UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAINE YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-309

MAINE YANKEE ATOMIC POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
License No. DPR-36

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Maine Yankee Atomic Power Company (the licensee) dated November 30, 1981 and April 12, 1982, as revised by a May 28, 1982 submittal and subsequent discussions, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.B(6)(b) of Facility Operating License No. DPR-36 is hereby amended to read as follows:

   (b) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 65, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on November 1, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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Operating Reactors Branch #3
Division of Licensing

Attachments:
Changes to the Technical
Specifications

Date of Issuance: October 28, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 65

TO FACILITY OPERATING LICENSE NO. DPR-36

DOCKET NO. 50-309

Revise Appendix A Technical Specifications as follows:

Remove	Insert
Table of Contents (2 pages)	Table of Contents (2 pages)
Definitions-pages 1 through 6	Definitions-pages 1 through 6
Page 3.0-1	Pages 3.0-1 and 2
3.1-1	3.1-1 and 2
3.2-1 and 2	3.2-1 and 2
3.3-1 through 3.3-3	3.3-1 through 3.3-3
3.4-1 through 3.4-13	3.4-1 through 3.4-11
3.5-1 and 2	3.5-1 and 2
3.6-1 through 3.6-3	3.6-1 through 3.6-3
3.7-1 and 2	3.7-1 and 2
3.8-1 through 3.8-3	3.8-1 through 3.8-4
3.9-1 through 3.9-5	3.9-1 through 3.9-5
3.10-1 through 3.10-8	3.10-1 through 3.10-8
Retain pages 3.10-9 through 3.10-15	
3.11-1 and 2	3.11-1 through 3.11-3
3.12-1 and 2	3.12-1 and 2
3.13-1 through 3	3.13-1 through 3
3.14-1 through 3	3.14-1 through 3
3.15-1	3.15-1
3.16-1 through 4	3.16-1 through 4
Retain pages 3.16-5 through 3.16-7	
3.17-1 through 3.17-9	3.17-1 through 3.17-8
3.18-1	3.18-1
3.19-1 and 2	3.19-1 and 2
3.20-1 and 2	3.20-1
Retain table 3.20-1 (5 pages)	
3.21-1	-
3.23-1 through 4	3.23-1 through 5
3.24-1	3.24-1
3.25-1 and 2	3.25-1 and 2

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Technical Specifications:

REACTOR OPERATING CONDITIONS

Refueling Shutdown Condition (Condition 1)

When the primary coolant is at refueling boron concentration and Tavg is less than 210°F.

Refueling Operations Condition (Condition 2)

Any operation involving movement of core components when the vessel head is unbolted or removed.

Cold Shutdown Condition (Condition 3)

When the primary coolant is at cold shutdown boron concentration, and Tavg is less than 210°F.

Transthermal Condition (Condition 4)

When the reactor is subcritical by 5% delta k/k and Tavg is between 210°F and 500°F inclusive.

Hot Shutdown Condition (Condition 5)

When the reactor is subcritical by 5% delta k/k and Tavg is greater than 500°F.

Hot Standby Condition (Condition 6)

The reactor is considered to be in a hot standby condition if the average temperature of the primary coolant (Tavg) is greater than 500°F and any of the control rods are withdrawn and the neutron flux power range instrumentation indicates less than 2% of the rated power.

Power Operation Condition (Condition 7)

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Higher Operating Condition

Operating condition with greater numerical index, re 1 to 7 above.

REACTOR STATUS

Refueling Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical under all refueling conditions.

Cold Shutdown Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical with all control rods in the core.

Hot Shutdown Boron Concentration

The boron concentration shall be sufficient to maintain the reactor at least 5% delta k/k subcritical with all control rods in the core.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power. The reactor is considered subcritical when it is not critical.

Low Power Physics Testing

Testing performed under approved written procedures to determine control rod worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of rated power, not including decay heat, and primary system temperature and pressure shall be in the range of 260°F to 550°F and 415 psia to 2300 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions in these Technical Specifications.

Power Range Physics Testing

Tests performed under approved written procedures to verify core nuclear design properties at power and plant response characteristics. Reactor power may be greater than 2% during these measurements. Primary system average temperature and pressure shall be in the range of 500°F to 580°F and between 1700 psia to 2300 psia, respectively. Certain deviations from normal operating practices which are necessary to enable the performance of some of these tests are permitted in accordance with specific provisions of these Technical Specifications.

Rated Power

A steady-state reactor core output of 2630 MWt.

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

$$\text{TILT} = \left[\frac{\text{Power in any quad}}{\text{Avg power of all quad}}\right] - 1$$

REACTOR PROTECTIVE SYSTEM

Instrument Channels

One of four independent measurement channels, complete with the. sensors, sensor power supply units, amplifiers, and trip modules provided for each safety parameter.

Reactor Trip

The de-energizing of the magnetic jack holding coils which releases the shutdown and regulating control elements (CEA's) and allows them to drop into the core.

Trip Module

A bistable unit in each of the instrument channels which is tripped when the parameter signal exceeds a specified limit. The relay contact outputs of the trip modules form the reactor protective system logic.

ENGINEERED SAFEGUARDS SYSTEMS

Subsystem

One of two or more redundant grouping of sensors, logic, and circuitry able to bring about automatic or manual initiation of an engineered safeguard.

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

INSTRUMENTATION SURVEILLANCE

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall, where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration/Channel Adjustment

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, interlocks or trip and shall include the channel functional test.

MISCELLANEOUS DEFINITIONS

Operable

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified functions(s). 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 are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Operating

A system or component is operating if it is performing its safeguard or operating functions.

Control Element Assemblies

All full-length shutdown and regulating control element assemblies (CEA's).

Partial-Length Control Element Assemblies

Control element assemblies (CEA) that contain neutron absorbing material only in the lower quarter of their length.

Containment Integrity

Containment integrity is defined to exist when all of the following are true:

a. All non-automatic containment isolation valves and blind flanges are closed.

b. The equipment hatch is properly closed and sealed.

c. At least one hatch in the personnel air lock is properly closed and sealed.

d. All automatic containment isolation valves are operable or are locked closed.

e. The uncontrolled containment leakage satisfies Specification 4.4 Section I.B.3.

Fire Suppression Water System

A fire suppression water system shall consist of: A water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

MISCELLANEOUS DEFINITIONS (Continued)

Radio Isotope Release Limits

The Maine Yankee radio isotope release limits are as defined in Technical Specification 3.16, paragraph A, item 2, for liquid releases and Technical Specification 3.17, paragraph A, item 2 for gaseous releases.

$\overline{E}$ - Average Disintegration Energy

$\overline{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in Mev) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Dose Equivalent I-131

Dose Equivalent I-131 is determined as that concentration of I-131 (micro Ci/gm) 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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

Reportable Occurrence

A reportable occurrence is defined in Section 5.9 of these specifications.

Remedial Action

Remedial Action is that part of a specification which prescribes corrective and/or compensatory measures required under designated conditions.

Noncompliance

Noncompliance with a Limiting Condition for Operation shall exist when neither the requirements of the Limiting Condition for Operation nor the associated Remedial Action (if any) are met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Remedial Action requirements is not required.

Nonconformance

Nonconformance with a specification shall exist when the requirements of the Limiting Condition for Operation are not met without reliance upon Remedial Action statements.

Frequency Notation

The frequency notation specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 0.1.

TABLE 0.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days if the plant is in the cold shutdown condition
SA	At least once per 6 months
A	At least once per year
R	At least once per 18 months
P	Prior to each reactor startup
N.A.	Not applicable

3.0 LIMITING CONDITIONS FOR OPERATIONS

Applicability:

Applies to section 3 of these Technical Specifications.

Objective:

To specify general regulatory requirements for compliance with these specifications and appropriate remedial actions when compliance cannot be attained.

Specification:

A. Nonconformance with a Limiting Condition for Operation:

If a Limiting Condition for Operation (LCO) in Section 3 of the Technical Specifications is not met, the following sequential remedial actions must be taken until conformance with the specification is achieved.

1. perform any remedial action permitted by the applied specification

   * See Note in Basis
2. commence a reactor shutdown within one hour and place the plant in a Hot Shutdown Condition within 6 hours after the discovery of the nonconforming condition or after any time period permitted by (1) above.
3. commence a reactor cooldown and place the plant in a Cold Shutdown Condition within 30 hours after the discovery of the nonconforming condition or after any time period permitted by (1) above.

B. Entry into a Higher Operating Condition:

Entry into a Higher Operating Condition shall not be made whenever the following exists:

1. The provisions of A.2 or A.3 above apply.

2. Any of the following specific LCO's is not met for the existing or higher condition without reliance upon the provisions contained in the remedial action statements:

   3.5-C; 3.6-A; 3.6-B; 3.9-A; 3.14-C

C. Operability of safety related components with emergency power sources:

If a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided:

1. its corresponding normal or emergency power source is operable;

and

2. all of its redundant system(s), train(s), component(s) and device(s) are operable, or likewise satisfy the requirements of this specification.

Exception: This specification is not applicable in the Cold Shutdown, Refueling Operations or Refueling Shutdown Condition.

Remedial Action: Unless both conditions C (1) and (2) are satisfied follow the actions specified in A above.

Basis:

Specification A assures compliance with 10 CFR 50.36 which states: "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification B assures that entry into a higher operating condition will not be made during periods of noncompliance when a plant shutdown or cooldown is required, or when designated specifications are not met without reliance upon any time permitted by the remedial action statements.

Specification C delineates additional conditions that must be satisfied to permit operation to continue by systems, subsystem, trains, components or devices required by these LCO's. It specifically prohibits operation when one ECCS train is inoperable because its normal or emergency power source is inoperable and a system, subsystem, component or device in the other train is inoperable for another reason.

*Note:

10 CFR 50.72 requires a licensee to notify the NRC Operations Center within one hour by telephone of the occurrence of any event requiring initiation of shutdown of a nuclear power plant in accordance with Technical Specification limiting conditions for operation. The Emergency Notification System should be used if it is available.

3.1 REACTOR CORE INSTRUMENTATION

Applicability:
Applies to the calibration of the ex-core symmetric offset protection system and the operability of the in-core instrumentation system.

Objective:
To specify the functional requirements which must be satisfied for the in-core instrumentation system to be considered operable as required for calibrating the ex-core symmetric offset protection system and for other purposes as required by Technical Specification 3.10.

Specification:

A. The ex-core symmetric offset protection system shall be re-calibrated monthly, utilizing the in-core instrumentation system whenever reactor power level is greater than 90% of the maximum power for 2 or 3 loop operation.

Remedial Action:

Power shall be limited to 90% of maximum power for 2 or 3 loop operation (whichever applies) if re-calibration of the ex-core symmetric offset protection system has not been accomplished within the previous 30 days.

B. For the in-core instrumentation system to be considered operable to meet the above specification or to meet the requirements of Specification 3.10:

1. At least 75% of all in-core detector octant positions shall be represented, and

2. a mimimum of 2 in-core detector locations per core quadrant shall be operable.

Note: An operable in-core detector octant position shall consist of a position with a minimum of three operable fixed detectors or where a moveable detector trace can be taken.

Basis:
The in-core detector system uses 45 radial locations throughout the core. A number of these locations have additional provision for moveable detectors, while the remainder have strings of fixed self powered detectors. This instrumentation can be used to determine the power balance between the top and bottom halves of the core in each of these locations. Moreover, a fixed detector string would still provide adequate capability with only three of its four rhodium detectors functioning. Thus the full system has more capability than would be needed for the calibration of the ex-core detectors.

After the ex-core system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due to fuel depletion and for changes in the detectors.

If the re-calibration is not performed, the mandated power reduction assures safe operation of the reactor since it will compensate for an error up to 10% in the ex-core detector system. Experience at Connecticut Yankee has shown that drift due to changes in the core or instrument channels is very slight. Thus limiting the operating levels to 90% of the maximum two or three loop power levels is very conservative for both operational modes.

3.2 REACTOR COOLANT SYSTEM ACTIVITY

Applicability:

Applies to measured maximum activity in the reactor coolant system.

Objective:

To ensure that the reactor coolant activity does not exceed a level commensurate with the safety of the plant personnel and the public.

Specification:

A. The specific activity of the primary coolant shall be limited to less than or equal to 1.0 micro Ci/gram DOSE EQUIVALENT I-131.

Remedial Action

1. If the specific activity of the primary coolant is greater than 1.0 micro Ci/gram Dose Equivalent I-131 for more than 800 cumulative hours in any period of 12 consecutive months, the reactor must be made subcritical within 48 hours.

2. If the specific activity of the primary coolant is greater than 1.0 micro Ci/gram Dose Equivalent I-131 for more than 500 cumulative hours in any period of 6 consecutive months, a report must be sent to the Commission within 30 days indicating the number of hours above this limit.

3. If the specific activity of the primary coolant is greater than 1.0 micro Ci/gram Dose Equivalent I-131 for more than 48 continuous hours or greater than 60 micro Ci/gram Dose Equivalent I-131 the reactor must be made subcritical with Tavg less than 500°F within the next 6 hours.

B. The specific activity of the primary coolant shall be limited to less than or equal to $100/\bar{E}$ micro Ci/gram.

Remedial Action: If the specific activity of the primary coolant is greater than $100/\bar{E}$ micro Ci/gram, the reactor must be made subcritical with Tavg less than 500°F within 6 hours.

C. With the specific activity of the primary coolant greater than 1.0 micro Ci/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ micro Ci/gram, the sampling and analysis requirements of Item 1 of Table 4.2-1 shall be performed until the specific activity of the primary coolant is restored to within its limits. A report shall be prepared and submitted to the Commission within 30 days, and shall contain the results of the specific activity analysis plus the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 micro Ci/gram DOSE EQUIVALENT I-131.

BASIS:

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the Part 100 limit following a steam generator tube rupture.

3.3 REACTOR COOLANT SYSTEM OPERATIONAL COMPONENTS

Applicability:

Applies to the operating status of the reactor coolant system equipment.

Objective:

To specify conditions of reactor coolant system components for reactor operation.

Specification:

A. Reactor Coolant Pumps

1. At least one reactor coolant pump or one low pressure safety injection pump operating in the residual heat removal mode shall be in operation providing flow through the reactor when the reactor coolant system boron concentration is being reduced.

2. At least one reactor coolant pump shall be in operation providing flow through the core with its steam generator capable of performing its heat transfer function whenever the reactor is in a critical condition. A second loop shall be maintained operable to perform its heat transfer function should the operating loop become inoperable.

3. At least three reactor coolant pumps shall be in operation providing flow through the core with their steam generators performing their heat transfer function whenever the reactor is in a power operation condition.

Exception: The requirement of 2 and 3 may be modified during initial testing to permit power levels not to exceed 10% of rated power with three loops operating on natural circulation.

B. Pressurizer Safety and Relief Valves

1. At least one pressurizer code safety valve shall be operable whenever fuel is in the reactor and the reactor coolant system is isolated from the residual heat removal system and the head is on the vessel.

2. At least two pressurizer code safety valves shall be operable whenever the reactor is critical.

3. One power operated relief valve (PORV) and its associated block valve shall be operable whenever the reactor coolant system temperature is greater than 210°F.

4. In the event either PORV or its associated block valve becomes inoperable, within six hours: either restore the PORV or block valve to operable status or close and remove power from the associated block valve.

C. Pressurizer

1. The pressurizer shall be operable with at least one bank of proportional heaters and a water level between 28 and 60 percent during normal system operation whenever the reactor coolant system Tavg is greater than 500°F.

2. The pressurizer spray system must be lined up to provide continuous pressurizer spray flow whenever the reactor is critical.

Basis:

Reactor coolant pump flow and steam generator heat transfer capabilities are specified to assure adequate core heat transfer capability under all operating conditions from criticality to full power. Three loop operation is specified to assure plant operation is restricted to conditions considered in the LOCA analyses.

The exception permits testing to determine decay heat removal capabilities of the primary system prior to higher power operation while on natural circulation.

Following a loss of offsite power, stored and decay heat from the reactor would normally be removed by natural circulation using the steam generators as the heat sink. Water supply to the steam generators is maintained by the auxiliary feedwater system. Natural circulation cooling of the primary system requires the use of the pressurizer heaters or high pressure safety injection pumps to maintain a suitable overpressure on the reactor coolant system. Alternatively, in the event that natural circulation in the reactor coolant system is interrupted, the feed and bleed mode of reactor coolant system operation can be used to remove decay heat from the reactor. This method of decay heat removal requires the use of the emergency core cooling system (ECCS) and the power-operated relief valves (PORV's) in the pressurizer.

The PORVs can be operated either manually or automatically in the Maine Yankee design. Block valves are provided upstream of the relief valves to isolate the valve in the event that a PORV valve fails.

When reactor coolant boron concentration is being reduced, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at a lower boron concentration which could result in a reactivity insertion.

Sufficient mixing of the reactor coolant is assured by one low pressure safety injection (LPSI) pump operating in the RHR mode. When operated in this mode it will circulate the reactor coolant system volume in less than 12 minutes. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. A continuous pressurizer spray flow will maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron. Without residual heat removal, the amount of steam which could be generated at safety valve lift pressure with the reactor subcritical would be less than half of one valve's capacity. One valve, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Overpressure protection is provided for all critical conditions. The safety valves are sized to relieve steam at a rate equivalent to the peak volumetric pressure surge rate. For this purpose one safety valve is sufficient; however, a minimum of two safety valves is required by Section III of the ASME Code.

3.4 COMBINED HEATUP, COOLDOWN AND PRESSURE-TEMPERATURE LIMITATIONS

Applicability:

Applies to temperature and pressure conditions during heatup and cooldown of the reactor coolant system.

Objective:

To maintain operational limits within design boundaries of the reactor coolant system.

Specification:

A. Reactor Coolant System

1. The reactor coolant system shall be operated within the limits set forth in Table 3.4-1 and the pressure-temperature limits derived from (2) below.

   Remedial Action: If the reactor coolant system is subject to conditions outside of the above limits the reactor shall be brought subcritical and an engineering analysis of the consequences shall be made prior to restoration of power operation.

2. The pressure-temperature limits for reactor coolant system operation shall be revised at each refueling using the following procedure.

   a. The pressure-temperature limits for reactor coolant system operation shall be as developed by superimposing fluence-dependent heatup and cooldown limits into the basic ASME Section 3 limits of operation (Figure 3.4-1). At each refueling the heatup and cooldown limits will be modified to account for material property changes in the reactor vessel projected through the next core cycle in accordance with the following procedure:

      1. Project the cumulative MWH(t) on the vessel through the next core cycle.

      2. Determine the associated fluence to the vessel from Figure 3.4-2.

      3. Determine the shift in RT_{NDT} at the 1/4t and 3/4t from Figure 3.4-3.

      4. The beginning of life heatup and cooldown limit lines in Figures 3.4-4 through 3.4-7 shall be shifted parallel to the temperature axis (horizontal) in the direction of increasing temperature, a distance equivalent to the shift in RT_{NDT} at the 1/4t and 3/4t as applicable.

The following table provides the shift parameter to be applied:

CURVE	SHIFT PARAMETER
Heatup, upper limit	1/4t
Heatup, all other rate limits	3/4t
Cooldown, all limits	1/4t

5. Superimpose the shifted Figures 3.4-4 through 3.4-7 onto Figure 3.4-1 to provide the appropriate operational limits for heatup and cooldown during normal and hydrostatic test operations.

B. Reactor Core

1. The reactor shall not be critical if the reactor coolant pressure is less than 400 psig or greater than 2400 psig.

2. The reactor shall not be critical (other than for the purposes of low power physics tests) if the temperature of the reactor coolant is:

   a. less than 111°F plus the shift in RT_{NDT} at the 1/4t (as determined in A.2.a.3), or

   b. within 40°F or less of the applicable heatup curve (as determined in A.2.a.4), or

3. The reactor shall not be critical without a steam bubble in the pressurizer.

4. The reactor shall not be critical during inservice leak or hydrostatic testing of the reactor coolant system.

C. Residual Heat Removal System

1. The residual heat removal system (RHRS) must be isolated whenever the reactor coolant system pressure exceeds 600 psig or the temperature exceeds 450°F.

D. Reactor Coolant System Low Temperature Overpressure Protection

1. The two power operated relief valves, aligned for the low pressure set point, and the two RHR spring relief valves shall be operable for RCS overpressure protection whenever the RCS is less than the minimum pressurization temperature and the RCS is not vented.

   Remedial Action: With one power operated relief valve or RHR spring relief valve inoperable, restore the relief valve to operable status within 7 days or depressurize and vent the RCS within the next 8 hours.

2. No more than one HPSI pump may be energized at RCS temperature below 220°F.

Exception: A second HPSI pump may be energized for up to 5 minutes for the purpose of rotating operating equipment.

3. Reactor Coolant Pumps may be started (or jogged) only if there is a steam bubble in the pressurizer with a maximum level of 80% or the steam generator temperature is less than 100°F above the reactor coolant temperature.

Basis:

The heatup and cooldown limit curves (Figures 3.4-4 through 3.4-7) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. Linear interpolation is permissible. The heatup and cooldown curves were prepared based on the beginning of life RT_{NDT} at the reactor vessel, and include adjustments for possible errors in the pressure and temperature sensing instruments.

The reactor vessel materials opposite the core have been tested to Appendix G of 10CFR50 to determine their RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in RT_{NDT}. As a result of irradiation tests of actual vessel materials, the shift in RT_{NDT} can be determined at the critical 1/4t and 3/4t locations from Figure 3.4-4. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area.

The pressure-temperature limit lines shown on Figures 3.4-4 through 3.4-7 for normal operation and inservice leak/hydrostatic testing, as well as the limits on criticality have been provided to assure compliance with the requirements of Appendix G to 10CFR50. The maximum NDTT for all reactor coolant system pressure retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit line shown on Figure 3.4-1 is based upon this NDTT since Article NB-2322 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code, requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. In addition, a 60°F margin is added to this for conservatism. Below this temperature, the system pressure must be limited to a maximum of 25% of this system's design pressure of 2485 psig.

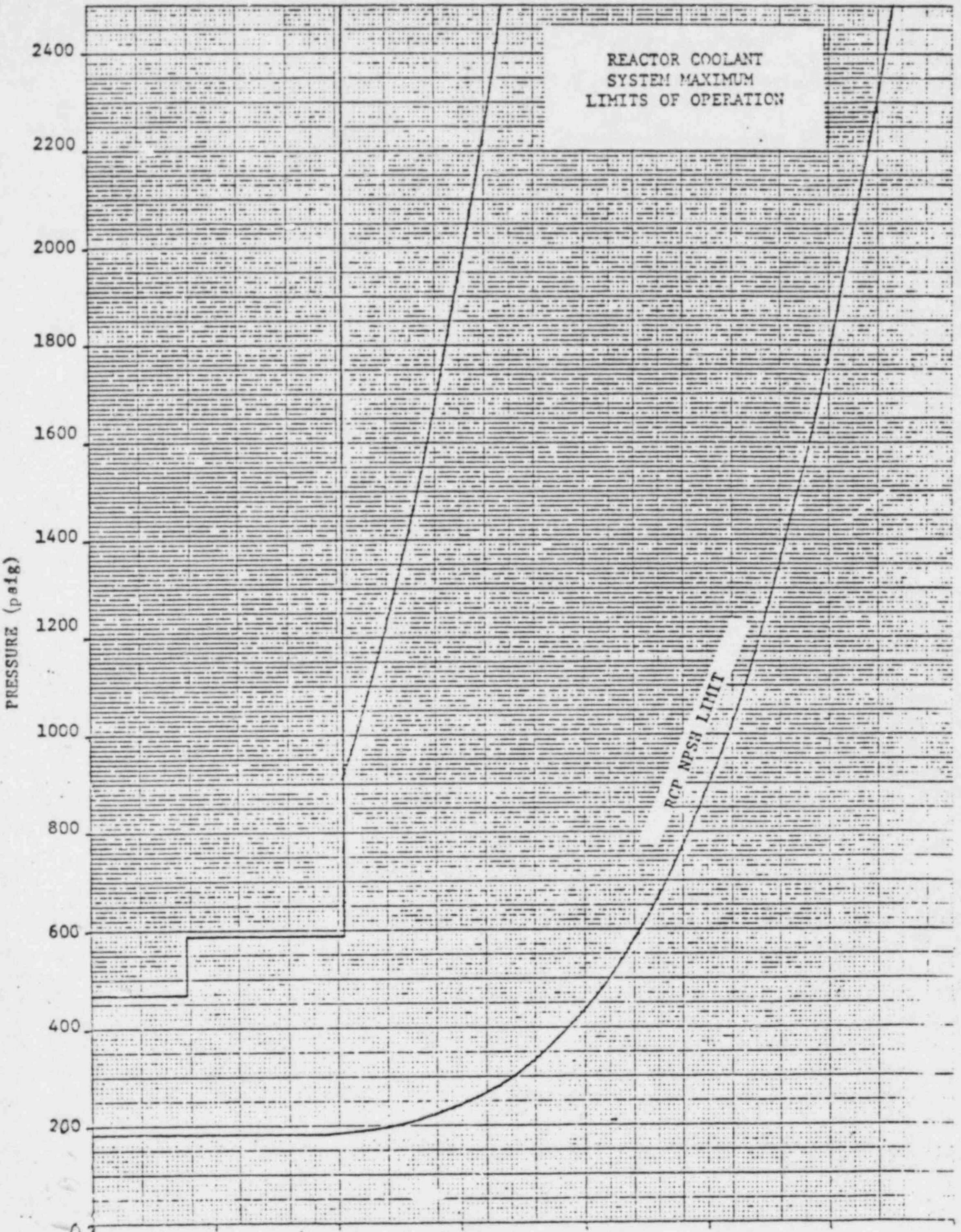
The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE 3.4.1

LIMITS OF OPERATION FOR THE REACTOR COOLANT SYSTEM

Limit	Reactor Vessel	Pressurizer	STEAM GENERATOR	
			Primary Side	Secondary Side
Maximum Heatup Rate ($^{\circ}$F in any one hour period)	100	100	100	--
Maximum Cooldown Rate ($^{\circ}$F in any one hour period)	100	200	100	--
Minimum Pressurization Temperature ($^{\circ}$F)	200	70	70	100
Maximum Pressure Below Minimum Pressurization Temp (psig)	621	500	500	230
Maximum Temperature Difference Between Operating Loops ($^{\circ}$F)	--	340	--	--

FIGURE 3.4-1

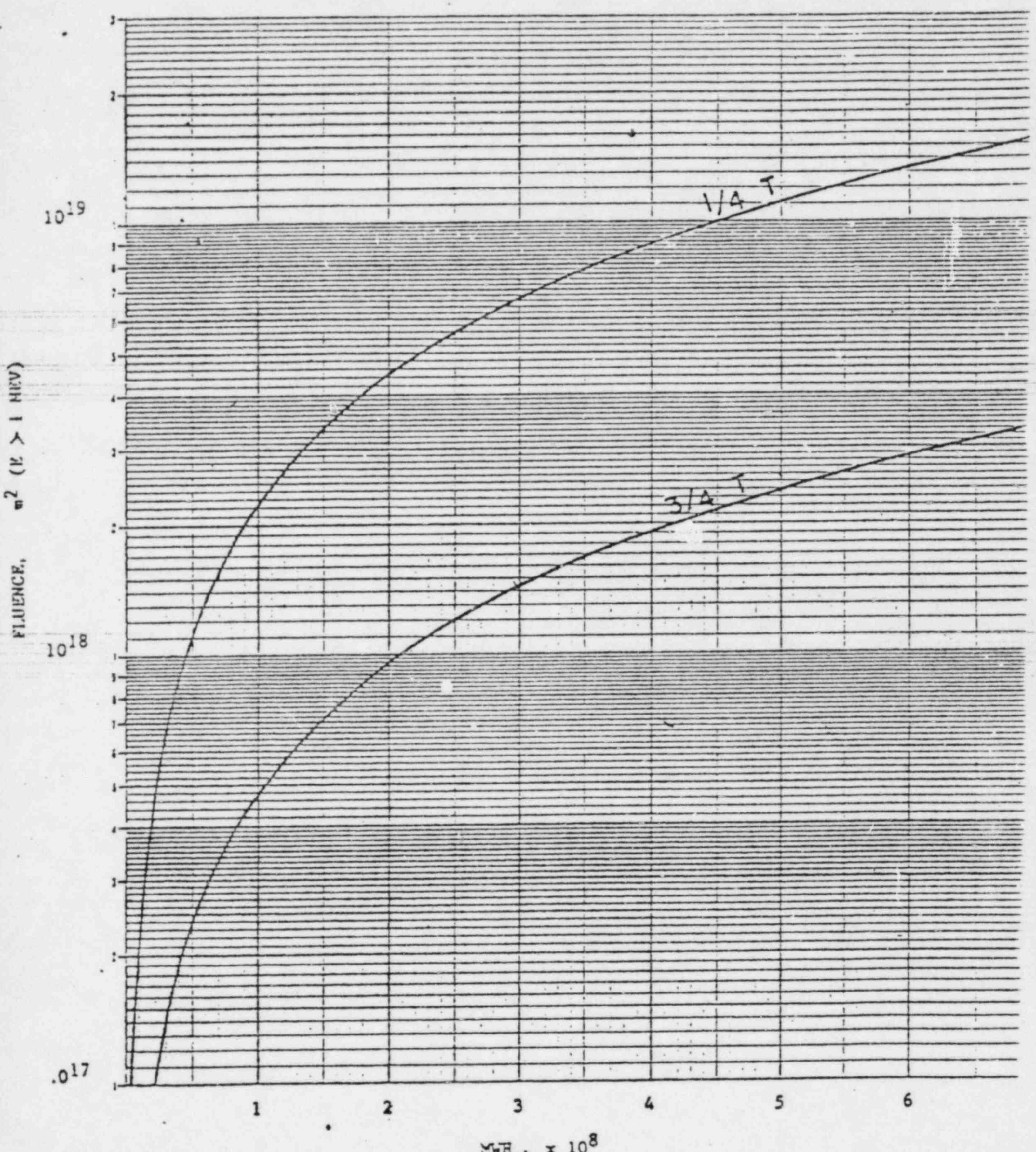


REACTOR COOLANT
SYSTEM MAXIMUM
LIMITS OF OPERATION
RCP NPSH LIMIT
PRESSURE (psig)
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
100
200
300
400
500
600
700
TEMPERATURE (°F)

Figure 3.4-2

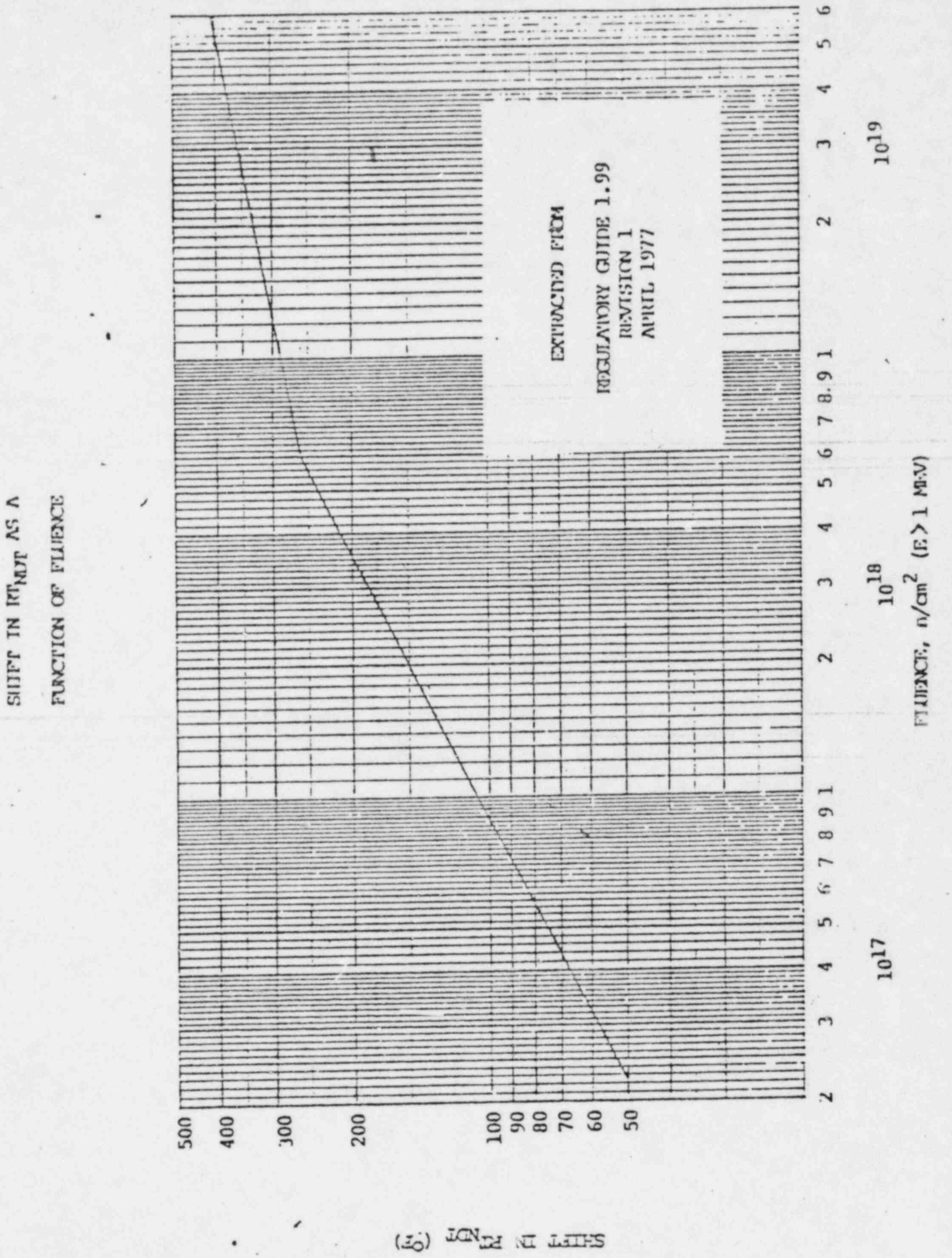
FLUENCE VS. BURNUP FOR 1/4T AND 3/4T POSITIONS

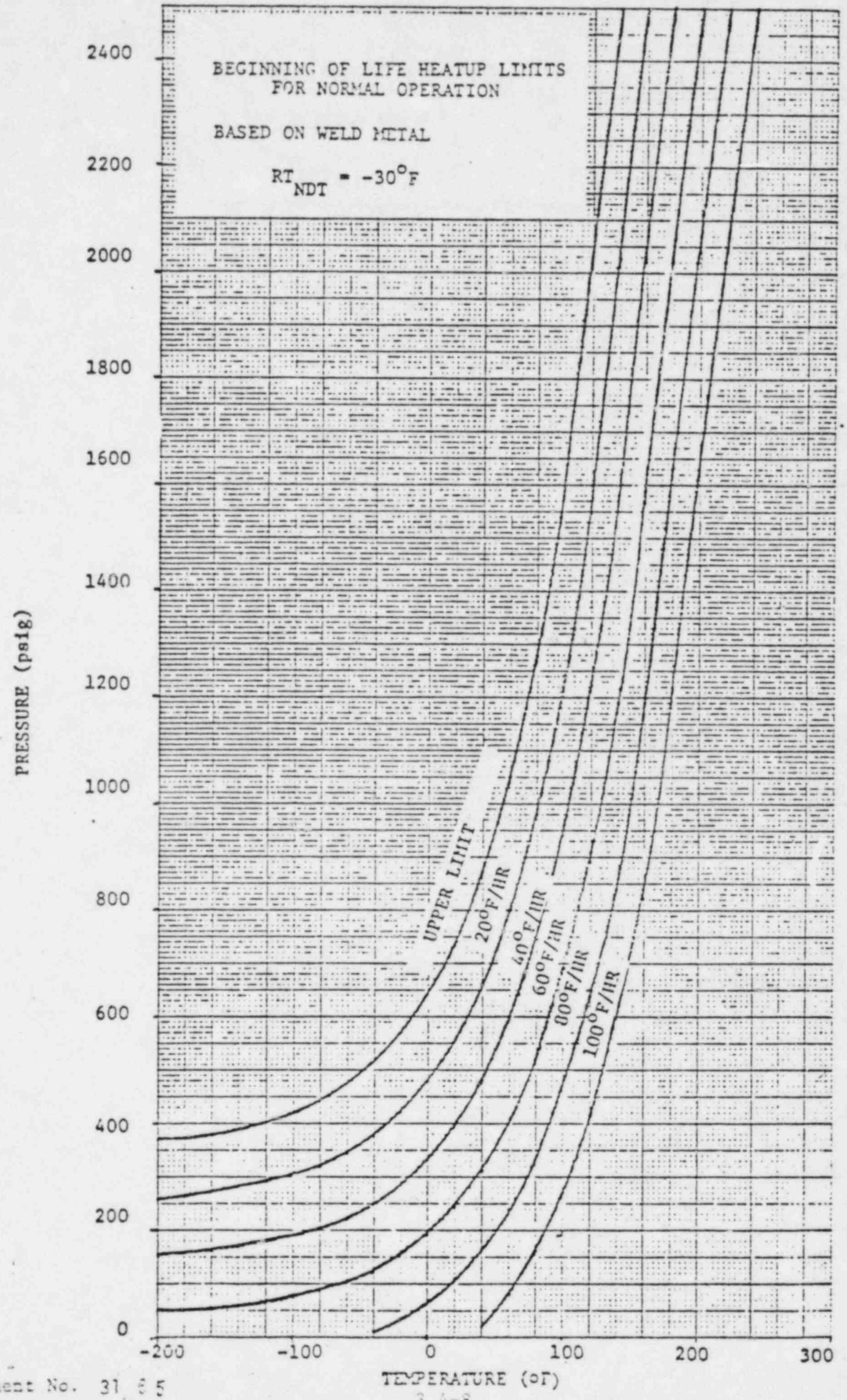
FOR MAINE YANKEE



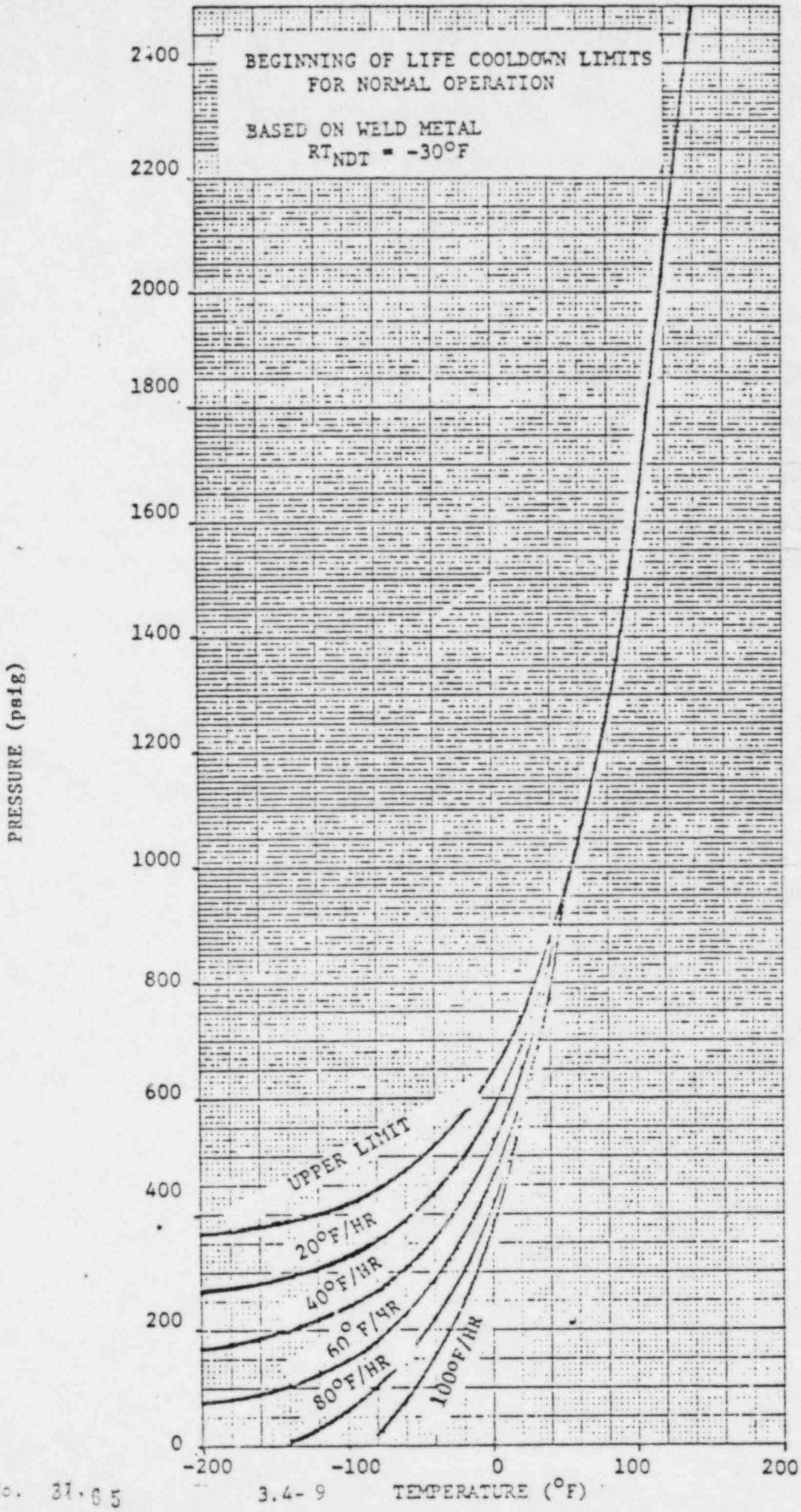
Amendment No. 27 58.5 F

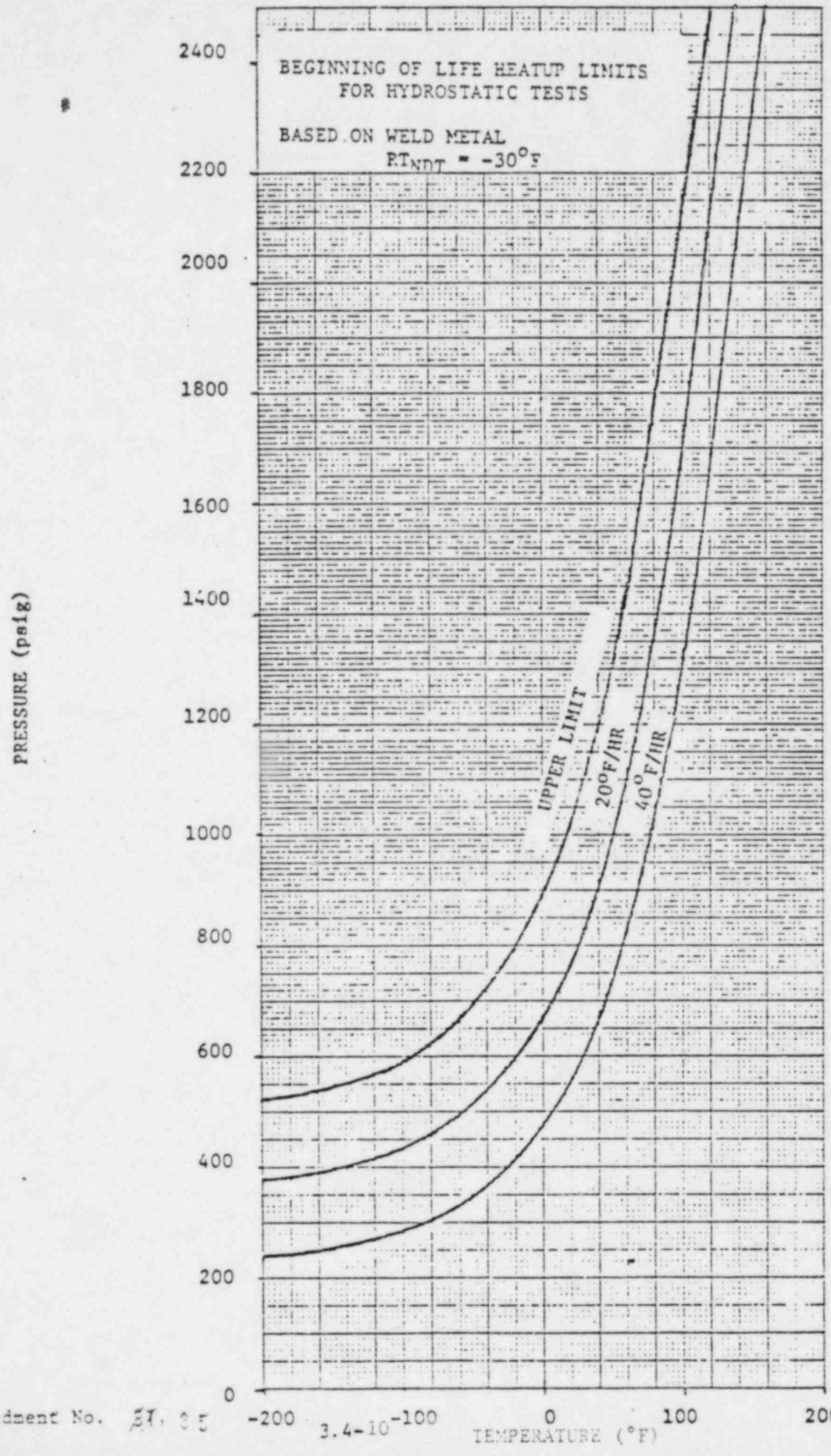
Figure 3,4-3

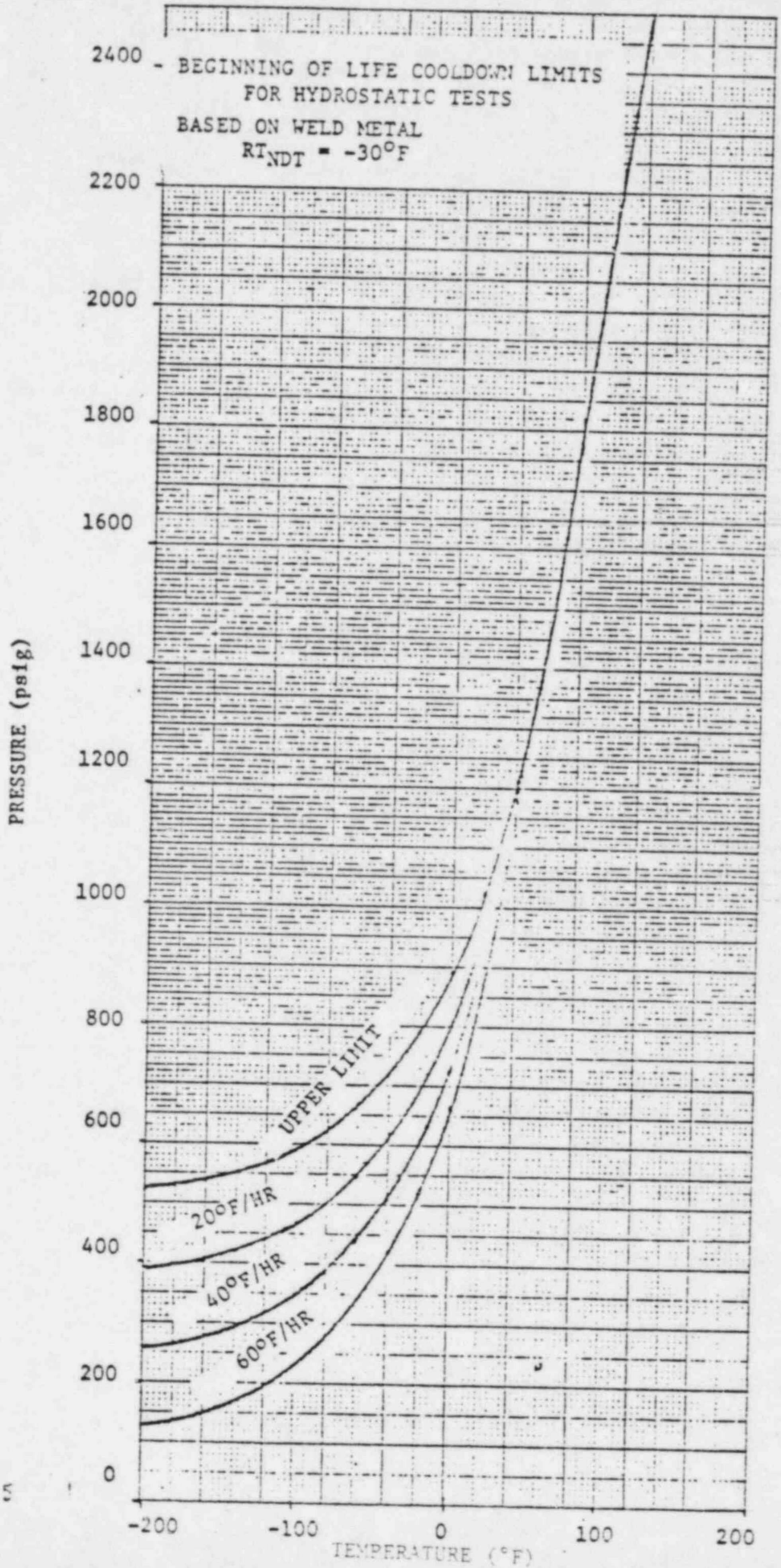

SHIFT IN RT_NDT AS A
FUNCTION OF FLUENCE
EXTRACTED FROM
REGULATORY GUIDE 1.99
REVISION 1
APRIL 1977
SHIFT IN RT_NDT (°F)
500
400
300
200
100
90
80
70
60
50
FLUENCE, n/cm² (E > 1 MeV)
10^17
10^18
10^19



BEGINNING OF LIFE HEATUP LIMITS FOR NORMAL OPERATION
BASED ON WELD METAL
RT NDT = -30°F
UPPER LIMIT
20°F/HR
40°F/HR
60°F/HR
80°F/HR
100°F/HR
PRESSURE (psig)
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
-200
-100
0
100
200
300
TEMPERATURE (°F)


BEGINNING OF LIFE COOLDOWN LIMITS
FOR NORMAL OPERATION
BASED ON WELD METAL
RT_{NDT} = -30°F
PRESSURE (psig)
TEMPERATURE (°F)
UPPER LIMIT
20°F/HR
40°F/HR
60°F/HR
80°F/HR
100°F/HR
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
-200
-100
0
100
200


BEGINNING OF LIFE HEATUP LIMITS
FOR HYDROSTATIC TESTS
BASED ON WELD METAL
RT_{NDT} = -30°F
UPPER LIMIT
20°F/HR
40°F/HR
PRESSURE (psig)
TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
-200
-100
0
100
200


BEGINNING OF LIFE COOLDOWN LIMITS
FOR HYDROSTATIC TESTS
BASED ON WELD METAL
$RT_{NDT} = -30^\circ F$
UPPER LIMIT
20°F/HR
40°F/HR
60°F/HR
PRESSURE (psig)
TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
-200
-100
0
100
200

3.5 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability:

Applies to the operational status of the chemical and volume control system when there is fuel in the reactor.

Objective:

To specify those limiting conditions for operation of the chemical and volume control system which must be met in order to ensure adequate boration capability is available.

Specification:

A. Whenever there is fuel in the reactor the boric acid storage tank or the refueling water storage tank shall contain sufficient boric acid solution to bring the Reactor Coolant System to the cold shutdown boron concentration. Solution temperatures shall be maintained at least 10°F above the concentration saturation temperature but not less than 40°F.

B. Whenever there is fuel in the reactor there shall be at least one operable path for boron injection consisting of system pumps, piping, heat tracing, valves, instrumentation and controls operable as to assure the capability of boron injection at a rate in excess of 250,000 ppm-gals/min. into the reactor coolant system.

Remedial Action: If no adequate operable boron injection path exists; no changes shall be made that may insert positive reactivity. An operable boron injection path must be established within 6 hours or the Commission shall be notified.

C. Whenever the reactor is critical there shall be at least two independent operable paths each meeting the requirements of B above.

Exception: The requirements may be modified during operation with an isolated loop to permit operation with one operable flow path for a period not to exceed 24 hours.

Remedial Action: With the reactor critical and only one of the boron injection flow paths required in C above OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours.

Basis:

The chemical and volume control system provides control of the reactor coolant boron inventory. Reduction of concentration is accomplished by dilution with unborated primary grade water or by boron removal through ion exchange. An increase in concentration may be accomplished by using either of the two operable charging pumps which have separate suction lines from the refueling water storage tank. An increase may also be accomplished using the auxiliary charging pump taking suction from the boric acid storage tank. Each of the three operable pumps can be lined up to discharge into the reactor coolant system through a separate flow path. Thus there are three operable flow paths normally available during operation. However, during periods of two loop operation, the loop fill header and the auxiliary charging pump may not be available, reducing the number of available flow paths to two. The exception provides time to restore redundancy should one flow path become inoperative. The rate specified is adequate to bring the reactor to a cold shutdown condition. It precludes the possiblity of the lower capacity auxiliary charging pump and the lower concentration refueling water storage tank being taken together as an available flow path. The allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

3.6 EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS

Applicability:

Applies to the operating status of the emergency core cooling and containment spray systems.

Objective:

To define the conditions under which components of the emergency core cooling and containment spray systems must be operable.

Specification:

A. The following equipment must be operable whenever the reactor coolant system temperature and pressure exceed 210°F and 400 psig:

1. Two safety injection tanks set for automatic initiation. Each tank shall contain 11,200 + 500 gallons of water borated to at least 1720 ppm and pressurized with nitrogen to 230 psig + 10 psi, - 25 psi.

2. One operable ECCS train consisting of the following subsystems of the train. Each subsystem includes the manual valves that are aligned and locked in the position required for safeguards operation, the automatically operated valves set for automatic operation or aligned and locked in the position required for safeguards operation, the controls set for automatic initiation where appropriate, and a pump powered from an engineered safeguards bus.

   a. One service water pump subsystem

   b. One component cooling pump subsystem

   c. One low pressure safety injection pump subsystem

   d. One high pressure safety injection pump subsystem

   f. One containment spray pump and RHR heat exchanger subsystem

3. Station service power in accordance with Technical Specification 3.12.A supplying the same operable ECCS train as in (2) above.

4. The refueling water storage tank and spray chemical addition tank are filled and available in accordance with Technical Specification 3.7.

5. The fill header motor operated root valves to two non-isolated loops.

Exception: The requirements may be modified with regard to the position of controls and valves during periods of hydrostatic testing.

Remedial Action: Restore required limiting condition within four hours.

B. Whenever the reactor coolant system boron concentration is less than that required for Hot Shutdown condition, two high pressure safety injection pump subsystems shall be operable.

   Remedial Action: If any of the component subsystems specified in B above becomes inoperable, the operable component subsystem performing the same function in the other train and its associated diesel generator shall be tested within two hours and the inoperable system must be restored to operable status within 72 hours of the discovery of the nonconforming condition.

C. The following equipment must be operable whenever the reactor is in a power operation condition.

   1. Three safety injection tanks set for automatic initiation and subject to the conditions specified in A.1 above.

   2. Two operable and redundant ECCS trains, each train consisting of the subsystems specified in A.2 above.

   3. Station service power in accordance with Technical Specification 3.12.B.

   4. The refueling water storage tank and the spray chemical addition tank filled and available in accordance with Technical Specification 3.7.

   5. The fill header motor operated root valves to three non-isolated loops.

   Exceptions:

   1. One safety injection tank may be isolated for a period not to exceed one hour.

   Remedial Actions:

   1. If any of the component subsystems specified in C.2 above becomes inoperable, the operable component subsystem performing the same function in the other train and its associated diesel generator shall be tested within two hours and the inoperable system must be restored to operable status within 72 hours of the discovery of the nonconforming condition.

   2. If any of the fill header motor operated root valves become inoperable both of the other root valves shall be tested operable within two hours and the inoperable valves shall be restored to operable status within 72 hours of the discovery of the nonconforming condition.

   3. If one of the safety injection tanks is found not to be within specifications it shall be restored to specification within four hours.

Basis:

Adequate core cooling and containment spray is provided for the entire break spectrum up to and including the design basis accident. This protection covers all modes of operation from shutdown to full power.

At full power minimum required safety injection includes three (3) operable safety injection tanks, and two complete ECCS trains consisting of the subsystems specified in A.2. The accident analysis considers that only 2/3 of the capacity of the operable equipment is effective for core cooling.

Containment peak accident pressure is maintained below design pressure and subsequent containment cooling requirements are adequate if one of the two containment spray pumps is operable.

Specification A provides a pressure and temperature limit above which ECCS must be operable. It recognizes the greatly decreased probability of a loss of coolant accident and the negligible amount of energy stored in the primary coolant.

Specification B ensures that a sufficient quantity of boron can be injected by the ECCS to maintain the reactor subcritical following the most limiting main steam line break accident with the concurrent failures of the highest worth CEA stuck out of the core and the failure of one ECCS train to function.

3.7 BORON AND SODIUM HYDROXIDE AVAILABLE FOR THE CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the concentration and volume inventory of borated water and spray chemical water.

Objective:

To ensure the availability of borated water for boron injection, core cooling and containment spray and the availability of sodium hydroxide solution for iodine absorption.

Specification:

A. Whenever the core cooling or containment spray systems are specified to be operable, the refueling water storage tank shall contain not less than 300,000 gallons, and shall have a boron concentration of between 1720 and 1900 ppm, the spray chemical addition tank shall contain not less than 15,400 gallons of sodium hydroxide solution at a concentration of between 8 and 11 percent. Solution temperature shall be maintained at least 10°F above the concentration saturation temperature but not less than 40°F.

B. Whenever the core cooling or containment spray systems are specified to be operable, the total boron available for mixing in the containment sump shall be limited according to the following equation:

$\frac{C_1M_1+C_2M_2+C_3M_3}{M_1+M_2+M_3+M_4}$ is less than or equal to 1890 ppm

where C_1 = boron concentration in refueling water storage tank, ppm

C_2 = boron concentration in reactor coolant system, ppm

C_3 = average boron concentration in safety injection tanks, ppm

M_1 = minimum mass of liquid transferred from refueling water storage tank = 1.67×10^6 lbs

M_2 = mass of liquid in reactor coolant system = 4.7×10^5 lbs

M_3 = mass of liquid in safety injection tanks, lbs

M_4 = minimum mass of liquid transferred from spray chemical tank = 7.5×10^4 lbs.

C. Whenever the refueling water storage tank is specified to be operable for boron injection it shall contain sufficient water, at a minimum boron concentration of not less than 1720 ppm, to bring the reactor coolant system to a cold shutdown boron concentration.

Remedial Action:

If A, B, or C are not met:

1. Restore the required minimum tank volumes and temperatures within four hours.

2. Restore the concentration to within ± 10% of required within four hours and to the value required within twenty four hours.

Basis:

The 300,000 gallons in the refueling water storage tank is based on allowing a minimum of 200,000 gallons to be transferred to the containment via spray and core cooling before recirculation is manually established. Automatic transfer to recirculation will occur after at least 200,000 gallons has been transferred from the tank leaving a minimum of 100,000 gallons which will insure adequate NPSH requirements for the engineered safeguards pumps.

The concentration of 1720 ppm is the highest value used in any of the safety analyses. By specifying this concentration the safety of the plant shown in Section 14 of the FSAR is assured. Analysis of loss-of-coolant incidents shows that 200,000 gallons will be sufficient to limit core temperatures and containment pressure for the full spectrum of pipe ruptures. These analyses are discussed in Section 14.14 of the FSAR.

The 15,400 gallons of sodium hydroxide solution is based on hydrostatically balancing a full refueling water storage tank.

The minimum and maximum sodium hydroxide and boron concentrations are based on maintaining the pH of the initial spray solution, the sump water at the start of recirculation, between 8.5 and 11. This will assure that the containment spray system will effectively remove iodine from the containment atmosphere.

The twenty-four hour grace period is necessary in order to allow time to adjust RWST boron concentration in a controlled manner. This involves initiating tank recirculation to thoroughly mix the tank to provide assurance that the samples obtained during the adjustment process are representative of the tank contents. Reliable sample data provides assurance against non-conformance due to over adjustment. It is unlikely that the boron concentration would be found to be appreciably below 1720 ppm or above 1900 ppm and it is therefore improbable that at any point in core lifetime the concentration would vary from that actually necessary.

3.8 REACTOR CORE ENERGY REMOVAL

Applicability:
Applies to the operating status of plant components for removal of reactor core energy.

Objective:
To specify conditions of the plant equipment necessary to ensure the capability to remove energy from the reactor core.

Specification:

A. Whenever there is fuel in the reactor, at least one of the following cooling mechanisms shall be in operation with a second mechanism operable:

1. RHR Train A
2. RHR Train B
3. Steam Generator No. 1
4. Steam Generator No. 2
5. Steam Generator No. 3
6. A minimum of 23 feet of water above the top of the core with the reactor head removed.

Remedial Action:

1. With only one operable cooling mechanism restore a second mechanism to operation within 72 hours or suspend all operations involving positive reactivity changes.
2. With no cooling mechanisms operable, suspend all operations involving positive reactivity changes, continuously monitor reactor coolant temperature, and restore one cooling mechanism to operation within 6 hours or notify the NRC (using the Emergency Notification System) within the next hour of plans to restore decay heat removal.

Exceptions:

1. The RHRS may be secured for a period not to exceed six hours to facilitate special maintenance, refueling functions or tests. During such periods reactor coolant temperatures shall be continuously monitored and initiation of core cooling shall be continuously available.
2. For purposes of inservice inspection testing, the RHRS may be secured provided that reactor coolant temperature is continuously monitored and two cooling mechanisms are continuously available.

B. The following conditions must be met for a steam generator to be considered operable for decay heat removal.

1. The reactor coolant system must be closed and pressurized to 100 psi above saturation pressure.

2. The steam generator must have both the cold and hot leg stop valves fully open.

3. The steam generator water level must be above the top of the tube bundle.

4. An inventory of over 100,000 gallons of primary grade feedwater must be available.

5. A feed pump must be operable.

C. The steam generators shall be demonstrated operable in accordance with specification 4.10 before the reactor coolant system T. Ave. can be increased above 210°F.

D. The reactor shall not be in a power operation condition which generates steam at a rate in excess of the on-line steam generator relieving capacity in accordance with figure 3.8-1.

E. The reactor shall not be maintained in a power operation condition unless the following conditions are met to assure post shutdown heat removal capability.

1. Two motor-driven steam generator auxiliary feed pumps are operable and set for automatic initiation.

2. An inventory of over 100,000 gallons of primary grade feedwater is available.

Remedial Action:

If either motor driven steam generator auxiliary feed pump becomes inoperable the operable feed pump is to be tested once a day, and the inoperable pump restored to operable status within seven days.

Basis:

Specification A assures that decay heat removal capability is always available.

A single steam generator is capable of removing core decay heat by natural or forced circulation provided the conditions specified in B are met.

A single cooling mechanism is sufficient to remove decay heat but single failure considerations require that two mechanisms be operable.

Specification C assures the structural integrity of the steam generator tubes which are a fission product barrier.

Specification D assures sufficient relieving capacity during either two loop or three loop power operation.

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generators is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the feedwater system.

In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety valves or the atmospheric steam dump valve. Either steam generator auxiliary feed pump can supply sufficient feedwater for removal of decay heat from the plant.

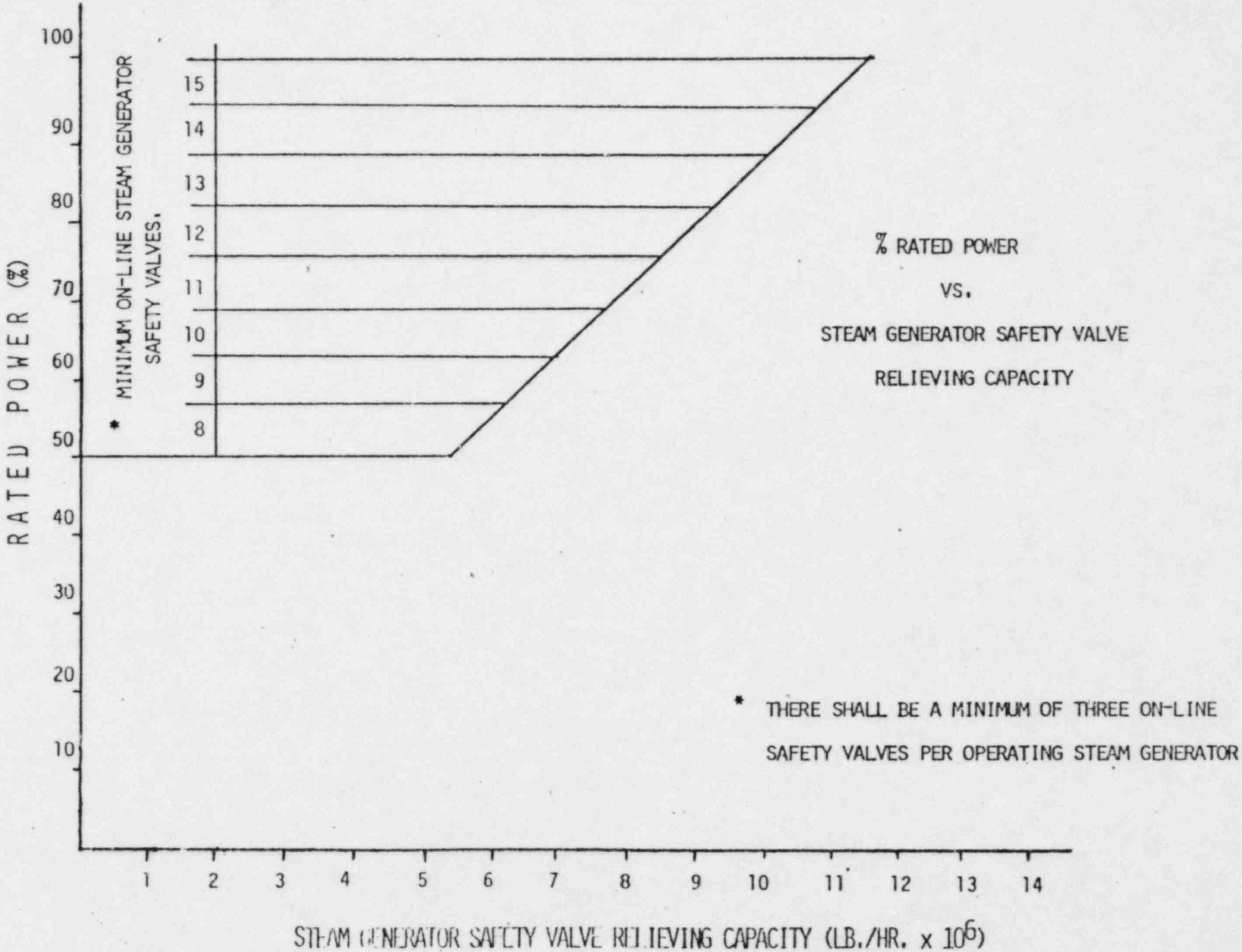

RATED POWER (%)
100
90
80
70
60
50
40
30
20
10
* MINIMUM ON-LINE STEAM GENERATOR SAFETY VALVES.
15
14
13
12
11
10
9
8
% RATED POWER
VS.
STEAM GENERATOR SAFETY VALVE
RELIEVING CAPACITY
* THERE SHALL BE A MINIMUM OF THREE ON-LINE
SAFETY VALVES PER OPERATING STEAM GENERATOR
1 2 3 4 5 6 7 8 9 10 11 12 13 14
STEAM GENERATOR SAFETY VALVE RELIEVING CAPACITY (LB./HR. x 10^6)

Figure 3.8-1

3.9 OPERATIONAL SAFETY INSTRUMENTATION, CONTROL SYSTEMS, AND ACCIDENT MONITORING INSTRUMENTATION

Applicability:
Applies to plant instrumentation system.

Objective:
To specify the conditions of the plant instrumentation and control systems necessary to ensure reactor safety.

Specification:

The operability of the plant instrument and control systems shall be in accordance with Tables 3.9-1, 3.9-2 and 3.9-3.

A. Power operation shall be permitted to continue with the limits as stated in Table 3.9-1 column entitled "Minimum Operable Channels" except as conditioned by the column entitled "Bypass Conditions".

Remedial Action: In the event that specification A above is not met, the plant shall be placed in a hot shutdown condition within 6 hours.

B. Whenever automatic initiation of engineered safeguards is required, the number of operable sensors shall not be less than the minimum specified in Table 3.9-2.

Exception: One subsystem can be removed from service during periods of maintenance or on-line testing for a period of 24 hours.

Remedial Action: In the event that specification B, including the exception, is not met, the plant shall be placed in a hot shutdown condition within 6 hours.

C. Whenever the reactor is at power the minimum Accident Monitoring Instrumentation listed in Table 3.9-3 shall be operable.

Remedial Action: In the event the number of operable accident monitoring instrumentation channels falls below the Minimum Channels Operable requirements in Table 3.9-3, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown condition in the next 6 hours.

Basis:
Reactor safety is assured by the instrumentation channels, logic circuitry, trip modules, and other equipment necessary in the reactor protective system. Selected nuclear steam supply system conditions are monitored and a rapid reactor shutdown is initiated if any one or a combination of conditions deviates from a pre-selected range. This system automatically initiates appropriate action to prevent exceeding established safety limits. Safety is not compromised by continuing operation with certain instrumentation channels or initiation circuits out

of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protection system when any one or more of the channels or circuits are out of service.

In the reactor protective system, four independent and redundant channels monitor each safety parameter. If any one of the four channels deviates from a pre-selected range, a trip signal is initiated. For any safety parameter, a trip signal from any two of the four protective channels will cause a reactor trip. If one of the four channels is taken out of service for maintenance, the protective system for that parameter is changed to a two out of three coincidence for a reactor trip by bypassing the removed channel. When a second channel is taken out of service, the trip module for that channel is placed in the trip mode, and the resultant logic for that parameter is one out of two. Thus, with one or two channels removed from service for that parameter, protective action is initiated when required and the effectiveness of the reactor protection system is retained.

The operating requirements for the reactor protective system are shown in Table 3.9-1.

Although no credit is taken for the high rate-of-change-of-power channel in the Maine Yankee accident analysis, operability of this channel at low power levels provides back up assurance against excessive power rate increases. Temperature feedback effects protect against excessive power rate increases at higher power levels.

Redundant sensors and logic are provided for the initiation of all engineered safeguards systems. In both the containment isolation and containment spray systems, two identical subsystems are used in each system. In the safety injection actuation systems diverse sensors are used for the initiation of two identical subsystems. Each of these three engineered safeguards systems may be operated as shown in Table 3.9-2 without jeopardizing safeguards initiation. One subsystem may be removed from service for a limited time for purposes of maintenance or testing because it is highly unlikely that a failure of the operable subsystem would occur concurrent with an accident requiring engineered safety features actuation.

The safety injection actuation system is initiated by two out of four pressure sensor channels. When three sensors are operable the degree of redundancy, as defined in the definitions section, is one. This degree of redundancy is also provided when two sensors are operable with a third sensor placed in a configuration which simulated the tripped condition.

The minimum number of operable channels for the accident monitoring instrumentation is given in Table 3.9-3. The accident monitoring instrumentation is used to evaluate and aid in mitigating the consequences of an accident.

TABLE 3.9-1

Instrumentation Operating Requirements

for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels (a)	Bypass Conditions
1	Manual (trip buttons)	1 set	None
2	High Rate-of-Change Power	2(c)	Below 10^{-4}% and Above 10% of Rat... Power (b)
3	High Power Level	2(c)	None
4	Thermal Margin/Low Pressurizer Pressure	2(c)	Below 10% of Rated Power (b)
5	High Pressurizer Pressure	2(c)	None
6	Low Reactor Coolant Flow	2(c)	Below 2% of Rated Power (b)
7	Low Steam Generator Water Level	2(c)	None
8	Low Steam Generator Pressure	2(c)	100 psi Above the Trip Setpoint
9	High Containment Pressure	2(c)	None
10	Axial Flux Offset	2(c)	Below 15% of Rated Power (b)

(a) The minimum degree of redundancy is one, except for manual trip which has a minimum degree of redundancy of zero.

(b) As indicated on Nuclear Instrumentation Channels.

(c) Providing one of the inoperable channels is placed in the trip positions, otherwise 3 channels is a minimum.

TABLE 3.9-2

Instrumentation Operating Requirements

for Engineered Safeguards Systems

No.	Functional Unit	Minimum Operable Sensors Per Subsystem	Bypass Conditions	Initiation Set Points
1	Safety Injection:			
	A. Manual	1	*	
	B. High Containment Pressure	3(a)	*	less than 5 psig
	C. Low Pressurizer Pressure	3(a)	*	greater than 1585 psig
2.	Containment Spray:			
	A. Manual	1	*	
	B. High Containment Pressure	2/set(b)	*	less than 20 psig
3	Containment Isolation:			
	A. Manual	1	*	
	B. Containment High Pressure	2/set(b)	*	less than 5 psig

(a) Two operable sensors is acceptable, provided one of the inoperable sensors is placed in a configuration which simulates the tripped condition.

(b) Each subsystem is initiated by two out of three pressure sensors. The minimum degree of redundancy in each subsystem is one.

* Reactor coolant pressure less than 1685 psig.

TABLE 3.9-3

Accident Monitoring Instrumentation

Instrument	Minimum Channels Operable
1. Pressurizer Water Level	1
2. Auxiliary Feedwater Flow Rate	1 per Steam Generator
3. Reactor Coolant System Subcooling Margin Monitor	1
4. PORV Position Indicator (Acoustic Flow Sensor)	1/valve
5. Safety Valve Position Indicator (Acoustic Flow Sensor)	1

3.10 CEA GROUP, POWER DISTRIBUTION, MODERATOR TEMPERATURE COEFFICIENT LIMITS AND COOLANT CONDITIONS

Applicability:

Applies to insertion of CEA groups and peak linear heat rate during operation.

Objective:

To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical CEA ejection, and (3) an acceptable core power distribution, moderator temperature coefficient, core inlet temperature, and reactor coolant system pressure during power operation.

Specification:

A. CEA Insertion Limits

1. When the reactor is critical, except for physics tests and CEA exercises, the shutdown CEA's (Groups A, B and C) shall be fully withdrawn and the regulating CEAS (groups 1 through 5) shall be no further inserted than the limits shown in Figure 3.10-1 for 3 loop operation.

2. CEA's shall be considered fully withdrawn when positioned such that:

   a. the rods are inserted within 4 steps from their upper electrical limit when the RCS boron concentration is greater than 100 ppm

   or

   b. the rods are at their upper electrical limit when the RCS boron concentration is less than or equal to 100 ppm.

3. When the reactor is critical, the shutdown margin with one CEA stuck out will not be less than that shown in Figure 3.10-7. During low power physics testing at the beginning of a cycle, CEA insertion is permitted such that the minimum shutdown margin is no less than 2% in reactivity.

4. Operation of the CEA's in the automatic mode is not permitted.

B. Power Distribution Limits

1. The peak linear heat rate with appropriate consideration of normal flux peaking, measurement- calculational uncertainty (8%), engineering factor (3%), increase in linear heat rate due to axial fuel densification and thermal expansion (0.3% for Types E, G, H & I only) and power measurement uncertainty (2%) shall not exceed:

Fresh Fuel 13.5 kw/ft $\frac{X}{L}$ greater than 0.50 and CAB less than or equal to 792 MWD/MTU

14 kw/ft $\frac{X}{L}$ greater than 0.50 and CAB greater than 792 MWD/MTU

16 kw/ft $\frac{X}{L}$ less than or equal to 0.50

Exposed Fuel: 14.0 kw/ft $\frac{X}{L}$ greater than 0.50

16.0 kw/ft $\frac{X}{L}$ less than or equal to 0.50

where $\frac{X}{L}$ is fraction of core height and CAB is cycle average burnup.

Should any of these limits be exceeded, immediate action will be taken to restore the linear heat rate to within the appropriate limit specified above.

2. The total radial peaking factor, defined as $F_R^T = F_R^P (1 + T_q)$, shall be evaluated at least once a month during power operation above 50% of rated full power.

2.1 F_R^P is the latest available unrodded radial peak determined from the incore monitoring system for a condition where all CEAs are at or above the 100% power insertion limit. T_q is given by the following expression:

$$T_q = 2\sqrt{\frac{(Pa-Pc)^2 + (Pb-Pd)^2}{(Pa+Pb + Pc+Pd)^2}}$$

Pi = relative quandrant power determined from incore system for quandrant i, when the incore system is operable and by Specification 3.10.B.4 otherwise.

2.2 If the measured value of F_R^T exceeds the value given in Figure 3.10-4, perform one of the following within 24 hours:

1. Reduce symmetric offset pre-trip alarm and trip band (Figure 2.1-2), thermal margin/low pressure trip limit (Figure 2.1-1 and Tech. Spec. 2.1), and excore LOCA monitoring limit (Figure 3.10-3) by a factor:

$$\geq \frac{F_R^T \text{ measured}}{F_R^T \text{ (Figure 3.10-4)}}$$

or

2. Reduce THERMAL POWER at a rate of at least 1%/hour to bring the combination of THERMAL power and % increase

in F_R^T to within the limits of Figure 3.10-5, while maintaining CEA's at or above the 100% power insertion limit; or

3. Be in at least HOT STANDBY.

3. Incore detector alarms shall be set at least weekly

Alarms will be based on the latest power distribution obtained, so that the peak linear heat rate does not exceed the linear heat rate limit defined in Specification 3.10.B.1. If four or more coincident alarms are received, the validity of the alarms shall be immediately determined and, if valid, power shall be immediately decreased below the alarm setpoint.

3.1 If the incore monitoring system becomes inoperable, perform one of the following within 4 E.F.P.H.

a. Initiate a power reduction to less than or equal to P at a rate of at least 1%/hour where P (% of rated Power) is given by:

$$P = 0.85 \left(\frac{\text{Linear heat rate permitted by Specification 3.10.B.1}}{\text{Latest measured peak linear heat rate corrected to 100\% Power}}\right) \times 100$$

while maintaining CEA's above the 100% power insertion limit and monitor symmetric offset once a shift to insure that it remains within $\pm$ 0.05 of the value measured at the time when the above equation is evaluated. This procedure may be employed for up to 2 effective full power weeks, or

b. Comply with the alarm band given in Figure 3.10-3. If a power reduction is required, reduce power at a rate of at least 1%/hour.

4. The azimuthal power tilt, Tq, shall be determined prior to operation above 50% of full rated power after each refueling and at least once per day during operation above 50% of full rated power.

Tq is given by the following expression:

$$Tq = 2\sqrt{\frac{(Da-Dc)^2 + (Db-Dd)^2}{(Da + Db + Dc + Dd)^2}}$$

Di = signal from excore detector channel i. Tq shall not exceed 0.03.

4.1 If the measured value of Tq is greater than 0.03 but less than or equal to 0.10, or an excore channel is inoperable, assure that the total radial peaking factor (F_R^T) is within the provisions of Specification 3.10.B.2 once per shift.

4.2 If the measured value of T_q is greater than 0.10, operation may proceed for up to 4 hours as long as F_q^T is maintained within the provisions of Specification 3.10.B.2. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided:

1. The THERMAL POWER level is restricted to less than or equal to 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination, and

2. Reduce setpoints in accordance with Specification 3.10.B.2.2.

5. The incore detector system shall be used to confirm power distribution, such that the peaking assumed in the safety analysis is not exceeded, after initial fuel loading and after each fuel reloading, prior to operation of the plant at 50% of rated power.

6. If the core is operating above 50% of rated power with one excore nuclear channel out of service, then the azimuthal power tilt shall be determined once per shift by at least one of the following means:

   a. Neutron detectors (at least 2 locations per quadrant).

   b. Core-exit thermocouples (at least 2 thermocouples per quadrant).

7. The pre-trip limits of Figure 2.1-2 constitute Limiting Conditions of Operation.

C. CEA Drop Times

1. At operating temperature and 3 pump flow, the requirement for the maximum drop time of each CEA shall be not greater than 2.7 seconds from the time the holding coil is de-energized until the rod reaches 90% of its full insertion.

D. Misaligned, Inoperable, Slow or Dropped CEA

1. A CEA is considered misaligned if it is out of position from the remainder of the bank by more than 8 inches.

   Every 24 hours, except during physics tests and CEA exercises, if a CEA is misaligned, linear heat rate and total radial peaking factors must be shown to be within design limits as specified in 3.10.B.1 and 3.10.B.3 using the latest unrodded radial peaking factors.

   If the CEA deviation alarms from both the computer pulse counting system and the reed switch indication system are not available, individual CEA positions shall be logged and misalignment checked every 4 hours.

2. A full length CEA is considered inoperable if it cannot be tripped. A CEA that cannot be driven in shall be assumed not able to be tripped until it is proven that it can be tripped. No more than one inoperable CEA is permitted during power operation, except during physics testing, or CEA exercises. The shutdown margin limitation specified in 3.10.A.3 must also be met by enough boration to compensate, if necessary for the inoperable CEA within 2 hours.

3. A CEA is considered to be a slow CEA if it does not meet the drop time requirement. Should a CEA exceed the required drop time, then the shutdown margin limitation specified in Specification 3.10.A.3 must be met by enough boration to compensate, if necessary, for the equivalent of 1.5 times the negative reactivity insertion which is delivered after 2.5 seconds.

4. Except during physics testing, in the event of a dropped or misaligned CEA which cannot be corrected within 4 hours of its identification:

   a. The remainder of the rods in its group will be aligned within 8 inches of the misaligned or dropped CEA while maintaining the allowable CEA sequence.

   b. Following realignment, the peak linear heat rate will be shown to be within the limit specified in 3.10.B.1 and the total radial peaking factor will be shown to be within the limit specified in 3.10.B.3 using the latest unrodded radial peaking factor.

E. Moderator Temperature Coefficient (MTC) shall be:

1. Less positive than 0.5×10^{-4} delta rho/°F whenever Thermal power is less than or equal to 70% of Rated Thermal Power.

2. Less positive than 0.0 whenever Thermal Power is greater than 70% of Rated Thermal Power.

F. Coolant Conditions

1. The reactor coolant pressure and the reactor coolant temperature at the inlet to the reactor vessel shall be maintained within the limits of Figure 3.10.6 under steady-state 3 loop operation.

2. The reactor coolant flow rate shall be maintained at or more than a nominal value of 360,000 gpm (indicated) during steady-state 100% power operation.

3. Except for low power physics testing, the reactor shall not be made critical unless the reactor coolant pressure and the reactor coolant temperature at the inlet to the reactor vessel is maintained within the limits of Figure 3.10.6

Basis:

The CEA insertion limit shown in Figure 3.10-1 assures that the individual CEA worths used for the CEA ejection analyses are not exceeded. The CEA insertions used for the CEA withdrawal accident are also not exceeded by this insertion limit. In addition, the limit ensures that the reactor can be brought to a safe hot shutdown condition even with the highest worth CEA not inserted. This restriction provides more shutdown margin than is required at BOL, since the modertor temperature coefficient is more negative at EOL. For this regulating group insertion limit, the peak linear heat rate will be well within the design values.

The limit applies also to two loop operation, in which case the power coordinate is rescaled to 100% of the rated two loop power. This ensures that the CEA induced peaking will not lead to worse thermal conditions than for 3 loop operation since the flow to power ratio is greater for two loop operation. This CEA insertion limit may be revised on the basis of physics calculations and physics data obtained during plant startup and subsequent operation.

Incore detector alarms are set based on the latest power distributions obtained from incore detector analyses. These techniques reflect actual radial and axial power distribution which exist in the core. Should the system become unavailable, continued operation is permitted under either the more conservative excore symmetric offset pretrip (alarm) envelope or at a power level consistent with maintaining a 15% margin to the peak linear heat rate assumed in the LOCA. Both these functions ensure that operation is within the limiting peak linear heat rates assumed as initial conditions for the Loss of Coolant Accident (LOCA). Further, since rod position information is not available to this excore system, this function assumes the most limiting radial power distributions permitted at each power level.

The split excore detectors monitor the axial component of the power distribution. The signal generated from the excore detectors is provided as input to both the Symmetric Offset and Thermal Margin/Low Pressure Trip Systems. Limiting Safety System Settings (LSSS) are, therefore, generated as a function of the excore detector response. The radial component of the power distribution is monitored as a Limiting Condition of Operation (LCO) by Technical Specification 3.10.B.3. Therefore, the intent of Technical Specification 3.10.B.3. is to monitor the radial component of the power distribution and to ensure that assumptions made in the generation of Reactor Protective System (RPS) LSSS remain valid. The LCO on the radial power distribution is specified in Figure 3.10-4 in the form of a steady-state unrodded total radial peak (F_R^T) and provides indication that the core power distribution is behaving as predicted. Figure 3.10-4 includes 10% for calculational uncertainties. The measured steady-state value of F_R^T, augmented by 8% for measurement uncertainty, is compared to this limit on a monthly basis. Should the measured steady-state unrodded total radial peak including uncertainties exceed the limit of Figure 3.10-4 at any time in the cycle specific action is to be taken to assure that the LSSS remain valid. The specific action includes a) the reduction of RPS LSSS and LCO by the ratio of F_R^T measured/F_R^T (Figure 3.10-4) to directly compensate for the higher radial peaks, or b) the imposition of additional restrictions on power and CEA position (Figure 3.10-5) to assure that the assumptions made in establishing the RPS LSSS and LCO remain valid. Figure 3.10-5 in conjunction with restricted CEA insertion allows for an increase in the steady-state unrodded total radial peak above the limits of 3.10-4 without a modification of the RPS LSSS. The allowed increase in radial peak is derived from the difference between the radial peaks assumed in the RPS setpoints for rodded conditions at reduced power and the radial peaks reflected in the CEA insertion limit at 100% power. This assures that the radial peaking factors vs. power assumed in the RPS LSSS remain valid.

The power distribution in the core can be determined in two ways. The normal method is through analysis of the fixed and movable neutron detector signals with the on-line computer. The alternative is to determine the radial and axial peaking factors by hand. The radial peaking factor can be determined from the core exit thermocouples, the fixed incore detectors or the movable incore detector traces. The axial peaking factor can be determined from the fixed incore detectors, the movable incore detector traces or the excore detectors. The requirement that the core power distribution be shown to be within the design limits after every refueling not only ensures that the reactor can be operated safely but will also provide reasonable verification that the core was properly loaded. The requirement for operability of incore instrumentation in the instance of an excore detector channel being out of service ensures that an unobserved quadrant power tilt will not occur.

The maximum CEA drop time specified is consistent with the values used in the safety analysis.

For a full length CEA, with misalignment up to 8 inches from the remainder of the bank, the peaking factors will be well within design limits. If a CEA is misaligned, the peak linear heat rate will be shown to be within design limits every 24 hours. The 24 hour time limit is short with respect to the probability of an independent incident occurring. The requirement that no more than one inoperable CEA is allowed and that the shutdown margin is maintained ensures that the reactor can be brought to a safe shutdown condition at any time.

Shutdown margin is assured within the required CEA drop time by conservatively borating to compensate for a slow CEA during operation, if necessary. CEA drop times, CEA core height vs. time, and CEA worth measurements are all made after initial loading and each refueling. Should a CEA drop time be in excess of 3.10.C.1, then the core height on that CEA at 2.5 seconds would be conservatively determined. Reactivity worth of the CEA from the above core height to the bottom of the core would then be determined. Sufficient boron would thereafter be added, if necessary, during power operation to compensate for 1.5 times the above measured reactivity in order to maintain adequate shutdown margin.

The requirement to align the dropped or misaligned CEA with the remainder of its bank assures that operation will not be under conditions which violate the assumptions used in the generation of the RPS setpoints.

The moderator temperature coefficient, coolant pressure, flow rate, and temperature specified are consistent with the value assumed in the safety analysis. The safety analysis assumes a maximum reactor inlet temperature of 554°F. The specified value includes 4°F for temperature measurement uncertainties.

3.11 CONTAINMENT

Applicability:

Applies to the operating status of the reactor containment.

Objective:

To ensure containment integrity.

Specification:

A. Containment integrity is defined to be operable when all the following are met:

1. All non-automatic containment isolation valves and blind flanges are closed.

2. The equipment hatch is properly closed and sealed.

3. At least one hatch in the personnel air lock is properly closed and sealed.

4. All automatic containment isolation valves are operable or are locked closed.

5. The uncontrolled containment leakage satisfied Specification 4.4 Section I.B.3.

B. Containment integrity shall be maintained whenever there is fuel in the reactor and

1. The reactor coolant system is above 210°F or

2. The reactor coolant boron concentration is less than Cold Shutdown Concentration with the reactor vessel head in place, or

3. The reactor coolant boron concentration is less than refueling concentration with the reactor vessel head removed.

Exception: On-line purging of containment is not a breach of containment integrity provided both valves on each line are automatically operable.

Remedial Action:

With one or more automatic containment isolation valves inoperable, maintain at least one automatic isolation valve operable in each affected penetration that is open and within 4 hours either:

1. restore the inoperable valve to operable status

or

2. isolate the affected penetration by use of at least one manual or automatic isolation valve locked in the closed position or by use of a blind flange.

C. The reactor shall not be critical if the containment internal pressure exceeds 3 psig.

D. Containment Weight of Air Monitoring System

1. The containment weight of air monitoring system shall be in operation whenever the reactor has been at power for more than 72 hours.

   Exception: The system need not be operational during periods of system maintenance or calibration, periods of recharging the containment pressure, or periods of containment on-line purging and 48 hours thereafter.

   Remedial Action

   If the containment weight of air monitoring system is out of service for more than ten days with the reactor critical, the Commission must be notified of plans to restore the system operability.

2. When the containment weight of air monitoring system indicates a daily air loss greater than the following, an evaluation shall be initiated to determine the validity of the indication.

   a. equivalent to 0.15 weight percent per day at 50 psig for seven consecutive days

   or

   b. equivalent to 0.5 weight percent per day at 50 psig for four consecutive days

   or

   c. equivalent to 1.0 weight percent per day at 50 psig for three consecutive days

3. The reactor shall be made subcritical within six hours if the evaluation required by D2:

   a. results in identification of the source of the leak and a determination that the known containment leak rate exceeds the equivalent of 0.15 weight percent per day at 50 psig through the containment integrity boundary

   or

   b. fails to identify the source of the leakage within ten days and the Containment Weight of Air Monitoring System indication persists at an average rate in excess of 0.15 weight percent per day at 50 psig.

Basis:

Specification A assures that the containment pressure boundary is defined while permitting maintenance of components necessary to integrity.Specification 4.4 Section 1.8.3 limits the uncontrolled containment leakage to assure that public exposure will be maintained well within the guidelines presented in 10 CFR 100 for the hypothethical accident described in Section 14.18 of the FSAR.

Specification B includes a limit of 210°F on main coolant temperature assures that no steam will be generated in the unlikely event of a main coolant system rupture and hence no driving force to release any fission products from the containment. The shutdown margins are selected based upon the type of activities that are being carried out. The higher value for refueling precludes criticality under all postulated incidents involving fuel movement. The lower value with the head in place will also preclude criticality for all postulated incidents.

There is about a 5 psig margin between the calculated peak accident pressure and the containment design pressure of 55 psig. The 3 psig maximum operating pressure permits a positive containment pressure which is necessary for successful operation of the continuous leakage monitoring system.

Specification D provides an added measure of assurance of containment integrity by specifying that the containment weight of air monitoring system be operational while recognizing the limitations of such systems to reliably measure very small changes in air mass and its operational limitations.

3.12 STATION SERVICE POWER

Applicability:

Applies to station service electrical power systems.

Objective:

To assure an adequate supply of electrical power during station operation.

Specification:

A. The following equipment must be operable whenever the reactor coolant system temperature and pressure exceeds 210°F and 400 psig.

1. One 115 kv incoming line in service.

2. Diesel generator DG-1A operable; 4160v emergency bus 5, 480 v emergency bus 7, and d-c distribution cabinet 1 in service,

   or

   Diesel generator DG-1B operable; 4160v emergency bus 6, 480v emergency bus 8, and d-c distribution cabinet 3 in service.

3. 10,000 gallons of diesel fuel oil in the fuel oil tanks.

Remedial Action: Restore required limiting condition within 4 hours.

B. The following equipment must be operable whenever the reactor is critical.

1. One 115 kv incoming line in service.

2. Diesel generator DG-1A operable; 4160v emergency bus 5, 480v emergency bus 7, and d-c distribution cabinet 1 in service.

3. Diesel generator DG-1B operable; 4160v emergency bus 6, 480v emergency bus 8, and d-c distribution cabinet 3 in service.

4. 19,600 gallons of diesel fuel oil in the fuel oil tanks.

Remedial Actions:

1. If the 115 kv incoming line becomes unavailable, the NRC shall be notified within 24 hours of the plans for restoration of service and the line must be restored to the available status within seven days.

2. If either diesel generator or its associated emergency buses or d-c distribution cabinet becomes unavailable, the operable diesel generator is to be tested once a day and the equipment restored to the available status within seven days.

3. If more than one of the power supplies in B above is not operable, follow Specification 3.0.A.

C. Under accident conditions the automatically connected load to either diesel generator shall not exceed the 2000 hour rating of 2850 kW.

Basis:

Availability of power to the engineered safeguards equipment is necessary when the reactor is at power. If the loss of both incoming lines, a diesel generator or its associated emergency buses occurs, a period of seven days operation is permitted while the situation is being assessed and full redundancy is being restored. This time period is justified because adequate sources of power remain available for the operation of the engineered safeguards equipment.

The fuel requirement of 19,600 gallons is made up as follows:

A. 17,827 gallons is the amount that will be required for the maximum expected engineered safeguards load for a period of seven days;

B. 10 percent of the above as a contingency for any non-engineered safeguard requirement during this period.

Specification A assures that an emergency power source is available whenever the reactor coolant system is above the specified pressure and temperature limit. It recognizes the decreased consequences of a loss of coolant accident if the reactor is subcritical.

3.13 REFUELING OPERATIONS

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To minimize the possibility of an accident occurring during refueling operations that could affect the health and safety of plant personnel and the public.

Specification:

A. The following conditions shall be satisfied during refueling operations:

1. The containment venting and purge system, including inlet and outlet trip valves that isolate the ventilation system in response to radiation monitors, shall be operable, with the discharge filtered through the high efficiency particulate air filters and charcoal absorbers.

2. Two radiation monitors that initiate isolation of the containment ventilation system, shall be tested and verified to both be operable immediately prior to fuel handling operations and remain operable during fuel handling operations. The two monitors shall be located on the containment fuel handling area level, shall be part of the plant area monitoring system, and shall employ one-out-of-two logic for isolation.

   Exception: The valve trip system may be bypassed for a period not to exceed 0.5 hours daily to facilitate routine testing of the radiation monitors.

   Remedial Action: Should one of the area monitors become inoperable, repairs shall be affected immediately and the logic shall revert to one-out-of-one for isolation. Refueling operations may continue for a maximum of 12 hours in this mode.

3. The capability of the containment purge trip valves to respond to a trip signal from the radiation monitors shall be tested immediately prior to fuel handling operations and weekly thereafter.

4. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

5. Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two wide range logarithm monitors, with each monitor providing continuous visual indication in the control room with at least one monitor generating an audible neutron count rate in the containment. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

6. At least one residual heat removal pump and heat exchanger shall be in operation.

   Exception: This system may be shut down for a maximum of 6 hours to facilitate upper guide structure assembly removal or other special maintenance operations. During such periods the reactor coolant temperature shall be continuously monitored and the initiation of core cooling flow shall be continuously available.

7. Both RHRS loops A and B shall be operable when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

8. During reactor vessel head removal and while refueling operations are being performed in the reactor, the refueling boron concentration shall be maintained in the reactor coolant system and shall be checked by sampling on each shift to insure that the boron concentration is such to maintain the core 5% delta k/k subcritical.

9. Direct communication between personnel in the control room and at the refueling station shall be operable whenever changes in core geometry are taking place.

10. A minimum of 23 feet of water above the top of the core shall be maintained whenever spent fuel is being handled.

11. Irradiated fuel shall not be handled until 72 hours after reactor shutdown.

Remedial Action: If any of the conditions in Specification A are not met, all refueling operations shall cease immediately; work shall be initiated to satisfy the required conditions, and no operations that may increase the reactivity of the core shall be made.

B. Prior to each refueling a complete checkout, including a load test, shall be conducted on fuel handling cranes that will be used to handle spent fuel assemblies.

C. Spent fuel storage racks may be moved only in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

Basis:

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above specifications and the design of the fuel handling equipment incorporating built-in interlocks and safeguards systems provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.

The exception to paragraph 3.13.A.2 permits routine testing of the radiation monitors without incurring unnecessary wear of the purge valve resilient seals. Weekly testing of these trip valves is sufficient to insure their operability.

Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal flow is used to remove core decay heat and maintain a uniform boron concentration.

A single cooling mechanism is sufficient to remove decay heat but single failure considerations require that two mechanisms be OPERABLE.

The shutdown margin as indicated will keep the core substantially subcritical, even if the highest worth CEAs were inadvertently withdrawn from the core without compensating boron addition.

Periodic checks of refueling water boron concentration insure the proper shutdown margin. Communication requirements allow the control room operator to inform the refueling station operator of any impending visual condition detected from the main control board indicators during fuel movement.

In addition to the above engineered safeguards systems, interlocks are utilized during refueling to insure safe handling. An excess weight interlock is provided to prevent excess loading of a fuel assembly, should it inadvertently become stuck.

In the analysis of the refueling accident conducted by the staff, 23 feet of water and 72 hours of decay time were used to limit exposures to 10% of 10 CFR 100. Valve alignment check sheets are completed to protect against sources of unborated water or draining of the system.

Procedures are required for movement of spent fuel racks to avoid unnecessary risk of spent fuel damage caused by dropping spent fuel racks.

3.14 PRIMARY SYSTEM LEAKAGE

Applicability:

Applies to limiting operation of the plant under varying rates and conditions of primary system leakage.

Objective:

To specify primary plant operability with primary system leakage.

Specification:

A. When the reactor is above 2% power, two reactor coolant leak detection systems of different operating principles shall be operating, with one of the two systems sensitive to radioactivity in the containment.

Remedial Action: If two reactor coolant leakage detection systems are operable but neither is sensitive to radioactivity, a system sensitive to radioactivity must be made operable within 48 hours.

B. Whenever reactor coolant system indicated leakage exceeds 1 gpm by any means available an investigation as to source and safety implications will be initiated as soon as practicable but no later than within four hours.

C. Reactor coolant system leakage shall not exceed any of the Specifications 1 through 5 below.

1. Leakage into the reactor containment of any magnitude that has been determined to be an indication of a deterioration of primary system pressure boundary strength welds or material.

2. Leakage into the reactor containment in excess of 1 gpm through bolted closures, valve packing, or other mechanical connections.

3. Leakage in excess of 1 gpm that is unexplained or unaccounted for.

4. Leakage in excess of 10 gpm to aerated or uncontained systems.

5. Total leakage through all steam generator tubes shall not exceed 1.0 gpm.

Remedial Actions:

1. If the leakage specified in C.1 above has been determined to be a deterioration of primary system pressure boundary strength welds or material, then the provisions of Specification 3.0.A.2 and 3 apply.

2. If reactor coolant system leakage exceeds any of the Specifications C.2 through C.5 above, the reactor shall be shut down within 24 hours.

Basis:

Reactor coolant system leakage may be indicated by one or more of the following methods:

1. Primary system water balance inventory.

2. Containment sump level.

3. Containment air particulate monitor and/or radio gas monitor.

4. Containment atmospheric humidity and/or temperature.

5. Steam generator blowdown monitor and/or air ejector effluent monitor.

Leakage may be indicated and/or identified by the following routine and special plant surveillance operations.

1. Primary system hot leak test. (Involves monitoring steady-state drop in Volume Control Tank Level).

2. Direct observation from accessible locations within the containment.

3. Sampling and analysis of containment atmosphere, steam generator blowdown and air ejector effluent for radioactive and non-radioactive tracers.

Reactor coolant system leakage will be maintained at the lowest practical value so that small leaks, with possible safety implications, will be more readily detected and identified.

A safety evaluation of a leak shall consider its magnitude, nature, and possible consequences. It shall assure that off-site radiation exposure from the primary coolant system activity is within the guidelines of 10 CFR 20.

For the purposes of determining a maximum allowable secondary coolant activity, the steam break accident is based on a postulated release of the contents of three steam generators to the atmosphere using a site boundary dose limit of 1.5 rem. The limiting dose for this accident results from iodine in the secondary coolant. The reactor coolant distribution of iodine isotopes with 1% failed fuel was used for this calculation. I-131 is the dominant isotope because of its low MPC in air and because the other iodine isotopes have shorter halflives and therefore cannot build up to significant concentration in the secondary coolant, given the limitations on primary system leak rate and activity. The steam generators which operate at a constant programmed level contain $131 m^3$ of water at standard conditions. One tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets.

The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{(C)\ (V)\ [B(t)]\ (X/Q)\ (DCF)}{10}$$

where:

C = Seconnary coolant sample activity 0.2 micro ci/cc = 0.2 Ci/M^3

V = Water volume in three steam generator = 131 M^3 at standard conditions

B (t) = Breathing rate (3.47×10^{-4} m^3/sec)

X/Q = 6.48×10^{-4} sec/m^3 (corresponding to Pasquill F stability and 1 m/sec wind speed)

DCF = 1.48×10^6 rem/Ci I-131 inhaled

The resulting thyroid dose is less than 1.5 rem.

3.15 REACTIVITY ANOMALIES

Applicability:
Applies to potential reactivity anomalies.

Objective:
To require evaluation of reactivity anomalies within the reactor.

Specification:

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the reactor coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of 1% in reactivity, the Nuclear Regulatory Commission shall be notified and an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission in accordance with Technical Specification 5.9.1.6.

Basis:
To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the CEA groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, ctual boron concentration can be compared with prediction and the r tivity status of the core can be continuously evaluated, and its occur nce would be thoroughly investigated and evaluated. The methods employ in calculating the reactivity of the core vs. burnup, and the reactivity worth of boron vs. burnup, are given in the FSAR.

3.16 RELEASE OF LIQUID RADIOACTIVE WASTE

Applicability:

Applies to the controlled release of all liquid waste discharged from the plant which may contain radioactive materials.

Objective:

To establish conditions for the release of liquid waste containing radioactive materials and to assure that all such releases are within the concentration limits specified in 10 CFR Part 20. In addition, to assure that the releases of radioactive material in liquid wastes (above background) to unrestricted areas meet the low as practicable concept, the following liquid release objectives shall apply:

A. The annual total quantity of radioactive materials in liquid waste, excluding tritium and dissolved gases, shall be less than 5 curies;

B. The annual average concentration of radioactive materials in liquid waste, excluding tritium and dissolved gases, shall not exceed 2×10^{-8} micro Ci/ml;

C. The annual average concentration of tritium in liquid waste shall not exceed 5×10^{-6} micro Ci/ml;

D. The annual average concentration of dissolved gases in liquid waste shall not exceed 2×10^{-6} micro Ci/ml.

Specification:

A. Release Quantities and Concentrations of Radioactive Materials in Liquid Waste

1. If the experienced release of radioactive materials in liquid wastes, when averaged over a calendar quarter, is such that these quantities if continued at the same release rate for a year would exceed twice the annual objectives the licensee will:

   a. make an investigation to identify the causes for such release rates;

   b. define and initiate a program of action to reduce such release rates to the design levels, and;

   c. describe these actions in a report to the Commission within 30 days.

2. If the experienced release of radioactive material in liquid waste, when averaged over a calendar quarter, is such that these quantities if continued at the same release rate for a year would exceed eight times the annual objectives, the licensee shall define and initiate a program of action to assure that such release rates are reduced, and shall submit a report to the Commission within 7 days describing the causes for such release rates and the course of action taken to reduce them.

3. The rate of release of radioactive materials in liquid waste from the plant shall be controlled such that the instantaneous concentration of radioactivity in liquid waste does not exceed the values listed in 10 CFR Part 20, Appendix B, Table II, Column 2.

B. Treatment and Monitoring

1. The equipment installed in the liquid radioactive waste system shall be maintained and operated with the intent of keeping releases within the objectives of these specifications.

2. At least one service water pump shall be in operation when liquid radioactive wastes are being released.

3. Liquid waste discharged from the test tanks shall be continuously monitored during release. The liquid effluent monitor reading shall be compared with the expected reading of each discharge batch. The monitor shall be tested daily and calibrated at refueling intervals. The calibration procedure shall consist of exposing the detector to a referenced calibration source in a controlled, reproducible geometry. The sources and geometry shall be referenced to the original monitor calibration which provides the applicable calibration curves.

4. The effluent control monitor shall be set to alarm and automatically close the waste discharge valve such that the requirements of the specification are met. In the event of a malfunction in the monitor, the alarm shall sound and automatically close the waste discharge valve.

5. Steam generator blowdown shall be continuously monitored, except that during periods when the monitor is not operating, daily grab samples shall be taken.

C. Sampling and Analysis

In addition to the above continuous monitoring requirements, liquid radioactive waste sampling and activity analysis shall be performed in accordance with Table 3.16-1. Records shall be maintained and reports of the sampling and analysis results shall be submitted in accordance with Sections 5.6 and 5.7 of these Specifications.

Basis:

It is expected that the releases of radioactive materials in liquid waste will be kept within the design objective levels and will not exceed the concentration limits specified in 10 CFR Part 20. These levels provide reasonable assurance that the resulting annual exposure to the whole body or any organ of an individual will not exceed 5 millirems per year. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that using this operational flexibility under unusual operating conditions, the licensee shall exert every effort to keep levels of radioactive material in liquid wastes as low as practicable and that annual releases will not exceed a small fraction of the annual average concentration limits specified in 10 CFR Part 20.

The design objectives have been developed taking into account a combination of variables including fuel failures, primary system leakage, primary-to-secondary leakage and the performance of the various waste treatment systems. The actual magnitude of these parameters are as follows:

A. Maximum expected reactor coolant corrosion product concentrations;

B. Reactor coolant fission product concentration corresponding to 0.1% fuel cladding defects;

C. Steam generator primary-to-secondary leak rate of 0.01 gpm;

D. Hydrogenated liquid waste generation rate of 1.75 gpm;

E. Aerated liquid waste generation rate of 0.475 gpm;

F. Steam generator blowdown rate of 5 gpm, of which 3 gpm is diverted to the waste disposal system for processing before discharge;

G. Decontamination factor of 10^4 for all radionuclides except tritium for the boron recovery and waste disposal evaporators;

H. Decontamination factor of 10 for Cs, Sr, Mo and Y for cesium demineralizer.

The application of the above estimates results in the radionuclide discharge concentrations and rates shown in Table 3.16-2. Also given in this table are the radionuclide concentrations in the reactor coolant and the secondary coolant, which are the "source terms" for releases from the primary and secondary systems, respectively. Liquid radioactive waste is mixed with service water in the plant discharge system prior to release. With four circulating water pumps in operation, the rated capacity of the system is 400,000 gpm. This is equivalent to a dilution multiple of 2.5×10^{-6} min/gal x the discharge rate in gal/min. Liquid radioactive waste from the waste treatment system is collected and stored in tanks until a quantity sufficient for processing has accumulated. The processed liquid waste is discharged through a recorder controller which provides a measure and control of volume of liquid released. The volume discharged and the analysis of the proportional composite sample provide the basis for reporting the quantity and concentration of activity released.

The operating manual will identify all equipment installed in the liquid waste handling and treatment systems and will specify detailed procedures for operating and maintaining this equipment.

The low as practicable liquid release objectives expressed in this specification are based on the guidelines contained in the proposed Appendix I of 10 CFR 50. Since these guidelines have not been adopted as yet, the release objectives of this Specification will be reviewed at the time Appendix I becomes a regulation to assure that this Specification is based upon the guidelines contained therein.

3.17 RELEASE OF GASEOUS RADIOACTIVE WASTE

Applicability:

Applies to the controlled release of all gaseous waste discharged from the plant which may contain radioactive materials.

Objective:

To establish conditions in which gaseous waste containing radioactive materials may be released and to assure that all such releases are within the concentration and dose limits specified in 10 CFR Part 20. In addition, to assure that the releases of gaseous radioactive wastes (above background) to unrestricted areas meet the as low as practicable concept, the following objectives shall apply:

A. Averaged over a yearly interval, the release rate of radioactive isotopes, except I-131 and particulate radioisotopes with half lives greater than 8 days, discharged at the plant stack, shall be limited as follows:

$$\sum \frac{Qi}{(MPC)i} \text{ less than or equal to } 800 \text{ m}^3/\text{sec}$$

where Qi is the annual controlled release rate (Ci/sec) of radioisotope i and (MPC)i (micro ci/cc) is defined for radioisotope i in column 1, Table II of Appendix B to 10 CFR 20.

B. Averaged over a yearly interval, the release rate of I-131 and other particulate radioisotopes with half lives longer than 8 days, discharged at the plant stack, shall be limited as follows:

$$\sum \frac{Qi}{(MPC)i} \text{ less than or equal to } 5.6 \text{ m}^3/\text{sec}$$

where Qi and (MPC)i are as defined above.

Specification:

A. Release Quantities and Concentrations of Radioacative Materials in Gaseous Waste

1. If the experienced rate of release of radioactive materials in gaseous wastes, when averaged over a calendar quarter is such that these quantities if continued at the same release rate for a year would exceed twice the annual objectives, the licensee will:

   a. make an investigation to identify the causes for such release rates;

   b. define and initiate a program of action to reduce such release rates to the design levels;

   c. describe these actions in a report to the Commission within 30 days.

2. If the experienced rate of release of radioactive material in gaseous wastes, when averaged over a calendar quarter, is such that these quantities if continued at the same release rate for a year would exceed eight times the annual objectives, the licensee shall define and initiate a program of action to assure that such release rates are reduced, and shall submit a report to the Commission within 7 days describing the causes for such release rates and the course of action taken to reduce them.

3. The rate of release of radioactive materials in gaseous waste from the plant (except I-131 and particulate radioisotopes with half lives greater than 8 days) shall be controlled such that the maximum release rate averaged over any one-hour period shall not exceed:

$$\sum \frac{Qi}{(MPC)i} = 3.1 \times 10^4 \ m^3/sec$$

B. Treatment and Monitoring

1. At least one exhaust fan shall be in operation when radioactive gaseous wastes are released to the stack.

2. During release of radioactive gaseous waste from the gaseous waste decay drums to the stack, the following conditions shall be met:

   a. 1. The gas decay drum effluent monitor and the stack sampling devices for halogens and particulates shall be operable.

      2. The normal response of the decay drum effluent monitor shall be verified by comparison with the prerelease sample analysis.

      3. The monitor shall be tested prior to any release of radioactive gas from a decay drum.

      4. The monitor shall be calibrated at refueling intervals. The calibration procedure shall consist of exposing the detector to a referenced calibration source in a controlled reproducible geometry. The source and geometry shall be referenced to the original monitor calibration which provides the applicable calibration curves.

   b. The gaseous waste from the decay drums shall be filtered through the high efficiency particulate air filters and the charcoal absorber provided.

3. a. During normal conditions of plant operation, radioactive gaseous waste from the hydrogenated waste gas system shall be provided a minimum average holdup of 60 days except for low radioactivity gaseous waste resulting from purge and fill operations associated with refueling and reactor startup.

Remedial Action: Holdup time less than that specified in B.3.a above shall be covered in the special effluent report to be included in the semi-annual report required by Section 5.7.B.(1)(a) of these specifications.

b. The maximum activity to be contained in one gas decay tank shall not exceed 88,400 curies of Xe-133 equivalent.

4. During the first indication of primary-to-secondary leakage, concurrent with sufficient fuel defects, a determination of the iodine partition factor for the blowdown tank shall be made.

5. During power operation the condenser air ejector discharge shall be continuously monitored for gross radiogas activity. Whenever this monitor is inoperable, grab samples shall be taken from the air ejector discharge and analyzed for gross radiogas activity daily.

6. Gases discharged through the stack shall be continuously monitored for gross noble gas and particulate activity. Whenever either of these monitors is inoperable, appropriate grab samples shall be taken and analyzed daily.

7. Reactor building purge shall be filtered through the high efficiency particulate air filters and charcoal absorbers whenever the average air concentration of iodine and particulate isotopes inside the reactor building exceeds the occupational MPC.

C. Sampling and Analysis

In addition to the above continuous sampling and monitoring requirements, gaseous radioactive waste sampling and activity analysis shall be performed in accordance with Table 3.17-1. Records shall be maintained and reports of the sampling and analysis results shall be submitted in accordance with sections 5.6 and 5.7 of these specifications.

Basis:

It is expected that the releases of radioactive materials in gaseous waste will be kept within the design objective levels and will not exceed on an instantaneous basis the dose rate limits specified in 10 CFR Part 20.

These levels provide reasonable assurance that the resulting annual exposure from noble gases to the whole body or any organ of an individual will not exceed 5 millirems per year. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that using this operational flexibility under unusual operating conditions, the licensee shall exert every effort to keep levels of radioactive material in gaseous wastes as low as practicable and that annual releases will not exceed a small fraction of the annual average concentration limits specified in 10 CFR Part 20. These efforts shall include consideration of meteorological conditions during releases.

The design objectives have been developed taking into account a combination of system variables including fuel failures, primary system leakage and the performance of radioisotope removal mechanisms. The values assumed for these variables include the following:

A. Reactor coolant fission product concentration corresponding to 0.1% fuel cladding defects;

B. Steam generator primary-to-secondary leak rate of 0.01 gpm;

C. Steam generator blowdown rate of 5 gpm;

D. Reactor coolant leakage to the containment building of 0.25 gpm and four containment vents per year;

E. Partition factor of 1000 for iodine in aerated drains tanks;

F. Gas decay drums average 60 days holdup;

G. Decontaminiation factor of 1000 for iodine in the degassifier;

H. Charcoal filter efficiency of 99% for iodine on the air ejector, aerated vent and gas decay drum systems.

The application of the above estimates result in the radiogas discharge rates shown in Table 3.17-2.

The noble gas release rate stated in the objectives is based on a X/Q value from the annual meteorological data. The dispersion factor used, 2.59×10^{-5} sec/m^3, is conservative and controls the release rate to a small fraction of 10 CFR Part 20 requirements at the site restricted area boundary (less than 10 mrem per year).

The I-131 and particulate release rate stated in the objectives limits the concentration at the restricted area boundary to less than 1% of the MPC listed in 10 CFR 20. The release rate also controls the expected concentrations at nearby commercial dairy farms to much less than 1/100,000 of the 10 CFR 20 requirements.

The maximum one-hour release rate limits the dose rate at the site boundary to less than 2 mrem/hour even during period of unfavorable meteorology. (Moderately stable conditions with 2 m/sec wind speed).

The maximim activity in a waste gas decay drum is specified as 68,400 curies of Xe-133 equivalent based on a postulated rupture that allows all of the contents to escape to the atmosphere. This specification limits the maximum offsite dose to well below the limits of 10 CFR 100.

The gaseous waste system is divided into two sections; aerated gases and hydrogenerated gases. Low activity, aerated gaseous wastes are discharged to the aerated gas header and through a high efficiency filter to the primary vent stack. Hydrogenated gaseous wastes flow from the surge drum and through the gas compressor which discharges to the waste gas decay drum. The drum is pressurized and then isolated for decay of the gaseous wastes before discharge to the primary vent stack. The gaseous discharge is continuously monitored both in the vent line to the primary auxiliary building fan suction and in the stack. Upon detection of high activity in the vent line or upon the loss of ventilation fan suction, the vent line flow control valve will close, terminating the release of gaseous waste.

The quantity and isotopic proportions of radioactive gases released into the reactor coolant system is dependent upon several factors including fuel leakage, burnup and power level. Changes in power level will affect gaseous generation rates temporarily. Gases are released from the reactor coolant to the gaseous waste system during degassifier treatment of the letdown and leakage water and also during venting of the system. This venting may occasionally be performed to degas the system and so control plant chemistry and/or reduce coolant radioactive gas concentrations to an acceptable value for the protection of plant personnel.

Gaseous waste holdup and decay occurs while it is retained in the reactor coolant system and in the surge drum of the gaseous treatment system. The gaseous waste holdup drums are of sufficient capacity to provide an additional average retention period of 60 days during normal operating conditions.

The low as practicable gaseous release objectives expressed in this specification are based on the guidelines contained in the proposed Appendix I of 10 CFR 50. Since these guidelines have not been adopted as yet, the release objectives of this Specification will be reviewed at the time Appendix I becomes a regulation to assure that this Specification is based upon the guidelines contained therein.

Specification 3.7 above describes when the reactor building purge shall be filtered through the high efficiency air filters and charcoal absorbers. The average air concentration of particulate isotopes may be measured by the Containment Air Particulate Detectors. The average air concentration of iodine may be measured by local sampling.

Table 3.17-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

A. Gas Decay Drum Releases

Sample Type	Sampling Frequency	Type of Activity Analysis	Sensitivity of Analysis (1)
Gas	Each Drum Release	Gross Gamma	10^{-5} μCi/cc
		Individual Gamma Emitters	10^{-4} μCi/cc (2)

B. Containment Venting Releases

Sample Type	Sampling Frequency	Type of Activity Analysis	Sensitivity of Analysis (1)
Gas	Each Vent	Gross Gamma	10^{-5} μCi/cc
		Individual Gamma Emitters	10^{-4} μCi/cc (2)
Dehumidified Sample	Each Vent	H-3	10^{-6} μCi/cc

C. Condenser Air Ejector Releases

Sample Type	Sampling Frequency	Type of Activity Analysis	Sensitivity of Analysis (1)
Gas	Monthly	Gross Gamma	10^{-4} μCi/cc
		Individual Gamma Emitters	10^{-3} μCi/cc (2)

Table 3.17-1 (cont'd)

D. Stack Releases

Sample Type	Sampling Frequency	Type of Activity Analysis	Sensitivity of Analysis (1)
Gas	Quarterly	Gross Gamma	10^{-6} µCi/cc
		Individual Gamma Emitters	10^{-5} µCi/cc (2)
Dehumidified Sample	Each Decay Drum Release	H-3	10^{-6} µCi/cc
Charcoal	Weekly	I-131, I-133, I-135	3×10^{-12} µCi/cc
Particulates	Weekly	Gross β, γ	3×10^{-12} µCi/cc
	Weekly	Ba-140, La-140, I-131	3×10^{-11} µCi/cc
	Monthly Composite of Weekly Samples	Gross β, γ	3×10^{-12} µCi/cc
		Individual Gamma Emitters	3×10^{-11} µCi/cc
	Quarterly Composite of Weekly Samples	Sr-89, Sr-90	1×10^{-11} µCi/cc
	One Weekly Sample/Quarter	Gross α	3×10^{-12} µCi/cc

NOTES:

(1) The above activity analysis sensitivities are based on the projected capability of laboratory instrumentation and techniques to be employed by Maine Yankee. In order to assure that actual Maine Yankee operating experience is utilized, a reevaluation will be performed within 2 years of initial full power operation of the plant.

(2) For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate the levels of such radionuclides using observed ratios with those radionuclides which are measurable.

Table 3.17-2

GASEOUS RADIOACTIVE RELEASES

Isotope	Reactor Coolant Concentration (μCi/ml @ 70^{o}F)	Release Rate, μCi/sec				
		Aerated Vents	Air Ejector	Containment Vent	Decay Drums	Total
I-131	2.99-1*	1.28-4	1.09-4	4.2-4	2.0-6	6.6-4
I-132	1.12-1	4.83-5	1.59-6	1.96-6	---	5.2-5
I-133	5.02-1	2.16-4	5.31-5	7.9-5	---	3.5-4
I-134	7.55-2	3.22-5	1.0-7	4.95-8	---	3.2-5
I-135	2.80-1	1.2-4	1.1-5	1.41-6	---	1.3-4
Kr-85	1.04	---	6.56-1	1.2+1	3.4+1	4.7+1
Kr-85m	1.9-1	---	1.2-1	6.27-3	---	1.26-1
Kr-87	1.08-1	---	6.8-2	1.05-3	---	6.9-2
Kr-88	3.26-1	---	2.06-1	6.8-3	---	2.1-1
Xe-131m	1.25-1	---	7.88-2	2.76-1	---	3.6-1
Xe-133	2.52+1	---	1.59+1	2.44+1	---	4.0+1
Xe-135	5.60-1	---	3.53-1	4.26-2	---	4.0-1

*2.99-1 = 2.99 x 10^{-1}

3.18 REACTOR COOLANT SYSTEM OXYGEN AND CHLORIDE/FLUORIDE CONCENTRATION

Applicability:

Applies to the measured maximum oxygen and chloride/fluoride concentrations in the reactor coolant system.

Objective:

To ensure that the oxygen and chloride/fluoride in the reactor coolant system do not exceed concentrations detrimental to the functional integrity of the system materials.

Specification:

Whenever the reactor coolant system temperature exceeds 210 F, the following chemical concentrations shall not be exceeded.

A. Oxygen shall not exceed 0.2 ppm.

B. Chloride plus floride shall not exceed 0.3 ppm.

C. Oxygen shall not exceed 0.1 ppm if chloride/floride exceeds 0.15 ppm.

D. Chloride/floride shall not exceed 0.15 ppm if oxygen exceeds 0.1 ppm.

Remedial Action: If the oxygen concentration or the chloride/fluoride concentration of the reactor system exceed the limits specified above, corrective action is to be initiated immediately and continued power operation is permitted for a maximum of 24 hours. If the system is not brought to within specifications in an additional 24 hour period, the system is to be brought to a cold shutdown condition and the cause of the out-of-specification condition accertained and corrected.

Basis:

By maintaining the oxygen and chloride/fluoride concentration in the reactor coolant within the limits as specified above, the functional integrity of the material in the Reactor Coolant System is assured under all operating conditions.

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, and further because of the time dependent nature of any adverse effects arising from oxygen and chloride/fluoride concentration in excess of the limits, it is unnecessary to shutdown immediately since the condition can be corrected. Thus the period of 24 hours for corrective action to restore the concentrations within the limits has been established. If the corrective action has not been effective at the end of the 24 hour period, then the reactor will be brought to the hot shutdown condition and the corrective action will continue. If at the end of a further 24 hour period, the corrective action has not been effective, long term corrective action could be required and the reactor will be brought to the cold shutdown condition.

3.19 SAFETY INJECTION SYSTEM

Applicability:

Applies to the condition of safety injection system isolation and loop stop valves.

Objective:

To define the condition of the safety injection system isolation and loop stop valves required during reactor operation.

Specification:

A. The reactor shall not be critical unless the following conditions are met:

1. The safety injection tank isolation valves (SIA-M-11, 21, 31) shall be disabled in the open position. This shall require the following:

   a. The breakers shall be locked and tagged open.

   b. The disconnect switches for each valve power operator shall be locked and tagged open.

   Exception: One safety injection tank isolation valve may be closed for a period of one hour.

2. The loop isolation valves (RC-M-11, 12, 21, 22, 31, 32) shall be disabled in the open position. This shall require the following:

   a. The breakers shall be locked and tagged open.

   b. The disconnect switches for each valve power operator shall be locked and tagged open.

3. The safety injection header isolation valves (HSI-16, 26, 36) shall not be closed.

4. The following ECCS check valve barriers shall have been determined to be intact in accordance with Technical Specification 4.6.A.2.f.

	Barrier	
Loop 1	a	HSI-17 and HSI-61
	b	LSI-12
Loop 2	a	HSI-27 and HSI-62
	b	LSI-22
Loop 3	a	HSI-37 and HSI-63
	b	LSI-32

Exception: If any of the ECCS check valve barriers specified above do not meet the acceptance criteria of Technical Specification 4.6.A.2.f, then the reactor may be made or remain critical in accordance with the provisions of Specification 4.6.A.2.f.

Basis:
The position restrictions on the loop isolation valves, safety injection header isolation valves, and the safety injection tank isolation valves are necessary to assure that plant operation is restricted to conditions considered in the loss-of-coolant accident analysis.

The three check valves in the ECCS line to each loop provides assurance that a valve failure will not result in unrestricted flow of pressurized reactor coolant into lower pressure connecting piping outside the containment. The valve integrity testing required by Technical Specification 4.6.A.2.f assures that the rate of flow under a valve failure condition will not exceed the pressure relief capacity of the line. It further provides periodic assurance that the check valves are intact.

The two check valves closest to the loop are grouped together as a single check valve barrier for test purposes. The first valve provides a thermal barrier preventing thermal distortion from affecting the tightness of the second valve. The third valve alone constitutes a check valve barrier.

In addition to the check valves the ECCS line to each loop contains a Motor Operated Valve (MOV) which is closed except for periodic monthly testing. The MOV and reactor side piping is designed for full system pressure and also capable of preventing an overpressure condition of connecting piping.

The exception permits time to schedule an orderly shutdown and maintenance of a defective valve while providing assurance that two separate intact barriers always exist.

3.20 SHOCK SUPPRESSORS (SNUBBERS)

Applicability:
Applies to those shock suppressors used on the primary coolant system and on other safety related systems or components.

Objective:
To define the condition of the above defined shock suppressors required for reactor operation.

Specification:

A. During all modes of operation except Cold Shutdown and Refueling, all safety-related snubbers listed on Table 3.20-1 shall be operable.

Remedial Action:

1. If any snubber listed on Table 3.20-1 is found to be inoperable, it must be repaired and made operable, or otherwise replaced with one which is operable within 72 hours.

2. If a snubber is determined to be inoperable while the reactor is in the shutdown or refueling mode, the snubber shall be made operable or replaced prior to reactor startup.

B. Snubbers may be added to safety related systems provided that a revision to Table 3.20-1 is included with the next license amendment.

Basis:
Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety related system or component be operable during reactor operation.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacement. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operation procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.20.A.2 prohibits startup with inoperable snubbers.

3.23 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the operating status of the plant installed fire protection systems.

Objective:

To define the operating status of the installed fire protection systems.

Specification:

A. As a minimum, the smoke detection instrumentation for each of the protected zones shown in Table 3.23-1 shall be operable whenever the equipment in that zone is required to be operable.

With the number of operable sensors in any zone less than required by Table 3.23-1:

1. Within 1 hour, establish a roving fire patrol to inspect the affected zone(s) at least once per hour, and

2. Restore the inoperable instrument(s) to operable status within 14 days, or prepare and submit a Special Report to the Commission within the next 30 days outlining the cause of the malfunction and plans for restoring the instrument to operable status.

   Exception: If the affected zone is Zone 5, the inspection required in A.1 above shall be performed at least once per 8 hours. If the affected zone is either Zone 11, 12 or 13, the inspection required in A.1 above shall be performed at least once per 24 hours, and the RCP pump bearing, winding and air temperatures shall be monitored once per hour.

B. The fire suppression water system shall be operable at all times with: two high pressure pumps each with a capacity of 2500 gpm with their discharge aligned to the fire suppression header, and automatic initiation logic for each pump, and a minimum of 7 ft of water in the fire pond.

Remedial Actions

1. With less than the above required equipment, restore the inoperable equipment to operable status within 7 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

2. With no fire suppression water system operable:

   a. Establish a backup fire suppression water system within 24 hours.

b. Notify the Commission by the telephone or telegraph within 24 hours.

c. Prepare and submit a Special Report to the Commission within the next 14 days outlining the cause of the malfunction and the plans for restoring the system to operable status,

C. The Cardox system shall be operable, with a minimum level of 75% and a minimum pressure of 250 psig in the storage tank, whenever the equipment in the protected areas is required to be operable.

With the Cardox system inoperable:

1. Within 1 hour, establish a roving fire watch to check the affected area(s) at least once per hour, and provide backup fire suppression equipment for the affected area(s).

2. Restore the system to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the cause of inoperability and the plans for restoring the system to operable status.

D. The fire hose stations in the following locations shall be operable whenever the equipment in the area is required to be operable:

Hose House #1 -- West side of RCA Storage Bldg.

Hose House #2 -- Near Demin. Water Storage Tank.

Hose House #3 -- South side of Turbine Bldg.

Hose House #6 -- North side of PAB, near Sta. Serv. Transformers.

The following fire hose stations shall be operable whenever the component cooling water system or service water system is required to be operable:

FS-68, 69, 72, 84, 85----Turbine Building 21 foot elevation
FS-81, 82, 83------------Turbine Building 39 foot elevation

With a hose station inoperable, route additional equivalent capacity hose to the unprotected area from an operable hose station within 1 hour.

E. Penetration fire barriers protecting safety related areas shall be functional at all times.

With a penetration fire barrier non-functional, within one hour a continuous fire watch shall be established on at least one side of the affected penetration.

F. The following spray and/or sprinkler systems shall be operable whenever the component cooling water system or service water system is required to be operable.

1. Turbine Lube Oil Reservoir Sprinkler

2. Seal Oil System Sprinkler

3. Sprinkler System directly under Turbine Building 39 foot elevation.

G. The following spray and/or sprinkler systems shall be operable whenever the diesel generators are required to be operable:

1. Turbine Lube Oil Storage Room Sprinkler

With one or more of the above required spray and/or sprinkler systems inoperable:

1. Within one hour establish a fire watch patrol with backup fire suppression equipment for the unprotected area(s) to inspect the unprotected area(s) at least once per hour.

2. Restore the system(s) to operable status within 14 days or, in lieu of any other report required by Specification 5.9.1, prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system(s) to operable status.

Basis:

Operability of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, sprinkler systems, CO_2, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppresion capability for the continued protection of the nuclear plant.

The functional integrity of the fire barrier penetration seals ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not functional, a fire patrol is required to frequently inspect in the vicinity of the affected seal until the seal is restored to functional status.

TABLE 3.23-1

SMOKE DETECTION INSTRUMENTS

ZONE	LOCATION	MINIMUM OPERABLE SENSORS
1	Service Bldg. Cable Vault	1
2	Protected Cable Tray Room	2
3	Unprotected Cable Tray Room	8
4	Containment Penetration & MCC Room (Outside)	2
5	Containment Penetration Room (Inside)	2
6	Protected Switchgear Room	2
7	Unprotected Switchgear Room	3
8	Diesel Generator Room (DG-1A)	1
9	Diesel Generator Room (DG-1B)	1
10	Computer Room	1
11	Reactor Coolant Pump P-1-1	1
12	Reactor Coolant Pump P-1-2	1
13	Reactor Coolant Pump P-1-3	1

3.24 SECONDARY COOLANT ACTIVITY

Applicability:

Applies to measured maximum activity in the secondary coolant system.

Objective:

To ensure that the secondary coolant activity does not exceed a level commensurate with the safety of the plant personnel and the public.

Specification:

The specific activity of the secondary coolant system shall be less than or equal to 0.10 micro Ci/gram DOSE EQUIVALENT I-131.

Basis:

The limitacions on secondary system specific activity insure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in a steam line rupture. This dose includes that contributed by a 0.1 gpm primary to secondary tube leak in the steam generator of the affected steam line.

Note: The secondary coolant activity surveillance requirements are given in Table 4.2-1, Item 7.

3.25 INSTALLED VENTILATION AND FILTER SYSTEMS

Applicability:

Applies to the operating status of the plant installed ventilation and filter systems.

Objective:

To define the operating status of the installed ventilation and filter systems required for plant operation.

Specification:

A. The containment hydrogen purge system shall be operable whenever the reactor is critical. With the containment hydrogen purge system inoperable, isolate the system and restore the hydrogen system to operable status within 30 days or be in Hot Shutdown within the next 12 hours.

B. One train of control room ventilation shall be operable whenever reactor coolant system temperature and pressure exceed 210°F and 400 psig. Two trains of control room ventilation shall be operable whenever the reactor is critical. With one control room ventilation system inoperable, restore the system to operable status within 14 days or be in Hot Standby.

C. Spent Fuel Pool Ventilation Requirements

1. When irradiated fuel which has decayed less than 60 days is in the spent fuel pool, the spent fuel ventilation system shall be operating and discharging through an HEPA and charcoal adsorber filter train during either:

   a. Fuel movement within the spent fuel pool, or

   b. Crane operation with loads over the spent fuel pool.

2. With C.1 above not satisfied, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

Basis:

The operability of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions.

Each control room ventilation system consists of one recirculation system at 3300 cfm and one breathing air supply system at 40 cfm. The operability of the control room ventilation system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The operability of this system in conjunction with

Basis: (Continued)

control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The operability of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.