Docket No. 50-423
B14477

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Removal of Component Lists from Technical Specifications
Marked Up Pages of Technical Specifications

September 1993

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ACTION 9 - ~~With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.~~

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the Turbine Control Valves.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 13A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is operable.

TABLE 3.3-3 (Continued)
TABLE NOTATIONS

#The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

~~**The Safety Injection Logic for this trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.~~

~~***The channel(s) associated with the protective functions derived from the out-of-service reactor coolant loop shall be placed in the tripped mode.~~

****Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - (not used).

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

March 14, 1989

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pumps	18.10	16.64	1.50	$\geq$ 18.10% of narrow range instrument span.	$\geq$ 17.11% of narrow range instrument span.
d. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps	N.A.	N.A.	N.A.	$\geq$ 2800V	$\geq$ 2720V
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA Trip Setpoints and Allowable Values.				
7. Control Building Isolation					
a. Manual Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.3	1.01	1.75	$\leq$ 3.0 psig	$\leq$ 3.8 psig
e. Control Building Inlet Ventilation Radiation	N.A.	N.A.	N.A.	$\leq 1.5 \times 10^{-5}$ μc/cc	$\leq 1.5 \times 10^{-5}$ μc/cc

μCi/cc

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

    1) Primary-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.

* Except that the surveillance related to the steam generator inspection, due no later than August 21, 1993, may be deferred until the next refueling outage or no later than September 30, 1993, whichever is earlier.

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.65% by weight of the containment air per 24 hours at P_a, 53.27 psia (38.57 psig);

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a; and

c. A combined leakage rate of less than or equal to 0.042 L_a for all penetrations ~~identified in Table 3.6-1 as~~ that are Enclosure Building bypass leakage paths when pressurized to P_a.

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 0.75 L_a, or the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, or the combined bypass leakage rate exceeding 0.042 L_a, restore the overall integrated leakage rate to less than 0.75 L_a, the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.042 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using methods and provisions of ANSI N45.4-1972 (Total Time Method) and/or ANSI/ANS 56.8-1981 (Mass Point Method):

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 53.27 psia (38.57 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

SURVEILLANCE REQUIREMENTS (Continued)

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the test by verifying that the supplemental test results, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 53.27 psia (38.57 psig), at intervals no greater than 24 months(*) except for tests involving:

   1) Air locks

e. The combined bypass leakage rate shall be determined to be less than or equal to 0.042 L_a by applicable Type B and C tests at least once per 24 months(*) except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 53.27 psia (38.57 psig), during each Type A test;

f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.

h. The provisions of Specification 4.0.2 are not applicable.

(*) The 24-month interval for Type B and Type C tests has been increased to 34 months for Cycle 4 only.

FOR INFO ONLY

January 25, 1991

TABLE 3.6-1

ENCLOSURE BUILDING BYPASS LEAKAGE PATHS

PENETRATION	DESCRIPTION	RELEASE LOCATION
14	N_2 to Safety Injection Tanks	Ground Release
15	Primary Water to Pressurizer Relief Tanks	Ground Release
35	Vacuum Pump Suction	Plant Vent
36	Vacuum Pump Suction	Plant Vent
37	Air Ejector Suction	Plant Vent
38	Chilled Water Supply	Plant Vent
45	Chilled Water Return	Plant Vent
52	Service Air	Turbine Building Roof Exhaust
54	Instrument Air	Turbine Building Roof Exhaust
56	Fire Protection	Ground Release
59	Fuel Pool Purification	Ground Release
60	Fuel Pool Purification	Ground Release
70	Demineralized Water	Ground Release
72	Chilled Water Supply	Plant Vent
85	Containment Purge	Ground Release
86	Containment Purge	Plant Vent
116	Chilled Water Return	Plant Vent
124	Nitrogen to Containment	Plant Vent

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PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does not exceed the design pressure of 60 psia during LOCA conditions. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature. The limits on the pressure and average air temperature are consistent with the assumptions of the safety analysis. The minimum total containment pressure of 10.6 psia is determined by summing the minimum permissible air partial pressure of 8.9 psia and the maximum expected vapor pressure of 1.7 psia (occurring at the maximum permissible containment initial temperature of 120°F).

THE ENCLOSURE BUILDING BYPASS LEAKAGE PATHS ARE LISTED IN OPERATING PROCEDURE 3273, "TECHNICAL REQUIREMENTS - SUPPLEMENTARY TECHNICAL SPECIFICATIONS." THE ADDITION OR DELETION OF THE ENCLOSURE BUILDING BYPASS LEAKAGE PATHS SHALL BE MADE IN ACCORDANCE WITH SECTION 50.59 OF 10 CFR 50 AND APPROVED BY THE PLANT OPERATION REVIEW COMMITTEE.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The molded case circuit breakers and unitized starters will be tested in accordance with Manufacturer's Instructions.

The OPERABILITY of the motor-operated valves thermal overload protection and integral bypass devices ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of the thermal overload protection are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

~~FSAR Tables 8.3-7, 8.3-8 and 8.3-9~~ list containment penetration conductor overcurrent protective devices and thermal overload protection bypassed only under accident conditions and thermal overload protection not bypassed under accident conditions. The addition or deletion of any device shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operation Review Committee.

OPERATING PROCEDURE 3273, "TECHNICAL REQUIREMENTS-SUPPLEMENTARY TECHNICAL SPECIFICATIONS,"

DOSE - RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The REMODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The REMODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The release rate specifications for radioiodines and radionuclides in particulate form and radionuclides other than noble gases are dependent upon the existing radionuclide pathways to man. The pathways that are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure to man.

3/4.11.~~4~~ 3 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190. For the purposes of the Special Report, it may be assumed that the dose commitment to any REAL MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Removal of Component Lists from Technical Specifications
Retyped Pages of Technical Specifications

September 1993

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.10 SPECIAL TEST EXCEPTIONS

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ACTION STATEMENTS (Continued)

ACTION 9 - (Not used)

ACTION 10 - With the number of OPERABLF channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the Turbine Control Valves.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 13A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is operable.

TABLE 3.3-3 (Continued)
TABLE NOTATIONS

#The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**** Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - (not used).

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pumps	18.10	16.64	1.50	$\geq$ 18.10% of narrow range instrument span.	$\geq$ 17.11% of narrow range instrument span.
d. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps	N.A.	N.A.	N.A.	$\geq$ 2800V	$\geq$ 2720V
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA Trip Setpoints and Allowable Values.				
7. Control Building Isolation					
a. Manual Actuation	N.A.	N.A.	N.A.	N.A	N.A.
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	N.A	N.A.
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.3	1.01	1.75	$\leq$ 3.0 psig	$\leq$ 3.8 psig
e. Control Building Inlet Ventilation Radiation	N.A.	N.A.	N.A	$\leq 1.5 \times 10^{-5}$ μci/cc	$\leq 1.5 \times 10^{-5}$ μci/cc

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

a. 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) Primary-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.

* Except that the surveillance related to the steam generator inspection, due no later than August 21, 1993, may be deferred until the next refueling outage or no later than September 30, 1993, whichever is earlier.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.65% by weight of the containment air per 24 hours at P_a, 53.27 psia (38.57 psig);

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a; and

c. A combined leakage rate of less than or equal to 0.042 L_a for all penetrations that are Enclosure Building bypass leakage paths when pressurized to P_a.

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 0.75 L_a, or the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, or the combined bypass leakage rate exceeding 0.042 L_a, restore the overall integrated leakage rate to less than 0.75 L_a, the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.042 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using methods and provisions of ANSI N45.4-1972 (Total Time Method) and/or ANSI/ANS 56.8-1981 (Mass Point Method):

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 53.27 psia (38.57 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

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PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

The enclosure building bypass leakage paths are listed in Operating Procedure 3273, "Technical Requirements - Supplementary Technical Specifications." The addition or deletion of the enclosure building bypass leakage paths shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operation Review Committee.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE (continued)

not exceed the design pressure of 60 psia during LOCA conditions. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature. The limits on the pressure and average air temperature are consistent with the assumptions of the safety analysis. The minimum total containment pressure of 10.6 psia is determined by summing the minimum permissible air partial pressure of 8.9 psia and the maximum expected vapor pressure of 1.7 psia (occurring at the maximum permissible containment initial temperature of 120°F).

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The molded case circuit breakers and unitized starters will be tested in accordance with Manufacturer's Instructions.

The OPERABILITY of the motor-operated valves thermal overload protection and integral bypass devices ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of the thermal overload protection are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

Operating Procedure 3273, "Technical Requirements - Supplementary Technical Specifications," list containment penetration conductor overcurrent protective devices and thermal overload protection bypassed only under accident conditions and thermal overload protection not bypassed under accident conditions. The addition or deletion of any device shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operation Review Committee.

DOSE - RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The REMODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The REMODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The release rate specifications for radioiodines and radionuclides in particulate form and radionuclides other than noble gases are dependent upon the existing radionuclide pathways to man. The pathways that are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure to man.

3/4.11.3 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190. For the purposes of the Special Report, it may be assumed that the dose commitment to any REAL MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.