ATTACHMENT 2

Marked-up Technical Specifications

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.

INSERT A e.

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.
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.

For Unit 1

INSERT A

e. 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 X.X volts. For tubes that will be administratively plugged, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as OD SCC.

OR REPAIRED 2.5

REACTOR COOLANT SYSTEM

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.

For Unit 1 IPC

d. Tubes in which the tube support plate elevation plugging limit have been applied shall be inspected during all future refueling outages.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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.

INSERT B HERE

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

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;

INSERT C HERE

REACTOR COOLANT SYSTEM

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.

INSERT D HERE

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,

INSERT B

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.

INSERT C

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.

INSERT D

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 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 ~~the~~ Catawba Unit 1 ~~steam generator inspections~~ 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 ~~0.4~~ 1.0 gpm. The basis for determining expected leak rates from the projected crack distribution is provided in SECL-92-282.

(INCLUDES OPERATIONAL AND ACCIDENT LEAKAGE)

2. A tube can remain in service with a bobbin coil signal amplitude greater than 1.0 volt but less than X.X 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 X.X 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 Limit Criteria as it has been determined that these tubes may collapse or deform following a postulated LOCA + SSE Event.

REACTOR COOLANT SYSTEM

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).

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

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,

0.4

c. ~~1~~ gpm total reactor-to-secondary leakage through all steam generators and ~~500~~ gallons per day through any one steam generator,

150

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.

REACTOR COOLANT SYSTEM

BASES

STEAM GENERATORS (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 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 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 = 500 gallons per day per steam generator). Cracks having a reactor-to-
150 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 500 gallons per day per steam generator can readily be detected ~~by radiation monitors of steam generator blowdown.~~ Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired.

REACTOR COOLANT SYSTEM

BASES

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.

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.

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.

REACTOR COOLANT SYSTEM

BASES

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. 1

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 limit of ~~1~~ 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 ~~1~~ gpm limit is consistent with the assumptions used in the analysis of these accidents. The ~~500~~ 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.

REACTOR COOLANT SYSTEM

BASES

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

0.4

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 ~~1.0~~ 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.

ATTACHMENT 3

No Significant Hazards Consideration
and
Environmental Impact Statement

NO SIGNIFICANT HAZARDS ANALYSIS

In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to: 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 any accident previously evaluated; or 3) involve a significant reduction in margin of safety.

Conformance of the proposed amendment to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

1) Operation of Catawba Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Testing of model boiler specimens for free span tubing (no tube support plate restraint) at room temperature conditions show burst pressures in excess of 5475 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 11 volts (Reference 1). Burst testing performed on pulled tubes from Catawba Unit 1 with up to a 1.5 volt indications show measured burst pressures in excess of 4800 psi at room temperature. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature, tube burst capability significantly exceeds the R.G. 1.121 criterion requiring 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 4.1 volts, 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 0.58 volts, and adding 20% NDE uncertainty of 0.2 volts (90% Cumulative Probability) to the interim plugging criterion of 1.0 volts results in an EOC voltage of 1.78 volts. The growth rate used to determine the projected EOC voltage is based on the review of growth rates for 541 TSP intersections. These indications were selected by Duke Power Company based on their largest amplitudes from the original analyses. The 541 indications were made up of 90, 117, 197, and 137 from steam generators A, B, C and D, respectively. This end of cycle voltage compares favorably with the Structural Limit 4.1 volt. The corresponding safety margin to the tube structural limit at end of cycle 7 upon implementation of the 1.0 volt steam generator tube interim plugging limit is 2.3 volts. The necessary plugging limit to meet tube structural limits is 2.5 volts.

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 are used to calculate the potential SLB leakage at the EOC-7 at Catawba Unit 1. The Monte Carlo analyses methods utilize the distributions for indications left inservice, NDE uncertainties, voltage growth and SLB leak rate. 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 0.0 gpm for volts less than or equal to 1.8 volts, 1 liter/hr for 1.8 to 3.5 volts, and 10 liter/hr for greater than 3.5 volts. Applying these leak rates to the projected EOC voltage distribution leads to a projected SLB leak rate of 0.54 gpm for steam generator D, the most limiting steam generator (3492 TSP elevation indications). The 0.54 gpm SLB leak rate compares 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. The projection indicates a maximum EPC-7 of 3.1 volts (90% cumulative probability). The analyses yields a negligible likelihood of a tube exceeding the 3.5 volt threshold for a 10 liter/hr SLB leak rate.

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.

2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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; no ODSCC is occurring outside the thickness of the tube support plates. 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 is anticipated during all plant conditions 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 4.1 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 4.1 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 0.10 gpm to 3 gpm, respectively, while lower 95% confidence level leak rates would range from about 0.015 gpm to 0.4 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 approximately 0.54 gpm. This amount of leakage does not result in unacceptable radiological consequ... 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.

3) The proposed license amendment does not involve a significant reduction in margin of safety.

Based on the analysis which shows the new leakage values proposed and the lealage 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 excessive 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 CFR100 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 address the combed 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 (Reference 3). 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.

CONCLUSION

Based on the preceding analysis, it is concluded that using the TSP elevation bobbin coil probe voltage-based interim steam generator tube plugging criterion for removing tubes from service at Catawba Unit 1 is acceptable and the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10 CFR 50.92.

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.