UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 81
License No. DPR-80

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

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated March 27, 1991, as supplemented October 5, 1992 and July 20, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 81 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 23, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. DPR-82

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

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated March 27, 1991, as supplemented October 5, 1992 and July 20, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 80 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of 30 days of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Howard W Clifford FOR

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 23, 1993

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages are also included, as appropriate.

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APPENDIX A TO LICENSE NOS. DPR-80 AND DPR-82
DIABLO CANYON NUCLEAR POWER PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS (NUREG-1151)

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet and two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. For Unit 1, Cycle 6:

Operating the PORV through one complete cycle of full travel, and

For Unit 1, Cycle 7 and after, and Unit 2, Cycle 6 and after:

Operating the PORV through one complete cycle of full travel during MODES 3 or 4 with the block valves closed, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 In addition to the requirements of Specification 4.0.5, each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The safety-related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Isolating and venting the normal air supply, and

b. Verifying that any leakage of the Class 1 Backup Nitrogen System is within its limits, and

c. Operating the PORVs through one complete cycle of full travel.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. Two Class 1 power-operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 323°F, MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned.

ACTION:

a. With one Class 1 PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within the next 8 hours.

b. With one Class 1 PORV inoperable in MODES 5 or 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned, restore the inoperable PORV to operable status within 24 hours or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches within the next 8 hours.

c. With both PORVs inoperable, depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within 8 hours.

d. In the event either the PORVs or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. Two Class 1 power-operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 323°F, MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned.

ACTION:

a. With one Class 1 PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within the next 8 hours.

b. With one Class 1 PORV inoperable in MODES 5 or 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned, restore the inoperable PORV to operable status within 24 hours or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches within the next 8 hours.

c. With both PORVs inoperable, depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within 8 hours.

d. In the event either the PORVs or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each Class 1 PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent shall be verified to be open when the vent is being used for overpressure protection at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise verify the vent pathway every 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. For Unit 1, Cycle 6:

Operating the PORV through one complete cycle of full travel, and

For Unit 1, Cycle 7 and after, and Unit 2, Cycle 6 and after:

Operating the PORV through one complete cycle of full travel during MODES 3 or 4 with the block valves closed, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 In addition to the requirements of Specification 4.0.5, each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The safety-related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Isolating and venting the normal air supply, and

b. Verifying that any leakage of the Class 1 Backup Nitrogen System is within its limits, and

c. Operating the PORVs through one complete cycle of full travel.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet and two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain power to the block valve(s), otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

   1. With only one Class 1 PORV OPERABLE, restore at least a total of two Class 1 PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no Class 1 PORVs OPERABLE, restore at least one Class 1 PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable Class 1 PORV or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour

   1. Restore the block valve(s) to OPERABLE status, or

   2. Close the PORV(s) and remove power from its associated solenoid.

   Also, comply with ACTION b, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generator as a heat removing component require that at least two RHR trains be OPERABLE.

The operation of one reactor coolant pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 323°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown.

SAFETY VALVES (Continued)

In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System (relief valves) provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

The functions of providing an RCS pressure boundary and manual RCS pressure control for mitigation of accidents such as steam generator tube rupture are the safety-related function of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of accidents is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure

RELIEF VALVES (Continued)

protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shut-off capability should a relief valve become inoperable. Operation with the block valves opened is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The OPERABILITY of the PORVs and block valves in MODES 1, 2, and 3 is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for accident mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for accident mitigation.

The non-Class 1 PORV and block valve are used only as a backup to the two redundant Class 1 PORVs and block valves to control RCS pressure for accident mitigation. Therefore continued operation with the non-Class 1 PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their safety functions. Surveillance Requirement 4.4.4.1 addresses the PORVs and Surveillance Requirement 4.4.4.2 addresses the block valves.

Surveillance Requirement 4.4.4.1.a provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS.

Surveillance Requirement 4.4.4.1.b provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in MODES 1, 2, and 3.

RELIEF VALVES (Continued)

Surveillance Requirements 4.4.4.1.c and 4.4.4.1.d provide assurance of operability of the Backup Air/Nitrogen system and that the Backup Air/Nitrogen system is capable of supplying sufficient air to operate the PORV(s) if they are needed for RCS pressure control and normal instrument air is not available.

Surveillance Requirement 4.4.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with ACTION statements "b" or "c." This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure 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-temperature limitations for the case in which a 1/4T 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, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest 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 such that 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.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operation limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of both Class 1 PORVs or an RCS vent opening of at least 2.07 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 less than or equal to 323°F. Either Class 1 PORV has adequate relieving capability to protect the RCS from overpressurization for all anticipated transients.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

The Maximum Allowed PORV Setpoint for the LTOPs will be modified, if required, 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 Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for the 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. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path 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 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.