ATTACHMENT 2

Catawba Units 1 and 2

Marked-Up Technical Specification Pages

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Steam Generator Water Level Low-Low		
a. Unit 1	≥10.7% of narrow range span ~~≥17% of span from 0% to 30% RTP* increasing linearly to ≥ 40.0% of span from 30% to 100% RTP*~~	≥ 9% of narrow range span ~~≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥38.3% of span from 30% to 100% RTP*~~
b. Unit 2	≥36.8% of narrow range span	≥35.1% of narrow range span
13. Undervoltage - Reactor Coolant Pumps	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
14. Underfrequency - Reactor Coolant Pumps	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
15. Turbine Trip		
a. Stop Valve EH Pressure Low	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	≥1% open	≥1% open
16. Safety Injection Input from ESF	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T' \leq$ 585.1°F (unit 1) and 590.8°F (unit 2) (Nominal T_{avg} allowed by Safety Analysis);

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report;

P = Pressurizer pressure, psig;

P′ = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the Core Operating Limits Report;

$f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.5% of Rated Thermal Power.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 590.8°F), (Unit 2)

585.1°F (Unit 1) and

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent ΔI that magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% (Unit 1) and 3.3% (Unit 2) of Rated Thermal Power.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	≥ 775 psig	≥ 744 psig
e. Steam Line Pressure-Negative Rate - High	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	83.9% ≤ ~~82.4%~~ of narrow range instrument span	85.6% ≤ ~~84.2%~~ of narrow range instrument span
2. Unit 2	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
c. T_{avg}-Low	≥ 564°F	≥ 561°F
d. Doghouse Water Level-High	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	≤ ~~82.4%~~ 83.9% of narrow range instrument span	≤ ~~84.2%~~ 85.6% of narrow range instrument span
2. Unit 2	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
7. Containment Pressure Control System		
a. Start Permissive	≤ 0.4 psid	≤ 0.45 psid
b. Termination	≥ 0.3 psid	≥ 0.25 psid
8. Auxiliary Feedwater		
a. Manual Initiation	N.A	N.A.
Automatic Actuation Logic and Actuation Relays	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)		
c. Steam Generator Water Level - Low-Low		
1) Unit 1	≥ 10.7% of narrow range span ~~≥ 17% of span from 0% to 30% RTP increasing linearly to ≥ 40.0% of span from 30% to 100% RTP~~	≥ 9% of narrow range span ~~≥ 15.3% of span from 0% to 30% RTP increasing linearly to ≥ 38.3% of span from 30% to 100% RTP~~
2) Unit 2	≥ 36.8% of narrow range span	≥ 35.1% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
e. Loss-of-Offsite Power	≥ 3500 V	≥ 3242 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low		
1) CAPS 5220, 5221, 5222	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	≥ 6.2 psig	≥ 5.2 psig
a. Unit 1	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low Coincident With Safety Injection	≥ 177.15 inches	≥ 162.4 inches
	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

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.~~

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

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

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

~~e. For Unit 1, implementation of the interim steam generator tube/tube support plate elevation plugging limit for Cycle 9 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. The determination of tube support plate intersections having OD SCC indications shall be based on the performance of at least 20 percent random sampling of tubes inspected over their full length. An inspection using the rotating pancake coil (RPC) probe is required in order to show operability of tubes with flaw like bobbin coil signal amplitudes greater than 1.0 volt but less than 2.7 volts. 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.

No Changes

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.

(after steam generator replacement on Unit 1)

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 at least 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.

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~~ plugging 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~~ plugging 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.~~

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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 1.0 volts, regardless of the depth of tube wall penetration, if, as a result, the projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 17.5 gpm (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 2.7 volts provided a rotating pancake coil (RPC) inspection does not detect degradation.

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

4. If as a result of leakage due to a mechanism other than OD SCC at the tube support plate intersection, or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of 4.4.5.4.a.13.2. If bobbin voltage is within expected limits the indication can remain in service. The expected repair limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

where:

V = measured voltage

V_{BOC} = voltage at beginning of cycle (BOC)

Δt = time period of operation to unscheduled outage

CL = cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)

V_{SL} = 4.5 volts for 3/4-inch tubes

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

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug ~~or repair~~ all tubes exceeding the plugging ~~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 plugged ~~repaired~~ in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

SURVEILLANCE REQUIREMENTS (Continued)

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,

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

   3) Identification of tubes ~~repaired~~ plugged.

c. ~~For Unit 2, r~~ 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 implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:~~

   ~~1. If the estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.~~

   ~~2. If circumferential crack like indications are detected at the tube support plate intersections.~~

   ~~3. If the indications are identified that extend beyond the confines of the tube support plate.~~

   ~~4. If the calculated conditional burst probability exceeds 1 X 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.~~

Table 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Four	Four
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

(after steam generator replacement on Unit 1)

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Table 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes* and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes* and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes*
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes* and inpsect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes*. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

~~*For Unit 1, defective tubes which fall under the alternate plugging criteria do not have to be plugged.~~

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)

Plugging

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~~ plugging limit of 40% of the tube nominal wall thickness. ~~For Unit 1, defective tubes which fall under the alternate tube plugging criteria do not have to be repaired. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service~~. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

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

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

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

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment vessel is designed and shall be maintained for a maximum internal pressure of 15 psig and a temperature of 328°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of cylindrical zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 5.0 weight percent U-235 with a maximum tolerance of ± .05 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material of which 102 inches shall be 100% boron carbide and remaining 40-inch tip shall be 80% silver, 15% indium, and 5% cadmium.

For Units 1 and 2, all control rods shall be clad with stainless steel tubing, except for Unit 2, a maximum of one Rod Cluster Control Assembly may have Inconel clad control rods.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
b. For a pressure of 2485 psig, and
c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

13,050 ± 100 cubic feet for Unit 1 and

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 ± 100 cubic feet at a nominal T_{avg} of 525°F.

for Unit 2

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1-1.

CORE OPERATING LIMITS REPORT (Continued)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, and Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002~~A~~, Rev. 1, "FSAR Chapter 15 System Transient Analysis Methodology," ~~November 1991~~. SER dated December, 1995.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P~~-A~~, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," ~~August 1994~~. SER dated December, 1995.

   (Modeling used in the system thermal-hydraulic analyses)

ATTACHMENT 3

Catawba Units 1 and 2

Typed Technical Specification Pages

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Steam Generator Water Level Low-Low		
a. Unit 1	≥ 10.7% of narrow range span	≥ 9% of narrow range span
b. Unit 2	≥ 36.8% of narrow range span	≥ 35.1% of narrow range span
13. Undervoltage - Reactor Coolant Pumps	≥ 77% of bus voltage (5082 volts) with a 0.7s response time	≥ 76% (5016 volts)
14. Underfrequency - Reactor Coolant Pumps	≥ 56.4 Hz with a 0.2s response time	≥ 55.9 Hz
15. Turbine Trip		
a. Stop Valve EH Pressure Low	≥ 550 psig	≥ 500 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T' ≤ 585.1°F (Unit 1) and 590.8°F (Unit 2) (Nominal T_{avg} allowed by Safety Analysis);

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For q_t - q_b between the "positive" and "negative" f_1 (ΔI) breakpoints as presented in the Core Operating Limits Report;

f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent ΔI that the magnitude of q_t - q_b is more negative than the f_1 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) For each percent ΔI that the magnitude of q_t - q_b is more positive than the f_1 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.5% of Rated Thermal Power.

NOTE 3: (Continued)

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 585.1°F (Unit 1) and 590.8°F (Unit 2)),

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" f_2 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the f_2 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the f_2 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% (Unit 1) and 3.3% (Unit 2) of Rated Thermal Power.

TABLE 3.3-4 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	≥ 775 psig	≥ 744 psig
e. Steam Line Pressure-Negative Rate - High	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	≤ 83.9% of narrow range instrument span	≤ 85.6% of narrow range instrument span
2. Unit 2	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
c. T_{avg}-Low	≥ 564°F	≥ 561°F
d. Doghouse Water Level-High	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	≤ 83.9% of narrow range instrument span	≤ 85.6% of narrow range instrument span
2. Unit 2	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
7. Containment Pressure Control System		
a. Start Permissive	≤ 0.4 psid	≤ 0.45 psid
b. Termination	≥ 0.3 psid	≥ 0.25 psid
8. Auxiliary Feedwater		
a. Manual Initiation	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)		
c. Steam Generator Water Level - Low-Low		
1) Unit 1	≥ 10.7% of narrow range span	≥ 9% of narrow range span
2) Unit 2	≥ 36.8% of narrow range span	≥ 35.1% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
e. Loss-of-Offsite Power	≥ 3500 V	≥ 3242 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low		
1) CAPS 5220, 5221, 5222	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	≥ 6.2 psig	≥ 5.2 psig
a. Unit 1	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low Coincident With Safety Injection	≥ 177.15 inches See Item 1. above for all Safety Injection Setpoints and Allowable Values.	≥ 162.4 inches

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. 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.

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.

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 (after steam generator replacement on Unit 1) shall be performed after at least 6 Effective Full Power Months but within 24 calendar months of initial criticality (after steam generator replacement on Unit 1). 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.

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

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

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

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging. The plugging limit is equal to 40% of the nominal tube wall thickness.

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;

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.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged 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,

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

   3) Identification of tubes plugged.

c. 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.

Table 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Four	Four
First Inservice Inspection (after Steam Generator Replacement on Unit 1)	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

TABLE NOTATIONS

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 Each of the other two steam generators not inspected (after steam generator replacement on Unit 1) during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug deflective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G.is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR 50.	N/A	N/A

S = 3 (N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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 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. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. 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.

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.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment vessel is designed and shall be maintained for a maximum internal pressure of 15 psig and a temperature of 328°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of cylindrical zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 5.0 weight percent U-235 with a maximum tolerance of ± .05 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material of which 102 inches shall be 100% boron carbide and remaining 40-inch tip shall be 80% silver, 15% indium, and 5% cadmium.

For Units 1 and 2, all control rods shall be clad with stainless steel tubing, except for Unit 2, a maximum of one Rod Cluster Control Assembly may have Inconel clad control rods.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,050 ± 100 cubic feet for Unit 1 and 12,040 ± 100 cubic feet for Unit 2 at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1-1.

CORE OPERATING LIMITS REPORT (Continued)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, and Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002, Rev. 1, "FSAR Chapter 15 System Transient Analysis Methodology," SER dated December, 1995.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," SER dated December, 1995.

   (Modeling used in the system thermal-hydraulic analyses)

ATTACHMENT 4

McGuire Unit 1

Marked-Up Technical Specification Pages

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 16.7% of span ~~≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40% of span at 100% of RATED THERMAL POWER~~	≥ 15% of span ~~≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 39.0% of span at 100% of RATE THERMAL POWER.~~
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1 x 10^{-10} amps	≥ 6 x 10^{-11} amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
2) P-13 Input	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = 585.1 $\leq$ ~~588.2~~°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_1 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_1 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_1 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "positive" slope presented in the Core Operating Limits Report.

T	=	As defined in Note 1,
T''	=	585.1 $\leq$ ~~588.2~~°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and

$f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	83.9% ≤ ~~82%~~ of narrow range instrument span each steam generator	85.6% ≤ ~~83%~~ of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	≥ 16.7% of span ~~≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.~~	≥ 15% of span ~~≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.~~
2) Start Turbine-Driven Pumps	≥ 16.7% of span ~~12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.~~	≥ 15% of span ~~≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.~~
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

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 4.4.5.4.a.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. In addition to the 3% sample, all F* tubes will be inspected.~~

c. ~~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.

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.

after the steam generator replacement

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

at least

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 following service under AVT conditions, 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 leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

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

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

   4) A main steam line or feedwater line break.

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;

Plugging

6) ~~Repair~~ Limit means the imperfection depth at or beyond which the tube ~~or sleeve~~ shall be removed from service by plugging ~~or repaired by sleeving~~ and is equal to 40% of the nominal tube ~~or sleeve~~ wall thickness. ~~This definition does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance. 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~~.

~~The Babcock & Wilcox process (or method) equivalent to the method described in Topical Report BAW-2045(P)-A will be used~~.

7) Unserviceable describes the condition of a tube ~~or sleeve~~ 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; and

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 ~~after the~~ prior to ~~field hydrostatic test and prior to~~ initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) F* Distance is the distance into the tubesheet from the top face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet) that has been conservatively chosen to be 2 inches.

11) F* TUBE is a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) in the F* distance.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug ~~or repair~~ all tubes exceeding the plugging ~~repair~~ limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged 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,

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

3) Identification of tubes plugged ~~or repaired~~.

c. The results of inspections of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:

1) Identification of F* tubes, and

2) Location and size of the degradation.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection After Steam Generator Replacement	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

after steam generator replacement

3 Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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,

c. ~~1~~ 0.27 gpm total primary-to-secondary leakage through all steam generators and ~~500~~ 135 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.

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. 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.

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 RCS 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, McGuire 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 .135 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-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 plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it

BASES

plugging

STEAM GENERATORS (Continued)

will be found during scheduled inservice steam generator tube examinations. ~~Repair~~ will be required for all tubes with imperfections exceeding the ~~repair~~ plugging limit of 40% of the tube nominal wall thickness. ~~Installed sleeves with imperfections exceeding 40% of the sleeve nominal wall thickness will be plugged. 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. ~~For tubes with degradation below the F* distance, and not degraded within the F* distance, repair is not required. 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~~.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS 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

Industry experience has shown that while a limited amount of leakage is expected from the RCS, 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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (continued)

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 RCS 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 accident analyses.

the applicable

and the 135 gpd leakage limit per generator 0.27

The total steam generator tube leakage limit of ~~1~~ gpm for all steam generators ~~not isolated from the RCS~~ 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~~ 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.

0.27 and 135 gpd limits are 135 for all FSAR Chapter 15 transients

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.

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 ~~1.0~~ gpm. The values for the limits on specific activity represent limits based upon a parametric

0.27

FUEL ASSEMBLIES Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material for Unit 1 control rods shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,050 ~~12,040~~ ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

CORE OPERATING LIMITS REPORT (Continued)

8. DPC-NE-3002~~A~~, Rev. 1, "FSAR Chapter 15 System Transient Analysis Methodology," ~~November 1991~~. SER dated December 1995

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P~~-A~~, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," ~~August 1994~~. SER dated December 1995

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ATTACHMENT 5

McGuire Unit 1

Typed Technical Specification Pages

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 16.7% of span	≥ 15% of span
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1×10^{-10} amps	≥ 6×10^{-11} amps

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = $\leq$ 585.1°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

T	=	As defined in Note 1,
T″	=	≤ 585.1°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and

f_2 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_2 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_2 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_2 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	≤ 83.9% of narrow range instrument span each steam generator	≤ 85.6% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	≥ 16.7% of span	≥ 15% of span
2) Start Turbine-Driven Pumps	≥ 16.7% of span	≥ 15% of span
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

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 4.4.5.4.a.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. 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.

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.

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 after the steam generator replacement shall be performed after at least 6 Effective Full Power Months but within 24 calendar months of initial criticality after the steam generator replacement. 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 following service under AVT conditions, 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 leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

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

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

   4) A main steam line or feedwater line break.

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

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

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

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is equal to 40% of the nominal tube wall thickness.

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; and

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.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged 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,

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

3) Identification of tubes plugged.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection After Steam Generator Replacement	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection after steam generator replacement shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3 Each of the other two steam generators not inspected during the first inservice inspections after steam generator replacement shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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,

c. 0.27 gpm total primary-to-secondary leakage through all steam generators and 135 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.

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. 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.

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 RCS 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 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 = 135 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 135 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 plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it

STEAM GENERATORS (Continued)

will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. 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 promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS 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

Industry experience has shown that while a limited amount of leakage is expected from the RCS, 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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (continued)

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 RCS 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 accident analyses.

The total steam generator tube leakage limit of 0.27 gpm for all steam generators and the 135 gpd leakage limit per generator ensures that the dosage contribution from the tube leakage will be limited to the applicable fraction of 10 CFR Part 100 dose guideline values for all FSAR Chapter 15 transients. The 0.27 gpm and the 135 gpd limits are consistent with the assumptions used in the analysis of these accidents. The 135 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.

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.

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 0.27 gpm. The values for the limits on specific activity represent limits based upon a parametric

FUEL ASSEMBLIES Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material for Unit 1 control rods shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,050 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

CORE OPERATING LIMITS REPORT (Continued)

8. DPC-NE-3002, Rev 1, "FSAR Chapter 15 System Transient Analysis Methodology," SER dated December 1995.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," SER dated December 1995.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ATTACHMENT 6

McGuire Unit 2

Marked-Up Technical Specification Pages

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 16.7% of span ≥ 12% of span ~~from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40% of span at 100% of RATED THERMAL POWER~~	≥ 15% of span ≥ 11% of span ~~from 0 to 30% of RATED THERMAL POWER, increasing to 39.9% of span at 100% of RATE THERMAL POWER.~~
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1 x 10^{-10} amps	≥ 6 x 10^{-11} amps

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

585.1

T' = $\le$ ~~588.2~~°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" f_1 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the f_1 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the f_1 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "positive" slope presented in the Core Operating Limits Report.

T = As defined in Note 1,

T″ = 585.1 ≤ ~~588.2~~°F Reference T_{avg} at RATED THERMAL POWER,

S = As defined in Note 1, and

f_2 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_2 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_2 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_2 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	83.9% ≤ ~~82%~~ of narrow range instrument span each steam generator	85.6% ≤ ~~83%~~ of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low 1) Start Motor-Driven Pumps	≥ 16.7% of span ~~≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.~~	≥ 15% of span ~~≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.~~
2) Start Turbine-Driven Pumps	≥ 16.7% of span ~~12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.~~	≥ 15% of span ~~≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.~~
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

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 4.4.5.4.a.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. In addition to the 3% sample, all F* tubes will be inspected.~~

c. ~~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.

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.

after the steam generator replacement

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

at least

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 following service under AVT conditions, 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 leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

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

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

   4) A main steam line or feedwater line break.

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;

Plugging

6) ~~Repair~~ Limit means the imperfection depth at or beyond which the tube ~~or sleeve~~ shall be removed from service by plugging ~~or repaired by sleeving~~ and is equal to 40% of the nominal tube ~~or sleeve~~ wall thickness. ~~This definition does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance. 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.~~

~~The Babcock & Wilcox process (or method) equivalent to the method described in Topical Report BAW-2045(P)-A will be used.~~

7) Unserviceable describes the condition of a tube ~~or sleeve~~ 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; and

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 ~~after the~~ prior to ~~field hydrostatic test and prior to~~ initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) F* Distance is the distance into the tubesheet from the top face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet) that has been conservatively chosen to be 2 inches.

11) F* TUBE is a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) in the F* distance.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug ~~or repair~~ all tubes exceeding the plugging ~~repair~~ limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged 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,

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

3) Identification of tubes plugged ~~or repaired~~.

c. The results of inspections of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:

1) Identification of F* tubes, and

2) Location and size of the degradation.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection After Steam Generator Replacement	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

after steam generator replacement

3 Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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,

c. ~~1~~ 0.27 gpm total primary-to-secondary leakage through all steam generators and ~~500~~ 135 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.

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. 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.

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 RCS 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, McGuire 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 135 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-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 plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it

STEAM GENERATORS (Continued)

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. ~~Installed sleeves with imperfections exceeding 40% of the sleeve nominal wall thickness will be plugged. 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. ~~For tubes with degradation below the F* distance, and not degraded within the F* distance, repair is not required. 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~~.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS 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

Industry experience has shown that while a limited amount of leakage is expected from the RCS, 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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (Continued)

the applicable

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 RCS 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 accident analyses.

and the 135 gpd leakage limit per generator 0.27

The total steam generator tube leakage limit of ~~1~~ gpm for all steam generators ~~not isolated from the RCS~~ 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~~ 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.

0.27

and 135gpd limits are

135

for all FSAR Chapter 15 transients

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.

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 ~~1.0~~ gpm. The values for the limits on specific activity represent limits based upon a parametric

0.27

FUEL ASSEMBLIES (Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material for Unit 2 control rods shall be 100% boron carbide (B_4C) for 102 inches and 80% silver, 15% indium, and 5% cadmium for the 40-inch tip. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

13,050

5.4.2 The total water and steam volume of the Reactor Coolant System is ~~12,040~~ ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

CORE OPERATING LIMITS REPORT

8. DPC-NE-3002~~A~~, Rev. 1, "FSAR Chapter 15 System Transient Analysis Methodology," ~~November 1991~~. SER dated December 1995

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P~~-A~~, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," ~~August 1994~~. SER dated December 1995

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ATTACHMENT 7

McGuire Unit 2

Typed Technical Specification Pages

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 16.7% of span	≥ 15% of span
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1 x 10^{-10} amps	≥ 6 x 10^{-11} amps

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = $\leq$ 585.1°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_1 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_1 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_1 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "positive" slope presented in the Core Operating Limits Report.

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

T	=	As defined in Note 1,
T″	=	≤ 585.1°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and

f_2 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_2 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_2 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_2 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	≤ 83.9% of narrow range instrument span each steam generator	≤ 85.6% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	$\geq$ 16.7% of span	$\geq$ 15% of span
2) Start Turbine-Driven Pumps	$\geq$ 16.7% of span	$\geq$ 15% of span
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	$\geq$ 2 psig	$\geq$ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

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 4.4.5.4.a.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. 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.

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.

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 after the steam generator replacement shall be performed after at least 6 Effective Full Power Months but within 24 calendar months of initial criticality after the steam generator replacement. 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 following service under AVT conditions, 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 leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

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

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

   4) A main steam line or feedwater line break.

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

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

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

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is equal to 40% of the nominal tube wall thickness.

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; and

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.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged 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,

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

3) Identification of tubes plugged.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection After Steam Generator Replacement	All[1]			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection after steam generator replacement shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3 Each of the other two steam generators not inspected during the first inservice inspections after steam generator replacement shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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,

c. 0.27 gpm total primary-to-secondary leakage through all steam generators and 135 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.

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. 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.

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 RCS 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 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 = 135 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 135 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 plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it

STEAM GENERATORS (Continued)

will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. 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 promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS 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

Industry experience has shown that while a limited amount of leakage is expected from the RCS, 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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (continued)

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 RCS 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 accident analyses.

The total steam generator tube leakage limit of 0.27 gpm for all steam generators and the 135 gpd leakage limit per generator ensures that the dosage contribution from the tube leakage will be limited to the applicable fraction of 10 CFR Part 100 dose guideline values for all FSAR Chapter 15 transients. The 0.27 gpm and the 135 gpd limits are consistent with the assumptions used in the analysis of these accidents. The 135 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.

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.

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 0.27 gpm. The values for the limits on specific activity represent limits based upon a parametric

FUEL ASSEMBLIES Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material for Unit 1 control rods shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,050 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

CORE OPERATING LIMITS REPORT (Continued)

8. DPC-NE-3002, Rev 1, "FSAR Chapter 15 System Transient Analysis Methodology," SER dated December 1995.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," SER dated December 1995.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.