ATTACHMENT 1

TO

Letter from C. R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

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PROPOSED TS AMENDMENT NO. 103

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TECHNICAL SPECIFICATIONS AND BASES

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability, that must be enforced to ensure safe operation of the facility.

e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0; and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The nonautomatic Containment System isolation valves and blind flanges are closed as required.

2. The Reactor Containment Vessel and Shield Building equipment hatches are properly closed.

3. At least ONE door in both the personnel and the emergency airlocks is properly closed.

4. The required automatic Containment System isolation valves are OPERABLE or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed.

5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.

6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.b.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq$ -5%	$\leq$ 140	~0
COLD SHUTDOWN	$\leq$ -1%	$\leq$ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	$\geq$ 540	~0
HOT STANDBY	< 0.25%	$\sim T_{oper}$	< 2
OPERATING	< 0.25%	$\sim T_{oper}$	$\geq$ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to Figure TS 3.10-1			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting off-gases from the primary coolant system and providing for delay or holdup for the purpose of reducing the total radioactivity released to the environment.

2. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

3. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

4. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, federal and state regulations and other requirements governing the disposal of the radioactive waste.

5. PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other OPERATING condition, in such a manner that replacement air or gas is required to purify the confinement.

6. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

7. SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

8. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

9. VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature atmospheric cleanup systems (i.e., Auxiliary Building special ventilation, Shield Building ventilation, spent fuel pool ventilation) are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

10. VENTING

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other OPERATING conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, as used in system names, does not imply a VENTING process.

11. RADIOLOGICAL ENVIRONMENTAL MONITORING MANUAL (REMM)

The REMM shall contain the current methodology and parameters used in the conduct of the radiological environmental monitoring program.

p. STANDARD SHUTDOWN SEQUENCE

When a LIMITING CONDITION FOR OPERATION is not met, and a plant shutdown is required except as provided in the associated action requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 36 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the specified time limits as measured from the time of determination of the failure to meet the LIMITING CONDITION FOR OPERATION. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable when the plant is in COLD or REFUELING SHUTDOWN.

q. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0361	I-132
0.2703	I-133
0.0169	I-134
0.0838	I-135

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the Operating status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.

2. Decay Heat Removal Capability

   A. At least TWO of the following FOUR heat sinks shall be operable whenever the average reactor coolant temperature is ≤ 350°F but > 200°F.

      1. Steam Generator 1A
      2. Steam Generator 1B
      3. Residual Heat Removal Train A
      4. Residual Heat Removal Train B

      If less than the above number of required heat sinks are operable, corrective action shall be taken immediately to restore the minimum number to the operable status.

B. TWO residual heat removal trains shall be operable whenever the average reactor coolant temperature is $\leq$ 200°F and irradiated fuel is in the reactor, except when in the REFUELING mode with the minimum water level above the top of the vessel flange $\geq$ 23 feet, one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

   a) ONE operable residual heat removal pump

   b) ONE operable residual heat removal heat exchanger

   c) An operable flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, corrective action shall be taken immediately to return it to the operable status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY modes, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[1]

C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

A. Two PORVs and their associated block valves shall be operable during HOT STANDBY and OPERATING modes.

1. If a pressurizer PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed and maintained closed by administrative procedures to prevent inadvertent opening.

2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the valve; otherwise the unit shall be placed in the HOT SHUTDOWN condition using normal operating procedures.

6. Pressurizer Heaters

A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is $> 350°F$.

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[1]Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position with their power breakers locked out.

7. Reactor Coolant Vent System

A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be operable and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.

B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:

1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.

2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

If operability is not restored within 30 days, then within one hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

C. If no Reactor Coolant System vent paths are operable, restore at least one vent path to operable status within 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2 for the service period up to 20 equivalent full-power years.

   A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   B. Figures TS 3.1-I and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

   A. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\bar{E}$ is the average sum of the beta and gamma energies in Mev per disintegration)

   whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

   A. With the specific activity of the reactor coolant > 1 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding the limit shown on Figure TS 3.1-3, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within 6 hours.

   B. With the specific activity of the reactor coolant > $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within 6 hours.

   C. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 *or* > $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item f, once every 4 hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 500 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable.

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is > 250°F.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee shall be made before starting.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is ≤ 250°F.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.

6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F.

f. Minimum Conditions for Criticality

1. Except during low-power physics tests, the reactor shall not be made critical unless the moderator temperature coefficient is negative.

2. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.

3. Except during low-power physics tests, when the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.

BASES - Operational Components (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

The requirement for at least one train of residual heat removal when in the REFUELING MODE is to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel < 140°F. The requirement to have two trains of residual heat removal operable when there is < 23 feet of water above the reactor vessel flange ensures that a single failure will not result in complete loss-of-heat removal capabilities. With the reactor vessel head removed and at least 23 feet of water above the vessel flange, a large heat sink is available. In the event of a failure of the operable train, additional time is available to initiate alternate core cooling procedures.

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[1] USAR Section 7.2.2

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

Pressure Isolation Valves (TS 3.1.a.4)

The Basis for the Pressure Isolation Valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.[2]

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the pressurizer pressure control system. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests[3] have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will assure the ability to maintain pressurizer pressure. Annual surveillance tests are performed to ensure heater operability.

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[2] Order for Modification of License dated 4/20/81

[3] Hot functional test (PT-RC-31)

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the high point vent system is to vent noncondensible gases from the high points of the RCS to assure that core cooling during natural circulation will not be inhibited. The operability of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.[4]

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties - (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code[5], and the calculation methods of Footnote[6]. The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

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, and are discussed in detail in Footnote[7].

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[4]Letter from E. R. Mathews to S. A. Varga dated 5/21/82

[5]ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[6]Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

[7]WCAP-13229, "Heatup and Cooldown Limit Curves for Normal Operation for Kewaunee," M. A. Ramirez and J. M. Chicots, March 1992 (Westinghouse Proprietary Class 3)

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\ K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{Im} is the stress intensity factor due to membrane (pressure) stress. K_{It} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{It} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{IR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan[8][9].

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 9878[10], weld metal Charpy test specimens from Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (235°F).

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[8] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[9] ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[10] S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

The results of Irradiation Capsules V, R, and P analyses are presented in WCAP 8908[11], WCAP 9878, and WCAP-12020[12], respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 20 effective full-power years.

Pressurizer Limits - (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Maximum Coolant Activity (TS 3.1.c)

This specification is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[13] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.[13]

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[11] S. E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

[12] S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

[13] USAR Section 14.2.4

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose,\ rem = 1/2\ \left[\overline{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10})\ \ (1.33 \times 10^{-11})\right]$$

Where: $\overline{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\overline{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9×10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 1 μCi/grams DOSE EQUIVALENT I-131, but within the allowable limit shown in Figure TS 3.1-3, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing average coolant to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

Leakage of Reactor Coolant (TS 3.1.d)[14]

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.5 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

[14]USAR Sections 6.5, 11.2.3, 14.2.4

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

   If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[15]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[16]. Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[17][18]

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

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[15] USAR Section 4.2

[16] USAR Section 9.2

[17] USAR Table 3.2-1

[18] USAR Figure 3.2-8

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[19] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.2 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[20]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

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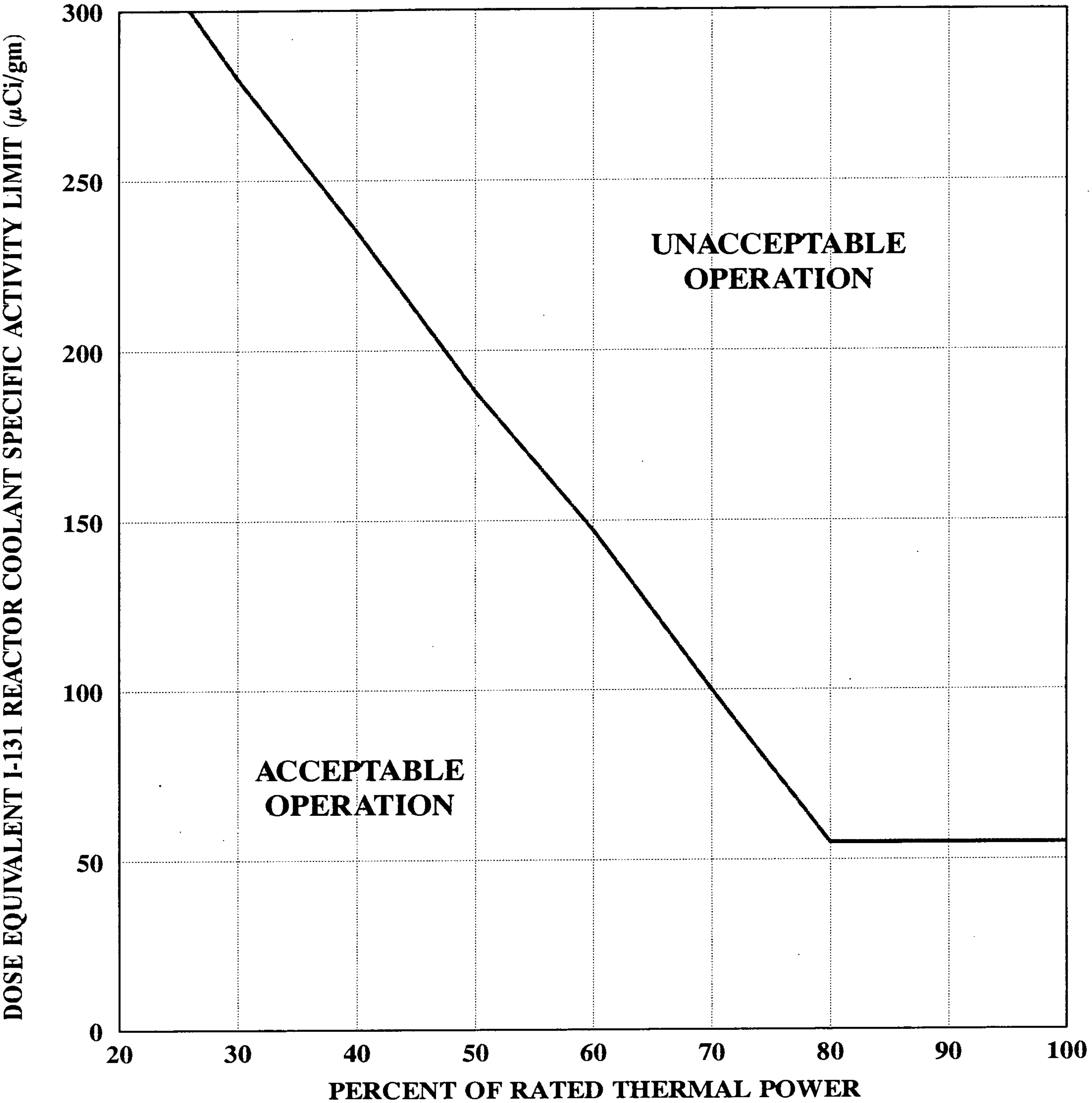
[19] USAR Figure 3.2-9

[20] USAR Table 3.2-1

FIGURE TS 3.1-3



DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1 μCi/GRAM DOSE EQUIVALENT I-131

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY	MAXIMUM TIME BETWEEN TESTS (DAYS)
1. Reactor Coolant Samples	a. Gross Radioactivity Determination (excluding tritium)	5/week	3
	b. DOSE EQUIVALENT I-131 Concentration	1/14 days[(1)]	17
	c. Tritium activity	Monthly	37
	d. Chemistry (Cl, F, O_2)*	3/week	4
	e. $\bar{E}$ Determination	1/6 months[(2)]	227
	f. RCS isotopic analysis for Iodine	Once per 4 hours in accordance with TS 3.1.c.2.C.	
2. Reactor Coolant Boron[(3)]	Boron Concentration*	2/week	5

[(1)] Sample required only when in the OPERATING MODE.

[(2)] Sample after a minimum of 2 EFPD and 20 days of OPERATING MODE operation have elapsed since the reactor was last subcritical for ≥ 48 hours.

[(3)] A reactor coolant boron concentration sample does not have to be taken when the core is completely unloaded.

* See TS 4.1.d

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY	MAXIMUM TIME BETWEEN TESTS (DAYS)
3. Refueling Water Storage Tank Water Sample[(4)]	Boron Concentration	Monthly[(5)]	37
4. Boric Acid Tanks	Boron Concentration	Weekly	8
5. Accumulator	Boron Concentration	Monthly	37
6. Spent Fuel Pool	Boron Concentration	Monthly[(6)]	37
7. Secondary Coolant	a. Gross Beta or Gamma Activity	Weekly	8
	b. Iodine Concentration	Weekly when gross beta or gamma activity ≥ 1.0 $\mu Ci/cc$	8

[(4)]A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during REFUELING outages.

[(5)]And after adjusting tank contents.

[(6)]Sample will be taken monthly when fuel is in the pool.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

a. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the USAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Reporting Requirements

Routine operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to March 1 of each year. Items reported in this category include:

A. Report of facility changes, tests or experiments required pursuant to 10 CFR 50.59(b).

B. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions,[1] e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and REFUELING. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

C. Challenges to and failures of the pressurizer power operated relief valves and safety valves.[2]

D. This report shall document the results of specific activity analysis in which the reactor coolant exceeded the limits of TS 3.1.c.1.A during the past year. The following information shall be included:

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations;

(3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and

(5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

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[1]This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

[2]Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (U.S. NRC) dated January 5, 1981.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

(1) The Annual Radiological Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by TS 7.7.2.

(2) The Annual Radiological Environmental Monitoring Reports shall include the results of analysis of radiological environmental samples and of environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the RADIOLOGICAL ENVIRONMENTAL MONITORING MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report when applicable.

(3) The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; legible maps covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by TS 7.7.3; discussion of all deviations from the sampling schedule of Table 7.3; and discussion of all analyses in which the LLD required by Table 8.5 was not achievable.

2. Semiannual Radioactive Effluent Release Report

A. Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

(1) Radioactive Effluent

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit following the format of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974.

(2) Radiation Dose Assessment

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.[3] This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. The assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed based on the calculational guidance, as presented in the OFF-SITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER(S) OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

(3) Solid Waste Shipped

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped off-site during the report period:

a) Container volume,

b) Total curie quantity (specify whether determined by measurement or estimate),

c) Principal radionuclides (specify whether determined by measurement or estimate),

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[3]In lieu of submission with the second half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

d) Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e) Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f) SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

(4) Unplanned Release

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

(5) PCP and ODCM Changes

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFF-SITE DOSE CALCULATION MANUAL (ODCM).

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.