Septemb 25, 1992

Docket Nos. 50-413
and 50-414

Distribution
See next page

Mr. M. S. Tuckman
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNIT 1 (TAC NO. M84221)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 102 to Facility Operating License NPF-35 and Amendment No. 96 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 24, 1992, as supplemented September 2, 4, 17, and 23, 1992.

The amendments change TS Sections 4.4.5.2, 3.4.6.2, and Bases 3/4.4.5, 3/4.4.6.2, and 3/4.4.8 to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations. The amendments also reduce the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day to 150 gallons per day. The total allowed primary-to-secondary operational leakage through all steam generators is reduced from .5 gallon per minute (720 gallons per day) to .4 gallon per minute (576 gallons per day). This change is only applicable for Unit 1 fuel Cycle 7.

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, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 102to NPF-35
2. Amendment No. 96to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

FILE NAME: C:CAT84221.AMD *SEE PREVIOUS CONCURRENCE

OFC	: PDII-3/LA	: PDII-3/PM	: OGC	: PDII-3/D	: EMCB/NRR*
NAME	: LBerry	: RMartin	:	: DMatthews	: JStrosnider
DATE	: 9/24/92	: 9/24/92	: 9/25/92	: 9/25/92	: 9/24/92

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

September 25, 1992

Docket No. 50-413

Mr. M. S. Tuckman
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNIT 1 (TAC NO. M84221)

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Sincerely,

Robert E. Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 102 to NPF-35
2. Amendment No. 96 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

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Catawba Nuclear Station

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DATED: SEPTEMBER 25, 1992

AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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

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. 102
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 August 24, 1992, as supplemented September 2, 4, 17, and 23, 1992, 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. 102 , 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

Robert E Martin
for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 25, 1992

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

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. 96
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 August 24, 1992, as supplemented September 2, 4, 17, and 23, 1992, 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. 96 , 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

Robert E Martin
for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 25, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 96

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 4-13a
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-15a	3/4 4-15a
3/4 4-16	3/4 4-16
3/4 4-16a	3/4 4-16a
---	3/4 4-16b
3/4 4-20	3/4 4-20
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

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 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 2.5 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.

SURVEILLANCE REQUIREMENTS (Continued)

Category	Inspection Results
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

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.

d. For Unit 1, tubes in which the tube support plate elevation IPC plugging limit have been applied shall be inspected during all future refueling outages.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;

4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective;

6) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving. It also means the imperfection depth at or beyond which a sleeved tube shall be plugged. The repair limit is equal to 40% of the nominal tube or sleeve wall thickness. For Unit 1, this definition does not apply to the region of the tube subject to the alternate tube plugging criteria.

If a tube is sleeved due to degradation in the F* distance, then any defects found in the tube below the sleeve will not necessitate plugging.

The Babcock & Wilcox process described in Topical Report BAW-2045(P)-A, Rev. 1 will be used for sleeving.

For Unit 1 also, this definition does not apply for tubes experiencing outer diameter stress corrosion cracking confirmed by bobbin probe inspection to be within the thickness of the tube support plates. See 4.4.5.4.a.13 for the plugging limit for use within the thickness of the tube support plate.

SURVEILLANCE REQUIREMENTS (Continued)

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg;

For Unit 1, for a tube in which the tube support plate elevation interim plugging (IPC) limit has been applied, the inspection will include all the hot leg intersections and all cold leg intersections down to and including, at least, the level of the last crack indication for which the interim plugging criteria limit is to be applied.

SURVEILLANCE REQUIREMENTS (Continued)

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 amended as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage/depth parameters as specified in Specification 4.4.5.2. Pending incorporation of the voltage verification requirement in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in Catawba Unit 1 for consistent voltage normalization.

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 1.0 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in SECL-92-282.

2. A tube can remain in service with a bobbin coil signal amplitude greater than 1.0 volt but less than 2.5 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 2.5 volts will be plugged or repaired.

Certain tubes as identified in SECL-92-282, 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 limit has been applied shall be reported to the Commission following the inspection and prior to the resumption of plant operation. The report shall include:

1. Listing of applicable tubes.

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

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 0.4 gpm total reactor-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

RELIEF VALVES (Continued)

reactor coolant system pressure except for limited periods where the PORV has been isolated due to excessive seat leakage and except for limited periods where the PORV and/or block valve is closed because of testing and is fully capable of being returned to its normal alignment at any time, provided that this evolution is covered by an approved procedure. This is a function that reduces challenges to the code safety valves for overpressurization events. 5) Manual control of a block valve to isolate a stuck-open PORV. Testing of the PORVs includes the emergency N_2 supply from the Cold Leg Accumulators. This test demonstrates that the valves in the supply line operate satisfactorily and that the nonsafety portion of the instrument air system is not necessary for proper PORV operation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The B&W process (or method equivalent) to the inspection method described in Topical Report BAW-2045(P)-A, Rev. 1, will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A, Rev. 1 with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, Catawba commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired.

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 criterion 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.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the Reactor Coolant System, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limits of 0.4 gpm for all steam generators not isolated from the Reactor Coolant System ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 0.4 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal Reactor Coolant System pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any Reactor Coolant System pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for Reactor Coolant System pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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 1.0 microCurie/gram DOSE EQUIVALENT I-131, 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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 96 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 August 24, 1992, as supplemented September 2, 4, 17, and 23, 1992, 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 revise TS Sections 3/4.4.5 and 3/4.4.6, and associated Bases 3/4.4.5, 3/4.4.6.2, and 3/4.4.8 to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations. The amendments also reduce the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day to 150 gallons per day. The total allowed primary-to-secondary operation leakage through all steam generators is reduced from .5 gallon per minute (720 gallons per day) to .4 gallons per minute (576 gallons per day). The changes are only applicable for Unit 1 fuel Cycle 7. The supplemental information provided clarifying information that did not change the initial proposed no significant hazards consideration determination found in 57 FR 39250.

2.0 BACKGROUND

Previous inservice inspections and examinations of the steam generator (SG) tubes at Catawba Unit 1 have identified intergranular stress corrosion cracking (IGSCC) on the outer diameter of the tubes at the tube support plate (TSP) intersections. The licensee refers to this particular form of IGSCC as outer diameter stress corrosion cracking (ODSCC).

Outer diameter stress corrosion cracking at TSP intersections is a common degradation phenomenon in SGs in a number of nuclear power plants. Approximately twenty (20) 3/4" OD tubes, including 42 tube-to-TSP intersections, have been removed from affected SGs across the industry for examination and testing. These include tubes from Catawba Unit 1 (including 9 TSP intersections). Each of these pulled tube TSP intersections was sectioned and metallographically examined. In general, these examinations have revealed multiple, segmented, and axial cracks with short lengths for the deepest penetrations. The ODSCC is generally confined to within the thickness of the

TSPs, consistent with the corrosion mechanism which involves the concentration of impurities, including caustics, in the tube-to-TSP crevices. There is some potential for shallow ODSCC for a short distance above or below the TSP. This has been observed in the TSP intersections of two 7/8" OD pulled tubes from another plant.

To date, the pulled tube specimens from Catawba Unit 1 have shown only limited intergranular attack (IGA) associated with the ODSCC. However, more significant IGA has been observed to occur with ODSCC on some pulled tube specimens from other plants. These results suggest that the degradation develops as IGA plus stress corrosion cracking (SCC), particularly when maximum IGA depths greater than 25% are found. A large number (greater than 100) of axial cracks around the circumference are commonly found on these tubes. The maximum depth of IGA is typically one-half to one-third of the SCC depth. Patches of cellular IGA/ODSCC formed by combined axial and circumferential orientation of microcracks are frequently found in pulled tube examinations. Axial crack segments have been the dominant flaw feature affecting the structural integrity of the pulled tube specimens as evidenced by results of burst tests of the pulled TSP intersections prior to sectioning.

Technical Specification 4.4.5.4.a.6, Plugging or Repair Limit, requires that tubes with imperfections exceeding 40% of the nominal tube wall thickness be repaired by sleeving or removed from service by plugging. The licensee believes this repair criterion will result in unnecessary repair or plugging of significant numbers of SG tubes. The proposed interim plugging criterion (IPC) would preclude this.

By letter dated August 24, 1992, the licensee requested interim modifications to the tube repair limit and primary-to-secondary leakage limit in the Technical Specifications for the 7th operating cycle only. The proposed modifications to the tube repair limits include a one volt repair criterion for flaws confined to the thickness of TSP in lieu of the currently applicable depth-based limit of 40%. This criterion would only apply to ODSCC degradation confined to within the thickness of the TSPs.

3.0 TECHNICAL SPECIFICATION CHANGES

Catawba Unit 1 Technical Specification 4.4.5.4.a.6, Plugging or Repair Limit, and Bases 3/4.4.5, Steam Generators, would be revised to specify that the repair limit at the TSP intersections for the seventh operating cycle is based on the analysis in WCAP-13494, "Catawba Unit 1, Technical Support for Steam Generator Interim Tube Plugging Criteria for Indications at Tube Support Plates," to maintain SG tube serviceability as described below:

a. An eddy current inspection using a bobbin probe for inspecting 100% of the hot leg TSP intersections and down to the lowest cold leg TSP intersections with known outside diameter stress corrosion cracking will be performed for tubes in service.

b. Degradation within the bounds of the TSP with a bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.

c. Degradation within the bounds of the TSP with a bobbin voltage greater than 1.0 volt will be repaired or plugged except as noted in d. below.

d. Tubes with indications of potential degradation within the bounds of the TSP with a bobbin voltage greater than 1.0 volt, but less than or equal to 2.5 volts, may remain in service if a rotating pancake coil (RPC) probe inspection does not detect or confirm the degradation. Tubes with indications of degradation with a bobbin voltage greater than 2.5 volts will be plugged or repaired.

e. Certain tubes as identified in SECL-92-282 will be excluded from application of the interim plugging criteria (IPC) limit as it has been determined that these tubes may collapse or deform following a postulated LOCA (loss of coolant accident) and SSE (safe shutdown earthquake) event.

Catawba Unit 1 Technical Specification 3.4.6.2 and Bases 3/4.4.5 would be revised to specify that, for the seventh operating cycle only, primary-to-secondary leakage through all SGs shall be limited to 0.4 gallons per minute (gpm) total reactor-to-secondary and 150 gpd through any one SG. Primary-to-secondary leakage during a steam line break (SLB) will not exceed 1 gpm.

4.0 EVALUATION

4.1 Inspection Issues

The proposed 1.0 volt criterion applies to voltages measured by bobbin type probes (i.e., the industry standard probes which are sensitive to axial flaws) using the 550/130 kHz mix differential channel. In support of the proposed interim repair limit, the licensee has utilized eddy current test guidelines that ensure the field bobbin indication voltage measurements are obtained in a manner consistent with how the voltage limit was developed. These guidelines define the calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators (SGs). In general the staff finds that these procedures were consistent with the generic guidelines developed by Westinghouse for 7/8" OD tubing which are contained in Westinghouse Reports WCAP-12871, Revision 2 (Proprietary Version) and WCAP-12872, Revision 2 (Non-Proprietary Version). In addition, the Catawba procedures are intended to be consistent with the procedures used to obtain the bobbin voltages utilized in the burst pressure versus bobbin voltage correlation. However, deviations from the generic guidelines were necessary to account for the different size tubing at Catawba Unit 1 (i.e., 3/4" OD tubing at Catawba versus 7/8" OD tubing at other facilities), including the following:

- the 550/130 kHz differential frequency mix was used to size the flaws at Catawba whereas the 400/100 kHz mix was used for the generic 7/8" OD tubing APC guidelines

- the probe diameter used at Catawba was primarily 0.610" whereas the 7/8" OD tubing APC probe size was 0.720"

In addition, several other differences between the Catawba procedures and the generic guidelines were noted by the staff. These differences include:

- the voltage level of the mix signal was set at 2.75 volts on the four 0.187" diameter 20% through-wall (TW) holes at Catawba whereas the APC for 7/8" OD tubing specifies 6.4 volts on the four 0.033" diameter 100% TW holes

- no wear standard was used at Catawba whereas the 7/8" OD tubing APC requires the use of a wear standard to monitor bobbin coil probe wear

- the 300 kHz rotating pancake coil (RPC) voltage was set at 10 volts on the 100% EDM notch at Catawba whereas the 7/8" OD tubing APC specifies 20 volts

When the voltage level of the mix signal is set to 6.4 volts on the four 0.033" diameter 100% TW holes for 7/8" tubing (which is consistent with the APC approach for 7/8" OD tubing), the voltage measured for the four 20% TW holes is approximately 2.75 volts. Therefore, the licensee used a reference calibration of 2.75 volts for the four 20% TW holes in the 550/130 kHz support plate suppression mix output. The licensee's calibration procedure is adequate and is consistent with the eddy current analysis guidelines used in the development of the burst pressure/bobbin voltage correlation; however, the staff believes that a more consistent calibration may be obtained using the four 0.033" diameter 100% TW holes rather than the four 20% TW holes due to the elimination of the difference in hole depth and the variations in the shape of the hole bottom.

The licensee did not use a wear standard during this inspection. Therefore, an assessment of the probe wear was conducted by comparing the data obtained during the American Society of Mechanical Engineers (ASME) standard calibration runs at the beginning and end of each data tape. The single 100% TW ASME hole was used to estimate the probe wear uncertainty by evaluating the variability in the 100% TW hole voltages after adjusting the voltages to correspond to readings when the four 20% TW ASME holes were set at 2.75 volts (consistent with the IPC development). As a result of this evaluation of probe wear, the licensee determined that the percent uncertainty in voltage measurements due to probe wear can be modeled as a normal distribution with a mean of 0% and a standard deviation of 16%. The staff believes that the voltage measurement uncertainty due to probe wear has been adequately characterized for puposes of supporting this IPC proposal. The use of a wear standard in future inspections can substantially reduce this uncertainty, and the staff, therefore, recommends that its use be included as part of any future proposal for voltage-based plugging criteria.

The normalization of the 300 kHz RPC signal at 10 volts vice 20 volts for the 100% EDM notch does not affect this application of the IPC since the voltage readings from the RPC are not used in the structural integrity analysis. However, the staff recommends that a consistent set of guidelines be used in order to facilitate the comparison of readings at later inspections.

The burst pressure versus bobbin voltage correlation developed by Westinghouse to support the IPC was developed using a 0.620" diameter probe for the voltage readings. The licensee used a 0.610" diameter probe during this inspection. The smaller diameter probe used by the licensee would tend to increase the amount of probe wobble and, therefore, would be equivalent to wear of a 0.620" diameter probe. Thus, the staff finds that the additional voltage measurement uncertainty associated with the use of a 0.610" diameter probe is implicitly accounted for in the licensee's probe wear assessment and that the use of the smaller diameter probe is adequate for the proposed IPC. However, additional confidence in the relationship of the voltage readings made in the field (using a 0.610" diameter probe) to the voltage readings made in the development of the burst pressure/bobbin voltage correlation (using a 0.620" diameter probe) would be obtained by using the same diameter probe in both applications, namely the 0.620" diameter probe.

The licensee's guidelines require that all signals indicative of degradation are reported regardlessof depth and with no minimum voltage threshold. However, the licensee's guidelines only required one analyst to make voltage calls on the indications with subsequent review by a two-person resolution team. The staff believes that the use of this method to resolve voltage calls is adequate for the proposed IPC, as supported by the licensee's assessment of analyst variability for a small population of indications. The assessment of analyst variability included evaluating the differences between the resolved voltage (i.e., a voltage that was determined through the above resolution process) and the voltage made by an independent analyst for 123 indications. The assessment showed the analyst variability to be bounded by 20% for indications above 0.8 volts. In addition the resolved voltages tended to be higher than the independent voltage call for voltages between 0.5 volts and 0.8 volts, whereas, at lower voltages (i.e., < 0.5 volts) the variation was higher with both plus and minus variations. From this assessment, the licensee concluded that the analyst variability could be modeled by a cumulative probability distribution with an upper limit on uncertainty at 20%. However, to add additional confidence in the voltage resolution process, the staff recommends any future proposals for voltage-based plugging criteria should incorporate voltage calls developed by two independent analysts with discrepancies being resolved with a resolution process.

The staff finds that the bobbin inspection program was consistent with the development of the voltage-based repair limit, namely, the establishment of the relationship between burst pressure and bobbin voltage. In addition, the licensee stated that a sample inspection of tubes using the RPC probe at tube support plate (TSP) intersections was performed. The RPC program included RPC inspection of all locations where the dent voltage was greater than five volts as measured by the bobbin probe and RPC inspection of TSP intersections with artifact indications. The RPC probe can provide improved resolution of flaw indications as compared to bobbin probe in the presence of dents and artifacts (non flaw-like residual signals) and is sensitive to both axial and circumferential flaws. The RPC inspection of the dented intersections (> 5.0 volts) and intersections with artifacts did not reveal any circumferential cracking. Small axial indications were observed in this sample (i.e., dented intersections and artifacts); however, they were repaired (i.e., plugged or sleeved).

The Catawba Nuclear Plant Unit 1 eddy current test guidelines required the RPC probe inspection of TSP intersections exhibiting bobbin indications exceeding 1.0 volt. The RPC probe inspections permitted better characterization of the indications found by the bobbin coil verifying that the flaws detected were applicable to the proposed interim repair limit (i.e., OD stress corrosion cracks as the dominant degradation mechanism with minor IGA involvement). The proposed limit is also based on the premise that any significant degradation is confined to the TSP. The licensee has stated that no unforeseen RPC probe findings relative to the characteristics of the flaws at the TSPs were detected. This includes any detectable circumferential indications or detectable indications extending outside the thickness of the TSP.

4.2 Tube Integrity Issues

The purpose of the Technical Specification 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 (SG) 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 would implicitly serve to ensure 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, tight limits on allowable primary-to-secondary leakage have been established in the Technical Specifications to ensure timely plant shutdown before adequate structural and leakage integrity of the affected tube is impaired.

The proposed interim tube repair limits for Catawba Unit 1 consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the proposed 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 volt (i.e., the proposed interim 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 steam line break (SLB) accidents. The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.3 and 4.4, respectively.

4.3 Structural Integrity

4.3.1 Burst Integrity

The licensee has developed a burst strength/voltage correlation to demonstrate that bobbin indications satisfying the proposed 1.0 volt interim repair criterion will retain adequate structural margins during Cycle 7 operation, consistent with the criteria of Regulatory Guide 1.121. The burst strength/voltage correlation includes the burst pressure versus field bobbin voltage data (pre-pull values) for 8 pulled tubes (10 TSP intersections) which includes 2 pulled tubes (3 TSP intersections) from Catawba Unit 1. This pulled tube data is supplemented by 47 data points from laboratory tube specimens containing outside diameter stress corrosion cracking (ODSCC) flaws produced in model boilers. The bobbin voltage data used to construct the burst pressure/voltage correlation have been normalized to be consistent with the calibration standard voltage set-ups and voltage measurement procedures described in WCAP-13494. The normalization is performed to ensure consistency among the voltage data in the burst pressure/voltage correlation and consistency between the voltage data in the correlation and the field voltage measurements at Catawba Unit 1.

The burst strength/voltage correlation consists of 57 data points from an available data set of 63 possible specimens. Six data points, from 4 Catawba Unit 1 pulled tubes, were excluded from the burst strength/voltage correlation since they included tubes that exhibited "incomplete bursts" or tubes that burst outside of the degraded region of interest (i.e., in an area where a grinding tool mark had been made for location purposes).

The 57 data points that were included in the burst pressure/bobbin voltage correlation included only 10 TSP intersections from pulled tubes. Of these 10 TSP intersections, 3 TSP intersections were from Catawba Unit 1, 6 from Doel Unit 4 (Belgium), and one from Asco-2 (Spain). The burst pressures of the Catawba pulled tube specimens were adjusted to account for only partial crack opening and incomplete burst. The pressure tests of the Catawba pulled tube specimens did not employ bladders, and the pressure tests were terminated upon initial leakage from the specimen prior to achieving the pressure necessary to achieve gross rupture of the tubing, which generally involves a fishmouth failure with significant bulging of the tube cross-section. The upward adjustments to the "incomplete" burst pressures were based on a study that compared the burst pressures obtained for EDM notches (the EDM notches were machined to simulate cracks that experienced "incomplete bursts") to the burst pressures obtained for "incomplete bursts" for the crack that the EDM notch was designed to simulate. Adjustment factors greater than 1.25 were considered unreliable by Westinghouse and resulted in the elimination of the data point from the correlation, as noted above. Although the pressure tests of these pulled tube specimens did not result in a fishmouth failure, these specimens did experience significant crack opening.

Six data points from Doel Unit 4 (Belgium) were included in the burst pressure/bobbin voltage correlation. Since the Belgians used different probes, equipment, calibrations and frequencies than was used in the development of the burst pressure/bobbin voltage correlation, a correlation between the Belgian and U.S. voltages at the APC frequencies and calibration was developed. The results from this correlation which did not involve the use of a transfer standard and was conducted using Belgian probes and a Belgian calibration standard (with EDM holes), indicated that the Belgian voltages should be increased by a factor of approximately 5.0 (based on Laborelec probes and calibration standards). To merge the Belgian data into a consistent industry population for 3/4" OD tubing a cross-calibration of the Belgian standard with the U.S. laboratory ASME standard for 3/4" OD tubing was performed. This cross-calibration process indicated that an additional adjustment of 1.809 might be necessary to the Belgian voltages based on the readings from the 4x100% TW holes in the calibration standard. Westinghouse has assumed a correction factor of 1.5 to the Belgian voltages pending the completion of additional studies being performed by Laborelec. The staff believes that there is presently an inadequate technical basis to support this assumption and that no additional correction factor, in excess of the factor of approximately 5.0 already included in the voltage readings, should be applied to the Belgian voltages until a complete evaluation has been performed by the licensee.

The Catawba eddy current data acquisition and analysis guidelines are intended to minimize voltage measurement variability by ensuring that the field data are acquired and analyzed on a consistent basis with each other and with the development of the burst pressure/voltage correlation. Westinghouse studies indicate that the residual variability in voltage measurements are dominated by probe wear and data analyst variability. Distribution functions have been developed to quantify these sources of uncertainty as was discussed earlier in Section 4.1.

Potential flaw growth between inspections has been evaluated based on the observed voltage amplitude changes during Cycle 5 (1990 to 1991) and Cycle 6 (1991-1992) at Catawba Unit 1. The Cycle 5 growth data was developed from 126 TSP indications which were plugged in 1991 (i.e., at the end-of-cycle (EOC) 5) and could be correlated with an indication that was present at the beginning-of-cycle (BOC); whereas, the Cycle 6 growth rate distribution was calculated from RPC confirmed indications plus indications not RPC tested at the EOC 6 that could be correlated with indications present at the BOC 6 (reference model). Consistent with the current industry practice of allowing tubes that have bobbin indications that have not been RPC confirmed to remain in service, the licensee excluded these indications from their growth rate assessment for Cycle 6 since these indications were considered too small or as false bobbin calls and, therefore, would not significantly degrade the structural integrity of the tube. However, an assessment of the growth rate distribution when all bobbin coil indications regardless of RPC confirmation are included resulted in a projected EOC 7 distribution consistent with the projected EOC distribution predicted by the reference model described above. The eddy current data from the 1990 and 1991 inspections were re-examined using a consistent data analysis procedure as was used during the latest 1992 inspection. The average percent changes in voltage, considering the entire

data set, was 13% between 1990 and 1991 with a standard deviation of 47% and 2% between 1991 and 1992 with a standard deviation of 42%. The licensee has utilized the Cycle 6 cumulative probability distribution of voltage growth as the projected voltage growth distribution for cycle 7 operation.

For the proposed 1 volt IPC, the projected EOC voltage is 1.66 volts assuming the 90% cumulative probability values of voltage measurement uncertainty and voltage growth and is 2.2 volts assuming the 99% cumulative probability values. A mid-cycle inspection (see Section 4.3.1.1 below) would be expected by the staff to significantly reduce these EOC voltage estimates. 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 4.1 volts. This assumes the 1.5 correction factor for the Belgian voltages in the burst pressure/voltage correlation. The staff notes that the maximum allowable voltage is 3.5 volts if no correction factor is applied to the Belgian voltages. Using the lower 99% prediction interval curve for burst pressure and based on the 1.5 correction factor for the Belgian voltages, the maximum allowable EOC voltage is 2.53 volts. The staff notes that this allowable voltage reduces to 1.9 volts (staff estimate) if no correction factor is applied.

For any specific individual tube, voltage measurement uncertainty and/or voltage growth may exceed the value assumed in the above deterministic basis, since the deterministic basis does not consider the full tails of the voltage measurement uncertainty and voltage growth distributions. Similarly, burst pressure for some tubes may be less than the 95% and/or the 99% lower prediction interval values in the burst pressure/voltage correlation. These distribution tails may involve sizable numbers of tubes in cases such as Catawba where thousands of tubes with indications are being accepted for continued service. The staff notes that Regulatory Guide 1.121 provides no guidance on the appropriate cumulative probability and prediction interval values to be employed in these types of analyses. The staff is continuing to evaluate this issue as part of the ongoing lead plant APC review. The licensee proposes that these uncertainties be directly accounted for by use of Monte Carlo methods to demonstrate that the probability of burst during SLB accidents is acceptably low for the distribution of voltage indications being left in service. This approach is not intended to demonstrate that all tubes will satisfy the Regulatory Guide burst criteria. Under this approach, the BOC indications left in service are projected to the EOC by randomly sampling the probability distributions for NDE uncertainties and voltage growth per cycle. For each EOC Monte Carlo sample of bobbin voltage, the burst pressure/voltage correlation (the correlation with the Doel correction factor of 1.5 applied) is randomly sampled to obtain a burst pressure. The 100,000 Monte Carlo samples are performed for the entire BOC distribution. The probability of tube burst at SLB is obtained as the sum of the samples resulting in burst pressures less than the SLB pressure differential of 2650 psi divided by the number of times the distribution of indications left in service is sampled.

This kind of Monte Carlo analysis was performed for the distribution of indications that are being left in service at the BOC 7 at Catawba Unit 1.

This analysis indicated that implementation of a 1.0 volt repair criterion at this time would yield a conditional probability of burst of 1.1×10^{-5}, given a SLB. The staff concurs that this is an extremely low probability, three orders of magnitude less than the value considered in a staff generic risk assessment for SGs (NUREG-0844). This estimate reflects the use of the 1.5 correction factor for the Belgian voltage data. The affect on this conditional probability estimate of assuming no correction factor has not been assessed. With a mid-cycle inspection (see Section 4.3.1.1), however, it is the staff's judgment that the estimated conditional probability of tube rupture would become smaller rather than larger.

Finally, the licensee is proposing as part of the interim repair criteria that indications with bobbin voltages greater than 1.0 volt, but less than or equal to 2.5 volts, remain in service if the RPC inspection does not confirm the indication. The staff notes that short and/or relatively shallow cracks that are detectable by the bobbin probe may sometimes not be detectable by the RPC, although the RPC 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/voltage correlation. The staff concludes that burst pressures for bobbin indications which were not confirmed by RPC inspection will tend to be at the upper end of the burst pressure distribution. The 2.5 volt cutoff, such that all bobbin indications would be plugged or repaired (with or without confirming RPC indications), provides additional assurance that all excessively degraded tubes will be removed from service. Thus, the staff finds the proposed exception to the 1.0 volt criterion to be acceptable.

4.3.1.1 Need for Mid-Cycle Inspection

The burst pressure/bobbin voltage correlations developed for 7/8-inch and 3/4-inch OD tubing are ultimately intended to support voltage-based APC limits of 3.6 volts, 4.0 volts, and 2.5 volts for Farley 1 and 2, Cook 1, and Catawba 1, respectively. The staff is conducting its lead plant review of proposed APCs for Farley 1 and 2. This lead plant review is still in progress since there are a number of technical issues remaining to be resolved as documented in NRC letters dated July 10 and August 18, 1992, to Southern Nuclear Operating Company. In the meantime, the staff has approved interim (voltage-based) plugging criteria (IPC) consisting of a 1.0 volt limit for Farley 2 and Cook 1. Both Farley and Cook have 7/8-inch OD tubing. The staff has concluded that pending completion of its lead plant APC review, a 1.0 volt interim limit at these plants is sufficiently conservative to provide adequate assurance that the tubes with indications accepted for continued service will continue to meet the burst pressure criteria of Regulatory Guide 1.121 throughout the next operating cycle. For Catawba, which has 3/4-inch OD tubing, the staff concludes that a mid-cycle inspection is necessary to ensure that the proposed 1.0 volt interim limit will provide a comparable level of conservatism for the reasons outlined below:

1. For Farley and Cook, the allowable EOC voltage consistent with the most limiting Regulatory Guide 1.121 criterion on burst is 6.2 volts. This is based on burst pressures evaluated at the lower 95% prediction interval as a function of voltage. For Catawba, the allowable EOC voltage is 3.5 volts if no correction factor is applied to the Belgian data as is discussed in Section 4.2. Thus, the 1 volt criterion for Farley and Cook provides a 5.2 volt allowance to accommodate voltage measurement uncertainty and voltage growth between inspections, whereas a 1 volt limit for Catawba provides only a 2.5 volt allowance.

2. Voltage measurement uncertainties are larger at Catawba than was the case for Farley or Cook, primarily due to the fact that a probe wear standard was not used at Catawba. The voltage measurement uncertainties at Farley were estimated to be 0.16 volts and 0.25 volts when evaluated at the 90% and 99% cumulative probability values, respectively. These compare to voltage measurement uncertainty estimates at Catawba of 0.22 volts and 0.42 volts, respectively.

3. Bounding values of voltage growth observed during previous cycles at Farley and Cook were 2.6 volts and 0.8 volts, respectively, based on the most recent information available at the time the staff approved the 1 volt IPC for these units (WCAP-12871, Rev. 2 and WCAP-131871). The staff's SERs noted that even if bounding values of voltage measurement uncertainty and voltage growth are simultaneously applied to a 1 volt indication being accepted for continued service, the resulting EOC voltage would be 3.9 volts for Farley and 2.0 volts for Cook, significantly less than the allowable EOC voltage of 6.2 volts. Pending completion of the lead plant APC review, the staff cited this as a basis for approving the interim 1 volt limit. For Catawba, the bounding value of voltage growth during the past operating cycle was 2.3 volts. Applying this value to a 1 volt indication being accepted for continued service leads to an EOC voltage of 3.3 volts which is close to the allowable EOC voltage of 3.5 volts. However, application of a voltage measurement uncertainty of greater than 0.2 volts and a voltage adjustment of approximately 0.5 volts for the increased length of Cycle 7 compared to Cycle 6 would increase this bounding estimate to beyond the allowable EOC voltage. The staff further notes that a 3.6 volt indication was actually found at Catawba during the current outage.

4. The staff concludes that a mid-cycle inspection may reasonably be expected to verify the growth of indications to values significantly less than 3.5 volts and that this should confirm a level of conservatism consistent with that which is inherent in IPCs that have been approved for other plants.

4.3.2 Combined Accident Loadings

The effects of combined safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) loads and SSE plus SLB loads on tube integrity, consistent with the General Design Criterion 2 (GDC-1) of 10 CFR Part 50, Appendix A have been evaluated. A combined LOCA plus SSE must be evaluated for potential yielding of the TSPs which could result in subsequent deformation of the

tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could open up which might create the potential for in-leakage (i.e, secondary-to-primary) under LOCA conditions. In-leakage during LOCA would pose a potential concern since it may cause an increase in the core peak clad temperature (PCT).

The most limiting accident conditions for tube deformation considerations result from the combination of SSE and LOCA loads. The seismic excitation defined for SGs is in the form of acceleration response spectra at the SG supports. In the seismic analysis, generic response spectra were used which envelope the Catawba-specific response spectra. A finite element model of the Model D SG was developed and the analysis was performed using the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam and pipe elements as well as general matrix input to represent the piping and support stiffness. Interactions at the TSP shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

Prior qualification of the Catawba Unit 1 primary piping for leak-before-break requirements resulted in the limiting LOCA event being the break of a minor branch line. The loads for the primary piping break were used as a conservative approximation. The principal tube loading during a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the SG tube U-bends. A differential pressure is created across the two legs of the tube which causes an in-plant horizontal motion of the U-bends and induces significant lateral loads on the tubes. The pressure-time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave induced loadings, the predominant motion of the U-bends is in the plane of the U-bend. Thus the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two TSPs.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the SG caused by the break hydraulics and reactor coolant loop motion. However, the resulting TSP loads from this motion are small compared to those due to the rarefaction wave induced motion.

To obtain the LOCA induced hydraulic forcing functions, a dynamic blowdown analysis is performed to obtain the system hydraulic forcing functions assuming an instantaneous (1.0 msec break opening time), double-ended guillotine break. The hydraulic forcing functions are then applied, along with the displacement time-history of the reactor pressure vessel (obtained from a separate reactor vessel blowdown analysis), to a system structural model that includes the SG, the reactor coolant pump, and the primary piping. This analysis yields the time-history displacements of the SG at its upper lateral and lower support nodes. These time-history displacements formulate

the forcing functions for obtaining the tube stresses due to LOCA shaking of the SG.

In calculating a combined TSP load, the LOCA rarefaction and LOCA shaking loads were combined directly, while the LOCA and SSE loads were combined using the square root of the sum of the squares. The overall TSP load was transferred to the SG shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding of the TSP at the wedge supports, causing some tubes in the vicinity of the wedge supports to be deformed. Utilizing results from recent tests and analysis programs, Catawba has shown that tubes will undergo permanent deformation if the change in diameter exceeds a minimum threshold value. This threshold for tube deformation is related to the concern for tubes with preexisting tight cracks that could potentially open during a combined LOCA plus SSE event. For Catawba Unit 1, the LOCA plus SSE loads (using large break forces) were determined to be of such magnitude that a limited number of the tubes (which are assumed to contain preexisting tight cracks) are predicted to exceed this deformation threshold value and, therefore, can lead to significant tube leakage. The IPC will not be applied to these tubes.

The effect of SSE bending stresses on the burst strength of tubes with axial cracks has been assessed. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously performed tests, it has been concluded that the burst strength of tubes with through-wall cracking is not affected by an SSE event.

Based on the above, it can be concluded that, at Catawba Unit 1, limited tube deformation can occur during an SSE plus LOCA event. However, the potential for in-leakage is nullified by not permitting the IPC to be applied to the subject tubes. In addition, burst strength of tubing with through-wall cracks is not affected by an SSE event.

4.4 Leakage Integrity

As discussed earlier, 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 proposed restrictive Technical Specification limits on allowable primary-to-secondary leakage as discussed in Section 4.5 of this Safety Evaluation. 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 the rate assumed in the Catawba Unit 1 design basis accident.

The licensee has identified in the Final Safety Analysis Report (FSAR), Chapter 15, accidents which result in secondary steam release, and thus whose consequences could be affected by the extent of primary-to-secondary leakage.

Of these accidents, the SLB, with respect to IPC, was determined to be the most limiting. In this case, since the SG in the faulted loop is subject to dryout, the activity release path is conservatively assumed to be direct to the environment, without any mitigation resulting from mixing with secondary liquid coolant in the SG.

For the purpose of supporting the interim repair limit proposal, the licensee has proposed that the maximum allowable primary-to-secondary leak rate during SLB be 1.0 gpm, which is consistent with the assumed leak rate in the FSAR design basis analysis. Therefore, there is no change in the off-site dose as a result of the use of the interim repair limit.

As the basis for estimating the potential leakage during SLB accidents, Westinghouse has correlated leakage test data obtained under simulated SLB conditions with the corresponding bobbin voltage amplitudes. For 3/4-inch OD tubing, the correlation is based on 32 data points from laboratory tube specimens containing ODSCC flaws produced in a model boiler facility. The correlation is based on a linear regression fit of the logarithms of the corresponding leak rates and 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.

Twenty-seven of the 32 specimens exhibited leakage under simulated SLB conditions (2650 psid, 616 degrees Fahrenheit). These specimens exhibited bobbin voltages exceeding 4.2 volts. The five non-leaking specimens exhibited voltages ranging from 2.8 to 5.7 volts. Destructive examination of the non-leaking specimens (performed after the leak test) revealed that the maximum crack depth in these specimens have ranged between 90% and 100% through-wall prior to leak testing.

The log-log SLB leakage/bobbin voltage correlation is highly sensitive to the treatment of the non-leakers. In past analyses for other units, Westinghouse assigned a leakage value of 0.0001 liters per hour (lph) to the non-leakers. This was done to establish a log-log slope of approximately 4 for the mean correlation, similar to the observed slope for leak rate versus crack length studies and consistent with the leak rate measuring instrumentation/method (capability to measure down to 0.001 to 0.0001 lph). More recently, Westinghouse has combined the leak rate versus crack length study with a study of bobbin voltage versus crack length to establish the nominal "predicted" relationship between leakage and voltage. This relationship is considered preliminary by Westinghouse and is conservative compared to most of the model boiler specimen leak rate versus voltage data for leak rates exceeding 0.7 lph. For Catawba, Westinghouse has assigned a leakage value of 0.001 lph to the non-leakers to establish a slope for the mean regression curve of the data which is consistent with that of the "predicted" relationship between leak rate and voltage.

Apart from the 5 non-leaking specimens included in the correlation, an additional 8 non-leaking model boiler specimens have not been included in the correlation. These latter specimens exhibited maximum crack depths of less than 90% through-wall and were associated with bobbin voltages ranging between

0.9 and 2.6 volts. Nor does the correlation include the SLB leakage data collected for the 9 TSP intersections from the Catawba pulled tube samples. The SLB leakage tests for the Catawba pulled tube samples were performed at room temperature. One of these 9 specimens, which exhibited a 1.8 volt bobbin indication in the field, exhibited leakage at the SLB pressure differential (the other 8 specimens were non leakers). A Westinghouse study indicates this one leaker to have been caused by damage incurred during the tube pulling process.

Westinghouse concludes that the combined SLB leak rate data base for 3/4-inch and 7/8-inch OD tubing supports applying a 2 volt cutoff to the above SLB leak rate model such that indications less than 2 volts can be assumed not to leak. For purposes of combining the 3/4-inch and 7/8-inch OD tubing data, Westinghouse has applied a scaling factor of 0.74 to the 7/8-inch OD tube voltages such that these voltages may be directly compared to 3/4 inch tubing voltages. This scaling factor is based on the ratio of the tube diameters and is needed since the ASME calibration standard hole size is the same for both tubes. However, the staff believes that the 7/8-inch OD tube voltage data should be adjusted by a total factor of approximately 0.5 in order to obtain approximately equal mean fits of the 3/4-inch and 7/8-inch OD tubing leakage data. Use of an adjustment factor of 0.5 leads to a leakage threshold on the order of 1 volt. Post-leak test destructive examinations show that lengths of 100% through-wall cracks must generally be beyond 0.1 inches long before the crack exhibits detectable leakage during the SLB leakage test. This is consistent with the observation that low voltage indications exhibit little or no leakage.

The calculated SLB leak rate is based on the same continuous frequency distribution of EOC bobbin voltages as that used to determine the probability of tube rupture during SLB (see Section 4.2 of this SER). Based on this projected EOC voltage distribution, Monte Carlo sampling of SLB leakage distribution (from the above SLB leakage/voltage correlation) has been performed to establish a cumulative probability distribution of total SLB leakage per steam generator. The SLB leak rate was evaluated at the 90% cumulative probability value. The calculated SLB leak rate for the most limiting steam generator was found to be approximately zero assuming a leakage threshold of 2 volts and 0.67 gpm when no leakage threshold is assumed.

The staff commented on the above methodology for predicting SLB leakage as part of its ongoing review of the APC proposal for Farley Units 1 and 2 (NRC letter dated July 10, 1992, to Southern Nuclear Operating Company). These comments were discussed with Westinghouse and Southern Nuclear Operating Company during a meeting on July 27, 1992. The staff and its consultants expressed the concern that the use of the 90% cumulative probability value as determined by Monte Carlo techniques may be non-conservative since it appears not to adequately account for the upper tail of the SLB leak rate distribution in the voltage correlation. The staff and its consultants noted that the upper tail of the distribution is likely to be the dominant contributor to the total SLB leakage in each SG. Thus the arithmetic mean of the distribution appears to be more appropriate for use than the 90% cumulative probability value.

During the July 27, 1992 meeting, the staff also commented on the treatment of non-leakers in the SLB leak rate/bobbin voltage correlation (see earlier discussion). The staff noted that this treatment (censoring of the data) appeared to be somewhat arbitrary; creating significant uncertainty in the correlation, particularly at relatively low voltages (< 4 volts). The recent preliminary development of a nominal "predicted" relationship between leakage and voltage is intended by Westinghouse to respond to this concern by serving as an independent check of the reasonableness of the correlation (as discussed earlier).

At the staff's request, the licensee submitted by letter dated September 23, 1992, a calculation of potential SLB leakage based on the use of the arithmetic mean of the SLB leak rate distribution as a function of voltage for each EOC indication. The assumed EOC distribution of indications was consistent with that used in the reference calculation above with the exception that the continuous frequency distribution of EOC indication voltages as predicted by Monte Carlo was integrated to produce a histogram of numbers of whole tubes falling into each voltage interval (the histogram was divided into 0.05 volt intervals for indications up to 1.8 volts and into intervals of 0.2 volts for indications beyond 1.8 volts). At the staff's request, this calculation also considered a 1 volt threshold for SLB leakage. With this alternate approach, the calculated leak rate for the most limiting SG is 0.049 gpm, which is well within the 1.0 gpm value assumed in the FSAR design basis analysis.

An alternate analysis, termed a bounding analysis by Westinghouse, has been provided in WCAP-13494, "Catawba Unit 1, Technical Support for Steam Generator Interim Tube Plugging Criteria for Indications at Tube Support Plates." This analysis considers the combined SLB leak rate data base for 3/4-inch and 7/8-inch OD tubing, where the voltages for 7/8-inch OD tubing have been revised downward to reflect the 0.74 scaling factor discussed earlier. This calculation bounds the combined data base by assuming zero leakage for EOC indications ranging to 2 volts and 1 lph for each indication larger than 2 volts but less than or equal to 3.5 volts. This calculation utilized the same EOC distribution indications as that discussed in the previous paragraph (maximum EOC voltage equal to 2.34 volts). On this basis, the calculated SLB leak rate is 0.01 gpm. The staff has repeated this calculation to reflect a scaling factor of 0.5 for 7/8-inch OD tube voltages. Consistent with this assumption, the staff assumed zero leakage for EOC indications ranging to 1 volt, 1 lph for each indication larger than 1 volt but less than or equal to 1.7 volts, and 10 lph for each indication larger than 1.7 volts but less than or equal to 2.9 volts. On this basis, the calculated SLB leak rate for the most limiting SG is 0.78 gpm, which is less than the 1.0 gpm value assumed in the FSAR design basis.

Both Westinghouse and the NRC staff are continuing to evaluate the SLB leak rate issues discussed at the July 27, 1992, meeting within the context of the Farley APC proposal. In the meantime, the staff concludes that potential leak rates during a postulated SLB assumed to occur at EOC have been reasonably bounded for Catawba Unit 1 and shown to be well within the value of 1.0 gpm assumed in the FSAR design basis analysis.

4.5 Proposed Interim Leakage Limits

4.5.1 Description

An interim change to the reactor coolant primary-to-secondary system leakage limit criteria in Technical Specification 3.4.6.2 that is applicable to the seventh operating cycle only is also proposed. The current 500 gpd limit for primary-to-secondary leakage through any one SG is changed to 150 gpd. In addition, the limit on total leakage through all SGs would be reduced from 1.0 gpm to 0.4 gpm.

4.5.2 Discussion

The current 500 gpd limit per SG is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of Regulatory Guide 1.121. The current 1.0 gpm limit for total primary-to-secondary leakage is consistent with the 1.0 gpm total leakage assumed for SLB analyses.

Development of the proposed 150 gpd limit per SG has utilized the extensive industry data base regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95% confidence interval for a given crack size, the proposed 150 gpd limit would be exceeded before the crack length reaches the critical crack length for SLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical length corresponding to three times normal operating pressure.

The proposed interim change is more restrictive than the existing limits and is intended to provide a greater margin of safety against rupture. The proposed interim limits are also intended to provide an additional margin to accommodate a rogue crack which might grow at much greater than expected rates, or unexpectedly extend outside the thickness of the TSP, and thus provide additional protection against exceeding SLB leakage limits.

4.6 Summary

Based on the above evaluation, it can be concluded that the proposed interim tube repair limits and leakage limits will ensure adequate structural and leakage integrity of the SG tubing at Catawba Unit 1, consistent with applicable regulatory requirements, until May 1, 1993. The staff concludes that a mid-cycle inspection is warranted for the reasons cited in section 4.3.1.1 of this report and due to the lack of unadjusted 3/4-inch OD pulled tube data used in the development of the burst pressure/bobbin voltage correlation as described in section 4.3.1. The staff concludes that a mid-cycle inspection is necessary by May 1, 1993, to ensure that the proposed 1.0 volt interim limit will provide a comparable level of conservatism as compared to the interim limits approved for plants with 7/8-inch OD SG tubing. The NRC staff finds the proposed changes to the TSs to be acceptable.

5.0 EXIGENCY CIRCUMSTANCES

The licensee, in its August 24, 1992, application, requested that the proposed TS change be approved on an exigency basis. The licensee states that, as a result of inspection of steam generator (SG) tubes during the current refueling outage on Unit 1, it was determined that an unforseen number of SG tubes (over 1000) would require repair during the current outage unless the Catawba TSs could be amended to permit the use of interim plugging criteria. The licensee recognized that an unanticipated number of SG tube indications were being found by August 5, 1992. The licensee completed SG tube inspections using the bobbin coil technique by August 8, 1992; completed a confirmatory inspection of the bobbin coil results with the motorized rotating pancake coil technique on August 10, 1992; and reached a decision on August 11, 1992, to request its vendor, Westinghouse Electric Corporation, to prepare analyses to support the application for changes to the TSs. A date of August 14, 1992, was initially set for submittal of the application, but due to the complexity of the issues involved, the application could not be submitted until August 24, 1992. Catawba Unit 1 is now scheduled to start up from its present refueling outage on September 25, 1992, and would need the proposed amendment prior to September 25, 1992, in order to permit startup.

The NRC staff concludes that, upon the determination that the resolution to the issue would require an amendment to the TS, the supporting analyses by Westinghouse and the licensee were performed in an expeditious fashion, and that the need for reviewing the application on an exigent basis could not than be avoided. Thus, pursuant to 10 CFR 50.91(a)(6), the staff finds that an exigent situation exists which would result in a delay in the startup of Unit 1.

6.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether a significant hazards consideration exists pursuant to 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from an accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The following evaluation in relation to the three standards demonstrates that the proposed amendment does not involve a significant hazards consideration.

The limiting burst pressure criterion from NRC Regulatory Guide 1.121 requires the maintenance of a margin of 3 times normal operating pressure differential on tube burst. The 3 times normal operating pressure differential for the Catawba Unit 1 steam generators corresponds to 3750 psi. Based on the existing data base, this criterion is satisfied with 3/4" diameter tubing with bobbin coil indications with signal amplitudes less than 3.5 volts, as explained in the staff's SER, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the

data. A 1.0 volt plugging criterion compares favorably with the structural limit considering the calculated growth rates for ODSCC within the Catawba Unit 1 steam generators. Considering a voltage increase of 0.44 volts (90% cumulative probability value), and adding 20% NDE uncertainty of 0.2 volts (90% cumulative probability value) to the interim plugging criterion of 1.0 volts results in an EOC voltage of 1.66 volts. The growth rate used to determine the projected EOC voltage is based on the review of voltage growth rates for all TSP intersections during Cycle 6 operation. This end of cycle voltage compares favorably with the structural limit 3.5 volt.

Pending completion of the NRC staff's lead-plant review (for Farley 1 and 2) of permanent voltage-based plugging limits, the staff has previously approved interim 1.0 volt limit for other plants based on consideration of the substantial degree of margin that had been demonstrated relative to meeting the Regulatory Guide criterion. For reasons stated in Section 4.3.1.1 of the SER, the staff has concluded that a mid-cycle inspection is necessary by May 1, 1993, to ensure that the proposed 1.0 volt interim limit will provide a comparable level of conservatism as compared to the interim limits which have been approved for other plants.

Only three indications of ODSCC have been reported to have operating leakage - all three have been in European plants. No field leakage has been reported at other plants from tubes with indications with a voltage level of under 6.2 volts (from 3/4" tubing). Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary to secondary leakage and steam release to the environment are: Feedwater System Malfunction, Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Uncontrolled Single Rod Withdrawal at Power, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. In support of implementation of the interim plugging criterion, it has been determined that the distribution of cracking indications at the tube support plate intersections at the end of cycle 7 are projected to be such that primary to secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines.

Monte Carlo analyses methods have been used to calculate the frequency distribution of indications as a function of voltage at the EOC-7 at Catawba Unit 1. These Monte Carlo analyses methods utilize the distributions for indications left inservice, NDE uncertainties, and voltage growth. The methods account for the tails of the distribution and yield eddy current voltages with an associated probability of occurrence and the cumulative probability of EOC voltages. The SLB leak rates applied to the Monte Carlo voltage distribution are represented by a correlation of SLB leak rate and bobbin coil voltage data obtained from tests of pulled tube specimens and laboratory tube specimens. SLB leak rates were evaluated using several different approaches including Monte Carlo sampling of the leak rate distribution as a function of voltage, and by deterministic approaches. The deterministic approaches included use of the arithmatic mean of the SLB leak rate distribution as a function of voltage, and use of bounding values of SLB leakage as a function of voltage. The calculated total SLB leak rate using each calculational method was found to compare favorably with the accident

analyses assumptions of 1.0 gpm in the affected steam generator identified in Table 15.3 of the Catawba Unit 1 Safety Evaluation Report.

Upon application of the interim plugging criterion, only a negligible increase in leakage above normal operating leakage would be expected during plant transients, other than steam line break, which have lower peak differential pressures.

Therefore, as steam generator tube burst capability and leaktightness during Cycle 7 operation following implementation of the proposed 1.0 volt interim plugging criterion remains consistent with the current licensing basis, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated with the Catawba Unit 1 FSAR.

Implementation of the proposed interim tube support plate elevation steam generator tube plugging criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations. A tube rupture event would not be expected in a steam generator in which the plugging criterion has been applied (during all plant conditions).

Upon application of the interim plugging criterion, no primary to secondary leakage during normal operating conditions is anticipated due to degradation at the tube support plate elevations in the Catawba Unit 1 steam generators. However, additional conservatism is built into the operating leakage limit with regard to protection against the maximum permissible single crack length which may be achieved during Cycle 7 operation due to the potential occurrence of through wall cracks at locations other than the tube support plate intersections.

Specifically, Duke Power Company will implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per steam generator to help preclude the potential for excessive leakage during all plant conditions. The currently proposed Cycle 7 Reload Technical Specification limits on primary to secondary leakage at operating conditions is a maximum of 0.5 gpm (720 gpd) for all steam generators, or, a maximum of 200 gpd for any one steam generator. The R.G. 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 3 against bursting at normal operating pressure differential. A voltage amplitude of 3.5 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% uncertainty limit on the burst correlation. Alternate crack morphologies can correspond to 3.5 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently,

typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 3 times normal operating pressure differential and SLB conditions are 0.48 inch and 0.76 inch, respectively. Nominal leakage for these crack lengths would range from about 0.24 gpm to 2.25 gpm, respectively, while lower 95% confidence level leak rates would range from about 0.04 gpm to 0.33 gpm, respectively. A leak rate of 150 gpd will provide for detection of 0.40 inch long cracks at nominal leak rates and 0.60 inch long cracks at the lower 95% confidence level leak rates.

Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

Application of the 1.0 volt interim steam generator tube plugging criterion at Catawba Unit 1 is not expected to result in tube burst during all plant conditions during Cycle 7 operation. Tube burst margins are expected to meet R.G. 1.121 acceptance criteria. The limiting consequence of the application of the interim plugging criterion is a potential for primary to secondary leakage of no greater than 0.78 gpm for expected voltage thresholds under SLB conditions. This amount of leakage does not result in unacceptable radiological consequences. No unacceptable leakage is anticipated at normal operating or RCP locked rotor conditions. Therefore, as the existing tube integrity criteria and accident analyses assumptions and results continue to be met, the proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

Based on the analysis which shows the new leakage values proposed and the leakage characteristics expected during accidents creating high differential pressures across the steam generator tubes (main steam line break) new dose analyses were run to determine offsite dose consequences. A new analysis of the Main Steam Line Break accident using pre-existing leakage's of 0.1 gpm per steam generator and leakage growth of 1.1 gpm in the faulted generator determined that the EAB and Low Population Zone doses remain well within 10% of the allowed 10 CFR100 values of 25 Rem whole body and 300 Rem thyroid. The most restrictive dose analysis is the Reactor Coolant Pump Locked Rotor accident which requires that total steam generator leakage remains less than 0.7 gpm. This is a new analysis which has been submitted to support Unit 1 Cycle 7. This accident does not create high differential pressure conditions across the steam generator tubes and by limiting the initial allowed primary to secondary leakage to 0.4 gpm total, 10% of 10 CFR 100 dose limits are again not exceeded. Reruns of the above accident dose analyses show that there is no significant increase in dose consequences.

The use of the voltage based bobbin probe interim tube support plate elevation plugging criterion at Catawba Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of Regulatory Guide 1.121. R.G. 1.21 describes a method acceptable to the NRC staff for meeting

GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criterion, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The end of cycle distribution of crack indications at the tube support plate elevations is calculated to result in minimal primary to secondary leakage during all plant conditions and radiological consequences are not adversely impacted.

In addressing the combined effects of LOCA + SSE on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Analyses results show that for the Catawba Unit 1 steam generators several tubes near wedge locations may significantly deform or collapse and secondary to primary inleakage may result. These tubes have been precluded from application of interim plugging criterion (WCAP-13494). For all other steam generator tubes, the possibility of secondary to primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary to primary leakage in the event of a LOCA + SSE is expected to be less than that associated with the application of this criterion, i.e., 150 gpd per steam generator. Secondary to primary inleakage would be less than primary to secondary leakage for the same pressure differential since the cracks would tend to close under a secondary to primary pressure differential. Additionally, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing R.G. 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criterion of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations, and rotating pancake coil inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate elevation plugging criterion will decrease the number of tubes which must be repaired or

taken out of service by plugging. The installation of steam generator tube plugs or sleeves reduces the RCS flow margin. Thus, implementation of the alternate plugging criterion will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any BASES of the plant Technical Specifications.

Based on the foregoing, the NRC staff has concluded that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the Commission has made a final determination that the proposed amendments do not involve a significant hazards consideration.

7.0 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.

8.0 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 and changes 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 (57 FR 39250 dated August 28, 1992). 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.

9.0 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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