December 16, 1993

Docket Nos. 50-413
and 50-414

Mr. David L. Rehn
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Distribution
Docket File
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S.Varga
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G.Hill(4) P1-37
C.Grimes 11F23
ACRS(10) P-135
PA 17F2
OC/LFMB MNB4702
E.Merschoff, RII

Dear Mr. Rehn:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 STEAM GENERATOR INTERIM PLUGGING CRITERIA FOR UNIT 1 CYCLE 8 (TAC NOS. M87840 AND M87841)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 111 to Facility Operating License NPF-35 and Amendment No. 105 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 5 and 14, 1993, as supplemented November 15 and December 14, 1993.

The amendments revise the TS to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
Robert E. Martin, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 111 to NPF-35
2. Amendment No. 105 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDII-3/LA	PDII-3/PM	OGC	PDII-3/D	
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DATE	12/14/93	12/15/93	12/15/93	12/16/93	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 16, 1993

Docket Nos. 50-413
and 50-414

Mr. David L. Rehn
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Rehn:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 STEAM GENERATOR INTERIM PLUGGING CRITERIA FOR UNIT 1 CYCLE 8 (TAC NOS. M87840 AND M87841)

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Robert Martin

Robert E. Martin, Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
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cc w/enclosures:
See next page

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Catawba Nuclear Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated October 5 and 14, 1993, as supplemented November 15 and December 14, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 111, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses for

Loren R. Plisco, Acting, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: December 16, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated October 5 and 14, 1993, as supplemented November 15 and December 14, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

    Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 105 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses for

Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: December 16, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.111

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO.105

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

	Remove Pages	Insert Pages
	3/4 4-13	3/4 4-13
	3/4 4-14	3/4 4-14
	3/4 4-16	3/4 4-16
	3/4 4-16a	3/4 4-16a
	3/4 4-16b	3/4 4-16b
	3/4 4-27	3/4 A4-27
		3/4 B4-27
	B 3/4 4-3a	B 3/4 4-3a
	B 3/4 4-5	B 3/4 4-5
Index	VII	VII

SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. For Unit 1, in addition to the 3% sample, all tubes for which the alternate plugging criteria has been previously applied shall be inspected in the tubesheet region.

d. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

e. For Unit 1, implementation of the interim steam generator tube/tube support plate elevation plugging limit for Cycle 8 requires a 100% bobbin probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (OD SCC) indications. An inspection using the rotating pancake coil (RPC) probe is required in order to show operability of tubes with flaw like bobbin coil signal amplitudes greater than 1.0 volt but less than 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as OD SCC.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Roll Expansion is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the tubesheet.

11) F* Distance is the minimum length of the roll expanded portion of the tube which cannot contain any defects in order to ensure the tube does not pull out of the tubesheet. The F* distance is 1.60 inches and is measured from the bottom of the roll expansion transition or the top of the tubesheet if the bottom of the roll expansion is above the top of the tubesheet. Included in this distance is a safety factor of 3 plus a 0.5 inch eddy current vertical measurement uncertainty.

12) Alternate tube plugging criteria does not require the tube to be removed from service or repaired when the tube degradation exceeds the repair limit so long as the degradation is in that portion of the tube from F* to the bottom of the tubesheet. This definition does not apply to tubes with degradation (i.e., indications of cracking) in the F* distance.

13) The Tube Support Plate Interim Plugging Criteria Limit is used for disposition of a steam generator tube for continued service that is experiencing outer diameter initiated stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin probe signal amplitude of flaw like indications. The plant specific guidelines used for all inspections shall be consistent with the eddy current guidelines in Appendix A of WCAP-13854 as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage parameters as specified in Specification 4.4.5.2.

    1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 1.0 volts, regardless of the depth of tube wall penetration, if, as a result, the projected end of cycle distribution of crack indications

is verified to result in total primary to secondary leakage less than 30.0 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in Attachment 4 of the Supplement to Technical Specification amendment dated December 14, 1993 (SG-93-12-006).

2. A tube can remain in service with a bobbin coil signal amplitude greater than 1.0 volt but less than 2.7 volts provided a rotating pancake coil (RPC) inspection does not detect degradation.

3. Indications of degradation with a flaw type bobbin coil signal amplitude of equal to or greater than 2.7 volts will be plugged or repaired.

Certain tubes as identified in WCAP-13494, Rev. 1, will be excluded from application of the Interim Plugging Criteria Limit as it has been determined that these tubes may collapse or deform following a postulated LOCA + SSE Event.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2. For Unit 1, tubes with defects below F* fall under the alternate tube plugging criteria and do not have to be plugged.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

SURVEILLANCE REQUIREMENTS (Continued)

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes repaired.

c. For Unit 2, results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Unit 1, the results of inspections for all tubes for which the alternate tube plugging criteria has been applied shall be reported to the Nuclear Regulatory Commission in accordance with 10 CFR 50.4, prior to restart of the unit following the inspection. This report shall include:

1) Identification of applicable tubes, and

2) Location and size of the degradation.

e. For Unit 1, the results of inspections performed under 4.4.5.2 for all tubes in which the tube support plate elevations interim plugging criteria has been applied shall be reported to the Commission following the inspection and prior to Cycle 8 operation. The report shall include:

1. Listing of applicable tubes.

2. Location (applicable intersections per tube) and extent of degradation (voltage).

3. Projected Steam Line Break (SLB) Leakage.

3/4.4.8 SPECIFIC ACTIVITY (FOR UNIT 1)

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 0.58 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross specific activity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5. (Unit 1)

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 0.58 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours;

b. With the gross specific activity of the reactor coolant greater than 100/E microCuries per gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

c. The provisions of Specification 3.0.4 are not applicable.

*With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY (FOR UNIT 2)

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross specific activity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5. (Unit 2)

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours;

b. With the gross specific activity of the reactor coolant greater than 100/E microCuries per gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

c. The provisions of Specification 3.0.4 are not applicable.

*With T_{avg} greater than or equal to 500°F.

STEAM GENERATORS (Continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. For Unit 1, defective tubes which fall under the alternate tube plugging criteria do not have to be repaired. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.13.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry, ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 0.4 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Catawba site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 0.58 microCurie/gram DOSE EQUIVALENT I-131 for Unit 1, and 1.0 microCurie/gram DOSE EQUIVALENT I-131 for Unit 2, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letters dated October 5 and 14, 1993, as supplemented November 15 and December 14, 1993, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations for Unit 1. Administrative changes were made to preserve current TS applicable to Unit 2. The November 15 and December 14, 1993, letters provided clarifying information and revisions to the reactor coolant specific activity that did not change the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The amendments revise, in part, TS Section 3/4.4.5 and associated Bases to allow the continuance of a voltage-based steam generator tube plugging criteria for defects located at the tube support plate elevations. All of the proposed changes are to be applicable to Cycle 8 operation only.

The staff reviewed a similar request that was applicable to Cycle 7 operation as documented in an amendment package dated September 25, 1992, "Issuance of Amendments - Catawba Nuclear Station, Unit 1 (TAC No. M84221)," and a Safety Evaluation dated July 30, 1993, "Catawba Nuclear Station, Unit 1 - Safety Evaluation Regarding Steam Generator Tube Interim Plugging Criteria Mid-Cycle Inspection (TAC No. M86116)" (referred to as Reference 1 and Reference 2, respectively). The staff concluded in Reference 1 that the proposed interim tube repair limits and leakage limits would ensure adequate structural and leakage integrity of the steam generator tubing at Catawba Nuclear Station Unit 1, consistent with applicable regulatory requirements, until May 1, 1993. The staff concluded that a mid-cycle inspection was necessary for the reasons specified in Reference 1. As a result of additional information provided by the licensee on margins to tube burst, bobbin coil voltage normalization, and main steam line break (MSLB) leakage, the staff concluded in Reference 2 that the proposed interim tube repair limits and leakage limits would ensure adequate structural and leakage integrity of the steam generator tubing at Catawba Nuclear Station Unit 1, consistent with applicable regulatory requirements, for the remainder of the Cycle 7 operation without performance of a mid-cycle inspection. This safety evaluation reflects additional information/operating experience that has been acquired since Reference 2 was issued.

The staff is currently developing a generic interim position on voltage-based limits for outside-diameter stress corrosion cracking at tube support plate elevations. The staff has recently published several tentative conclusions regarding voltage-based plugging criteria in draft NUREG-1477; however, the staff is continuing to evaluate an acceptable generic position which takes into consideration public comments received on draft NUREG-1477 and additional data which has been made available from European nuclear power plants. The staff currently plans to document its final position in a generic letter with the associated technical basis being documented in the final version of NUREG-1477.

In the meantime, pending completion and issuance of the staff's generic position on voltage-based interim plugging criteria (IPC), the staff is continuing to evaluate IPC proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety. The staff's current evaluation, documented herein, is, for the most part, consistent with the staff's previous case-specific evaluation of the Catawba Unit 1 IPC application. One noteworthy exception is that the potential steam generator tube leakage during a postulated MSLB is calculated in accordance with the methodology described in draft NUREG-1477.

3.0 BACKGROUND

The modifications to the tube repair limits, as documented in References 1 and 2, included a one-volt repair criterion for axially oriented outside-diameter stress corrosion cracking (ODSCC) flaws confined to within the thickness of the tube support plate in lieu of the depth-based limit of 40-percent. The staff's review concluded that the interim tube repair limits and leakage limits would ensure adequate structural and leakage integrity of the steam generator tubing at Catawba Nuclear Station, Unit 1, consistent with applicable regulatory requirements, for Cycle 7 operation. The licensee's current proposal is applicable to Cycle 8 operation and is similar to the licensee's previous proposal which was approved as documented in References 1 and 2.

3.1 Tube Integrity Issues

The purpose of the TS tube repair limits is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria 14, 15, 31, and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits. The traditional strategy for accomplishing these objectives has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowance for eddy current measurement error and flaw growth between inspections has been added to the minimum wall thickness requirements (consistent with the Regulatory Guide) to arrive at a depth-based repair limit. Enforcement of a minimum wall thickness requirement implicitly ensures leakage integrity (during normal operation and accidents), as well as structural integrity. It has been recognized, however, that defects, especially cracks, will occasionally grow entirely through-wall and develop small leaks. For this reason, limits on allowable primary-to-secondary leakage have been established in the TS to ensure timely plant shutdown before adequate structural and leakage integrity of the affected tube is impaired.

The interim tube repair limits for Catawba Unit 1 consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide data base from the pulled tube examinations show that for bobbin indications at or near 1.0 volt (i.e., the IPC repair limit), maximum crack depths range between 20% and 98% through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For indications at or near 2.0 volts, the maximum crack depths have been found to generally range between 50% and 100% through-wall. Clearly, many of the tubes which will be found to contain non-repairable indications under the proposed interim criteria may develop through-wall and near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents. The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 3.2 and 3.3, respectively. Section 3.4 contains the staff's evaluation of several inspection issues and Section 3.5 addresses the assessment of the overall IPC methodology to be performed following this refueling outage at Catawba Unit 1.

3.2 Structural Integrity

In support of the 1.0 volt repair limit approved in References 1 and 2 for Cycle 7 operation, the licensee developed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 1.0 volt interim repair criterion would retain adequate structural margins during Cycle 7 operation, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled tube data and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized to be consistent with the calibration standard voltage set-ups and voltage measurement procedures described in WCAP-13494 (Revision 1) and WCAP-13854. The normalization was performed to ensure consistency among the voltage data in the burst pressure/bobbin voltage correlation and consistency between the voltage data in the correlation and the field voltage measurements at Catawba Unit 1.

For any specific individual tube, voltage measurement uncertainty and/or voltage growth may exceed the value assumed in the previously mentioned Regulatory Guide 1.121 deterministic analysis since the deterministic analysis does not consider the full tails of the voltage measurement uncertainty and voltage growth distributions. Similarly, the burst pressure for some tubes may be less than the 95% lower prediction interval values in the burst pressure/bobbin voltage correlation. These distribution tails may involve sizable numbers of tubes in instances where a large number of tubes with indications are being accepted for continued service. To directly account for these uncertainties, Monte Carlo methods have typically been used to demonstrate that the probability of burst during a postulated MSLB accident is acceptably low for the distribution of voltage indications being left in service. Under this approach, the beginning-of-cycle (BOC) indications left in service are projected to the end-of-cycle (EOC) by randomly sampling the non-destructive examination (NDE) uncertainty probability distribution and the voltage growth per cycle probability distribution. For each EOC Monte Carlo sample of bobbin voltage, the burst pressure/bobbin voltage correlation is randomly sampled to obtain a burst pressure. A number of Monte Carlo samples (e.g., 100,000) are performed for the entire BOC distribution. The probability of tube burst under postulated MSLB differential pressures is obtained as the sum of the samples resulting in burst pressures less than the MSLB pressure differential of 2560 psi divided by the number of times the distribution of indications left in service is sampled.

For the proposed 1.0 volt IPC, the projected deterministic EOC voltage is 1.82 volts assuming the 95% cumulative probability values of voltage measurement uncertainty (with probe wear standard) and voltage growth (for Cycle 7). Using the lower 95% prediction interval curve for burst pressure as a function of voltage, the maximum allowable EOC voltage that will satisfy the limiting burst pressure criterion in Regulatory Guide 1.121 is approximately 4.2 volts. That staff notes that in cases such as Catawba Unit 1 where thousands of indications are being found, that the above deterministic analysis assures that the vast majority, but not all, of the indications will meet the burst pressure criteria of Regulatory Guide 1.121. As a result of the limitations of the deterministic analysis, the Monte Carlo analyses referenced above for calculating the probability of rupture given a MSLB is performed.

The staff concludes that the proposed 1.0 volt interim criterion will provide adequate assurance that the vast majority of tubes with indications which are accepted for continued service during Cycle 8 operation will meet the burst pressure criteria of Regulatory Guide 1.121. However, as discussed in WCAP-13494 Revision 1 and in Reference 2, application of the IPC will not be applied to tubes which may collapse or deform following a postulated loss-of-coolant accident with a concurrent safe shutdown earthquake. These tubes were identified in WCAP-13494 Revision 1.

The licensee's current submittal permits bobbin indications greater than 1.0 volt but less than 2.7 volts to remain in service if a motorized rotating pancake coil (MRPC) probe inspection does not detect a flaw, and it requires flaw indications with a bobbin voltage greater than 2.7 volts to be plugged or repaired regardless of MRPC probe findings. The staff notes that the 2.7 volts reflects an alternate plugging criteria (APC) voltage limit that was derived in WCAP-13854.

During the staff's review, the staff noted that several data points were excluded from the database used in determination of the 2.7 volt limit. As a result, the staff requested the licensee to submit their technical basis for not including all the data points contained within Table 4-1 of WCAP-13854. The licensee responded with a letter dated November 15, 1993. In the licensee's letter, one specimen (i.e., model boiler specimen 598-1) was identified as being an outlier; however, there was no specific error in either the burst pressure test or voltage measurement for this outlier. In NUREG-1477, the staff reached the tentative conclusion that unless the test (e.g., burst pressure, voltage measurement) itself can be shown to be invalid, then the data should be included in the burst pressure/bobbin voltage correlation. Therefore, the staff believes this data point should be included in the burst pressure correlation. At the staff's request, the licensee provided an assessment of the change in the voltage limit as a result of including this data point in the correlation. The inclusion of model boiler specimen 598-1 in the burst pressure correlation had a negligible change on the voltage limit (i.e., 2.7 volts).

In addition to model boiler specimen 598-1, pulled tube specimen R28C41 from plant "S" was excluded from the burst pressure database. This specimen was excluded because it had exhibited an incomplete burst during the pressure test. The staff is continuing to review the appropriateness of eliminating this data point from the correlation as part of its on-going generic review effort; however, for this IPC application, the staff believes that even if this data point were to be included in the database that the change in the voltage limit (i.e., 2.7 volts) would be minor.

With respect to the proposed exception to the 1.0 volt criterion,the staff notes that short and/or relatively shallow cracks that are detectable by the bobbin coil may sometimes not be detectable by the MRPC probe, although the MRPC probe is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance. The staff further notes that burst strength is not a unique function of voltage, rather for a given voltage there is a statistical distribution of possible burst strengths as indicated in the burst pressure/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the MRPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). The 2.7 volt cutoff, such that all bobbin indications would be plugged or repaired (with or without confirming MRPC indications), provides assurance that all excessively degraded tubes will be removed from service. The staff further notes that the projected leakage from these tubes (i.e., tubes with bobbin voltages between 1.0 and 2.7 volts which exhibited no detectable degradation during the MRPC inspection) will be considered in the leak rate assessment performed by the licensee prior to plant restart. Thus, the staff finds the proposed exception to the 1.0 volt criterion to be acceptable.

To calculate the conditional probability of rupture given an MSLB, the Monte Carlo analysis described previously was performed for the projected distribution of indications at the EOC 7 based on the actual distribution of indications left in service at the BOC 7 at Catawba Unit 1. This analysis indicated that implementation of a 1.0 volt repair criterion at that time would have yielded a conditional probability of burst given a MSLB of approximately 1.1×10^{-5}. In Reference 2, however, the staff noted that the conditional probability of burst given a MSLB, referenced above, should be calculated from a BOC distribution that includes: 1) tubes that were left in service between 1.0-volt and the APC repair limit (2.5 volts at that time) that were not confirmed by MRPC to be degraded, and 2) non-detected ODSCC indications. In addition, the staff noted in draft NUREG-1477 and in Reference 2 that the burst pressure correlation should include all data unless a specific error in either the burst pressure test or voltage measurement occurred. As a result, the staff estimated in Reference 2 that taking these considerations into account resulted in a conditional probability of burst given a MSLB of approximately 4×10^{-5}. These values indicate an extremely low probability of burst given a MSLB, approximately three orders of magnitude less than the value considered in the staff's generic risk assessment for steam generators contained in NUREG-0844.

In WCAP-13854, the licensee described an additional analysis for calculating the probability of rupture given an MSLB based, in part, on the expected tube support plate deflection and the crack length. The licensee has stated that this analysis demonstrates that for Catawba Unit 1 it is not necessary to perform burst probability analyses for indications left in service for Cycle 8 IPC implementation. Furthermore, the licensee concluded that tube burst can be considered negligible and ignored for the alternate repair criteria at Catawba Unit 1 for ODSCC at tube support plate elevations. Since the staff is continuing to review tube support plate displacement analyses as part of the on-going generic review effort, the licensee has committed to calculate the probability of rupture given a MSLB with the methodology described in NUREG-1477. The results of this analysis are to be submitted to the NRC following completion of the refueling outage.

3.3 Leakage Integrity

A number of the indications satisfying the proposed interim 1.0 volt repair limit can be expected to have, or to develop, through-wall and/or near through-wall crack penetrations during the next cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. The staff finds that adequate leakage integrity during normal operating conditions is assured by the TS limits on allowable primary-to-secondary leakage. These limits were incorporated into the TS upon issuance of Reference 1. Specifically, a primary-to-secondary operational leakage limit of 150 gallons per day (gpd) through any one steam generator and a total allowable primary-to-secondary operational leakage limit through all steam generators of 0.4 gallons per minute were adopted in Reference 1. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded.

As the basis for estimating the potential leakage during MSLB accidents, Westinghouse has correlated leakage test data obtained under simulated MSLB conditions with the corresponding bobbin voltage amplitudes. The correlation is based on a linear regression fit of the logarithms of the corresponding leak rates and bobbin voltages. The leak rate data exhibits considerable scatter relative to the mean correlation. Thus, prediction intervals for leak rate at a given voltage have been established to statistically define the range of potential leak rates. As part of the on-going review of the APC, the staff is continuing to review the correlation of the leak rate data to bobbin voltage. Until the issue of the leak rate versus voltage correlation is resolved, the staff has concluded that a voltage-based approach can be used if these non-conservatisms are accounted for and sufficient conservatisms are included in the analysis. Therefore, at the staff's request, the licensee has committed to provide a calculation of potential MSLB leakage by a methodology designed to address the staff concerns. The methodology that the licensee will use to calculate the MSLB leakage is described in draft NUREG-1477.

For purposes of this IPC application, the MSLB leakage analysis should be performed with the most recent leak rate data for 3/4-inch outside diameter tubing. In the leak rate database submitted by the licensee, several data points were excluded for various reasons. Similar to the evaluation of the burst pressure database, the staff believes the leak rate database should include data points where there was no specific error in either the leakage measurement or the voltage measurement. Therefore, model boiler specimens 598-1 and 598-3 should be included in the database for purposes of this IPC application. The staff notes that the leakage from pulled tube R28C41 from plant "S" was not accounted for in the database since the leakage from this tube exceeded the capability of the leak rate measurement facility. As a result of this and the geometry of the crack, the staff believes that significant leakage may have come from this crack and that this data point should be accounted for in the database. Pending further experimental and/or analytic analyses for this data point (to demonstrate the structural and leakage characteristics under postulated accident conditions), the staff believes that an appropriate leakage value to be assigned to this data point is the value predicted by the CRACKFLO computer code.

The staff concludes that calculating the primary-to-secondary leakage in accordance with the methodology described in NUREG-1477 is acceptable for this IPC application. The staff notes that the database used in this assessment should also include the three specimens referenced above. In addition, the voltage growth distribution used in predicting the primary-to-secondary leakage under postulated accident conditions should (1) consider the most recent voltage growth data (i.e., Cycle 7), and (2) be adjusted for the planned Cycle 8 duration. Evaluation of the acceptability of the estimated primary-to-secondary leakage rate for postulated accident conditions should be based on current staff positions regarding calculational methods and limits for offsite doses.

The staff notes that the licensee has modified the procedures for adjusting the leak rate measurements for various differential pressure conditions. The changes were made to reconcile NRC staff comments expressed in draft NUREG-1477. The staff has not completed its review of these modifications;

however, the staff has concluded that the changes made will have a minor effect on the predicted leak rate under postulated accident conditions when performed at 2560 psi. The staff will continue to review the modifications to the leak rate measurement adjustment procedures as part of its on-going generic review effort.

3.4 Inspection Issues

In support of the proposed interim repair limit, the licensee proposes to utilize the eddy current test guidelines provided in Appendix A of WCAP-13854 to ensure the field bobbin indication voltage measurements are obtained in a manner consistent with how the voltage limit was developed. These guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators. Appendix A of WCAP-13854 contains, in part, requirements to:

1. Record all indications regardless of voltage amplitude.

2. Perform MRPC inspections of 100 tubes, including tubes with dent indications exceeding 5 volts as measured by the bobbin coil and also including tube support plate intersections with artifact indications. Expansion of this sample, if required, will be based on the nature and number of the flaws discovered.

3. Perform MRPC examinations of all tubes with bobbin voltages in excess of 1.0 volt unless the tube is to be plugged or sleeved.

In its letter of December 14, 1993, the licensee discussed several aspects of its inspection program including the results of the inspection. The licensee stated that: 1) all hot leg tube support plate indications whose dent voltage exceeded 4 volts were inspected with an MRPC probe, 2) no indications at the flow distribution baffle were identified, 3) no unexpected inspection findings, relative to the assumed characteristics of the flaws at the tube support plate elevations, were identified, and 4) no detectable circumferential indications or detectable indications extending outside the thickness of the tube support plate were identified.

In Reference 1, the staff made several observations on the licensee's inspection program. Some of these observations are listed below:

1. A reference calibration of 2.75 volts for the four 20% through-wall holes in the 550/130 kHz support plate suppression mix was used. The staff concluded that this calibration procedure was adequate and is consistent with the eddy current analysis guidelines used in the development of the burst pressure/bobbin voltage correlation; however, the staff stated that a more consistent calibration may be obtained using the four 0.033-inch diameter 100% through-wall holes rather than the four 20% through-wall holes due to the elimination of the difference in hole depth and the variations in the shape of the hole bottom.

2. A probe wear standard was not used during the inspection. The staff recommended that all future inspections should include the use of a wear standard.

3. The 300 kHz MRPC signal was calibrated to 10 volts vice 20 volts for the 100% EDM notch. The staff concluded that since the MRPC voltages are not used in the structural integrity analysis that this did not affect the IPC application; however, consideration should be given to using a consistent set of guidelines in order to facilitate the comparison of readings at later inspections.

4. A 0.610-inch diameter probe was used during the inspection vice a 0.620-inch diameter probe. The staff noted that additional confidence in the relationship of the field voltage readings to the burst pressure correlation voltage readings would be obtained if the same diameter probes were used in both applications.

5. Voltage calls were being made by one analyst with subsequent review by a two-person resolution team. The staff recommended that all future inspections in which the IPC was implemented should include voltage calls being developed by two independent analysts with discrepancies being resolved with a resolution process.

To address several of these concerns, the licensee has:

1. Implemented the use of a probe wear standard as described in WCAP-13854.

2. Revised the normalization process of the 400 kHz MRPC signal to 20 volts for the 100% EDM notch.

3. Specified that the 0.610-inch probe is the appropriate probe to be used during the inspection.

4. Revised the voltage resolution process to require analysis by a resolution analyst if the voltage values called by the independent analysts deviate by more than 20% and one or both of the calls exceeds 1.0 volts. In addition, the licensee stated (during a teleconference on December 6, 1993), that the voltage resolution process at Catawba involved resolution of signals whose voltages differed by 0.1 volts.

The licensee's current calibration procedure still requires calibration on the four 20% through-wall holes. The staff believes that a more consistent calibration can be obtained using the four 0.033-inch diameter 100% through-wall holes rather than the four 20% through-wall holes due to the elimination of the difference in hole depth and the variations in the shape of the hole bottom. This position was elaborated in draft NUREG-1477. The staff is currently reviewing public comments pertaining to the calibration procedure; however, pending completion of the review the staff believes that for purposes of this IPC application that calibration on the four 20% through-wall holes will not introduce a significant amount of error.

3.5 Overall Assessment

The staff notes that the methodologies described in WCAP-13854 and in this safety evaluation for predicting MSLB leakage and the probability of rupture given a MSLB depend largely, in part, on the ability to accurately predict an EOC voltage distribution. An assessment of the effectiveness of the methodology described in WCAP-13494 and in this safety evaluation for predicting the EOC voltage distribution is warranted to confirm the adequacy of the methodology used. In WCAP-13854, the licensee has committed to provide such an assessment to the NRC staff. The assessment for Catawba Unit 1 will be made in a manner consistent with the methodology described in WCAP-13494 Revision 1 and in WCAP-13854. The staff notes that this assessment should address any discrepancies between the predicted and actual EOC voltage values. The staff requests the following information be included in this assessment in both tabular and graphical form:

a. EOC 6 voltage distribution - all indications found during the inspection regardless of MRPC confirmation
b. Cycle 6 growth rate (i.e., from BOC 6 to EOC 6)
c. EOC 6 repaired indications voltage distribution - distribution of indications presented in (a) above that were repaired (i.e., plugged or sleeved)
d. Voltage distribution for indications left in service at the BOC 7 regardless of MRPC confirmation - obtained from (a) and (c) above
e. Voltage distribution for indications left in service at the BOC 7 that were confirmed by MRPC to be crack-like or not MRPC inspected
f. Non-destructive examination uncertainty distribution used in predicting the EOC 7 voltage distribution
g. Projected EOC 7 voltage distribution using the methodology in WCAP-13494 Revision 1
h. Actual EOC 7 voltage distribution - all indications found during the inspection regardless of MRPC confirmation
i. Cycle 7 growth rate (i.e., from BOC 7 to EOC 7)
j. EOC 7 repaired indications voltage distribution - distribution of indications presented in (h) above that were repaired (i.e., plugged or sleeved)
k. Voltage distribution for indications left in service at the BOC 8 regardless of MRPC confirmation - obtained from (h) and (j) above
l. Voltage distribution for indications left in service at the BOC 8 that were confirmed by MRPC to be crack-like or not MRPC inspected
m. Non-destructive examination uncertainty distribution used in predicting the EOC 8 voltage distribution
n. Projected EOC 8 voltage distribution using the methodology in WCAP-13494 Revision 1

The staff recognizes that compilation of this confirmatory assessment on the overall IPC methodology may not be possible until after completion of the refueling outage. The assessment should be provided to the NRC staff as soon as possible following completion of the refueling outage. The staff recognizes that some of the information referenced above has been submitted to the NRC in the past. As a result, reference to the appropriate source (including page number) would be acceptable.

4.0 SUMMARY

Based on the above, it can be concluded that adequate structural integrity of the steam generator tubing can be ensured for Cycle 8 at Catawba Unit 1, consistent with applicable regulatory requirements. In addition, the staff concludes that the methodology described herein for determining the expected primary-to-secondary leakage during a postulated MSLB at the end of fuel Cycle 8 for Catawba Unit 1 is acceptable.

The staff's approval of the proposed interim repair limit is based on the licensee being able to demonstrate that the primary-to-secondary leakage during the postulated MSLB will be acceptable. In addition, the licensee has agreed to report, prior to Cycle 8 operation, the results of this leakage analysis.

5.0 RADIOLOGICAL ASSESSMENT

The licensee proposed to change the value of total primary to secondary leakage in TS 4.4.5.4 from the value of 1.0 gpm to 30.0 gpm and to change the value of the allowable primary coolant activity level in TS 3.4.8.a from 1.0 microCurie per gram dose equivalent I-131 to 0.58 microCuries per gram. To demonstrate the acceptability of the proposed change the licensee presented the results of their analysis of the dose consequences of a main steamline break accident using these parameters. The results of the licensee's analysis are presented below.

Consequences of Main Steamline Break Accident
Licensee's Calculations (Dose in Rems)

	EAB	LPZ
Thyroid (Accident Initiated Spike)	30	21
Thyroid (Pre-existing Spike)	102	47
Whole Body (Pre-existing Spike)	0.013	0.0063

The staff has independently calculated the doses resulting from a main steamline break accident. The results of the staff's calculations confirm the licensee's conclusions that the doses would be within the limits established by Standard Review Plan (SRP) 15.1.5, Appendix A. It should be noted that the staff performed these calculations in accordance with the methodology associated with SRP 15.1.5, Appendix A, and that the staff did not credit the licensee with the removal associated with the letdown flow and the removal of radioiodine by the letdown demineralizer. The latter assumption was made by the licensee in its calculations. However, the staff concluded that assuming credit for such removal was inappropriate because the letdown demineralizers are never tested to demonstrate iodine removal capability. Furthermore, the letdown demineralizers are not safety-related components.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 57849 dated October 27, 1993). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

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Date: December 16, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 107 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated October 25, 1993, as supplemented December 3 and 6, 1993, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would reduce the required minimum measured reactor coolant system (RCS) flow from 385,000 gallons per minute (gpm) to 382,000 gpm. The reasons for this request are that the degrading of the steam generator tubes in Catawba Unit 1 and McGuire Units 1 and 2 have necessitated that tubes be plugged or sleeved, which reduces the available flow area in the steam generators and consequently reduces flow through the core. In addition, a hot leg temperature streaming phenomenon has affected the ability to accurately measure flow. As a result of these effects, it was difficult to ensure meeting the TS minimum flow requirements to maintain 100% power operation. The December 3 and 6, 1993, letters provided clarifying information that did not change the scope of the October 25, 1993, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The following TS were modified to reflect the reduction in RCS flow:

1) Figure 2.1-1, Reactor Core Safety Limits - Four Loops in Operation,

2) Figure 3.2-1, Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation, and

3) The overtemperature delta T (OTΔT) and overpower delta T (OPΔT) setpoint equation constants in Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints.

These revisions are applicable to McGuire Units 1 and 2 and to Catawba Unit 1. The TS changes for McGuire will be reflected in an amendment to the McGuire facility operating licenses. The TS changes discussed above are not applicable to Catawaba Unit 2, because the steam generators in Unit 2 have not required tube plugging or sleeving to the extent of the other three units. However, a change was also made to Catawba Unit 2 relating to the OPΔT allowable values due to a minor error discovered as stated in Section 2.1. In addition, there were editorial changes.

The NRC staff is continuing its review of the licensee's proposal to modify the Limiting Condition for Operation of TS 2.1-1 to make the DNBR and centerline fuel temperature (CFT) limits consistent with the Babcock and Wilcox Improved Standard Technical Specification. The licensee amended its submittal by letter dated December 6, 1993 (Reference 3), to address only the changes required by the reduction in the required measured minimum RCS flow for this amendment.

2.0 EVALUATION

2.1 Revision of OTΔT and OTΔP Parameters in Table 2.2-1

To support the reduction in measured minimum RCS flow (MMF), changes were required for the OPΔT setpoints for McGuire Units 1 and 2 and Catawba Unit 1. These changes involved recalculation of the TS allowable values of the trip functions. The revised core thermal limits were generated to reflect the reduced MMF of 382,000 gpm. Based on these new protection limits, the OTΔT setpoint constants (Note 1 of Table 2.2-1), and the OPΔT setpoint equation constants (Notes 2 and 3 of Table 2.2-1 for McGuire and Catawba, respectively) were revised to reflect the necessary changes. The impact of the reduced flow on the coefficients was partially offset by a reduction in the margin assumed in the calculation of the coefficients.

The revised OPΔT allowable values are more restrictive than the existing values. In the course of the these calculations, a minor error was discovered by DPC that affected the existing allowable values for all four units. This resulted in a recalculation, of the allowable value for Catawba Unit 2, as well as the three units affected by the flow reduction.

The revision required for the McGuire OTΔT allowable value is less restrictive than the existing value and the Catawba value is unchanged by the reduction in flow. To improve clarity, the maximum trip setpoint limit in Notes 2 and 4 of TS Table 2.2-1 will be expressed in percent of rated thermal power (RTP) instead of percent instrument span.

In response to a request for additional information, DPC responded (Reference 2) with information which provided the approved methodology (Reference 4) for the changes made relating to OPΔT and OTΔT. The staff, therefore, finds these changes to be acceptable.

2.2 The Effect of Reduced Flow on the Final Safety Analysis Report Analyses

Duke Power performed analyses to justify reduction in the minimum RCS flow to 382,000 gpm. These analyses were to show that the reduced flow rate will not have a significant impact on any accident analyses presented in the Final Safety Analysis Report (FSAR) Chapters 4, 6, or 15.

2.2.1 Thermal Hydraulic Design, FSAR Section 4.4

The thermal hydraulic design for the McGuire and Catawba units was analyzed by DPC with the reduction in RCS MMF to 382,000 gpm. The reduced flow rate resulted in a slight reduction of the margin in the core DNB limits. TS Figure 3.2-1, Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation, was revised to reflect the lower allowable flow rate. For the changes made in RCS flow at reduced power, DPC stated (Reference 2) that the RCS flow values were determined using the same 2% power per 1% flow reduction factor used in the existing TS figure. The axial Flux Difference Limits, TS Section 3.2.1, are unchanged and all the current thermal hydraulic design criteria are satisfied at the reduced flow conditions.

2.2.2 Mass and Energy Releases for Containment Analyses, FSAR Chapter 6

Duke Power stated that the reduction in MMF flow affects the mass and energy releases for containment analysis only through a change in the RCS temperature input assumption. As the RCS average temperature will remain unchanged with the change in MMF, the RCS initial fluid and metal stored energy will remain unchanged. Also, a constant RCS average temperature implies that the driving temperature difference for primary to secondary heat transfer will remain unchanged. These two parameters, initial energy content and rate of energy transfer, are the means by which mass and energy releases influence containment response for the transients analyzed in Chapter 6 of the FSAR. Since the reduction in MMF is being made with a negligible change in RCS temperature, DPC stated that the mass and energy releases calculated in FSAR Chapter 6 will not be affected.

2.2.3 Accident Analyses, FSAR Chapter 15

All of the FSAR Chapter 15 accident analyses which are applicable to the McGuire and Catawba Nuclear Stations were explicitly analyzed by DPC with an initial RCS flow assumption which corresponds to an MMF of 382,000 gpm, or have been evaluated to determine the impact of a reduction in MMF of 3,000 gpm.

The following analyses were reanalyzed by DPC with an initial RCS flow assumption which is less than or equal to an MMF flow of 382,000 gpm.

15.1.5	Steam System Piping Failure
15.2.3b	Turbine Trip - Peak Primary Pressure
15.2.6	Loss of Non-emergency AC Power
15.2.7	Loss of Normal Feedwater Flow
15.2.8	Feedwater System Pipe Break

15.3.1 Partial Loss of Reactor Coolant System Flow
15.3.2 Complete Loss of Reactor Coolant System Flow
15.3.3 Locked Rotor
15.4.1 Uncontrolled Bank Withdrawal from Subcritical
15.4.2 Uncontrolled Bank Withdrawal at Power
15.4.3 Rod Assembly Misoperation
15.4.8 Rod Ejection
15.6.3 Steam Generator Tube Rupture
15.6.5 Loss of Coolant Accident

Events that were not reanalyzed included those that are bounded by other more limiting events as stated in DPC topical report DPC-NE-3002-A and events which are analyzed with the acceptance criteria of no departure from nucleate boiling.

As noted above, DPC has performed reanalyses or has made evaluations that determine that the reduction in MMF will not adversely affect the steady state or transient analyses documented in Chapters 4, 6, and 15 of the Catawba and McGuire FSARs. Duke Power stated (Reference 2) that the reanalyses used approved codes (References 5 to 9). Therefore, the staff finds the decrease in the MMF from 385,000 gpm to 382,000 gpm in the Catawba and McGuire TS to be acceptable.

The staff has reviewed the licensee's submittal to support the reduction in the required minimum measured reactor coolant system flow and finds the TS changes to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 59747 dated November 10, 1993). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

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REFERENCES

1. Letter from M. S. Tuckman, DPC, to USNRC, dated October 25, 1993.
2. Letter from M. S. Tuckman, DPC, to USNRC, dated December 3, 1993.
3. Letter from M. S. Tuckman, DPC, to USNRC, dated December 6, 1993.
4. Letter from T. C. McMeekin, DPC, to USNRC, dated April 26, 1993.
5. Kabadi, J. N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266P-A, Rev. 2, March 1987.
6. N. Lee, et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054P-A, August 1985.
7. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.
8. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991.
9. DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.