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 RATED 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

9410070183 940930
PDR ADOCK 05000369
P PDR

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	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Negative Steam Line Pressure Rate - High	$\leq$ 100 psi with a rate/lag function time constant $\geq$ 50 seconds	$\leq$ 120 psi with a rate/lag function time constant $\geq$ 50 seconds
e. Steam Line Pressure - Low	$\geq$ 775 psig	$\geq$ 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)	$\leq$ 83.9% of narrow range instrument span each steam generator	$\leq$ 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 $\leq$ SP/T $\leq$ 0.4 PSIG	0.25 $\leq$ SP/T $\leq$ 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	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump (Note 1)	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	≥ 3200 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

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

at least

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections 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

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 field hydrostatic test and prior to~~ 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 ~~repair~~ 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. ~~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 the 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.

after steam generator replacement

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.

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. 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.

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Loop A	Loop B	Loop C	Loop D	LIFT SETTING (± 3%)*	ORIFICE SIZE
1.	SV 20	SV 14	SV 8	SV 2	1170 psig	12.174 in^2
2.	SV 21	SV 15	SV 9	SV 3	1190 psig	12.174 in^2
3.	SV 22	SV 16	SV 10	SV 4	1205 psig	16.00 in^2
4.	SV 23	SV 17	SV 11	SV 5	1210 ~~1220~~ psig	16.00 in^2
5.	SV 24	SV 18	SV 12	SV 6	1220 ~~1225~~ psig	16.00 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

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 between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = ~~500~~ 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 ~~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

3/4.4.5 STEAM GENERATORS (Continued)

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 will be found during scheduled inservice steam generator tube examinations. Plugging ~~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.~~

STEAM GENERATORS (Continued)

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.

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 [redacted] gpm for all steam generators and the 135 gpd Leakage limit per generator [redacted] ensure that the dosage contribution from the tube leakage will be limited to the applicable [redacted] fraction of 10 CFR Part 100 dose guideline values [redacted] The [redacted] gpm [redacted] consistent with the assumptions used in the analysis of these accidents. The [redacted] 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

OPERATIONAL LEAKAGE (Continued)

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~~ 0.27 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the McGuire site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

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

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 The new and spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in Section 9.1.2.3.1 of the FSAR, and

b. A nominal 21-inch center-to-center distance between fuel assemblies placed in the new fuel storage vault racks, and

c. A nominal 10.4-inch and 9.125-inch center-to-center distance between fuel assemblies placed in Region 1 and Region 2 storage racks, respectively, in the spent fuel storage pool.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft. 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 spaces in Region 1 and 1177 spaces in Region 2) having an initial enrichment less than or equal to 4.0 weight percent U-235.

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for FQ Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," September 1989 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011P, "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, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," March 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-NE-2010P, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," April 1984 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

8. DPC-NE-3002 Rev 1, "FSAR Chapter 15 System Transient Analysis Methodology," August 1991.

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

CORE OPERATING LIMITS REPORT

9. DPC-NE-3000, Rev. ~~1~~ 3, "Thermal-Hydraulic Transient Analysis Methodology," May 1989.

   (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.)

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.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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

Attachment 3

Proposed Revision to Technical Specification Table 2.2-1

The low-low steam generator water level reactor trip setpoint is modified from a variable setpoint that is proportional to nuclear power to a constant level setpoint of 16.7% of narrow range span.

Technical Justification

These setpoints were chosen to maximize the plant operating region while still ensuring that reactor trip on low-low level would occur following a feedline break inside containment. The new low-low level setpoints are consistent with all reanalyzed licensing basis safety analyses. All of the reanalyzed transients that take credit for this trip function meet the applicable acceptance criteria.

Proposed Revision to Technical Specification Table 3.3-4

a) The high-high steam generator water level setpoint is changed to 83.9% of narrow range span.

b) The low-low steam generator water level auxiliary feedwater actuation setpoint will be modified from a variable setpoint that is proportional to nuclear power to a constant level setpoint of 16.7% of narrow range span.

Technical Justification

a) This setpoint was chosen to maximize the plant operating region while still ensuring that feedwater isolation on high-high level would occur prior to the actual water level in the generator reaching the flood point. The new high-high level setpoint is consistent with all reanalyzed licensing basis safety analyses. The results of the increase in feedwater flow analysis, which is the only FSAR Chapter 15 transient which relies on this trip function, show that all applicable acceptance criteria are met.

b) Refer to the technical justification for the proposed revision to Technical Specification Table 2.2-1.

Proposed Revision to Technical Specification 3/4.4.4

The Surveillance Requirements are changed to delete repair methods which are no longer applicable after the replacement of the steam generators. References to F*, and sleeving are deleted. Clarification is added to the surveillance requirements on

performing initial inspections after replacement of the steam generators and when they will be performed.

Technical Justification

This proposed change to the Technical Specifications deletes repair criteria which will no longer be applicable after the replacement of the steam generators. References to F* criteria and sleeving are removed because these methods of repair were approved specifically for use on the current steam generators. Clarification is also added to show that initial inspections will be performed after replacement of the steam generators and that the baseline inspection of the tubing will be performed after the manufacturer performs the hydrostatic test. These changes will not alter the way surveillance's are performed, and continue to meet the current intent of the requirements.

Proposed Revision to Technical Specification 3/4.4.6.2

This proposed Technical Specification revision reduces the allowable primary-to-secondary system leakage to a total of 0.27 gpm through all steam generators and 135 gallons per day through any one generator.

Technical Justification

The primary-to-secondary leak rate has a major impact on the results of the offsite dose calculation for the locked rotor, single uncontrolled rod withdrawal, and rod ejection events. This leakage is the mechanism for the release of the fission products. The taller tubes in the feedring steam generators potentially result in a longer period of tube bundle uncovery during the transient; this necessitated the recalculation of the offsite doses. In order to ensure that the doses do not exceed the acceptance criterion, the allowable leak rate must be reduced.

Proposed Revision to Technical Specification Table 3.7-3

The steam line safety valve lift settings for banks 4 and 5 are changed to 1210 and 1220 psig, respectively.

Technical Justification

A preliminary analysis showed the peak secondary system pressure for the turbine trip event to exceed the acceptance criterion of 110% of design pressure. The design basis of these safety valves is specifically to prevent such an overpressurization. Primarily due to the increased lift setting tolerance of ±3% assumed in the analysis, the final two

banks of safety valves were not fully open at the point in the analysis when the peak system pressure occurred. Reducing the lift settings for these two safety banks results in acceptable turbine trip transient results.

Proposed Revision to Technical Specification 5.4.2

The volume of the Reactor Coolant System changes from 12,040 $\pm$ 100 cubic feet to 13050 $\pm$ 100 cubic feet.

Technical Justification

The mass and energy release for postulated loss of coolant accidents inside containment is analyzed to ensure that the peak containment pressure limit is not exceeded. Since the Reactor Coolant System volume is greater, the total mass released into containment will be greater. In addition, during the depressurization of the RCS, the steam generators actually function as heat sources. Since the feedring steam generator full power liquid mass is greater than that of the Model D steam generators, the total energy available for removal by the RCS is increased. Both of these effects have the potential to yield more severe mass and energy release results. This event and all other reanalysis which was required for the replacement steam generators has been done assuming the new reactor coolant system volume. The results of these analyses show that the applicable acceptance criteria continue to met.

Proposed Changes to Technical Specification 6.9.1.9

The references to DPC-NE-3000 and DPC-NE-3002 have been updated to reflect the use of the most current approved revision to the topical reports for the replacement steam generators.

Technical Justification

This revision, which reflects the use of the most current revision to the above topical reports, is administrative in nature.

Attachment 4

No Significant Hazards Analysis
McGuire Nuclear Station

The following analysis, required by 50.91, concludes that the proposed amendment will not involve significant hazards considerations as defined by 10 CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Operation of McGuire Nuclear Station in accordance with the proposed changes to the Technical Specifications will not involve a significant increase in the probability or consequences of an accident previously evaluated. The low-low steam generator water level reactor trip setpoint, the high-high steam generator water level setpoint for turbine trip and feedwater isolation, and the low-low steam generator water level setpoint for auxiliary feedwater initiation are changing to support operation with the replacement steam generators. These setpoints were chosen both to optimize plant operation, and ensure that all applicable acceptance criteria are met for licensing basis safety analysis. These setpoints do not contribute to the initiation of any accident evaluated in the McGuire FSAR and have no adverse impact on system operation, therefore it can be concluded that these changes will not significantly increase the probability or consequences of an accident evaluated in the FSAR.

The reduction in the primary to secondary leakage rate for McGuire will not increase the probability of an accident evaluated in the FSAR. This lower limit will require corrective action more quickly than is currently required in the event that there is a steam generator tube leak. This change will not significantly affect the consequences of an accident previously evaluated. The allowable leakage is being lowered because this leakage has a major impact on the results of the offsite dose calculation for the locked rotor, single uncontrolled rod withdrawal, and rod ejection events. The taller tube bundle in the replacement steam generators will potentially result in a longer period of tube bundle uncovery during the above transients. The revised allowable leakages of 0.27

gpm through all steam generators and 135 gallons per day through any one generator ensure that the dose analysis results are within the applicable fraction 10 CFR 100 limits.

The increase in Reactor Coolant System volume due to the replacement steam generators will not increase the probability or consequences of an accident previously evaluated. The increase in volume has no effect on the probability of occurrence of any accident evaluated in the FSAR. The mass and energy release due to postulated loss of coolant accidents inside containment has been analyzed to ensure that the peak containment pressure limit is not exceeded. All Chapter 15 reanalysis which was required due to the replacement steam generators assumed the new Reactor Coolant System volume. Since the results of these analyses show the applicable acceptance criteria continue to be met, it can be concluded that the consequences of an accident previously evaluated are not significantly increased due to this change.

The changes in the steam line safety valve lift settings to 1210 and 1220 psig respectively ensure that the peak secondary system pressure for the limiting ANS Condition II event, turbine trip, does not exceed the acceptance criterion of 110% design pressure. The design basis of these valves is to prevent such an overpressurization. Since reducing these lift setpoints results in acceptable turbine trip transient results by ensuring that the valves perform their design basis function, it can be concluded that the probability or consequences of an accident previously evaluated is not significantly increased.

Operation of McGuire Nuclear Station in accordance with the proposed changes to the Technical Specification will not create the possibility of a new or different accident from any accident previously evaluated. The proposed changes to revise the low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater initiation ensure that the appropriate acceptance criteria for FSAR Chapter 15 transients which rely on these functions are met for operation with the replacement steam generators. The proposed change to lower primary to secondary leakage for operation with the replacement steam generators will require that corrective action be taken more quickly in the event that steam generator tube leakage is experienced during operation. As discussed in the technical justification, this will cause the dose results for transients affected by tube bundle uncovery to be within acceptable limits. The proposed change to Table 3.7-3 to reduce steam line safety valve lift settings allows the valves to perform their design basis function of ensuring that the peak secondary system pressure of 110% design is not exceeded for the turbine trip event, which is the limiting ANS Condition II event. The increase in Reactor Coolant System volume is taken into account in the analysis of the mass and energy release due to a postulated loss of coolant inside containment and Chapter 15 events which have been reanalyzed due to

replacement of the steam generators. As discussed above, the proposed changes will not introduce the possibility of a new or different accident from any previously evaluated; they will ensure that transients that take credit for these functions and dose analyses meet applicable acceptance criteria for operation with the replacement steam generators.

Operation of McC ire Nuclear Station in accordance with the proposed changes to the Technical Specifications will not involve a significant reduction in a margin of safety. The proposed changes are being made to ensure that transients that rely on low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater actuation meet applicable acceptance criteria. The reduction in allowable primary to secondary leak rate will ensure that transients with dose analyses which are affected by the replacement steam generators meet the current acceptable limits. The reduction in the steam line safety valve lift settings will ensure that the design basis of these valves is met. The proposed change in the Reactor Coolant System volume will not involve a significant reduction in a margin of safety. The increased volume affects the mass and energy release due to a postulated loss of coolant accident inside containment and the other Chapter 15 events which were reanalyzed due to replacement of the steam generators. These events have been analyzed and the results are within current acceptable limits. As discussed above, the acceptance criteria for FSAR transients which are affected by these proposed changes continue to be met, therefore there is no significant reduction in the margin of safety.

Changes to the steam generator surveillance requirements will simply delete inspection requirements and repair methods which are no longer applicable after installation of the replacement steam generators. The only exception to this is Surveillance Requirement 4.4.5.4.a.9. This requirement is modified to clarify that the manufacturer will perform the hydrostatic test for the replacement steam generators. This change will not affect the probability or consequences of an accident previously evaluated, the purpose of the preservice inspection is to establish the baseline condition of the tubing. The baseline condition of the tubing in the replacement steam generators will be established prior to installation. The possibility of a new or different accident from any previously evaluated will not be created. No new accident initiation mechanisms will be introduced by this change, and the intent of the requirement, to establish the baseline condition of the tubing, will be met. Since the baseline condition of the tubing will be obtained for use in the monitoring of tubing degradation, as is currently required by the surveillance requirement, there will not be a significant reduction in the margin of safety.

The changes to Technical Specification 6.9.1.9 are administrative in nature. These changes are being made to reflect the most recent revisions of DPC-NE-

3002 and DPC-NE-3000, which include changes associated with the replacement steam generators. These topical reports revisions will be reviewed and approved for use regarding Catawba and McGuire Nuclear Stations. Since these changes are administrative in nature, no significant hazards considerations are involved.

Attachment 5