DUKE ENERGY®

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-15-058

June 12, 2015

10 CFR 50.90

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendments
TS 3.4.1, Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Pursuant to 10 CFR 50.90, Duke Energy is requesting amendments to the Catawba Units 1 and 2 Facility Operating Licenses (FOLs) NPF-35 and NPF-52, respectively, and the associated TS. This request is to modify the subject TS to allow lower minimum values of RCS flowrate. Specifically, Duke Energy proposes to modify TS Table 3.4.1-1, "RCS DNB Parameters", Parameter 3, "RCS Total Flow Rate", Limit as follows:

Unit 1: From "≥ 388,000 gpm and ≥ the limit specified in the COLR (Unit 1)", to "≥ 384,000 gpm and ≥ the limit specified in the COLR (Unit 1)"
Unit 2: From "≥ 390,000 gpm and ≥ the limit specified in the COLR (Unit 2)", to "≥ 387,000 gpm and ≥ the limit specified in the COLR (Unit 2)"

The contents of this amendment request package are as follows:

Attachment 1 provides the technical and regulatory evaluations associated with the proposed changes. Attachment 2 provides the marked-up TS page showing the proposed changes. The retyped (clean) TS page will be provided to the NRC immediately prior to issuance of the approved amendments. No changes to the corresponding TS Bases are required in conjunction with this amendment request.

Duke Energy is requesting NRC review and approval of this amendment request within one year of the date of submittal.

Duke Energy is requesting a 60-day implementation period in conjunction with this amendment request. Implementation of the approved amendments will require changes to the Updated Final Safety Analysis Report (UFSAR). Necessary UFSAR changes will be submitted to the NRC in accordance with 10 CFR 50.71(e), with approved exemptions.

A001
NRR

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

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

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

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

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

Executed on June 12, 2015.

Very truly yours,

Kelvin Henderson
Vice President, Catawba Nuclear Station

LJR/s

Attachments

xc (with attachments):

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

G.A. Hutto III
Senior Resident Inspector (Catawba)
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

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

S.E. Jenkins
Manager
Radioactive & Infectious Waste Management
Division of Waste Management
South Carolina Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

ATTACHMENT 1

TECHNICAL AND REGULATORY EVALUATIONS

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendments
TS 3.4.1, Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

1. DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy is requesting amendments to the Catawba Units 1 and 2 Facility Operating Licenses (FOLs) NPF-35 and NPF-52, respectively, and the associated TS. This request is to modify the subject TS to allow lower minimum values of RCS flowrate. Specifically, Duke Energy proposes to modify TS Table 3.4.1-1, "RCS DNB Parameters", Parameter 3, "RCS Total Flow Rate", Limit as follows:

Unit 1: From "≥ 388,000 gpm and ≥ the limit specified in the COLR (Unit 1)", to "≥ 384,000 gpm and ≥ the limit specified in the COLR (Unit 1)"
Unit 2: From "≥ 390,000 gpm and ≥ the limit specified in the COLR (Unit 2)", to "≥ 387,000 gpm and ≥ the limit specified in the COLR (Unit 2)"

2. PROPOSED CHANGES

The proposed changes reduce the required TS 3.4.1 minimum measured RCS flow rate from 388,000 gpm to 384,000 gpm for Catawba Unit 1 and from 390,000 gpm to 387,000 gpm for Catawba Unit 2. Although the methods of evaluation are similar for both units, each is evaluated separately below for clarity.

The safety and quality of operation of Catawba Nuclear Station will not be compromised by the implementation of these proposed amendments.

The methodologies used in the evaluations that follow are documented in the approved Core Operating Limits Report (COLR) methodology reports listed below:

1) DPC-NE-3001-P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology", Revision 0a
2) DPC-NE-3000-P-A, "Thermal-Hydraulic Transient Analysis Methodology", Revision 5a
3) DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology", Revision 4b
4) DPC-NE-2005-P-A, "Thermal Hydraulic Statistical Core Design Methodology", Revision 4a

Tables for all Updated Final Safety Analysis Report (UFSAR) transients discussed within each section below are included below. Tables are included for Unit 1 (Table 1) and for Unit 2 (Table 2) separately. These tables summarize the disposition of each UFSAR transient discussed below. The tables also include references for the various approved methodologies used for each analysis.

3. BACKGROUND

Over a number of years, a decrease in RCS flow has been observed at Catawba Unit 1 and Catawba Unit 2. The actual amount of the flow decrease has been small, but nevertheless it has resulted in a loss of margin relative to the minimum values allowed by Catawba TS 3.4.1.

RCS flow data for the last eight fuel cycles spanning approximately eleven years is shown in the tables below. Data shown are the average values over the first month of full power operation for each fuel cycle.

Fuel design has remained constant throughout this period with the use of Westinghouse Robust Fuel Assembly (RFA) and RFA-2.

On Unit 1, the Loop A reactor coolant pump was replaced during the End of Cycle (EOC) 19 Refueling Outage (RFO). The new pump resulted in a significant increase in flow in Loop A. Loop A had experienced a significant decrease in flow in the previous fuel cycle, so part of the increase for Cycle 20 was likely attributed to the recovery of some of the previous flow decrease.

During Unit 2 Cycle 18, a Loop B flow transmitter was recalibrated which resulted in approximately a 500 gpm increase in Loop B and total flow rate. This is reflected in the Cycle 19 flow values.

Crud deposits on the fuel cladding and possibly the steam generator tubes appears to be a significant contributor to the observed changes in RCS flow. A Unit 1 reactor trip in Cycle 16 resulted in a step increase in RCS flow following the return to power, indicating that some crud had been removed from the fuel cladding and/or the steam generator tubes. A Unit 1 reactor trip in Cycle 20 did not have the same effect.

Zinc injection into the RCS began on Unit 1 during Cycle 17 and on Unit 2 during Cycle 15. Zinc is added to reduce outage dose rates for steam generator work by replacing activated material in the steam generator tubes with zinc. It was expected that some of this displaced material would adhere to the fuel cladding; ultrasonic fuel cleaning during outages was started before beginning zinc injection. Following zinc injection, RCS flow increased in the last thirty to sixty days of the fuel cycles (for Unit 1 Cycles 17, 18, and 19, and for Unit 2 Cycles 15, 16, 17, and 18). The flow increases along with radiochemistry analysis indicate that some crud is being released from the fuel cladding surface at the end of the fuel cycle. This phenomenon has not been observed for the most recent fuel cycles, however. The most recent fuel cycles also have not experienced significant decreases in RCS flow; in fact, RCS flow has increased from Unit 1 Cycle 19 and Unit 2 Cycle 18.

Catawba Unit 1

	RCS Flow Data, gpm					Change from Previous Cycle, gpm					Percent of TS Required Flow (Last Three Cycles)
Cycle	Total	Loop A	Loop B	Loop C	Loop D	Total	Loop A	Loop B	Loop C	Loop D	
C1C15	391,405	98,036	98,049	97,489	97,836						
C1C16	390,974	97,846	97,909	97,485	97,741	-431	-190	-140	-4	-95	
C1C17	390,224	97,844	97,734	97,232	97,446	-750	-2	-175	-253	-295	
C1C18	389,948	97,840	97,564	97,059	97,491	-275	-3	-170	-173	45	
C1C19	389,794	97,105	97,692	97,213	97,774	-155	-736	128	154	283	100.3
C1C20	390,731	98,110	97,730	97,240	97,646	938	1,005	39	27	-127	100.6
C1C21	390,523	98,083	97,561	97,253	97,611	-209	-26	-169	13	-36	100.6
C1C22	390,860	98,221	97,626	97,350	97,653	337	138	66	97	42	
					Total Change:	-545	185	-423	-138	-183	

Catawba Unit 2

	RCS Flow Data, gpm					Change from Previous Cycle, gpm					Percent of TS Required Flow (Last Three Cycles)
Cycle	Total	Loop A	Loop B	Loop C	Loop D	Total	Loop A	Loop B	Loop C	Loop D	
C2C13	394,685	97,285	98,696	99,704	99,009						
C2C14	394,631	97,666	98,474	99,672	98,837	-54	380	-223	-32	-172	
C2C15	394,563	97,604	98,336	99,678	98,941	-68	-62	-138	6	104	
C2C16	394,396	97,535	98,243	99,691	98,939	-167	-69	-93	13	-2	
C2C17	393,091	97,184	98,055	99,210	98,640	-1305	-351	-187	-480	-298	
C2C18	391,846	97,169	97,480	98,591	98,618	-1245	-16	-575	-620	-22	100.5
C2C19	393,514	97,287	98,249	99,178	98,807	1668	119	769	587	188	100.8
C2C20	393,807	97,409	98,160	99,578	98,668	293	122	-89	400	-138	100.9
C2C21	393,761	97,437	98,287	99,237	98,792	-49	27	127	-342	123	
					Total Change:	-925	152	-409	-468	-217	

NOTE: Total flow does not equal the sum of the loop flows because of the method used to retrieve archived data.

In summary, the following insights were noted concerning the observed RCS flow decrease on Unit 1 and Unit 2:

- The flow decrease is not due to drifting components or an instrumentation calibration issue.
- The flow decrease could not be confirmed with an independent indication due to the small magnitude of the change as compared to the data resolution of other indications (e.g., RCS loop delta temperature and reactor coolant pump motor parameters).
- The following potential causes were evaluated, and as noted, eliminated.
    - Reactor coolant boron concentration changes (eliminated)
    - Boron concentration effects in the elbow tap tubing runs (eliminated)
    - RCS chemistry transients (eliminated)
    - RCS chemistry pH and zinc addition programs (eliminated)
    - Shutdown crud burst pump configuration with hydrogen peroxide addition (eliminated)
    - Steam generator outage eddy current inspections (eliminated)
    - Core effects from bypass flow changes or temperature increases (eliminated)
    - Density effects from elbow tap tubing run temperature differences between the high and low taps (eliminated)
    - Nitrogen bubbles in the high tap tubing runs due to inadequate venting following nitrogen purge evolutions (eliminated)
    - Crud buildup in the tubing runs (eliminated)

RCS flow may eventually approach the minimum values allowed by Catawba TS 3.4.1. The purpose of justifying a lower RCS flow for Unit 1 and Unit 2 is to accommodate any additional flow decrease that may occur during the course of future fuel cycles. The supporting analyses for this LAR submittal substantiate a reduction in the minimum RCS flow allowed by TS 3.4.1 to 384,000 gpm for Unit 1 and to 387,000 gpm for Unit 2.

4. TECHNICAL EVALUATION

Basis for Proposed Changes (Catawba Unit 1)

The following summarizes the effect on UFSAR analyses of the requested reduction in Catawba Unit 1 required minimum measured RCS flow rate from 388,000 gpm to 384,000 gpm.

For the purposes of this evaluation, the UFSAR accident analyses are divided into three categories. For each category, the evaluation of the proposed RCS flow rate reduction is approached differently.

Category 1: Transients bounded by current RCS flow assumption

In the first category of analyses, the RCS flow rate assumed in the current analysis of record (AOR) is based on either the mechanical design flow of 420,000 gpm (where maximum RCS flows are conservative) or the thermal design flow of 382,000 gpm (where minimum RCS flows are conservative). Therefore, a change in the minimum RCS total flow rate limit from 388,000 gpm to 384,000 gpm would have no impact on the analysis.

1A. LOCA Blowdown Reactor Vessel and Loop Forces (UFSAR Section 3.6.4.1 and 3.9.1.5)
1B. Containment Functional Design (UFSAR Section 6.2.1)
1C. Feedwater System Malfunction Causing an Increase in Feedwater Flow (zero power case) (UFSAR Section 15.1.2)
1D. Turbine Trip (UFSAR Section 15.2.3)
1E. Loss of Non-Emergency AC Power to the Station Auxiliaries (UFSAR Section 15.2.6)
1F. Anticipated Transients Without Trip (UFSAR Section 15.8)

Category 2: Inapplicable transients/transients bounded by another transient/transients insensitive to RCS flow

In the second category of analyses, it is determined that the analysis itself is not applicable to Catawba (e.g., BWR transients) or is addressed in the UFSAR as being bounded by other analyzed transients. The proposed RCS flow reduction has no impact on these analyses.

In addition, many of the analyses are not sensitive to the reduction in RCS total flow, and a conclusion is reached that the 4000 gpm flow reduction does not affect the transient. Therefore, the following events are either not applicable, bounded by another accident, or are unaffected by the decrease of minimum RCS total flow rate to 384,000 gpm because RCS flow is not a factor in the event:

2A. Feedwater System Malfunctions that Result in a Reduction in Feedwater Temperature (UFSAR Section 15.1.1) The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Section 15.1.2) or the increase in secondary steam flow event (Section 15.1.3). Based on results

presented in Sections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

2B. Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve (UFSAR Section 15.1.4) The analyses performed assuming a rupture of a main steam line are given in Section 15.1.5. The main steam line break analysis bounds the inadvertent opening of a steam generator relief or safety valve analysis.

2C. Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (UFSAR Section 15.2.1, not applicable)

2D. Loss of External Load (UFSAR Section 15.2.2) A loss of external load event results in a Nuclear Steam Supply System (NSSS) transient that is less severe than a turbine trip event (see Section 15.2.3).

2E. Inadvertent Closure of Main Steam Isolation Valves (UFSAR Section 15.2.4) The longer closing time of the MSIVs, relative to the turbine stop valve closure time, offsets the effects of the smaller steam piping volume, and therefore the MSIV closure event is less severe than a turbine trip event. Turbine trips are discussed in Section 15.2.3.

2F. Loss of Condenser Vacuum and Other Event Causing a Turbine Trip (UFSAR Section 15.2.5) Since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum.

2G. Feedwater System Pipe Break (UFSAR Section 15.2.8) (short-term) The short-term results for the feedwater line break analysis are bounded by the analysis for a loss of normal feedwater flow (UFSAR Section 15.2.7).

2H. Reactor Coolant Pump Shaft Break (UFSAR Section 15.3.4) The consequences of a reactor coolant pump shaft break are similar to those calculated for the locked rotor incident (see UFSAR Section 15.3.3).

2I. Dropped Rod Cluster Control Assembly (RCCA) Bank (UFSAR Section 15.4.3b) The results for a dropped RCCA bank are bounded by the analysis presented for one or more dropped rods.

2J. Statically Misaligned RCCA (UFSAR Section 15.4.3c) The results for a statically misaligned RCCA are bounded by the analysis presented for one or more dropped rods (see UFSAR Section 15.4.3a) and for the Single RCCA Withdrawal (UFSAR Section 15.4.3d). DNBR does not fall below the limit value for the RCCA misalignment incident.

2K. BWR Transient (UFSAR Section 15.4.5, not applicable)

2L. Chemical Volume and Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant System (UFSAR Section 15.4.6, unaffected by change in RCS flow)

2M. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (UFSAR Section 15.4.7, unaffected by change in RCS flow)

2N. BWR Transient (UFSAR Section 15.4.9, not applicable)

2O. Inadvertent Operation of Emergency Core Cooling System (ECCS) During Power Operation (UFSAR Section 15.5.1) The DNB result of this transient is bounded by the inadvertent opening of a pressurizer safety or relief valve transient (see UFSAR Section 15.6.1).

2P. Chemical Volume and Control System Malfunction that Increases Reactor Coolant Inventory (UFSAR Section 15.5.2) An increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS is analyzed in

Section 15.4.6. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1.

2Q. A Number of BWR Transients (UFSAR Section 15.5.3, not applicable)

2R. Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment (UFSAR Section 15.6.2) The limiting case for this accident is a break occurring in the Chemical Volume and Control System (CVCS). The transient results would be unaffected by the proposed change in RCS flow.

2S. Spectrum of BWR Steam System Piping Failures Outside Containment (UFSAR Section 15.6.4, not applicable)

2T. A Number of BWR Transients (UFSAR Section 15.6.6, not applicable)

2U. All Radiological Accidents (UFSAR Sections 15.7, 15.9, and 15.10) These accidents are not impacted by changes in RCS flow.

Category 3: Evaluated Transients

The following summarizes the conclusions for the final category of safety analyses that have been re-evaluated with the reduced minimum RCS total flow rate of 384,000 gpm. Each of the safety analyses was individually reviewed to determine if the results of the analyses are affected by a 4000 gpm reduction in RCS total flow.

In a subgroup of this category of analyses, the current AOR assumes a conservatively high core bypass flow of 8.5% of the total RCS flow. In the AOR for these transients, the flow passing through the core is 355,020 gpm (388,000 gpm minus the 8.5% bypass flow). For efficiency, each of these AORs was performed to bound all Duke Energy units with the Babcock and Wilcox feedring steam generator design (McGuire Units 1 and 2 and Catawba Unit 1). This 8.5% bypass flow assumption conservatively bounds the unit-specific core bypass flows for all three of these units. Therefore, each accident has a single AOR for this category of transients, which applies to all three of these units.

However, the unit-specific calculated core bypass flow for Catawba Unit 1 is 6.49%. At a total RCS flow rate of 384,000 gpm, the flow passing through the core is 359,078 gpm (384,000 gpm minus the 6.49% core bypass flow). These flow calculations are summarized below:

Catawba Unit 1	Total RCS Flow [gpm]	Core Bypass Flow Assumed [%]	Net RCS Flow Through Core [gpm]
Existing AOR	388,000	8.5	355,020
With revised RCS flow	384,000	6.49	359,078

For the analyses listed below, core cooling is a concern and lower core flows are conservative. For these events, when the Catawba Unit 1 specific core bypass flow is considered, the existing AOR (with a net core flow of 355,020 gpm) remains bounding for Catawba Unit 1 with an RCS flow rate of 384,000 gpm (net core flow of 359,078 gpm). Therefore, no re-analysis was necessary since the existing AOR remains bounding:

3A. Feedwater Malfunction Causing an Increase in Feedwater Flow (full power case) (UFSAR Section 15.1.2)

3B. Partial Loss of Coolant Flow (UFSAR Section 15.3.1)
3C. Complete Loss of Coolant Flow (UFSAR Section 15.3.2)
3D. Reactor Coolant Pump Shaft Seizure – Locked Rotor (DNB) (UFSAR Section 15.3.3)
3E. Uncontrolled RCCA Bank Withdrawal at Power (DNB) (UFSAR Section 15.4.2)
3F. Dropped RCCA Rod (UFSAR Section 15.4.3a)
3G. Single Rod Withdrawal (UFSAR Section 15.4.3d)
3H. Spectrum of Rod Cluster Control Assembly Ejection Accidents (UFSAR Section 15.4.8)
3I. Inadvertent Opening of a Pressurizer Safety or Relief Valve (UFSAR Section 15.6.1)
3J. Steam Generator Tube Rupture (DNB) (UFSAR Section 15.6.3)

For the balance of the analyses, other evaluations were performed as described below:

3K. Excessive Increase in Secondary Steam Flow (UFSAR Section 15.1.3)
3L. Steam System Piping Failure (UFSAR Section 15.1.5)
3M. Loss of Normal Feedwater Flow (UFSAR Section 15.2.7)
3N. Feedwater System Pipe Break (long-term) (UFSAR Section 15.2.8)
3O. Reactor Coolant Pump Shaft Seizure – Locked Rotor (peak RCS pressure) (UFSAR Section 15.3.3)
3P. Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR Section 15.4.1)
3Q. Uncontrolled RCCA Bank Withdrawal at Power (peak RCS pressure) (UFSAR Section 15.4.2)
3R. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (UFSAR Section 15.4.4)
3S. Steam Generator Tube Rupture (UFSAR Section 15.6.3)
3T. Loss of Coolant Accidents (UFSAR Section 15.6.5)

These events are summarized below:

3K. Excessive Increase in Secondary Steam Flow (UFSAR Section 15.1.3)
The increase in steam flow accident was originally evaluated at an RCS flow of 382,000 gpm, and it was concluded that approximately 6% margin to the overpower delta temperature trip existed during the event. Likewise, the cases analyzed at 388,000 gpm also concluded that at least 6% margin to the overpower delta temperature trip existed. Therefore, the analysis is not sensitive to RCS flow. The proposed minimum RCS flow rate of 384,000 gpm is determined to be acceptable.

3L. Steam System Piping Failure (UFSAR 15.1.5)
This event was re-analyzed at a minimum RCS flow rate of 384,000 gpm to demonstrate DNB does not occur. The minimum DNBR was calculated to be 1.784, which is well above the W-3S Critical Heat Flux (CHF) correlation limit of 1.45.

3M. Loss of Normal Feedwater Flow (UFSAR Section 15.2.7)
The Loss of Normal Feedwater Flow short-term core cooling analysis was performed at the proposed RCS flow rate of 384,000 gpm with a conservatively

high 8.5% core bypass flow assumption. The minimum DNBR was calculated to be 2.131, which is well above the WRB-2M CHF correlation limit of 1.45.

The long-term core cooling analyses have significant margin to the acceptance criteria, which is based on maintaining adequate decay heat removal. The long-term core cooling AOR assumes an RCS flow rate of 388,000 gpm, with auxiliary feedwater flows conservatively bounding the actual Catawba capacity by at least 10%. In the analysis, subcooling of at least 40 degrees F was maintained throughout the transient. Therefore, it is concluded that the proposed reduction in RCS flow rate to 384,000 gpm will have an inconsequential impact on this transient.

3N. Feedwater System Pipe Break (UFSAR Section 15.2.8)
The feedwater line break short-term analysis is bounded by the analysis documented in UFSAR Section 15.2.7, which was satisfactorily analyzed at 384,000 gpm.

The long-term core cooling analysis indicates adequate hot leg subcooling ensuring long-term core cooling capability. The key parameters in this analysis are the heat balance and the ability of auxiliary feedwater to maintain adequate hot leg subcooling. The proposed change in RCS flow does not alter either the heat sources (core decay heat and reactor coolant pump heat) or heat sink parameters (auxiliary feedwater flow and temperature). Therefore, it is concluded that the current AOR will not be significantly impacted by the proposed change in RCS flow rate.

3O. Reactor Coolant Pump Shaft Seizure – Locked Rotor (peak RCS pressure) (UFSAR Section 15.3.3)
This event was re-analyzed for a minimum RCS flow rate of 384,000 gpm to calculate the peak RCS pressure for this accident. The resulting peak RCS pressure of 2529 psig is well below the design value of 2735 psig.

3P. Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR Section 15.4.1)
This analysis addresses adequate core cooling (Case 1) and peak RCS pressure (Case 2) acceptance criteria. Case 1 assumes RCS flow with three reactor coolant pumps operational based on nominal flow of 388,000 gpm. The delivered flow per pump increases to between 104% and 107% with one pump not running. The 1% reduction in nominal full flow to 384,000 gpm is therefore adequately accommodated by the three-pump assumption. Case 2 was re-analyzed with a total assumed RCS flow of 384,000 gpm. The results of the current AOR show about 130 psi of margin relative to the design value of 2735 psig.

Therefore, it can be concluded that the current AORs for this transient are not impacted by the proposed reduction in RCS flow.

3Q. Uncontrolled RCCA Bank Withdrawal at Power (peak RCS pressure) (UFSAR Section 15.4.2)

This event was re-analyzed for a minimum RCS flow rate of 384,000 gpm to calculate the peak RCS pressure for this accident. The resulting peak RCS pressure of 2708 psig is below the design value of 2735 psig.

3R. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (UFSAR Section 15.4.4)
This analysis is initiated from an assumed initial power level of 50%. The analysis assumes RCS flow with three reactor coolant pumps initially operating. The current AOR is based on the calculated three-pump flow, starting from a nominal full power and four-pump flow of 388,000 gpm. A historical analysis with different fuel types was based on a nominal full power RCS flow of 382,000 gpm. Both cases showed similar core responses and considerable margin to the limiting DNBR in both cases. Based on the similarity of results between the two cases, it is concluded that the analysis results are insensitive to RCS flow. The proposed reduction in total RCS flow to 384,000 gpm will have no significant impact on the analysis results.

3S. Steam Generator Tube Rupture (UFSAR Section 15.6.3)
This event is examined for DNB, radiological consequences, and steam generator overfill. The overfill analysis determined the assumed RCS flow rate to be inconsequential. The DNB analysis was performed at 388,000 gpm and is evaluated above (item 3J). The dose input analysis was performed at 390,000 gpm plus uncertainty. Prior to trip, the marginal reduction in flow has an inconsequential impact on the analysis. Upon manual reactor trip, a loss of offsite power is assumed to trip the reactor coolant pumps and the impact of the small change in initial RCS flow has no impact on the balance of the transient.

3T. Loss of Coolant Accidents (UFSAR Section 15.6.5)
The Large Break LOCA (LBLOCA) analysis was evaluated by Westinghouse for the proposed reduction in RCS total flow rate. This analysis, which was applicable for both Catawba units, determined that the variations in the global model calculations are such that the 95th percentile peak clad temperature is not impacted.

The significant factors in the Small Break LOCA (SBLOCA) analysis are decay heat, RCS mass, break flow, and ECCS delivery. Three of these variables are completely unrelated to initial RCS flow, and the fourth (RCS mass) is insignificantly affected. Changes in initial RCS flow may possibly change the thermal mass in the steam generators, and thus the operating pressure, which would in turn affect the time at which the safety valves would lift. However, this second order effect is considered insignificant to the SBLOCA outcome.

Refer to the summary Table 1 at the end of this attachment for a synopsis of the analysis results.

Basis for Proposed Changes (Catawba Unit 2)

The following summarizes the effect on UFSAR analyses of the requested reduction in Catawba Unit 2 required minimum measured RCS flow rate from 390,000 gpm to 387,000 gpm.

For the purposes of this evaluation, the UFSAR accident analyses are divided into three categories. For each category, the evaluation of the proposed RCS flow rate reduction is approached differently.

Category 1: Transients bounded by current RCS flow assumption

In the first category of analyses, the RCS flow rate assumed in the current AOR is based on either the mechanical design flow of 420,000 gpm (where maximum RCS flows are conservative) or the thermal design flow of 382,000 gpm (where minimum RCS flows are conservative). Therefore, a change in the minimum RCS total flow rate limit from 390,000 gpm to 387,000 gpm would have no impact on the analysis.

4A. LOCA Blowdown Reactor Vessel and Loop Forces (UFSAR Section 3.6.4.1 and 3.9.1.5)
4B. Containment Functional Design (UFSAR Section 6.2.1)
4C. Feedwater System Malfunction Causing an Increase in Feedwater Flow (zero power case) (UFSAR Section 15.1.2)
4D. Excessive Increase in Secondary Steam Flow (UFSAR Section 15.1.3)
4E. Turbine Trip (UFSAR Section 15.2.3)
4F. Loss of Non-Emergency AC Power to the Station Auxiliaries (UFSAR Section 15.2.6)
4G. Loss of Normal Feedwater Flow (long-term analysis) (UFSAR Section 15.2.7)
4H. Feedwater System Pipe Break (long-term analysis) (UFSAR Section 15.2.8)
4I. Anticipated Transients Without Trip (UFSAR Section 15.8)

Category 2: Inapplicable transients/transients bounded by another transient/transients insensitive to RCS flow

In the second category of analyses, it is determined that the analysis itself is not applicable to Catawba (e.g., BWR transients) or is addressed in the UFSAR as being bounded by other analyzed transients. The proposed RCS flow reduction has no impact on these analyses.

In addition, many of the analyses are not sensitive to the reduction in RCS total flow, and a conclusion is reached that the 3000 gpm flow reduction does not affect the transient. Therefore, the following events are either not applicable, bounded by another accident, or are unaffected by the decrease of minimum RCS total flow rate to 387,000 gpm because RCS flow is not a factor in the event:

5A. Feedwater System Malfunctions that Result in a Reduction in Feedwater Temperature (UFSAR Section 15.1.1) The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Section 15.1.2) or the increase in secondary steam flow event (Section 15.1.3). Based on results

presented in Sections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

5B. Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve (UFSAR Section 15.1.4) The analyses performed assuming a rupture of a main steam line are given in Section 15.1.5. The main steam line break analysis bounds the inadvertent opening of a steam generator relief or safety valve analysis.

5C. Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (UFSAR Section 15.2.1, not applicable)

5D. Loss of External Load (UFSAR Section 15.2.2) A loss of external load event results in an NSSS transient that is less severe than a turbine trip event (see Section 15.2.3).

5E. Inadvertent Closure of Main Steam Isolation Valves (UFSAR Section 15.2.4) The longer closing time of the MSIVs, relative to the turbine stop valve closure time, offsets the effects of the smaller steam piping volume, and therefore the MSIV closure event is less severe than a turbine trip event. Turbine trips are discussed in Section 15.2.3.

5F. Loss of Condenser Vacuum and Other Event Causing a Turbine Trip (UFSAR Section 15.2.5) Since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum.

5G. Feedwater System Pipe Break (UFSAR Section 15.2.8) (short-term) The short-term results for the feedwater line break analysis are bounded by the analysis for a loss of normal feedwater flow (UFSAR Section 15.2.7).

5H. Reactor Coolant Pump Shaft Break (UFSAR Section 15.3.4) The consequences of a reactor coolant pump shaft break are similar to those calculated for the locked rotor incident (see Section 15.3.3).

5I. Dropped RCCA Bank (UFSAR Section 15.4.3b) The results for a dropped RCCA bank are bounded by the analysis presented for one or more dropped rods.

5J. Statically Misaligned RCCA (UFSAR Section 15.4.3c) The results for a statically misaligned RCCA are bounded by the analysis presented for one or more dropped rods (see UFSAR Section 15.4.3a) and for the Single RCCA Withdrawal (UFSAR Section 15.4.3d). DNBR does not fall below the limit value for the RCCA misalignment incident.

5K. BWR Transient (UFSAR Section 15.4.5, not applicable)

5L. Chemical Volume and Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant System (UFSAR Section 15.4.6, unaffected by change in RCS flow)

5M. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (UFSAR Section 15.4.7, unaffected by change in RCS flow)

5N. BWR Transient (UFSAR Section 15.4.9, not applicable)

5O. Inadvertent Operation of ECCS During Power Operation (UFSAR Section 15.5.1) The DNB results of this transient are bounded by the inadvertent opening of a pressurizer safety or relief valve transient (see UFSAR Section 15.6.1).

5P. Chemical Volume and Control System Malfunction that Increases Reactor Coolant Inventory (UFSAR Section 15.5.2) An increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS is analyzed in Section 15.4.6. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1.

5Q. A Number of BWR Transients (UFSAR Section 15.5.3, not applicable)

5R. Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment (UFSAR Section 15.6.2) The limiting case for this accident is a break occurring in the CVCS. The transient results would be unaffected by the proposed change in RCS flow.
5S. Spectrum of BWR Steam System Piping Failures Outside Containment (UFSAR Section 15.6.4, not applicable)
5T. A Number of BWR Transients (UFSAR Section 15.6.6, not applicable)
5U. All Radiological Accidents (UFSAR Sections 15.7, 15.9, and 15.10) These accidents are not impacted by changes in RCS flow.

Category 3: Evaluated Transients

The following summarizes the results of the final category of safety analyses that have been either re-evaluated or re-analyzed with the reduced minimum RCS total flow rate of 387,000 gpm. Each of the safety analyses was individually reviewed to determine if the results of the analyses are affected by a 3000 gpm reduction in RCS total flow.

In a subgroup of this category of analyses, the current AOR for Catawba Unit 2 assumes a conservatively high core bypass flow of 7.5% of the total RCS flow. In the AOR for each of these transients, the flow passing through the core is 360,750 gpm (390,000 gpm minus the 7.5% bypass flow).

However, the unit-specific calculated core bypass flow for Catawba Unit 2 is 6.71%. At a total RCS flow rate of 387,000 gpm, the flow passing through the core is 361,032 gpm (387,000 gpm minus the 6.71% core bypass flow). These flow calculations are summarized below:

Catawba Unit 2	Total RCS Flow [gpm]	Core Bypass Flow Assumed [%]	Net RCS Flow Through Core [gpm]
Existing AOR	390,000	7.5	360,750
With revised RCS flow	387,000	6.71	361,032

For the analyses listed below, core cooling is a concern and lower core flows are conservative. For these events, when the Catawba Unit 2 specific core bypass flow is considered, the existing AOR (with a net core flow of 360,750 gpm) remains bounding for Catawba Unit 2 with an RCS flow rate of 387,000 gpm (net core flow of 361,032 gpm). Therefore, no re-analysis was necessary since the existing AOR remains bounding:

6A. Feedwater Malfunction Causing an Increase in Feedwater Flow (full power case) (UFSAR Section 15.1.2)
6B. Loss of Normal Feedwater Flow (short-term analysis) (UFSAR Section 15.2.7)
6C. Partial Loss of Coolant Flow (UFSAR Section 15.3.1)
6D. Complete Loss of Coolant Flow (UFSAR Section 15.3.2)
6E. Reactor Coolant Pump Shaft Seizure – Locked Rotor (DNB) (UFSAR Section 15.3.3)
6F. Uncontrolled RCCA Bank Withdrawal at Power (DNB) (UFSAR Section 15.4.2)
6G. Dropped RCCA Rod (UFSAR Section 15.4.3a)

6H. Single Rod Withdrawal (UFSAR Section 15.4.3d)
6I. Spectrum of Rod Cluster Control Assembly Ejection Accidents (UFSAR Section 15.4.8)
6J. Inadvertent Opening of a Pressurizer Safety or Relief Valve (UFSAR Section 15.6.1)
6K. Steam Generator Tube Rupture (DNB) (UFSAR 15.6.3)

For the balance of the analyses, other evaluations were performed as described below:

6L. Steam System Piping Failure (UFSAR Section 15.1.5)
6M. Reactor Coolant Pump Shaft Seizure – Locked Rotor (peak RCS pressure) (UFSAR Section 15.3.3)
6N. Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR Section 15.4.1)
6O. Uncontrolled RCCA Bank Withdrawal at Power (peak RCS pressure) (UFSAR Section 15.4.2)
6P. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (UFSAR Section 15.4.4)
6Q. Steam Generator Tube Rupture (dose and overfill) (UFSAR Section 15.6.3)
6R. Loss of Coolant Accidents (UFSAR Section 15.6.5)

These events are summarized below:

6L. Steam System Piping Failure (UFSAR Section 15.1.5)
This event was re-analyzed at a minimum RCS flow rate of 387,000 gpm to demonstrate DNB does not occur. The minimum DNBR was calculated to be 1.987, which is well above the W-3S CHF correlation limit of 1.45.

6M. Reactor Coolant Pump Shaft Seizure – Locked Rotor (peak RCS pressure) (UFSAR Section 15.3.3)
This event was re-analyzed for a minimum RCS flow rate of 387,000 gpm to calculate the peak RCS pressure for this accident. The resulting peak RCS pressure of 2536 psig is well below the design value of 2735 psig.

6N. Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR Section 15.4.1)
This analysis addresses adequate core cooling (Case 1) and peak RCS pressure (Case 2) acceptance criteria. Both Case 1 and Case 2 have sufficient margin to their respective acceptance criteria. The minimum DNBR for Case 1 for the accident analysis at zero power is 3.395, which is well above the acceptance criteria of 1.45. The results of Case 2 showed about 150 psi of margin relative to the design value of 2735 psig.

It is concluded that a reduction in RCS flow from 390,000 gpm to 387,000 gpm would not have an appreciable impact on the results in either case due to the amount of margin between the existing AOR results and the acceptance criteria.

6O. Uncontrolled RCCA Bank Withdrawal at Power (peak RCS pressure) (UFSAR Section 15.4.2)

This event was re-analyzed for a minimum RCS flow rate of 387,000 gpm to calculate the peak RCS pressure for this accident. The resulting peak RCS pressure of 2709 psig is below the design value of 2735 psig.

6P. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (UFSAR Section 15.4.4)
This analysis is initiated from an assumed initial power level of 50%. The analysis assumes RCS flow with three reactor coolant pumps initially operating. The current AOR is based on the calculated three-pump flow, starting from a nominal full power and four-pump flow of 388,000 gpm. A historical analysis with different fuel types was based on a nominal full power RCS flow of 382,000 gpm. Both cases showed similar core responses and considerable margin to the limiting DNBR in both cases. Based on the similarity of results between the two cases, it is concluded that the analysis results are insensitive to RCS flow. The proposed reduction in total RCS flow to 387,000 gpm will have no significant impact on the analysis results.

6Q. Steam Generator Tube Rupture (UFSAR Section 15.6.3)
This event is examined for DNB, radiological consequences, and steam generator overfill. The overfill analysis determined the assumed RCS flow rate to be inconsequential. The DNB analysis was performed at 388,000 gpm and is evaluated above (item 6K). The dose input analysis was performed at 390,000 gpm plus uncertainty. Prior to trip, the marginal reduction in flow has an inconsequential impact on the analysis. Upon manual reactor trip, a loss of offsite power is assumed to trip the reactor coolant pumps and the impact of the small change in initial RCS flow has no impact on the balance of the transient.

6R. Loss of Coolant Accidents (UFSAR Section 15.6.5)
The LBLOCA analysis was evaluated by Westinghouse for the proposed reduction in RCS total flow rate. (The Catawba Unit 2 proposed RCS flow reduction is only 3000 gpm, so the Westinghouse evaluation is applicable for both Catawba units.) It was determined that the variations in the global model calculations are such that the 95th percentile peak clad temperature is not impacted by this proposed RCS flow reduction.

The significant factors in the SBLOCA analysis are decay heat, RCS mass, break flow, and ECCS delivery. Three of these variables are completely unrelated to initial RCS flow, and the fourth (RCS mass) is insignificantly affected. Changes in initial RCS flow may possibly change the thermal mass in the steam generators, and thus the operating pressure, which would in turn affect the time at which the safety valves would lift. However, this second order effect is considered insignificant to the SBLOCA outcome.

Refer to the summary Table 2 at the end of this attachment for a synopsis of the analysis results.

<u>Other UFSAR analyses with common evaluations for Units 1 and 2</u>

<u>Best Estimate/Thermal/Mechanical Design Flows (UFSAR Section 5.1)</u>

The RCS Best Estimate Design Flow, Thermal Design Flow, and Mechanical Design Flow values for Catawba Units 1 and 2 are discussed in Section 5.1 of the UFSAR. (Unit-specific values are given in Table 5-1.) None of these design flows are impacted by the proposed reduction in the TS 3.4.1 minimum RCS flows discussed here.

Operation at 98% thermal power/99% RCS total flow rate (TS 3.4.1, Required Action B.1)

Catawba is allowed by TS 3.4.1 to operate at RCS flows between 99% and 100% of the limit specified in the COLR, as long as Reactor Thermal Power is reduced to $\leq$ 98% of the Rated Thermal Power level. The proposed reduction in TS 3.4.1 minimum flows would therefore allow operation at RCS flows as low as 380,160 gpm (Unit 1) and 383,130 gpm (Unit 2) as long as the required reduction in Reactor Thermal Power occurs.

This "stair step" flow reduction was considered as part of the evaluation of the entire list of accident analyses described above for Catawba Units 1 and 2. The only accidents where operation with the 2%/1% "stair step" reduction in RCS flow/Reactor Thermal Power may represent a bounding condition have been evaluated for the additional reduction in RCS minimum flow proposed here. There is no impact on the TS 3.4.1 Required Action B.1 requirement due to the proposed reduction in RCS minimum flow.

Conclusion

The UFSAR transient and accident analyses have been reviewed for a decrease in the TS RCS minimum measured flow rate from 388,000 gpm to 384,000 gpm for Catawba Unit 1, and for a decrease from 390,000 gpm to 387,000 gpm for Catawba Unit 2. Analysis revisions were performed for those events sensitive to a decrease in RCS flow. The acceptance criteria were met for all events. Therefore, the reductions in RCS minimum measured flow described above are justified.

5. REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36(b) states that the TS will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34. The minimum allowable RCS flow rates for Unit 1 and Unit 2 that are located in the TS are supported by existing safety analyses that demonstrate that the units can meet safety analysis acceptance criteria at the revised flow rates. All applicable safety analyses either are not affected by the proposed changes or have been reanalyzed at the proposed new values. 10 CFR 50, Appendix A, General Design Criterion 10 (Reactor design) states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. 10 CFR 50, Appendix A, General Design Criterion 15 (Reactor coolant system design) states that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The reanalyzed safety analyses demonstrate that these two criteria will continue to be met with the proposed changes.

5.2 Precedent

On December 19, 2003 (ADAMS Accession Numbers ML033570127 and ML033580393), the NRC approved Amendments 210 and 204 for Catawba Units 1 and 2, respectively. These amendments changed the TS for both units to relocate certain RCS cycle-specific parameter limits from the TS to the Core Operating Limits Report (COLR), and revised the minimum allowable RCS flow rate for Unit 1 from 390,000 gpm to 388,000 gpm. The approved amendments reference all of the Duke Energy correspondence associated with this license amendment request. This correspondence included the following:

Initial submittal dated March 24, 2003 (ADAMS Accession Number ML030920393)

Supplemental submittal dated June 25, 2003 (ADAMS Accession Number ML032130497)

Supplemental submittal dated October 15, 2003 (ADAMS Accession Number ML033100103)

5.3 No Significant Hazards Consideration

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no

significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. Involve a significant reduction in a margin of safety.

First Standard

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

Response: No.

The reduction in Catawba Unit 1 Reactor Coolant System (RCS) minimum measured flow from 388,000 gpm to 384,000 gpm and the reduction in Catawba Unit 2 RCS minimum measured flow from 390,000 gpm to 387,000 gpm will not change the probability of actuation of any Engineered Safeguard Feature or any other device. The consequences of previously analyzed accidents have been found to be insignificantly different when these reduced flow rates are assumed. The system transient response is not affected by the initial RCS flow assumption unless the initial assumption is so low as to impair the steady state core cooling capability or the steam generator heat transfer capability. This is clearly not the case with the small proposed reductions in RCS flow. The proposed changes will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed changes. The proposed amendments will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the Updated Final Safety Analysis Report (UFSAR). Therefore, the proposed amendments do not result in the increase in the probability or consequences of an accident previously evaluated.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

These changes do 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 NRC approval of this amendment request. No changes are being made to the facility which would introduce any new accident

causal mechanisms. This amendment request does not impact any plant systems that are accident initiators.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

Implementation of these amendments would not involve a significant reduction in the margin of safety. The decreases in Catawba Unit 1 and Unit 2 RCS minimum measured flow have been analyzed and found to have an insignificant effect on the applicable transient analyses as described in the UFSAR. Margin of safety is related to the confidence of the fission product barriers being able to perform their accident mitigating functions. These fission product barriers include the fuel cladding, the RCS, and the containment. The proposed amendments will have no impact upon the ability of these barriers to function as designed. Consequently, no safety margins will be impacted.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendments do not involve a significant hazards consideration.

5.4 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 Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of these amendments will have no adverse impact upon the Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, these amendments to the Catawba TS meet the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7. REFERENCES

7.1 Letter from Sean Peters, NRC to D.M. Jamil, Duke Energy, "CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB8359 AND MB8360)", December 19, 2003, ADAMS Accession Numbers ML033570127 and ML033580393.

TABLE 1
Summary of UFSAR Transient Information for Use in Proposed Catawba Unit 1 RCS Flow Reduction (384,000 gpm)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(1) 15.1.1	Feedwater System Malfunction that Results in a Reduction in Feedwater Temperature	2A Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			Transient results are less severe than 15.1.2 and 15.1.3 – not analyzed since transient is bounded
(2) 15.1.2	Feedwater System Malfunction Causing an Increase in Feedwater Flow	1C (zero power) Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low	353,350 gpm (zero power case)	NA	Zero power case – 382,000 gpm total RCS flow rate - bounds proposed RCS flow rate of 384,000 gpm total flow
		3A (full power) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm (full power case)	359,078 gpm (full power case)	Full power case – Evaluated
(3) 15.1.3	Excessive Increase in Secondary Steam Flow	3K Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
(4) 15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	2B Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			Results are bounded by those in Main Steam Line Break Analysis (15.1.5)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3) Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
(5) 15.1.5	Steam System Piping Failure	3L Evaluated	References 6 and 10	References 7 and 10	Low	337,590 gpm	337,590 gpm	Reanalyzed with RCS total flow of 384,000 gpm minus flow uncertainty, stairstep flow reduction, and bounding high core bypass flow
(6) 15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	2C Not applicable to Catawba	NA	NA	NA			NA
(7) 15.2.2	Loss of External Load	2D Bounded by other transient results	Reference 8	Reference 9	Low			Results are bounded by those in Turbine Trip (15.2.3)
(8) 15.2.3	Turbine Trip	1D Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low (Pk primary)	(Peak primary pressure) 345,576 gpm	NA	Peak primary pressure: 382,000 gpm (total)
					High (Pk secondary)	(Peak secondary pressure) 388,500 gpm	NA	Peak secondary pressure: 420,000 gpm (total)
(9) 15.2.4	Inadvertent Closure of Main Steam Isolation Valves	2E Bounded by other transient results	NA	NA	Low			Results are bounded by those in Turbine Trip (15.2.3)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(10) 15.2.5	Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	2F Bounded by other transient results	NA	NA	Low			Results are bounded by those in Turbine Trip (15.2.3)
(11) 15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	1E Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low	345,576 gpm	NA	Current AOR assumes conservatively low total RCS flow of 382,000 gpm
(12) 15.2.7	Loss of Normal Feedwater Flow	3M Evaluated	Reference 1	References 2, 3, 4, and 5	(Short-term) Low	(Short-term) 351,360 gpm	(Short-term) 351,360 gpm	(Short-term) Reanalyzed with 384,000 gpm total RCS flow
					(Long-term) Low	(Long-term) 343,415 gpm	(Long-term) 343,415 gpm	(Long-term) Current AOR assumes 388,000 gpm total RCS flow, but has substantial margin to success criteria
(13) 15.2.8	Feedwater System Pipe Break	2G (Short-term) Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	(Short-term) Low			(Short-term) Results are bounded by those in Loss of Normal Feedwater Flow (15.2.7)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
		3N (Long-term) Evaluated	Reference 1	References 2, 3, 4, and 5	(Long-term) Low	(Long-term) 352,813 gpm	(Long-term) 352,813 gpm	(Long-term) Current AOR assumes 390,000 gpm total RCS flow, but results are insensitive to assumed RCS flow
(14) 15.3.1	Partial Loss of Forced Reactor Coolant Flow	3B Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
(15) 15.3.2	Complete Loss of Forced Reactor Coolant Flow	3C Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
(16) 15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	3D (DNB) Evaluated	T&H: Reference 1 Dose: Reference 10	T&H: References 2, 3, 4, and 5 Dose: Reference 10	Low	355,020 gpm	359,078 gpm	(DNB) Evaluated
		3O (peak RCS pressure) Evaluated	Reference 1	Reference 2, 3, 4, and 5	Low	341,001 gpm	NA	(peak RCS pressure) Reanalyzed with 384,000 gpm total RCS flow
(17) 15.3.4	Reactor Coolant Pump Shaft Break	2H Bounded by other transient results	NA	NA	Low			Results are similar to those calculated in Locked Rotor (15.3.3)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(18) 15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition	3P (DNB) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	(3 RCP) 274,146 gpm	(3 RCP) 274,146 gpm	(3 RCP - core cooling) – AOR has total RCS flow rate based on 388,000 gpm and 3 pump operation at HZP; proposed decrease in RCS flow has no impact on 3 RCP case
		3P (Peak RCS pressure) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	(Peak RCS pressure) 337,591 gpm	(Peak RCS pressure) 337,591 gpm	(Peak RCS pressure) – Reanalyzed with RCS total flow of 384,000 gpm
(19) 15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	3E (DNB) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
		3Q (Peak RCS pressure) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	337,590 gpm	NA	(Peak RCS pressure) – Reanalyzed with RCS total flow of 384,000 gpm
(20) 15.4.3	Rod Cluster Control Assembly Misoperation (System Malfunction or	15.4.3a: 3F Evaluated	Reference 6	Reference 7	Low	355,020 gpm	359,078 gpm	15.4.3a: (One or more dropped RCCA rods): Evaluated

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3) Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
	Operator Error)	15.4.3b: 2I Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			15.4.3b: (Dropped RCCA bank): Results are bounded by case with one or more dropped rods (15.4.3a and 15.4.3c)
		15.4.3c: 2J Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			15.4.3c: (Statically misaligned RCCA): Results are bounded by the analysis of one or more dropped rods, or by single RCCA withdrawal (SUCR)
		15.4.3d: 3G Evaluated	Reference 1	References 2, 3, 4, and 5	Low	358,900 gpm	359,078 gpm	15.4.3d: (SUCR): Flow through the core: Evaluated
(21) 15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	3R Evaluated	Reference 1	References 2, 3, 4, and 5	Low	Based on 339,980 gpm initial core flow	Based on 339,980 gpm initial core flow	AOR has total RCS flow rate based on 388,000 gpm; proposed decrease in RCS flow has no impact for analysis with 3 RCP initial condition
(22) 15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor	2K (BWR Transient)	NA	NA	NA			NA

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
	Coolant Flow Rate							
(23) 15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	2L (Unaffected)	Reference 1	References 2, 3, 4, and 5	NA			NA
(24) 15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	2M (Unaffected)	Reference 1	References 2, 3, 4, and 5	NA			NA
(25) 15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	3H Evaluated	T&H: Reference 6 Dose: Reference 10	T&H: Reference 7 Dose: Reference 10	Low	343,621 gpm	344,655 gpm	Evaluated
(26) 15.4.9	Spectrum of Rod Drop Accidents (BWR)	2N (BWR Transient)	NA	NA	NA			NA
(27) 15.5.1	Inadvertent Operation of Emergency Core Cooling System During Power Operation	2O (Bounded by other transients)	Reference 1	References 2, 3, 4, and 5	NA			NA
(28) 15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	2P (Bounded by other transients)	NA	NA	NA			NA
(29) 15.5.3	A Number of BWR Transients	2Q (BWR Transient)	NA	NA	NA			NA

						Evaluated Cases (see Note 3)		
UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
(30) 15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	3I Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
(31) 15.6.2	Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment	2R (Unaffected)	Reference 10	Reference 10	NA			NA
(32) 15.6.3	Steam Generator Tube Failure	3J (DNB) Evaluated	T&H: Reference 1	T&H: References 2, 3, 4, and 5	Low	355,020 gpm	359,078 gpm	Evaluated
		3S (overfill, dose) Evaluated	Dose: Reference 10	Dose: Reference 10	Low	358,900 gpm (overfill) 343,414 gpm (dose)	NA	For the SG overfill and dose consequences of SGTR, the analysis results are not sensitive to the assumed RCS flow
(33) 15.6.4	Spectrum of BWR Steam System Piping Failures Outside Containment	2S (BWR Transient)	NA	NA	NA			NA
(34) 15.6.5	Loss-of-Coolant Accidents	3T Evaluated	T&H: References 11 and 12 Dose: References 13, 14, 15, 16, and 17	T&H: References 18 and 19 Dose: Reference 20	Low	LBLOCA/ SBLOCA: Based on 390,000 gpm total RCS flow rate	LBLOCA/SBLOCA: Based on 384,000 gpm total RCS flow rate	LBLOCA/SBLOCA consequences are insensitive to the proposed change in RCS flow

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3) Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
(35) 15.6.6	A Number of BWR Transients	2T (BWR Transient)	NA	NA	NA			NA
(36) 15.7.1	Radioactive Gas Waste System Leak or Failure	2U (Unaffected)	Reference 21	Reference 20	NA			NA
(37) 15.7.2	Radioactive Liquid Waste System Leak or Failure	2U (Unaffected)	Reference 10	Reference 10	NA			NA
(38) 15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failures	2U (Unaffected)	Reference 10	Reference 10	NA			NA
(39) 15.7.4	Fuel Handling Accidents in the Containment and Spent Fuel Storage Buildings	2U (Unaffected)	References 22, 23, 24, and 25	References 25 and 26	NA			NA
(40) 15.8	Anticipated Transients Without Trip	1F Bounded by current AOR	Reference 27	Reference 28	NA	Based on nominal total RCS flow of 377,600 gpm	NA	Current AOR assumes conservatively low total RCS flow of 377,600 gpm
(41) 15.9	Formerly Chapter 15 Appendix A - Models Used for Calculation of Accident Doses	2U (Unaffected)	NA	NA	NA			NA
(42) 15.10	Formerly Chapter 15 Appendix B - Supplementary Radiological Analyses	2U (Unaffected)	References 29, 30, and 31	References 9 and 32	NA			NA

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC, or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(43) 6.2.1.1.3.1 6.2.1.1.3.3 6.2.1.1.3.4 6.2.1.3.2 6.2.1.4	Containment Performance Analyses	1B Bounded by current AOR	Reference 33	References 34 and 35	High	388,500 gpm	NA	Current AOR assume conservatively high RCS flow of 420,000 gpm
(44) 6.2.1	Postulated Secondary System Pipe Rupture Outside Containment	1B Bounded by current AOR	References 33 and 36	References 34, 35, and 37	High	388,500 gpm	NA	Current AOR assumes conservatively high RCS flow of 420,000 gpm
(45) 3.6.2.2.1 3.9.1.4	LOCA Blowdown Reactor Vessel and Loop Forces	1A Bounded by current AOR	NA	NA	High	Based on total RCS flow rate of 398,000 gpm	NA	Current AOR assume conservatively high RCS flow of 398,000 gpm

NOTE 1: LAR Designations listed by Category pages B-1 to B-5.
NOTE 2: "Core Flow" represents net flow through core in analysis, defined as Total RCS flow minus Core Bypass Flow.
NOTE 3: Transients which are bounded by other transient results, not applicable for Catawba Unit 1, or unaffected by the proposed RCS flow reduction have shaded fields in this column.

REFERENCES:

1. DPC-NE-3002-A, Revision 4b, "McGuire and Catawba Nuclear Station UFSAR Chapter 15 System Transient Analysis Methodology", September 2010

2. Letter from Tim Reed (NRC) to H. B. Tucker (Duke) dated November 15, 1991, "Safety Evaluation on Topical Report DPC-NE-3002, "FSAR Chapter 15 System Transient Analysis Methodology," (TAC No. 66850)"

3. Letter from Robert Martin (NRC) to M. S. Tuckman (Duke) dated December 28, 1995, "Safety Evaluation for Revision 1 to Topical Report DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology" McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M89944, M89945, and M89946)"

4. Letter from Herbert Berkow (NRC) to M. S. Tuckman (Duke) dated April 26, 1996, "Safety Evaluation on Change to Topical Report DPC-NE-3002-A on Opening Characteristics of Safety Valves - McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M94405, M94406, M94407, and M94408)"

5. Letter from Chandu Patel (NRC) to G. R. Peterson (Duke) dated April 6, 2001, "Catawba Nuclear Station, Units 1 and 2 RE: Revision 4 to the Duke Energy Corporation Topical Report DPC-NE-3002-A, "UFSAR Chapter 15 Transient Analysis Methodology" (TAC Nos. MA8928 and MA8929)"

6. DPC-NE-3001-PA, Revision 0a, "McGuire and Catawba Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology", May 2009

7. Letter from Timothy Reed (NRC) to H. B. Tucker (Duke) dated November 15, 1991, "Safety Evaluation on Topical Report DPC-NE-3001, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" (TAC Nos. 75954/75955/75956/75957)"

8. Technical Information Document (TID) 14844, Calculation of Distance Factors for Power and Test Reactors, U.S. Atomic Energy Commission, 1962

9. 10 CFR Part 100, Section 100.11

10. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors"

11. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best-Estimate LOCA Analysis," March 1998.

12. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

13. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term," March 20, 2008.

14. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas: LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term Revision to Control Room Atmospheric Dispersion Factors," March 20, 2008.

15. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term. Response to Request for Additional Information," October 6, 2008.

16. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Response to Request for Additional Information related to the License Amendment Request for Implementation of Alternative Source Term," December 17, 2008.

17. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Response to Request for Additional Information related to the License Amendment Request (LAR) for Implementation of Alternative Source Term (AST)," February 12, 2009.

18. Letter from R. C. Jones (NRC) to N. J. Liparulo (Westinghouse) dated June 28, 1996, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Analysis".

19. Letter from R. C. Jones (NRC) to E. P. Rahe (Westinghouse) dated May 21, 1985, "Acceptance for Referencing of Licensing Topical Report WCAP-10054 (P), Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code".

20. Letter from John Stang (USNRC) to B.H. Hamilton (Duke), "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Adoption of the Alternative Source Term Radiological Analysis Methodology (TAC Nos. MD8400 and MD8401)," March 30, 2009.

21. Regulatory Guide (RG) 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I"

22. Letter from G. R. Peterson (Duke) to U.S. NRC dated December 20, 2005, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations"

23. Letter from G. R. Peterson (Duke) to U.S. NRC dated May 4, 2006, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations. Response to Request for Additional Information"

24. Letter from G. R. Peterson (Duke) to U.S. NRC dated August 31, 2006, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations. Additional Commitments Regarding Containment Closure Administrative Controls."

25. ISG-5, Revision 1 - "Confinement Evaluation", Spent Fuel Project Office, NRC

26. Letter from John Stang (NRC) to G. R. Peterson (Duke) dated December 22, 2006, "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Implementation of Alternative Source Term Methodology (TAC Nos. MC9746 AND MC9747)"

27. Letter from T. M. Anderson (Westinghouse) to S. H. Hanauer (NRC) dated December 30, 1979, "NS-TMA-2182, ATWS Submittal"

28. Letter from Darl S. Hood (NRC) to H. B. Tucker (Duke) dated November 6, 1987, "ATWS Rule (10 CFR 50.62) for McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 59081/59111/59112/64535)"

29. Letter from Hal Tucker (Duke) to Harold Denton (NRC) dated February 17, 1984, "McGuire Nuclear Station Docket Nos. 50-369, 50-370", Spent Fuel Pool Re-rack LAR

30. Letter from Hal Tucker (Duke) to Harold Denton (NRC) dated March 20, 1984, "McGuire Nuclear Station Docket Nos. 50-369, 50-370", Safety & Environmental Analysis for Spent Fuel Pool Re-rack

31. Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)"

32. Letter from Elinor Adensam (NRC) to Hal Tucker (Duke) dated September 24, 1984, "Issuance of Amendment No.35 to Facility Operating License NPF-9 and Amendment No. 16 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2"

33. DPC-NE-3004-PA, Revision 1, McGuire and Catawba Mass and Energy Release and Containment Response Methodology, December 2000

34. Letter from NRC to M. S. Tuckman (Duke) dated September 6, 1995, "Safety Evaluation for Topical Report DPC-NE-3004-P, "Mass and Energy Release and Containment Response Methodology", McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station , Units 1 and 2 (TAC Nos. M90646, M90647, and M90648)"

35. Letter from NRC to H. B. Barron (Duke) dated February 29, 2000, "McGuire Nuclear Station and Catawba Nuclear Station RE: Review of Topical Report DPC-NE-3004-PA, Rev. 1, Regarding Proposed Finer Nodalization of Ice Condenser (TAC Nos. MA5511, MA5512, MA5517, and MA5518)"

36. Letter from M. S. Tuckman (Duke) to U. S. NRC dated March 15, 1996, "Catawba Nuclear Station, Units 1 and 2, Docket Nos. 50-413 and 414; McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 370; Response to Request for Additional Information"

37. Letter from Victor Nerses (NRC) to H. B. Barron (Duke) dated May 5, 1997, "Issuance of Amendments – McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M90590 and M90591)"

TABLE 2
Summary of UFSAR Transient Information for Use in Proposed Catawba Unit 2 RCS Flow Reduction (387,000 gpm)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(1) 15.1.1	Feedwater System Malfunction that Results in a Reduction in Feedwater Temperature	5A Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			Transient results are less severe than 15.1.2 and 15.1.3 – not analyzed since transient is bounded
(2) 15.1.2	Feedwater System Malfunction Causing an Increase in Feedwater Flow	4C (zero power) Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low	353,350 gpm (zero power case)	NA	Zero power case – 382,000 gpm total RCS flow rate - bounds proposed RCS flow rate of 387,000 gpm total flow
		6A (full power) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	358,800 gpm (full power case)	361,032 gpm (full power case)	Full power case – Evaluated
(3) 15.1.3	Excessive Increase in Secondary Steam Flow	4D Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low	356,125 gpm	361,032 gpm	Current AOR assumes RCS total flow rate of 385,000 gpm
(4) 15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	5B Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			Results are bounded by those in Main Steam Line Break Analysis (15.1.5)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(5) 15.1.5	Steam System Piping Failure	6L Evaluated	References 6 and 10	References 7 and 10	Low	353,343 gpm	353,343 gpm	Reanalyzed with RCS flow of 387,000 gpm minus flow uncertainty, stairstep flow reduction, and bounding core bypass flow
(6) 15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	5C Not applicable to Catawba	NA	NA	NA			NA
(7) 15.2.2	Loss of External Load	5D Bounded by other transient results	Reference 8	Reference 9	Low			Results are bounded by those in Turbine Trip (15.2.3)
(8) 15.2.3	Turbine Trip	4E Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low (Peak primary)	(Peak primary pressure) 345,576 gpm	NA	Peak primary pressure: 382,000 gpm (total)
					High (Peak secondary)	(Peak secondary pressure) 388,500 gpm	NA	Peak secondary pressure: 420,000 gpm (total)
(9) 15.2.4	Inadvertent Closure of Main Steam Isolation Valves	5E Bounded by other transient results	NA	NA	Low			Results are bounded by those in Turbine Trip (15.2.3)
(10) 15.2.5	Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	5F Bounded by other transient results	NA	NA	Low			Results are bounded by those in Turbine Trip (15.2.3)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(11) 15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	4F Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	Low	345,576 gpm	NA	Current AOR assumes conservatively low RCS flow of 382,000 gpm
(12) 15.2.7	Loss of Normal Feedwater Flow	(Short-term) 6B Evaluated	Reference 1	References 2, 3, 4, and 5	(Short-term) Low	(Short-term) 360,750 gpm	(Short-term) 361,032 gpm	(Short-term) Evaluated
		(Long-term) 4G Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	(Long-term) Low	(Long-term) 348,290 gpm	(Long-term) NA	(Long-term) Current AOR assumes 385,000 gpm
(13) 15.2.8	Feedwater System Pipe Break	5G (Short-term) Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	(Short-term) Low			(Short-term) Results are bounded by those in Loss of Normal Feedwater Flow (15.2.7)
		4H (Long-term) Bounded by current AOR	Reference 1	References 2, 3, 4, and 5	(Long-term) Low	(Long-term) 339,972 gpm	(Long-term) NA	(Long-term) Current AOR assumes 382,000 gpm
(14) 15.3.1	Partial Loss of Forced Reactor Coolant Flow	6C Evaluated	Reference 1	References 2, 3, 4, and 5	Low	358,800 gpm	361,032 gpm	Evaluated
(15) 15.3.2	Complete Loss of Forced Reactor Coolant Flow	6D Evaluated	Reference 1	References 2, 3, 4, and 5	Low	358,800 gpm	361,032 gpm	Evaluated

						Evaluated Cases (see Note 3)		
UFSAR Section	**Analysis Title**	**LAR Designation and Evaluation Type (see Note 1)**	**Approved by NRC or Conducted Using Methods or Processes Approved by NRC**	**Reference for NRC Approval**	**High RCS Flow/ Low RCS Flow Limiting**	**Core Flow in Current AOR (see Note 2)**	**Core Flow in Evaluation with Reduced RCS Flow**	**Comment**
(16) 15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	6E (DNB) Evaluated	T&H: Reference 1 Dose: Reference 10	T&H: References 2, 3, 4, and 5 Dose: Reference 10	Low	360,750 gpm	361,032 gpm	Evaluated
		6M (peak RCS pressure) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	350,099 gpm	NA	(Peak RCS pressure) – Reanalyzed with RCS total flow of 387,000 gpm
(17) 15.3.4	Reactor Coolant Pump Shaft Break	5H Bounded by other transient results	NA	NA	Low			Results are similar to those calculated in Locked Rotor (15.3.3)
(18) 15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition	6N (DNB, peak RCS pressure) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	(3 RCP) 357,142 gpm	NA	(3 RCP - core cooling) – AOR has total RCS flow rate based on 390,000 gpm; proposed decrease in RCS flow has no impact on 3 RCP case

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3) Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
						(Peak RCS pressure) based on 349,285 gpm	NA	(Peak RCS pressure) – AOR has total RCS flow rate based on 390,000 gpm; proposed decrease in RCS flow has no impact on 3 RCP case
(19) 15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	6F (DNB) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	357,142 gpm	357,421 gpm	Evaluated
		6O (Peak RCS pressure) Evaluated	Reference 1	References 2, 3, 4, and 5	Low	346,597 gpm	NA	Reanalyzed with RCS total flow of 387,000 gpm
(20) 15.4.3	Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)	15.4.3a: 6G Evaluated	15.4.3a: Reference 6 Dose: Reference 10	15.4.3a: Reference 7 Dose: Reference 10	(All cases) Low	358,800 gpm	361,032 gpm	15.4.3a: (One or more dropped RCCA rods): Evaluated
		15.4.3b: 5I Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			15.4.3b: (Dropped RCCA bank): Results are bounded by case with one or more dropped rods (15.4.3a and 15.4.3c)

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
		15.4.3c: 5J Bounded by other transient results	Reference 1	References 2, 3, 4, and 5	Low			15.4.3c: (Statically misaligned RCCA): Results are bounded by cases with one or more dropped rods, or by single RCCA withdrawal (SUCR)
		15.4.3d: 6H Evaluated	Reference 1	References 2, 3, 4, and 5	Low	360,750 gpm	361,032 gpm	15.4.3d: (SUCR): Flow through the core: Evaluated
(21) 15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	6P Evaluated	Reference 1	References 2, 3, 4, and 5	Low	Based on 339,980 gpm initial core flow	Based on 339,980 gpm initial core flow	(3 RCP - core cooling) – AOR has total RCS flow rate based on 388,000 gpm; proposed decrease in RCS flow has no impact on 3 RCP case
(22) 15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	5K (BWR Transient)	NA	NA	NA			NA
(23) 15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	5L (Unaffected)	Reference 1	References 2, 3, 4, and 5	NA			NA

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(24) 15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	5M (Unaffected)	Reference 1	References 2, 3, 4, and 5	NA			NA
(25) 15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	6I Evaluated	T&H: Reference 6 Dose: Reference 10	T&H: Reference 7 Dose: Reference 10	Low	343,621 gpm	349,558 gpm	Evaluated
(26) 15.4.9	Spectrum of Rod Drop Accidents (BWR)	5N (BWR Transient)	NA	NA	NA			NA
(27) 15.5.1	Inadvertent Operation of Emergency Core Cooling System During Power Operation	5O (Unaffected)	Reference 1	References 2, 3, 4, and 5	NA			NA
(28) 15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	5P (Bounded by other transients)	NA	NA	NA			NA
(29) 15.5.3	A Number of BWR Transients	5Q (BWR Transient)	NA	NA	NA			NA
(30) 15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	6J Evaluated	Reference 1	References 2, 3, 4, and 5	Low	355,020 gpm	361,032 gpm	Evaluated

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(31) 15.6.2	Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment	5R (Unaffected)	Reference 10	Reference 10	NA			NA
(32) 15.6.3	Steam Generator Tube Failure	6K (DNB) Evaluated	T&H: Reference 1	T&H: References 2, 3, 4, and 5	Low	360,750 gpm	361,032 gpm	Evaluated
		6Q (overfill, dose) Evaluated	Dose: Reference 10	Dose: Reference 10	Low	345,576 gpm (dose) 356,125 gpm (overfill)	NA	For the SG overfill and dose consequences of SGTR, the analysis results are not sensitive to the assumed RCS flow
(33) 15.6.4	Spectrum of BWR Steam System Piping Failures Outside Containment	5S (BWR Transient)	NA	NA	NA			NA
(34) 15.6.5	Loss-of-Coolant Accidents	6R Evaluated	T&H: References 11 and 12 Dose: References 13, 14, 15, 16, and 17	T&H: References 18 and 19 Dose: Reference 20	Low	LBLOCA/ SBLOCA: Based on 390,000 gpm total RCS flow rate	LBLOCA/SBLOCA: Based on 387,000 gpm total RCS flow rate	LBLOCA/SBLOCA consequences are insensitive to the proposed change in RCS flow
(35) 15.6.6	A Number of BWR Transients	5T (BWR Transient)	NA	NA	NA			NA

						Evaluated Cases (see Note 3)		
UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	Comment
(36) 15.7.1	Radioactive Gas Waste System Leak or Failure	5U (Unaffected)	Reference 21	Reference 21	NA			NA
(37) 15.7.2	Radioactive Liquid Waste System Leak or Failure	5U (Unaffected)	Reference 10	Reference 10	NA			NA
(38) 15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failures	5U (Unaffected)	Reference 10	Reference 10	NA			NA
(39) 15.7.4	Fuel Handling Accidents in the Containment and Spent Fuel Storage Buildings	5U (Unaffected)	References 22, 23, 24, and 25	References 25 and 26	NA			NA
(40) 15.8	Anticipated Transients Without Trip	4I (Bounded by current AOR)	Reference 27	Reference 28	NA	Based on nominal total RCS flow of 377,600 gpm	NA	Current AOR assumes conservatively low RCS flow of 377,600 gpm
(41) 15.9	Formerly Chapter 15 Appendix A - Models Used for Calculation of Accident Doses	5U (Unaffected)	NA	NA	NA			NA
(42) 15.10	Formerly Chapter 15 Appendix B - Supplementary Radiological Analyses	5U (Unaffected)	References 29, 30, and 31	References 9 and 32	NA			NA

UFSAR Section	Analysis Title	LAR Designation and Evaluation Type (see Note 1)	Approved by NRC or Conducted Using Methods or Processes Approved by NRC	Reference for NRC Approval	High RCS Flow/ Low RCS Flow Limiting	Evaluated Cases (see Note 3)		Comment
						Core Flow in Current AOR (see Note 2)	Core Flow in Evaluation with Reduced RCS Flow	
(43) 6.2.1.1.3.1 6.2.1.1.3.3 6.2.1.1.3.4 6.2.1.3.2 6.2.1.4	Containment Performance Analyses	4B (Bounded by current AOR)	Reference 33	References 34 and 35	High	388,500 gpm	NA	Current AOR assume conservatively high RCS flow of 420,000 gpm (Unit 1 analysis is used since results bound those for Unit 2)
(44) 6.2.1	Postulated Secondary System Pipe Rupture Outside Containment	4B (Bounded by current AOR)	References 33 and 36	References 34, 35, and 37	High	388,500 gpm	NA	Current AOR assumes conservatively high RCS flow of 420,000 gpm
(45) 3.6.2.2.1 3.9.1.4	LOCA Blowdown Reactor Vessel and Loop Forces	4A (Evaluated)	NA	NA	High	Based on total RCS flow rate of 398,000 gpm	NA	Current AOR assume conservatively high RCS flow of 398,000 gpm

NOTE 1: LAR Designations listed by Category pages B-8 to B-11.
NOTE 2: "Core Flow" represents net flow through core in analysis, defined as Total RCS flow minus Core Bypass Flow.
NOTE 3: Transients which are bounded by other transient results, not applicable for Catawba Unit 2, or unaffected by the proposed RCS flow reduction have shaded fields in this column.

REFERENCES:

1. DPC-NE-3002-A, Revision 4b, "McGuire and Catawba Nuclear Station UFSAR Chapter 15 System Transient Analysis Methodology", September 2010

2. Letter from Tim Reed (NRC) to H. B. Tucker (Duke) dated November 15, 1991, "Safety Evaluation on Topical Report DPC-NE-3002, "FSAR Chapter 15 System Transient Analysis Methodology," (TAC No. 66850)"

3. Letter from Robert Martin (NRC) to M. S. Tuckman (Duke) dated December 28, 1995, "Safety Evaluation for Revision 1 to Topical Report DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology" McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M89944, M89945, and M89946)"

4. Letter from Herbert Berkow (NRC) to M. S. Tuckman (Duke) dated April 26, 1996, "Safety Evaluation on Change to Topical Report DPC-NE-3002-A on Opening Characteristics of Safety Valves - McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M94405, M94406, M94407, and M94408)"

5. Letter from Chandu Patel (NRC) to G. R. Peterson (Duke) dated April 6, 2001, "Catawba Nuclear Station, Units 1 and 2 RE: Revision 4 to the Duke Energy Corporation Topical Report DPC-NE-3002-A, "UFSAR Chapter 15 Transient Analysis Methodology" (TAC Nos. MA8928 and MA8929)"

6. DPC-NE-3001-PA, Revision 0a, "McGuire and Catawba Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology", May 2009

7. Letter from Timothy Reed (NRC) to H. B. Tucker (Duke) dated November 15, 1991, "Safety Evaluation on Topical Report DPC-NE-3001, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" (TAC Nos. 75954/75955/75956/75957)"

8. Technical Information Document (TID) 14844, Calculation of Distance Factors for Power and Test Reactors, U.S. Atomic Energy Commission, 1962

9. 10 CFR Part 100, Section 100.11

10. Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

11. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best-Estimate LOCA Analysis," March 1998.

12. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

13. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term," March 20, 2008.

14. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas: LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term Revision to Control Room Atmospheric Dispersion Factors," March 20, 2008.

15. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 License Amendment Request for Full Scope Implementation of the Alternative Source Term. Response to Request for Additional Information," October 6, 2008.

16. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Response to Request for Additional Information related to the License Amendment Request for Implementation of Alternative Source Term," December 17, 2008.

17. Letter from B.H. Hamilton (Duke) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Response to Request for Additional Information related to the License Amendment Request (LAR) for Implementation of Alternative Source Term (AST)," February 12, 2009.

18. Letter from R. C. Jones (NRC) to N. J. Liparulo (Westinghouse) dated June 28, 1996, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Analysis".

19. Letter from R. C. Jones (NRC) to E. P. Rahe (Westinghouse) dated May 21, 1985, "Acceptance for Referencing of Licensing Topical Report WCAP-10054 (P), Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code".

20. Letter from John Stang (USNRC) to B.H. Hamilton (Duke), "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Adoption of the Alternative Source Term Radiological Analysis Methodology (TAC Nos. MD8400 and MD8401)," March 30, 2009.

21. Regulatory Guide (RG) 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I"

22. Letter from G. R. Peterson (Duke) to U.S. NRC dated December 20, 2005, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations"

23. Letter from G. R. Peterson (Duke) to U.S. NRC dated May 4, 2006, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations. Response to Request for Additional Information"

24. Letter from G. R. Peterson (Duke) to U.S. NRC dated August 31, 2006, "License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations. Additional Commitments Regarding Containment Closure Administrative Controls."

25. ISG-5, Revision 1 - "Confinement Evaluation", Spent Fuel Project Office, NRC

26. Letter from John Stang (NRC) to G. R. Peterson (Duke) dated December 22, 2006, "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Implementation of Alternative Source Term Methodology (TAC Nos. MC9746 AND MC9747)"

27. Letter from T. M. Anderson (Westinghouse) to S. H. Hanauer (NRC) dated December 30, 1979, "NS-TMA-2182, ATWS Submittal"

28. Letter from Darl S. Hood (NRC) to H. B. Tucker (Duke) dated November 6, 1987, "ATWS Rule (10 CFR 50.62) for McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 59081/59111/59112/64535)"

29. Letter from Hal Tucker (Duke) to Harold Denton (NRC) dated February 17, 1984, "McGuire Nuclear Station Docket Nos. 50-369, 50-370", Spent Fuel Pool Re-rack LAR

30. Letter from Hal Tucker (Duke) to Harold Denton (NRC) dated March 20, 1984, "McGuire Nuclear Station Docket Nos. 50-369, 50-370", Safety & Environmental Analysis for Spent Fuel Pool Re-rack

31. Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)"

32. Letter from Elinor Adensam (NRC) to Hal Tucker (Duke) dated September 24, 1984, "Issuance of Amendment No.35 to Facility Operating License NPF-9 and Amendment No. 16 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2"

33. DPC-NE-3004-PA, Revision 1, McGuire and Catawba Mass and Energy Release and Containment Response Methodology, December 2000

34. Letter from NRC to M. S. Tuckman (Duke) dated September 6, 1995, "Safety Evaluation for Topical Report DPC-NE-3004-P, "Mass and Energy Release and Containment Response Methodology", McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station , Units 1 and 2 (TAC Nos. M90646, M90647, and M90648)"

35. Letter from NRC to H. B. Barron (Duke) dated February 29, 2000, "McGuire Nuclear Station and Catawba Nuclear Station RE: Review of Topical Report DPC-NE-3004-PA, Rev. 1, Regarding Proposed Finer Nodalization of Ice Condenser (TAC Nos. MA5511, MA5512, MA5517, and MA5518)"

36. Letter from M. S. Tuckman (Duke) to U. S. NRC dated March 15, 1996, "Catawba Nuclear Station, Units 1 and 2, Docket Nos. 50-413 and 414; McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 370; Response to Request for Additional Information"

37. Letter from Victor Nerses (NRC) to H. B. Barron (Duke) dated May 5, 1997, "Issuance of Amendments – McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M90590 and M90591)"

ATTACHMENT 2

MARKED UP TS PAGE

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in Table 3.4.1-1.

APPLICABILITY: MODE 1.

-------------------------------NOTE-------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer pressure or RCS average temperature DNB parameters not within limits.	A.1 Restore DNB parameter(s) to within limit.	2 hours
B. RCS total flow rate ≥ 99%, but < 100% of the limit specified in the COLR.	B.1 Reduce THERMAL POWER to ≤ 98% RTP.	2 hours
	AND	
	B.2 Reduce the Power Range Neutron Flux – High Trip Setpoint below the nominal setpoint by 2% RTP.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS total flow rate < 99% of the value specified in the COLR.	C.1 Restore RCS total flow rate to ≥ 99% of the value specified in the COLR.	2 hours
	OR	
	C.2.1 Reduce THERMAL POWER to < 50% RTP.	2 hours
	AND	
	C.2.2 Reduce the Power Range Neutron Flux - High Trip Setpoint to ≤ 55% RTP.	6 hours
	AND	
	C.2.3 Restore RCS total flow rate to ≥ 99% of the value specified in the COLR.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

NO CHANGES THIS PAGE.
FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify RCS average temperature is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	Verify RCS total flow rate is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.4	Perform CHANNEL CALIBRATION for each RCS total flow indicator.	In accordance with the Surveillance Frequency Control Program

Table 3.4.1-1 (page 1 of 1)
RCS DNB Parameters

PARAMETER		INDICATION	No. OPERABLE CHANNELS	LIMITS
1.	Indicated RCS Average Temperature – Unit 1	meter	4	≤ the value specified in the COLR
		meter	3	≤ the value specified in the COLR
		computer	4	≤ the value specified in the COLR
		computer	3	≤ the value specified in the COLR
	Indicated RCS Average Temperature – Unit 2	meter	4	≤ the value specified in the COLR
		meter	3	≤ the value specified in the COLR
		computer	4	≤ the value specified in the COLR
		computer	3	≤ the value specified in the COLR
2.	Indicated Pressurizer Pressure	meter	4	≥ the value specified in the COLR
		meter	3	≥ the value specified in the COLR
		computer	4	≥ the value specified in the COLR
		computer	3	≥ the value specified in the COLR
3.	RCS Total Flow Rate		384,000	≥ ~~388,000~~ gpm and ≥ the limit specified in the COLR (Unit 1);
			387,000	≥ ~~390,000~~ gpm and ≥ the limit specified in the COLR (Unit 2)