Mr. Gregory M. Ru[illegible]er
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P. O. Box 770000
San Francisco, California 94106

March 9, [illegible]5

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M90260) AND UNIT NO. 2 (TAC NO. M90261)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. DPR-80 and Amendment No. 97 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated August 17, 1994.

These amendments relocate TS 3/4.4.2.1, "Safety Valves - Shutdown," 3/4.4.7, "Chemistry," 3/4.4.9.2, "Pressurizer (Temperature Limits)," 3/4.4.10, "Structural Integrity," and 3/4.4.11, "Reactor Vessel Head Vents," in accordance with the Commission's Final Policy Statement for relocation of current TS that do not meet any of the screening criteria for retention. As part of the relocation of TS 3/4.4.2.1, TS 3/4.4.2.2, "Safety Valves - Operating," would be revised to require that the pressurizer safety valves be operable in Mode 4 with the reactor coolant system cold-leg temperature greater than the low-temperature overpressure protection system enable temperature, and TS 6.8, "Procedures and Programs," would be revised to include the reactor coolant pump flywheel inspection program.

A copy of the related Safety Evaluation is enclosed. A notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By
Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 98 to DPR-80
2. Amendment No. 97 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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DOCUMENT NAME: DC90260.AMD * See previous concurrence

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

March 9, 1995

Mr. Gregory M. Rueger
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P. O. Box 770000
San Francisco, California 94106

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M90260) AND UNIT NO. 2 (TAC NO. M90261)

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Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 98 to DPR-80
2. Amendment No. 97 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

Mr. Gregory M. Rueger
Pacific Gas and Electric Company

Diablo Canyon

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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. 98
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated August 17, 1994, 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. 98 , are hereby incorporated in the license. Pacific Gas and 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 90 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie A. Miller

Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 9, 1995

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. 97
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated August 17, 1994, 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. 97, are hereby incorporated in the license. Pacific Gas and 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 90 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie A. Miller

Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 9, 1995

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 97 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.

REMOVE	INSERT
vii	vii
viii	viii
xv	xv
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-22	3/4 4-22
3/4 4-23	---
3/4 4-24	---
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3/4 4-38	3/4 4-38
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6-15c	6-15c

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 2, 3 and 4.#

ACTION:

a. With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN with all RCS cold leg temperatures less than or equal to 323°F within the following 6 hours.

b. The provisions of Specification 3.0.4 may be suspended for up to 18 hours per valve for entry into and during operations in MODE 3 and 4# for the purpose of setting the pressurizer Code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#When all RCS cold leg temperatures are greater than 323°F.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
1. 8948 A, B, C, and D	Accumulator, RHR and SIS first off check valves from RCS cold legs
2. 8819 A, B, C, and D	SIS second off check valves from RCS cold legs
3. 8818 A, B, C, and D	RHR second off check valves from RCS cold legs
4. 8956 A, B, C, and D	Accumulator second off check valves from RCS cold legs
5. 8701* and 8702*	RHR suction isolation valves
6. 8949# A, B, C, and D	RHR and SIS first off check valves from RCS hot legs
7. 8905# A, B, C, and D	SIS second off check valves from RCS hot legs
8. 8740# A and B	RHR second off check valves from RCS hot legs
9. 8802*# A and B	SIS to RCS hot legs isolation valves
10. 8703*#	RHR to RCS hot legs isolation valve

*Testing per Specification 4.4.6.2.2c. not required.

#For flowpaths with 3 pressure isolation valves in series, at least 2 of the 3 valves shall meet the requirements of Specification 3.4.6.2f.

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3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Diablo Canyon site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 should be limited since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month. Alternatively, gamma spectroscopy may be used.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Presure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

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

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. Deleted

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI. Allowable pressures and temperatures for inservice leak and hydrostatic tests are given in Figure 3.4-2.

6. The criticality limit on Figure 3.4-2 is based on the minimum allowable temperature of 295°F for an inservice hydrostatic test of 110% of operating pressure.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1966 Edition for Unit 1 and the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil ductility reference temperature, RT_{NDT}, at the end of 8 effective full power years (EFPY) of service life. The 8 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region

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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.

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.

PROCEDURES AND PROGRAMS (Continued)

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31 -day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

a. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR part 50,

9) Limitations on the annual and quarterly doses to MEMBERS OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the RMCP, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ERMP,

PROCEDURES AND PROGRAMS (Continued)

h. Radiological Environmental Monitoring Program

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in the environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

i. Reactor Coolant Pump Flywheel Inspection

Inspect each reactor coolant pump flywheel in accordance with the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter of August 17, 1994, Pacific Gas and Electric Company (or the licensee) submitted a request for changes to the Technical Specifications (TS). The proposed amendments would relocate TS 3/4.4.2.1, "Safety Valves - Shutdown," 3/4.4.7, "Chemistry," 3/4.4.9.2, "Pressurizer (Temperature Limits)," 3/4.4.10, "Structural Integrity," and 3/4.4.11, "Reactor Vessel Head Vents," in accordance with the Commission's Final Policy Statement for relocation of current TS that do not meet any of the screening criteria for retention. These TS limiting conditions for operation would be relocated to Diablo Canyon Nuclear Power Plant Administrative Controls and the Updated Final Safety Analysis Report (UFSAR) by reference, such that future changes to these requirements would be made pursuant to 10 CFR 50.59.

As part of the relocation of TS 3/4.4.2.1, TS 3/4.4.2.2, "Safety Valves - Operating," would be revised to require that the pressurizer safety valves be operable in Mode 4 with the reactor coolant system cold-leg temperature greater than the low-temperature overpressure protection (LTOP) system enable temperature, and TS 6.8, "Procedures and Programs," would be revised to include the reactor coolant pump flywheel inspection program.

2.0 EVALUATION

Section 50.36 of Title 10 of the Code of Federal Regulations established the regulatory requirements related to the content of TS. The rule requires that TS include items in specific categories, including safety limits, limiting conditions for operation, and surveillance requirements; however, the rule does not specify the particular requirements to be included in a plant's TS. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), to determine which design conditions and associated surveillances are "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety" and need to be located in the TS. Briefly, those criteria are (1) detection of abnormal degradation of the reactor coolant pressure boundary, (2) boundary conditions for design basis accidents and transients, (3) primary success paths to prevent or mitigate design basis accidents and

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transients, and (4) functions determined to be important to risk or operating experience.[1] The Commission's Final Policy Statement acknowledged that its implementation may result in the relocation of existing TS requirements to licensee controlled documents and programs.

<u>TS 3/4.4.2.1, "Safety Valves - Shutdown"</u>

The existing requirements in TS 3/4.4.2.1 and TS 3/4.4.2.2 specify limiting conditions for operation (LCOs) and surveillance requirements for reactor coolant system (RCS) safety valves in Modes 4 and 5. The licensee has proposed to combine the requirements for Mode 4 with those for the operating modes by revising "Safety Valves - Operating" to include a requirement that the pressurizer safety valves be operable in Mode 4 with the reactor coolant system (RCS) cold-leg temperature greater than the LTOP system enable temperature. The requirements for RCS safety valve operability and surveillances in Mode 5 would be relocated to plant procedures.

The safety valves, together with the reactor protection system, protect the RCS from being pressurized above its safety limit of 2735 psig. In the event that no safety valves are operable, an operating residual heat removal 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 temperature.

The proposed changes to the TS establish operability and surveillance requirements for the safety valves which are consistent with the RCS overpressure design basis. The existing TS requirements for the safety valves in Mode 5, when the RCS is below the pressure and temperature conditions associated with overpressure transients, do not constitute initial conditions that challenge the RCS or a primary success path to mitigate a design basis accident or transient. Therefore, these requirements do not satisfy the criteria for TS and the licensee has proposed to relocate these provisions to the plant procedures and, because these functions are described in the FSAR, future changes to these requirements would be subject to 10 CFR 50.59.

The staff has concluded that 10 CFR 50.36 does not require inclusion of the RCS safety valve operability and surveillance requirements in Mode 5. Further, the changes to combine the LCOs and surveillance requirements for the safety valves in Modes 1, 2, 3, and 4 is purely administrative. The staff has, therefore, concluded that the proposed changes are acceptable.

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[1] The Commission recently promulgated a proposed change to 10 CFR 50.36, pursuant to which the rule would be amended to codify and incorporate these criteria (59 FR 48180). The Commission's Final Policy Statement specified that Reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip are included in the TS under Criterion 4. In the proposed change to §50.36, the Commission specifically requested public comments regarding application of Criterion 4. Until additional guidance has been developed, Criterion 4 will not been applied to add TS restrictions other than those indicated above.

TS 3/4.4.7, "Chemistry"

The reactor coolant chemistry program provides limits on particular chemical properties of the primary coolant, and surveillance practices to monitor those properties, to ensure that degradation of the reactor coolant pressure boundary is not exacerbated by poor chemistry conditions. However, degradation of the reactor coolant pressure boundary is a long-term process, and there are other, direct means to monitor and correct the degradation of the reactor coolant pressure boundary which are controlled by regulations and TS (e.g., in-service inspection and primary coolant leakage limits). Therefore, requirements related to the chemistry program do not constitute initial conditions that are assumed in any design basis accident or transient related to the RCS integrity, nor does the reactor coolant chemistry program constitute a primary success path or risk-significant safety function warranting TS requirements under the criteria in the Final Policy Statement described above.

The reactor coolant chemistry requirements are maintained in the licensee's Chemistry Program, which is described in the updated Final Safety Analysis Report. Any changes to these chemistry requirements would be evaluated under the licensee's Chemistry Program and, if the changes are determined to involve an unreviewed safety question, the licensee must submit a license amendment to obtain prior NRC review and approval in accordance with 10 CFR 50.59, in accordance with the criteria in the NRC's policy on technical specification improvements.

The staff has concluded, therefore, that requirements for RCS chemistry limits (a) are not specifically required by 10 CFR 50.36 or other regulations, (b) are not required to avert an immediate threat to the public health and safety, and (c) are not necessary because changes that are deemed to involve an unreviewed safety question will require prior NRC approval by a license amendment as provided by 10 CFR 50.59(c).

TS 3/4.4.9.2, "Pressurizer (Temperature Limits)"

Pressure and temperature limits are placed on the pressurizer to be consistent with the requirements of the American Society of Mechanical Engineers (ASME) Code. As described in the Bases for the existing TS requirements, the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, and operational limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements. Pressurizer integrity is a design capability maintained by ASME Code design and component cyclic/transient limit requirements under 10 CFR 50.55a. Further, these limits are associated with long-term effects on the material properties of the pressurizer; therefore, these operational limits are not necessary to ensure immediate protection of the public health and safety.

On this basis, the staff has concluded that the pressurizer pressure and temperature limits need not be controlled by TS because they (a) are adequately controlled by §50.55a and §50.59, (b) are not specifically required by 10 CFR 50.36, and (c) are not required to avert an immediate threat to the public health and safety. Therefore, the proposed changes to the TS are acceptable.

TS 3/4.4.10, "Structural Integrity"

This specification provides inspection requirements for the ASME Code Class 1, 2, and 3 components to ensure that the structural integrity and operational capability 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, in accordance with 10 CFR 50.55a. As part of the relocation of TS 3/4.4.10, TS 6.8 would be revised to include the reactor coolant pump flywheel inspection program, in accordance with the Westinghouse Standard TS (NUREG-1431), which is not covered under §50.55a.

The inspection program associated with the existing TS requirements is performed on systems required to function to mitigate a design basis accident. However, the TS include separate operability and surveillance requirements for these systems. The requirements in TS 3/4.4.10 relate to long-term maintenance of the structural design margins which are not relied on to avert an immediate threat to public health and safety. Structural integrity is a design capability maintained by ASME Code under 10 CFR 50.55a. Further, these limits and surveillance requirements are associated with long-term effects on the material properties; therefore, these operational limits are not necessary to ensure immediate protection of the public health and safety.

On this basis, the staff has concluded that these structural integrity requirements need not be controlled by TS because they (a) are adequately controlled by §50.55a and §50.59, (b) are not specifically required by 10 CFR 50.36, and (c) are not required to avert an immediate threat to the public health and safety. Therefore, the proposed changes to the TS are acceptable.

TS 3/4.4.11, "Reactor Vessel Head Vents"

The RCS vents are provided to exhaust noncondensible gases and/or steam from the RCS that could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long-term cooling, such as a loss-of-coolant accident. The valve redundancy of the RCS vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of the vent valve power supply or control system does not prevent isolation of the vent path. Their function, capabilities, and testing requirements are consistent with the severe core damage assumptions of item II.B.1 of NUREG-0737, "Clarification of Three Mile Island Action Plan Requirements" (November 1980). As such, this capability is not part of a primary success path to mitigate a design basis accident or transient, nor is it relied on to avert an immediate threat to public health and safety. Therefore, the reactor vessel head vents do not satisfy any of the Final Policy Statement criteria and need not be included in the TS.

The staff has concluded, therefore, that requirements for reactor vessel head vents need not be controlled by TS because they (a) are not specifically required by 10 CFR 50.36, (b) are not required to avert an immediate threat to the public health and safety, and (c) are not necessary because changes that are deemed to involve an unreviewed safety question will require prior NRC approval by a license amendment as provided by 10 CFR 50.59(c).

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 51621). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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