Docket File

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

March 2, 1992

Docket No. 50-334
and 50-412
Serial No. BV-92-001

Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: CHANGE TO TECHNICAL SPECIFICATION BASES SECTION 3/4 4.9 PRESSURE/TEMPERATURE LIMITS (TAC NOS. M81395 AND M81396)

By letter dated August 12, 1991, Duquesne Light Company (DLC) proposed a change to the Beaver Valley Power Station, Unit 1 and 2 Technical Specifications (TS) Bases 3/4 4.9. The proposed change would add pressure/temperature limits that would be applicable to reactor coolant loop components when a loop is isolated from the reactor vessel (Figure B 3/4 4-3). The revised Bases would identify the steam generator channel head to tubesheet region as the limiting ferritic component in an isolated loop.

This change was requested to resolve an issue resulting from an event which occurred at Unit 2 on September 7-8, 1990. During this event, the pressure in the isolated portion of Loop C exceeded the limits of TS 3.4.9.1. DLC performed an engineering review of the event and concluded that TS 3.4.9.1 did not apply because the limiting component for that TS is the reactor vessel. The reactor vessel was not affected during the event. However, TS 3.4.9.1 is worded so as to apply to the entire reactor coolant system except the pressurizer, and no recognition is given to the possibility that at times a loop may be isolated from the reactor vessel and may be at substantially different pressure and/or temperature. The addition of Figure B 3/4 4-3 resolves the question as to the correct pressure-temperature limits to apply to the isolated portion of a loop.

To evaluate the pressure-temperature limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

The staff has reviewed the proposed pressure-temperature limits for the isolated loop as shown in proposed Figure B 3/4 4-3. The staff finds that the pressure-temperature limits are conservative and satisfy SRP 5.3.2 and RG 1.99, Rev. 2. Therefore, the staff has no objection to your proposed

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pressure-temperature limits in Bases Section 3/4/ 4.9 and the addition of Figure 3/4 4-3. This resolves Unresolved Issue 50-412/90-01.

Enclosed is a copy of revised Bases pages B 3/4 4-10, B 3/4 4-10a, and B 3/4 4-11 for Beaver Valley, Unit 1 TS, and revised Bases pages B 3/4 4-14, B 3/4 4-14a, and B 3/4 4-15 for Beaver Valley, Unit 2 TS.

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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Mr. J. D. Sieber
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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BEAVER VALLEY POWER STATION, UNIT 1

TECHNICAL SPECIFICATION BASES

Replace the following pages of Appendix A, Technical Specification Bases with the enclosed pages as indicated. The revised pages contain vertical lines indicating the areas of change.

Remove	Insert
B 3/4 4-10	B 3/4 4-10
-	B 3/4 4-10a
B 3/4 4-11	B 3/4 4-11

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4.5-3 to assure compliance with the requirements of Appendix H to 10 CFR 50.

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.

Pressure-temperature limit curves shown in Figure B 3/4 4-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop.

The OPERABILITY of two PORV's or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq$ 275°F. Either PORV has adequate relieving capability to protect the RCS from over-pressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq$ 25°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

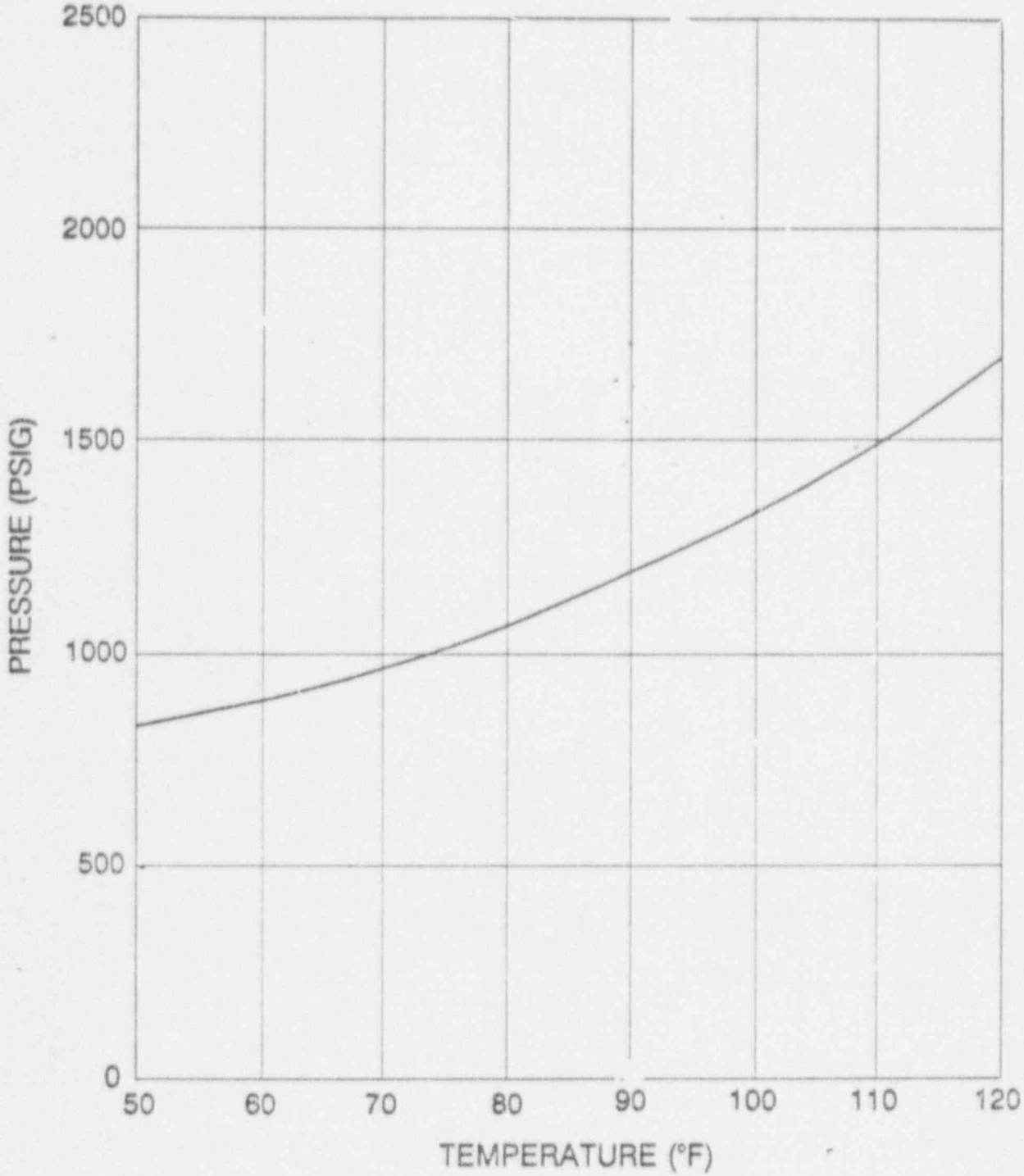

FIGURE B 3/4 4-3

ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE

Added by NRC letter
dated 3/2/92

3/4.4.11 RELIEF VALVES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.12 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space, ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

BEAVER VALLEY POWER STATION, UNIT 2

TECHNICAL SPECIFICATION BASES

Replace the following pages of Appendix A, Technical Specification Bases with the enclosed pages as indicated. The revised pages contain vertical lines indicating the areas of change.

Remove	Insert
B 3/4 4-14	B 3/4 4-14
-	B 3/4 4-14a
B 3/4 4-15	B 3/4 4-15

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature.

During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temeprature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 5.3-6 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary 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.

Pressure-temperature limit curves shown in figure B 3/4 4-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

OVERPRESSURE PROTECTION SYSTEMS

The Maximum Allowed PORV Setpoint for the Overpressure Protection Systems (OPPS) is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that nominal 10 EPPY Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; and (4) single failure.

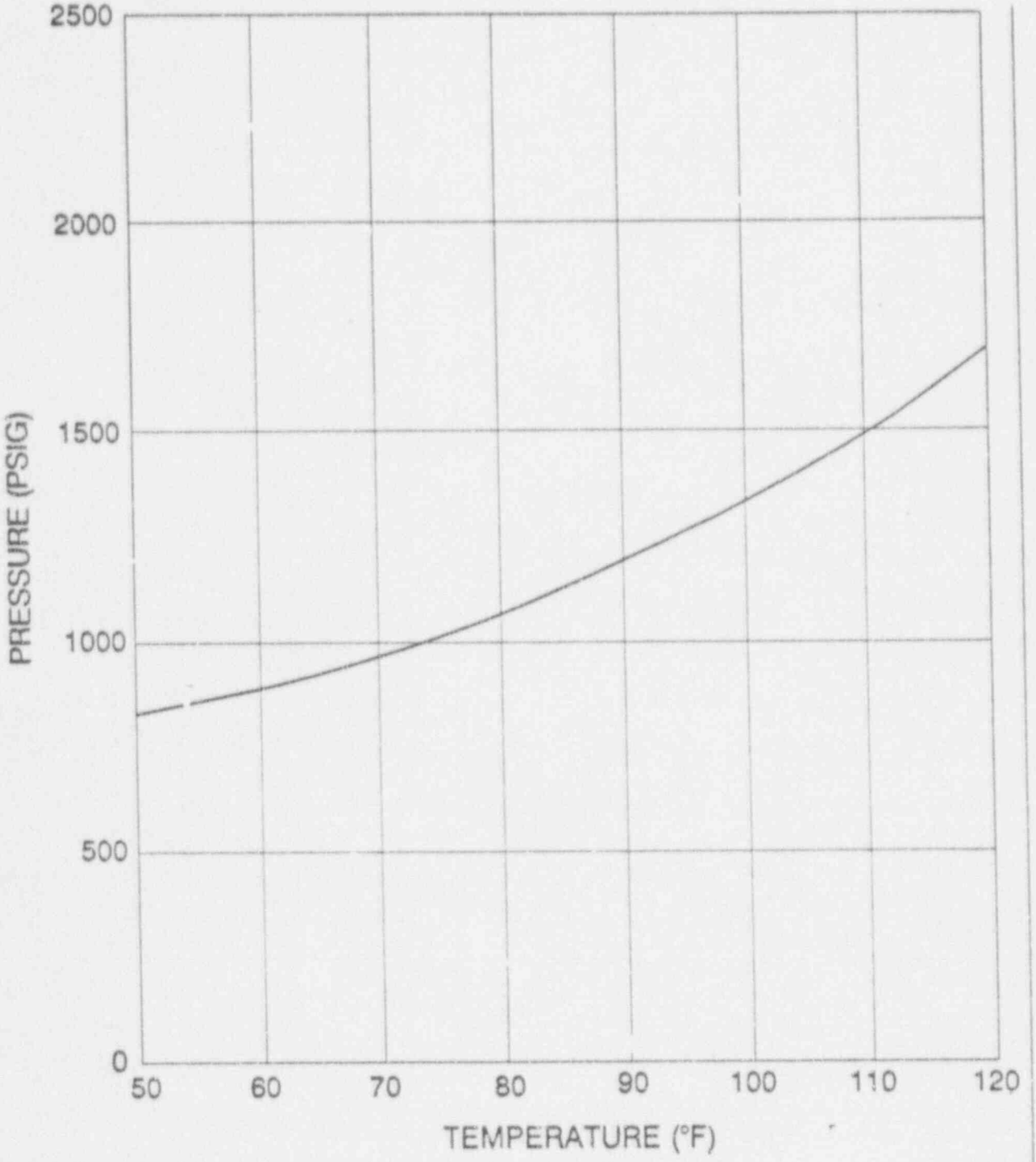

FIGURE B 3/4 4-3

ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE

Added by NRC letter dated 3/2/92

OVERPRSSURE PROTECTION SYSTEMS (Continued)

To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE is allowed for up to 4 hours. As shown by analysis LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single LHSI pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one LHSI pump). For temperatures above 325°F, an overpressure event occurring as a result of starting these two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

The maximum allowed PORV setpoint for the Overpressure Protection System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H and in accordance with the schedule in UFSAR Table 5.3-6.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The operability of at least one PORV will ensure the additional capability to vent the pressurizer steam space via the PORV s.

3/4.4.12 REACTOR COOLANT SYSTEM HEAD VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head or the pressurizer steam space via the PORV's ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System Head vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.