excessive steam leakage. Efforts are underway to repair the valves. However, it will require greater than 90 days to procure part and services to perform the repairs. Therefore, this evaluation is being performed per NSD 228.

Summary

Currently, both units have two atmospheric dump valves (1SV-44, 1SV-54, 2SV-36, and 2SV-44) isolated due to excessive steam leakage. Plans are underway to repair the valves and return them to service. However, it will require greater than 90 days to procure the parts and services to repair the valves and return them to service. Therefore, the isolation of the atmospheric dump valves must be evaluated against the criteria of 10CFR50.59 per NSD 228. This 10CFR50.59 evaluation will evaluate the isolation of up to any three atmospheric dumps to bound this and any future temporary alterations to support maintenance.

The atmospheric dump valves are part of the Turbine Bypass System. The Turbine Bypass System (TBS) is designed to reduce the magnitude of nuclear system transients following large turbine load reductions by dumping main steam directly to the main condenser and/or to the atmosphere, thereby creating an artificial load on the reactor. The reactor can accept a 10% step change or a 5% per minute ramp change in load without tripping or steam dump. For load reductions greater than 10% step or 5% per minute ramp, the dump system is controlled to give the required ramp load change to prevent reactor trip and insertion of the shutdown control rods.

The steam dump system is not essential for safe shutdown of the unit, thus it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Failure of the steam dump system will not preclude operation of any essential system. They are not credited in any in UFSAR Chapter 15 accidents.

However, Duke credited these valves as justification for bypassing the anticipatory reactor trip on turbine trip for power levels below 70% . The anticipatory reactor trip on turbine trip is a post-TMI modification that Duke committed to implementing . The purpose of this anticipatory trip was to prevent opening of a pressurizer power operated relief valve (PORV) after a turbine trip, and a subsequent small break LOCA if it were to stick open. It was justified blocking this trip below 70% power based on an analysis that took credit for the correct operation of steam dumps, rod control, feedwater control, and pressurizer pressure control. This analysis demonstrated that below 70% power, a turbine trip without reactor trip would not cause pressurizer pressure to rise above the PORV setpoint of 2335 psig.

There are no Technical Specifications that explicitly require the atmospheric steam dumps to be operable. However, the basis for Technical Specification 3.3.1 (Reactor Trip System Instrumentation) states "Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System." As stated above, three atmospheric dump valves may be removed from service and maintain the validity of the analysis that formed the basis for the P-9 setpoint.

Isolation of up to three atmospheric dump valves per unit will not require addition of any new technical specifications because it does not meet the criteria of 10 CFR 50.36 (c)(2)(ii).

The atmospheric steam dump system is a standby system that only actuates in response to a large load rejection. Excessive leakage up to complete failure of an atmospheric valve would cause an increase in secondary steam flow. Isolation of atmospheric dump valves that are leaking excessively by closing the associated block valve reduces the probability of such an occurrence by removing them from the main steam system pressure boundary. Therefore, isolation of up to three atmospheric dump valves per unit will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The anticipatory reactor trip on turbine trip is a post-TMI modification that Duke committed to implementing. The purpose of this anticipatory trip was to prevent opening of a pressurizer power operated relief valve after a turbine trip, and a subsequent small break LOCA if it were to stick open. It has been justified blocking this trip below 70% power based on an analysis that took credit for the correct operation of steam dumps, feedwater control, and pressurizer pressure control. This analysis assumed 70% steam dump capacity. This analysis was incorporated by reference into Section 7.2.1.1.2 of the UFSAR. As stated in the activity description, isolation of up to three atmospheric dump valves maintains the validity of the 70% steam dump capacity in the original analysis. Thus, the original analysis remains bounding. Therefore, isolation of up to three atmospheric dump valves per unit will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The atmospheric steam dumps are not used to mitigate the consequences of any accidents in the UFSAR. In the loss of external load and turbine trip events, no credit is taken for any of the condenser or atmospheric dump valves with regard to overpressure protection of the main steam system. Excessive leakage from the atmospheric dump valves could contribute to offsite dose in the event of a stuck open main steam isolation valve during a steam generator tube rupture. Isolation of known leaking valves eliminates this potential leak path. Therefore, isolation of leaking valves would be expected to either decrease or result in no change in offsite dose. Therefore, isolation of up to three atmospheric dump valves per unit will not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The anticipatory reactor trip on turbine trip is a post-TMI modification that Duke committed to implementing. The purpose of this anticipatory trip was to prevent opening of a pressurizer power operated relief valve after a turbine trip, and a subsequent small break LOCA if it were to stick open. The atmospheric dump valves are not safety related and are not used to mitigate the consequences of any malfunctions of any SSCs important to safety. Therefore, the response/consequences to a malfunction of any SSCs important to safety would be unchanged from that evaluated in the UFSAR.

There are three possible failure scenarios of the atmospheric dump valves:

1. Failure of the pressure boundary of the valve (including failing open) resulting in an increase in steam flow. Isolation of degraded valves decreases the probability of this event.
2. Partial or complete loss of steam dump capability. Overpressure protection provided by main steam safety/relief valves.
3. Failure to actuate after a turbine trip below the P-9 setpoint resulting in opening of a pressurizer PORV. As stated above, up to three atmospheric dump valves may be isolated and maintain the validity of the original analysis.

All of these scenarios are evaluated in UFSAR. Therefore, isolation of up to three atmospheric dump valves will not create a possibility for an accident of a different type than previously evaluated in the UFSAR.

The UFSAR analysis takes no credit for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure in the maximum primary system pressure case. Pressurizer safety valves are available. The reactor is tripped by the high pressurizer pressure trip function. The steam generator safety valves limit the Main System pressure below 110 percent of the design value. The pressurizer safety valves limit the primary system pressure below 110 percent of the design value. Since the design basis limit for the reactor coolant pressure boundary is maintained with taking no credit for the operation of the steam dump system, isolation of up three atmospheric dump valves will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

Isolation of up to three atmospheric steam dumps does not involve any evaluation methodology. The original analysis remains valid and bounding with no changes.

A/R Number: 00446137

Facility: CATAWBA NUCLEAR STATION

Unit(s): 1, 2

Activity Title: Revise the UFSAR section 5.2.2 and TS Bases 3.4.12 to acknowledge the commonality of power supplies between the NC PORV s and normal and excess letdown valves and the use administrative controls to prohibit credit of NC32B when excess letdown is in service and credit of NC34A when normal letdown is in service during LTOP applicability.

Summary

Revise the UFSAR and Bases section of TS 3.4.12 to acknowledge the commonality of power supplies between the NC PORV's and normal and excess letdown valves; and then describe how procedural controls are used to prohibit credit of the associated PORV's under the identified configurations where a power supply failure would result in both a loss of letdown and a LTOP PORV. Once UFSAR section 5.2.2.2 and the bases Section of TS 3.4.12 are revised, the OBDN condition, in PIP C-12-01241, can be closed provided procedural controls are maintained.

UFSAR section 5.2.2.2 addresses overpressure protection of the Reactor Coolant (NC) System. At NC temperatures > 2100 F, the NC System pressure boundary is protected from over pressurization via the NC pressurizer safety valves. At low RCS temperature conditions, < 2100 F, brittle fracture of the reactor vessel is of concern. The latter condition is termed Low Temperature Over Pressure (LTOP). The LTOP condition is addressed in Technical Specification (TS) 3.4.12 , Low Temperature Over Pressure (LTOP) System . The Limiting Condition for Operation (LCO) for TS 3.4.12 is required to meet criterion 2 of 10 CFR 50.36 (c)(2)(ii). Per the Bases section of TS 3.4.12, the LCO provides NC system overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but two pumps incapable of injection into the RCS, isolating the accumulators, and limiting reactor coolant pump operation at low temperatures. The pressure relief capacity requires two redundant relief paths. The NC relief path is the overpressure protection device that acts to terminate an increasing pressure event. The LTOP relief paths can either be two LTOP Power Operated Relief Valves (PORV's) actuating at the LTOP setpoint or two Residual Heat Removal (ND) pump suction relief valves with the associated ND suction isolation valves open and de-energized or a combination of one LTOP PORV and one ND suction relief valve.

NUREG -0138 provides NRC staff discussion of 15 NRR concerns. The 15th item of NUREG -0138 addresses concerns regarding a number of industry events where pressure limits were exceeded during low temperature shutdown operation. NUREG 800 Branch technical Position 5-2 concerning reactor coolant pressurization at low pressure conditions addresses the NUREG -0138 concerns with over-pressurization. NUREG 800 Branch technical Position 5-2 , section B, item 3, states that the low temperature overpressure protection system should be able to perform its function assuming any single active failure. Furthermore item 3 states "The cause for initiation of the event (e.g. operator error, component malfunction) should not be considered as the single failure". In addressing Branch Technical position 5.2 item B 3, UFSAR Section 5.2.2.2, Design Evaluation, states that the normal letdown isolation valves and the LTOP PORV's are powered from different DC power sources such that failure of a DC power supply will not cause a loss of letdown, which could result in an LTOP challenge, and simultaneously result in loss of a LTOP protection relief path. Contrary to this statement Reference 11, identifies several commonalities between the Normal or Excess Letdown isolation valves and the LTOP PORV's or LTOP permissive' s such that a single power supply failure could result in the isolation of an in-service letdown path and a LTOP PORV. As a result of this nonconformance, the LTOP system, required by Technical Specification 3.4.12, is Operable But Degraded Nonconforming (OBDN). The OBDN condition is documented in PIP C-12-01241. The following compensatory actions have been implemented to address the OBDN condition:

- During LTOP applicability on either unit NC-34A must be declared inoperable per TS 3.4.12 when normal letdown is in service.
- During LTOP applicability on either Unit NC-32B must be declared inoperable per TS 3.4.12 when excess letdown is in service.
- During LTOP applicability on either unit NC-32B must be declared inoperable per TS 3.4.12 when normal letdown is in service with only NV pump B in operation.

Elimination of the commonalities between the normal letdown isolation valves and NC34A and excess letdown isolation valves and NC32B involve new cable routes from the Auxiliary Building to Containment where the letdown valves and PORV's are located. Changing the power supplies on either of these circuits requires significant cable modifications through containment penetrations in order to maintain channel separation. This makes engineering changes to eliminate the commonalities between NC34A and normal letdown and NC32B and excess letdown cost prohibitive. Elimination of the loss of charging interlock commonality with NC32B also requires a substantial wiring revision due to channel separation requirements.

Therefore, it is proposed to revise the UFSAR and Bases section of TS 3.4.12 to acknowledge the commonality of power supplies between the NC PORV's and normal and excess letdown valves; and then use procedural controls to prohibit credit of the associated PORV's under the identified configurations where a power supply failure would result in both a loss of letdown and a LTOP PORV. The change to the UFSAR recognizes that the design feature of diverse power supplies between letdown and the LTOP PORV's does not exist. This design feature is being replaced with procedural controls to ensure the LTOP PORV's are not credited for meeting TS 3.4.12 during letdown configurations when separation of power supplies between the letdown isolation valves and PORV's does not exist. Once UFSAR section 5.2.2.2 and the bases Section of TS 3.4.12 are revised, the OBDN condition can be closed provided procedural controls are maintained.

It is concluded that the proposed activity does not require a change to the Technical Specifications; none of the eight (8) Evaluation criteria were met (thus an LAR is not required); and the activity does not require prior NRC approval.

A/R Number: 00448537

Facility: CATAWBA NUCLEAR STATION

Unit(s): 1, 2

Activity Title: EC 105974 / Installation of SSPS Boards Containing Complex Programmable Logic Device (CPLD) Technology

Summary

The proposed activity is an Engineering Change (EC) 105974 that allows for the replacement of SSPS original design printed circuit boards containing Motorola High Threshold Logic (MHTL) components with the Westinghouse new design boards containing Complex Programmable Logic Device (CPLD) technology in the Solid State Protection System (SSPS). The SSPS boards are being replaced due to obsolescence issues of the original design boards and to improve reliability.

The NRC has reviewed the Westinghouse Topical Report (WCAP-17867-P, Revision 1, "Westinghouse SSPS Board Replacement Licensing Summary Report") and issued a Final Safety Evaluation Report (SER). This SER concludes that the new design SSPS boards are equivalent to original SSPS boards, provided that Licensees review their existing design and licensing bases to ensure that the testing conditions provided in the WCAP bound plant conditions. Catawba has utilized the 10CFR50.59 Evaluation template provided by Westinghouse as the foundation for this 10CFR50.59 Evaluation and has concluded that Catawba satisfies the criteria for which the new SSPS boards may be installed. In addition to the EC, Catawba is revising its UFSAR to reference the WCAP.

This 10CFR50.59 Evaluation also satisfies the requirements set forth by NRC Enforcement Guidance Memorandum (EGM) 14-002, Dispositioning Westinghouse Pressurized Water Reactor Licensee Noncompliance with 10 CFR 50.59, "Changes, Tests, and Experiments," for the Installation of Complex Programmable Logic Device (CPLD) Based Solid State Protection System (SSPS) Cards.

It is concluded that the proposed activity does not require a change to the Technical Specifications; none of the eight (8) Evaluation criteria were met (thus an LAR is not required); and the activity does not require prior NRC approval.