REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required coolant loop in operation, restore the required loops to operation within 72 hours or open the Reactor Trip System breakers.

c. With no reactor coolant loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loops to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours.

4.4.1.2.2 At least 3 reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

(4 hours on a one time basis for natural circulation testing following steam generator replacement)

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required coolant loop in operation, restore the required loops to operation within 72 hours or open the Reactor Trip System breakers.

c. With no reactor coolant loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loops to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours.

4.4.1.2.2 At least 3 reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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*All reactor coolant pumps may be deenergized for up to 1 hour (4 hours on a one time basis for natural circulation testing following steam generator replacement) provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

Attachment 2

Proposed Revision to Technical Specifications 3.4.1.2

Revise the footnote to the LCO statement for the affected Technical Specification:

"All reactor coolant pumps may be deenergized for up to 1 hour (4 hours on a one time basis for natural circulation testing following steam generator replacement) provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature."

Background

The Catawba Unit 1 steam generators were replaced during the recently completed 1EOC9 outage. One desirable post-outage test was re-measurement of the natural circulation capability of the Reactor Coolant System. This test was previously performed during startup testing following initial fuel loading of the unit. The replacement steam generators (RSG's) have design features (greater tube bundle surface area and a taller tube bundle) that are expected to enhance the natural circulation capabilities of the unit. In order to get actual plant performance data, it is desirable to perform a natural circulation test under realistic plant conditions. The test data will be utilized to validate analysis and simulator models. Plant operators will also receive valuable experience from performance of the test. This natural circulation test is planned for Catawba Unit 1 only. McGuire Units 1 and 2 are also replacing their steam generators with identical Babcock & Wilcox International RSG's. The Catawba test data will also be utilized to validate analysis and simulator models for McGuire.

A natural circulation test was performed for Catawba Unit 1 during the initial startup testing program as described in UFSAR Section 14.5.30. This test utilized low reactor power to simulate core decay heat. The reactor was maintained at approximately 3% power during the test. A Special Test Exception (3/4.10.4) was included in the Technical Specifications to allow natural circulation testing in Mode 2. Review of this initial test indicated that a more realistic and useful test could be performed in Mode 3 using actual core decay heat. Performance of the test in Mode 3 would have the nuclear safety advantage that the reactor would be sub-critical throughout the test. As now written, Technical Specification 3.4.1.2 allows all reactor coolant pumps to be deenergized for up to 1 hour. From previous experience with natural circulation events and testing, 1 hour is not sufficient time to stabilize plant parameters and collect sufficient data. The previous test method (reactor critical in Mode 2) had the further drawback that the test actually resulted in violation of Technical Specification 3.10.3 (Physics Tests) when a reactor coolant pump could not be re-started for a period of time at the completion of the test due to a high stand pipe level. This Technical Specification could not be violated if the test were performed in Mode 3.

The planned one time natural circulation test will be performed with the unit in a stable Mode 3 condition. It is anticipated that the test will be performed prior to entering the 1EOC10 refueling outage. However, other opportunities may be available prior to that time. The test will not be performed with the unit in a degraded condition that could adversely affect reactor operator response in conducting the test. A written and approved test procedure will be used for conducting the test

With the Unit in Mode 3 (~557°F, 2235 psig), all four reactor coolant pumps will be simultaneously tripped. Establishment of natural circulation will be verified by observation of wide range loop temperatures as well as core exit thermocouples. Stable natural circulation will be maintained for a minimum of 30 minutes while data are gathered to verify that the acceptance criteria have been met. Specific test termination criteria will be provided to the reactor operators, such as:

Lowest RCS Subcooling Margin	⌠ 15°
Core outlet Temperature (TS 3.4.1.2)	< 10° F below saturation temperature
S/G Level (N/R)	< 11%
Pressurizer level	<17% or >5% unexplained decrease
RCS T-hot (W/R) or any valid Incore T/C	>610°F

RCS ΔT in any loop >65°F

Dilution of RCS (controlled or uncontrolled) (TS 3.4.1.2)

Reactor Coolant Pumps de-energized for greater than 4 hours

Recovery from the test will be performed using existing operating procedures (OP/1/A/6200/01, Chemical and Volume Control System; OP/1/A/6150/02A, Reactor Coolant Pump Operation; OP/1/A/6100/02, Controlling Procedure for Unit Shutdown.

Technical Justification

The current Catawba Unit 1 Technical Specification 3.10.4 allows performance of a post steam generator replacement natural circulation test in Mode 2 with no time limit for restarting a reactor coolant pump. Considering nuclear safety issues, the proposed test method is preferred in that the reactor would be subcritical at the start of the test and no trip setpoint changes are needed or bypassed to perform the test. It is estimated that approximately two hours is needed to stabilize plant conditions following initiation of the test and to gather test data. Therefore four hours is considered to be a reasonable limit for contingencies and for re-starting a reactor coolant pump.

NO SIGNIFICANT HAZARDS EVALUATION

The execution of a natural circulation test in Mode 3 associated with the proposed change has been evaluated against the standards of 10 CFR 50.92. The results of the evaluation of this activity are as follows:

1) *The activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed natural circulation test would be performed in Mode 3 with the reactor subcritical. This transient is bounded by the transient analyzed in UFSAR Section 15.2.6, Loss of Non-Emergency AC Power to the Station Auxiliaries. For this ANS Condition II event, the reactor is assumed to be operating at 102% power, the turbine driven auxiliary feedwater pump is assumed unavailable and each steam generator is assumed to have 18% of the steam generator tubes plugged. By contrast, the planned natural circulation test would be performed with the reactor subcritical, less than 0.1% of the tubes plugged in each steam generator, and all support systems such as auxiliary feedwater, operable for the test. Therefore, the proposed natural circulation test would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) *The activity does not create the possibility of a new or different type of accident from any accident previously evaluated.*

The proposed change does not involve a physical alteration of the unit (i.e., no new or different equipment will be installed), nor will the function of equipment be changed. The change will allow for a one time performance of a natural circulation test in Mode 3 which will provide useful data on the natural circulation capabilities of the new Babcock and Wilcox International (BWI) steam generators that were recently installed at Catawba Unit 1. The test data will be utilized to validate analysis and simulator models. Plant operators will also receive valuable experience from performance of the test. The test will be conducted using written and approved procedures. An Emergency procedure (EP/1/A/5000/ECA-0.1) is also available to the Operators for this test. This test is bounded by the Loss of Non-Emergency AC Power to the Station Auxiliaries event in Section 15.2.6 of the Catawba UFSAR. For these reasons, the planned natural circulation test will not create the possibility of a new or different type of accident from any previously evaluated.

3) *The activity does not involve a significant reduction in the margin of safety.*

Margin of safety is associated with confidence in the ability of the fission product barriers (the fuel and fuel cladding, the Reactor Coolant System pressure boundary, and the containment) to limit the level of radiation doses to the public. As demonstrated by the bounding UFSAR analysis in Section 15.2.6, none of the fission product barriers are adversely impacted by the proposed one-time change. The proposed change does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. For these reasons, the activity does not involve a significant reduction in the margin of safety.

Environmental Impact Assessment

This change to the Technical Specifications will allow performance of a one-time natural circulation test following steam generator replacement. It has been determined that this amendment will not involve a significant hazards consideration, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and that there is no significant increase in the individual or cumulative occupational radiation exposure. This amendment request therefore meets the criteria of 10 CFR 51.22.(c)(9) for categorical exclusion from an environmental impact statement.