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February 15, 2005

U.S. Nuclear Regulatory Commission
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Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Technical Specification Bases Changes

Pursuant to 10CFR 50.4, please find attached changes to the Catawba Nuclear Station Technical Specification Bases. These Bases changes were made according to the provisions of 10CFR 50.59.

Any questions regarding this information should be directed to L. J. Rudy, Regulatory Compliance, at (803) 831-3084.

I certify that I am a duly authorized officer of Duke Energy Corporation and that the information contained herein accurately represents changes made to the Technical Specification Bases since the previous submittal.

R. Michael Glenn /for

Dhiaa M. Jamil

Attachment

xc: W. D. Travers, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II

S. E. Peters, Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation, Mail Stop 0-8-G9

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February 15, 2005

Re: Catawba Nuclear Station
Technical Specifications (TS) Manual

Please replace the corresponding pages in your copy of the Catawba Technical Specifications Manual as follows:

REMOVE THESE PAGES	INSERT THESE PAGES
List of Effective Pages	
Page 16	Page 16
Page 32	Page 32
Tab 3.3.1 Bases	
B 3.3.1-49 - B 3.3.1-50	B 3.3.1-49 – B 3.3.1-50
Tab 3.9.4 Bases	
B 3.9.4-1 – B 3.9.4-4	B 3.9.4-1 – B 3.9.4-4

If you have any questions concerning the contents of this Technical Specification update, contact Jill Ferguson at (803) 831-3938.

Lee A. Keller
Manager, Regulatory Compliance

Page Number	Amendment	Revision Date
B 3.3.1-19	Revision 0	9/30/98
B 3.3.1-20	Revision 0	9/30/98
B 3.3.1-21	Revision 0	9/30/98
B 3.3.1-22	Revision 0	9/30/98
B 3.3.1-23	Revision 0	9/30/98
B 3.3.1-24	Revision 0	9/30/98
B 3.3.1-25	Revision 0	9/30/98
B 3.3.1-26	Revision 0	9/30/98
B 3.3.1-27	Revision 0	9/30/98
B 3.3.1-28	Revision 0	9/30/98
B 3.3.1-29	Revision 0	9/30/98
B 3.3.1-30	Revision 1	8/13/99
B 3.3.1-31	Revision 1	8/13/99
B 3.3.1-32	Revision 0	9/30/98
B 3.3.1-33	Revision 0	9/30/98
B 3.3.1-34	Revision 0	9/30/98
B 3.3.1-35	Revision 1	7/29/03
B 3.3.1-36	Revision 1	7/29/03
B 3.3.1-37	Revision 0	9/30/98
B 3.3.1-38	Revision 0	9/30/98
B 3.3.1-39	Revision 0	9/30/98
B 3.3.1-40	Revision 0	9/30/98
B 3.3.1-41	Revision 0	9/30/98
B 3.3.1-42	Revision 0	9/30/98
B 3.3.1-43	Revision 0	9/30/98
B 3.3.1-44	Revision 0	9/30/98
B 3.3.1-45	Revision 1	2/18/02
B 3.3.1-46	Revision 1	2/18/02
B 3.3.1-47	Revision 0	9/30/98
B 3.3.1-48	Revision 0	9/30/98
B 3.3.1-49	Revision 1	11/24/04
B 3.3.1-50	Revision 1	4/22/02

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B 3.8.9-7	Revision 0	9/30/98
B 3.8.9-8	Revision 0	9/30/98
B 3.8.9-9	Revision 1	2/26/99
B 3.8.9-10	Revision 1	2/26/99
B 3.8.10-1	Revision 0	9/30/98
B 3.8.10-2	Revision 0	9/30/98
B 3.8.10-3	Revision 2	7/29/03
B 3.8.10-4	Revision 1	7/29/03
B 3.9.1-1	Revision 0	9/30/98
B 3.9.1-2	Revision 1	7/29/03
B 3.9.1-3	Revision 1	7/29/03
B 3.9.2-1	Revision 2	6/21/04
B 3.9.2-2	Revision 2	6/21/04
B 3.9.2-3	Revision 2	6/21/04
B 3.9.3-1	Revision 2	4/23/02
B 3.9.3-2	Revision 1	4/23/02
B 3.9.3-3	Revision 1	4/23/02
B 3.9.3-4	Revision 1	4/23/02
B 3.9.3-5	Revision 1	4/23/02
B 3.9.4-1	Revision 0	9/30/98
B 3.9.4-2	Revision 3	11/24/04
B 3.9.4-3	Revision 2	11/24/04
B 3.9.4-4	Revision 2	11/24/04
B 3.9.5-1	Revision 0	9/30/98
B 3.9.5-2	Revision 2	7/29/03
B 3.9.5-3	Revision 1	7/29/03
B 3.9.5-4	Revision 1	7/29/03
B 3.9.6-1	Revision 0	9/30/98
B 3.9.6-2	Revision 0	9/30/98
B 3.9.6-3	Revision 0	9/30/98
B 3.9.7-1	Revision 0	6/21/04

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical.

SR 3.3.1.16 and SR 3.3.1.17

SR 3.3.1.16 and SR 3.3.1.17 verify that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the UFSAR (Ref. 1). Individual component response times are not modeled in the analyses.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. In addition, while not specifically identified in the WCAP, ITT Barton 386A and 580A-0 sensors were compared to sensors which were identified. It was concluded that the WCAP results could be applied to these two sensor types as well. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant and component cooling water through the RHR heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

BASES

LCO (continued)

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. The operability of the operating RHR train and the supporting heat sink is dependent on the ability to maintain the desired RCS temperature. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room and there are no restraints to placing the equipment in service.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the minimum required RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

The acceptability of the LCO and the LCO Note is based on preventing boiling in the core in the event of the loss of RHR cooling. However, it has been determined that when the upper internals are in place in the reactor vessel there is insufficient communication with the water above the core for adequate decay heat removal by natural circulation. As a result, boiling in the core could occur in a relatively short time if RHR cooling is lost. Therefore, during the short period of time that the upper internals are installed, administrative processes are implemented to reduce the risk of core boiling. The availability of additional cooling equipment, including equipment not required to be OPERABLE by the Technical Specifications, contributes to this risk reduction. The plant staff assesses these cooling sources to assure that the desired minimal level of risk is maintained. This is commonly referred to as defense-in-depth. This strategy is

BASES

LCO (continued)

consistent with NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Ref. 3).

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

BASES

ACTIONS (continued)

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The RCS temperature is determined to ensure the appropriate decay heat removal is maintained. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. UFSAR, Section 5.5.7.

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management."