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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

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.

Dhiaa M. Jamil

Attachment

A001

xc: L. A. Reyes, 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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November 20, 2003

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
	Tab 3.2.4	
B 3.2.4-5 - B 3.2.4-7		B 3.2.4-5 - B 3.2.4-7
	Tab 3.3.2	
B 3.3.2-17 – B 3.3.2-24		B 3.3.2-17 – B 3.3.2-24
	Tab 3.4.11	
B 3.4.11-1 – B 3.4.11-4		B 3.4.11-1 – B 3.4.11-4
	Tab 3.8.4	
B 3.8.4-5 – B 3.8.4-9		B 3.8.4-5 – B 3.8.4-10

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

Lee A. Keller
Lee A. Keller
Manager, Regulatory Compliance

BASES

ACTIONS (continued)

reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the more restrictive of the power level limit determined by Required Action A.1 or A.2 is exceeded. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the more restrictive limit of Required Action A.1 or A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.7 is modified by a Note that states that the peaking factor surveillances must be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.6). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.7 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by three Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. Note 3 states that the SR is not required to be performed until 12 hours after exceeding 50% RTP. This is necessary to establish core conditions necessary to provide meaningful calculation.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The

BASES

SURVEILLANCE REQUIREMENTS (continued)

Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

The QPTR alarm is inoperable for the duration of excore channel calibrations performed for agreement with incore detector measurements.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore tilt. Therefore, incore tilt can be used to confirm that QPTR is within limits.

With one or more NIS channel inputs to QPTR inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred,

BASES

SURVEILLANCE REQUIREMENTS (continued)

which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

requires two channels to be OPERABLE. Individual valves may also be closed using individual hand switches in the control room. The LCO requires four individual channels to be OPERABLE.

b. Steam Line Isolation-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation-Containment Pressure-High High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. The Containment Pressure-High High function is described in ESFAS Function 2.C.

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Steam Line Isolation-Steam Line Pressure

Steam Line Pressure channels provide both protection and control functions. The protection functions include: Steam Line Pressure-Low and Steam Line Pressure-Negative Rate functions. The control functions include: Digital Feedwater Control System (DFCS) which controls SG level.

(1) Steam Line Pressure-Low

Steam Line Pressure—Low provides closure of the MSIVs in the event of an SLB to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump.

DFCS receives steam pressure inputs from three separate protection channels for each SG. The three inputs are median selected for each SG, with the resultant output being used by the automatic control algorithm. The median select feature prevents the failure of an input signal from affecting the control system. A loss of two or more input signals will place the control system in manual and alert the operator. DFCS will maintain a steady control function during the switch to manual operation; therefore, a failure of one or more input signals will not cause a control system action that would result in a condition requiring protective actions. Thus, three OPERABLE channels on each steam line, with a two-out-of-three logic on each steam line, are sufficient to satisfy protective requirements.

Steam Line Pressure-Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure-High High. Stuck valve transients and outside containment SLBs will be

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

terminated by the Steam Line Pressure-Negative Rate-High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure-Negative Rate-High

Steam Line Pressure-Negative Rate-High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure-Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate-High signal is automatically enabled. DFCS receives steam pressure inputs from three separate protection channels for each SG. The three inputs are median selected for each SG, with the resultant output being used by the automatic control algorithm. The median select feature prevents the failure of an input signal from affecting the control system. A loss of two or more input signals will place the control system in manual and alert the operator. DFCS will maintain a steady control function during the switch to manual operation; therefore, a failure of one or more input signals will not cause a control system action that would result in a condition requiring protective actions. Thus, three OPERABLE channels on each steam line, with a two-out-of-three logic on each steam line, are sufficient to satisfy protective requirements.

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, stop the excessive flow of feedwater into the SGs, and to limit the energy released into containment. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows. Feedwater Isolation serves to limit the energy released into containment upon a feedwater line or steam line break inside containment.

The Functions are actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

Turbine Trip and Feedwater Isolation signals are both actuated by SG Water Level-High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. A Feedwater Isolation signal is also generated by a reactor trip (P-4) coincident with T_{avg}-Low and on a high water level in the reactor building doghouse. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

a. Turbine Trip

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(1) Turbine Trip-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

(2) Turbine Trip-Steam Generator Water Level-High High (P-14)

This signal prevents damage to the turbine due to water in the steam lines. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. The setpoints are based on percent of narrow range instrument span.

(3) Turbine Trip-Safety Injection

Turbine Trip is also initiated by all Functions that initiate SI. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements. Item 5.a.(1) is referenced for the applicable MODES.

The Turbine Trip Function must be OPERABLE in MODES 1 and 2. In lower MODES, the turbine generator is not in service and this Function is not required to be OPERABLE.

b. Feedwater Isolation

(1) Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(2) Feedwater Isolation-Steam Generator Water Level-High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. The setpoints are based on percent of narrow range instrument span.

(3) Feedwater Isolation-Safety Injection

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements. Item 5.b.(1) is referenced for the applicable MODES.

(4) Feedwater Isolation - RCS T_{avg} - Low coincident with Reactor Trip (P-4)

This signal provides protection against excessive cooldown, which could subsequently introduce a positive reactivity excursion after a plant trip. There are four channels of RCS T_{avg} - Low (one per loop), with a two-out-of-four logic required coincident with a reactor trip signal (P-4) to initiate a feedwater isolation. The P-4 interlock is discussed in Function 8.a.

(5) Feedwater Isolation – Doghouse Water Level – High High

This signal initiates a Feedwater Isolation. The signal terminates forward feedwater flow in the event of a postulated pipe break in the main feedwater

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

piping in the doghouses to prevent flooding safety related equipment essential to the safe shutdown of the plant. The level instrumentation consists of two level switches (one per train) in each of the two reactor building doghouses. A high-high level detected by one-out-of-two switches, in either the inboard or outboard doghouse, will initiate a doghouse isolation. This signal initiates Feedwater Isolation for the specific doghouse where the High-High level is detected and trips both main feedwater pumps thus causing a main turbine trip.

The Feedwater Isolation Function must be OPERABLE in MODES 1 and 2 and also in MODE 3 (except for the functions listed in Table 3.3.2-1). Feedwater Isolation is not required OPERABLE when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In lower MODES, the MFW System is not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal and accident operation. The normal source of water for the AFW System is the condensate storage system (not safety related). A low suction pressure to the AFW pumps will automatically realign the pump suctions to the Nuclear Service Water System (NSWS)(safety related). The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

a. Auxiliary Feedwater-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Auxiliary Feedwater-Steam Generator Water Level-Low Low

SG Water Level-Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level-Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with two-out-of-four logic. The setpoints are based on percent of narrow range instrument span.

SG Water Level—Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level—Low Low in any two operating SGs will cause the turbine driven pumps to start.

c. Auxiliary Feedwater—Safety Injection

An SI signal starts the motor driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater-Loss of Offsite Power

A loss of offsite power to the service buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each essential service bus. Loss of power to either essential service bus will start the turbine driven and motor driven AFW pumps to ensure that at least two SGs contain enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

Functions 6.a through 6.d must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. These Functions do not have to be OPERABLE in MODES 5 and 6

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Three PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has three PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

BASES

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for manual RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are assumed to operate in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV automatic actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. Events that assume this condition include uncontrolled bank withdrawal at power, uncontrolled bank withdrawal from subcritical, and single rod withdrawal at power (Ref. 2). (This statement clarifies that worst case DNBR calculations are analyzed by assuming automatic PORV operation. This statement is to bound the worst case DNBR calculations based on all possible plant conditions and is not a requirement for automatic PORV operation for system OPERABILITY.)

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. Three PORVs are required to be OPERABLE to meet RCS pressure boundary requirements. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2.

BASES

APPLICABILITY (continued)

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4 ≤ 285°F, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.

A.1

With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2). (This statement simply describes normal plant operations.)

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

If one or two PORVs are inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. (This statement addresses the requirement to defeat automatic PORV operation with an inoperable block valve.) The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours

ACTIONS (continued)

the loss of the channel DC power and the associated DG DC power, the load center power for the train is inoperable and the Condition(s) and Required Action(s) for the Distribution Systems must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 9).

SR 3.8.4.2

Verifying battery individual cell voltage while on float charge for the DG batteries ensures each cell is capable of supporting their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. For this surveillance two different cells shall be tested each month. The 7 day Frequency is consistent with manufacturer recommendations.

SR 3.8.4.3

For the DC channel batteries, visual inspection to detect corrosion of the battery terminals and connections, or measurement of the resistance of each intercell, interrack, intertier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of visible corrosion does not necessarily represent a failure of this SR, provided an evaluation determines that the visible corrosion does not affect the OPERABILITY of the battery.

SURVEILLANCE REQUIREMENTS (continued)

For the DG batteries, visual inspection to detect corrosion of the battery terminals and connections provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of visible corrosion does not necessarily represent a failure of this SR, provided an evaluation determines that the visible corrosion does not affect the OPERABILITY of the battery.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.8.4.4

For the DC channel batteries, visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

For the DG batteries, visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. Since the DG battery cell jars are not transparent, a direct visual inspection of the cell plates cannot be performed. Instead, the cell plates are inspected for physical damage and abnormal deterioration by: 1) visually inspecting the jar sides of each cell for excessive bowing and/or deformation, and 2) visually inspecting the electrolyte of each cell for abnormal appearance.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.4.5 and SR 3.8.4.6

Visual inspection and resistance measurements of intercell, interrack, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material, as recommended by the manufacturer for the DG batteries, is used to help ensure good electrical

SURVEILLANCE REQUIREMENTS (continued)

connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.5.

For the DG batteries, the cell-to-cell terminal pole screws should be set from 14 to 15 foot-pounds of torque. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.4.7

This SR requires that each battery charger for the DC channel be capable of supplying at least 200 amps and at least 75 amps for the DG chargers. All chargers shall be tested at a voltage of at least 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.8

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4. The DC channel batteries are tested to supply a current ≥ 522.14 amps for the first minute, then ≥ 267.71 amps for the next 9 minutes, ≥ 376.15 amps for the next 10 minutes, and ≥ 281.94 amps for the next 100 minutes. Terminal voltage is required to remain ≥ 110.4 volts during this test. The

SURVEILLANCE REQUIREMENTS (continued)

DG batteries are tested to supply a current ≥ 218.5 amps for the first minute, then ≥ 42.5 amps for the next 10 minutes, then ≥ 121.8 amps for the next minute, then ≥ 42.5 amps for the remaining 108 minutes. Terminal voltage is required to remain ≥ 105 volts during this test.

Except for performing SR 3.8.4.8 for the DC channel batteries with the unit on line, the Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10), which states that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a performance discharge test that is augmented to include the high-rate, short duration discharge loads (during the first minute and 11-to-12 minute discharge periods) of the service test. The duty cycle of the modified performance test must fully envelope the duty cycle of the service test if the modified performance discharge test is to be used in lieu of the service test. Since the ampere-hours removed by the high-rate, short duration discharge periods of the service test represents a very small portion of the battery capacity, the test rate can be changed to that for the modified performance discharge test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rates of the duty cycle). This will often confirm the battery's ability to meet the critical periods of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.9

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined

SURVEILLANCE REQUIREMENTS (continued)

by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.8. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.9; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.9 while satisfying the requirements of SR 3.8.4.8 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9). This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 18 months. However (for DC vital batteries only), if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its average capacity on the previous performance tests or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 9). This SR is modified by a Note which is applicable to the DG batteries only. The reason for the Note is that performing the Surveillance would perturb the associated electrical distribution system and challenge safety systems.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1971 and 1974.
4. UFSAR, Chapter 8.
5. IEEE-485-1983, June 1983.
6. UFSAR, Chapter 6.

REFERENCES (continued)

7. UFSAR, Chapter 15.

8. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

9. IEEE-450-1975 and/or 1980.

10. Regulatory Guide 1.32, February 1977.