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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-275

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
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 June 5, 1991, as supplemented by letter dated May 19, 1992, 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 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. 71 , 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 becomes effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

H. Rood 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 1, 1992

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-323

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
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 June 5, 1991, as supplemented by letter dated May 19, 1992, 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 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. 70 , 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 becomes effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Rood 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 1, 1992

ATTACHMENT TO LICENSE AMENDMENT NOS. 71 AND 70

FACILITY OPERATING LICENSE NOS. DPR-80 AND 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 PAGE	INSERT PAGE
v	v
B 2-1	B 2-1
B 2-1a	B 2-1a
3/4 2-5	3/4 2-5
3/4 2-6	3/4-2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
3/4 2-12	3/4 2-12
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.3 INSTRUMENTATION (continued)

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during steady-state operation, normal operational transients, and anticipated transients is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 for LOPAR fuel and the WRB-2 for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with a 95 percent confidence level that DNB will not occur when the minimum DNBR is at or greater than the DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with a 95 confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Diablo Canyon Units, the design DNBR values are 1.33 and 1.37 for thimble and typical cells, respectively, for LOPAR fuel, and 1.30 for thimble and 1.32 for typical cells for the VANTAGE 5 fuel. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.44 for thimble and 1.48 for typical cells for LOPAR fuel, and 1.68 and 1.71 for thimble and typical cells, respectively, for VANTAGE 5 fuel in performing safety analyses.

The curves in Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis DNBR limits, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on limiting enthalpy hot channel factors, $F^N_{\Delta H}$ for LOPAR fuel and for VANTAGE 5 fuel, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expressions:

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} \, [1 + PF_{\Delta H}(1 - P)] \text{ for LOPAR fuel}$$

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} \, [1 + PF_{\Delta H}(1 - P)] \text{ for VANTAGE 5 fuel}$$

where P is the fraction of RATED THERMAL POWER, and $F^{RTP}_{\Delta H}$ are the limiting enthalpy hot channel factors at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT (COLR) for each fuel type, and where $PF_{\Delta H}$ are the power factor multipliers specified in the COLR.

The 4% measurement uncertainty associated with $F^{N}_{\Delta H}$ is accounted for in the DNBR design limit.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the Setpoints to provide protection consistent with core Safety Limits.

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) < \left[\frac{F_Q^{RTP}}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{F_Q^{RTP}}{0.5}\right] [K(Z)] \text{ for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

and $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z)$ = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limits by:

a. Using the moveable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \le 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RPT}, K(z), and W(z) are specified in the COLR.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or
2. At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \; \frac{F_Q^M(z)}{K(z)}$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

---

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2.c, or

2) $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \frac{F_Q^M(z)}{K(z)} \text{ is not increasing.}$$

f. With the relationship specified in Specification 4.2.2.2.c above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds it limit by the following expression:

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]\right) - 1\right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)}\right]\right) - 1\right\} \times 100 \quad \text{for } P < 0.5$$

2. Either one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in Specification 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(z)$ exceeds its limit, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated.

g. The limits sperified in Specification 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 for four loop operation.

Where:

a. $$R = \frac{F^{N}_{\Delta H}}{F^{RTP}_{\Delta H}\ [1.0 + PF_{\Delta H}\ (1.0 - P)]}$$ for LOPAR fuel

$$R = \frac{F^{N}_{\Delta H}}{F^{RTP}_{\Delta H}\ [1.0 + PF_{\Delta H}\ (1.0 - P)]}$$ For VANTAGE 5 fuel

b. $$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

c. $F^{N}_{\Delta H}$ = Measured values of $F^{N}_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^{N}_{\Delta H}$ shall be used to calculate R since Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include measurement uncertainties of 2.4% for flow and 4% for incore measurement of $F^{N}_{\Delta H}$.

d. $F^{RTP}_{\Delta H}$ = The $F^{N}_{\Delta H}$ limits at Rated Thermal Power (RTP) specified in the Core Operating Limits Report (COLR).

e. $PF_{\Delta H}$ = The Power Factor Multipliers specified in the COLR.

APPLICABILITY: MODE 1.

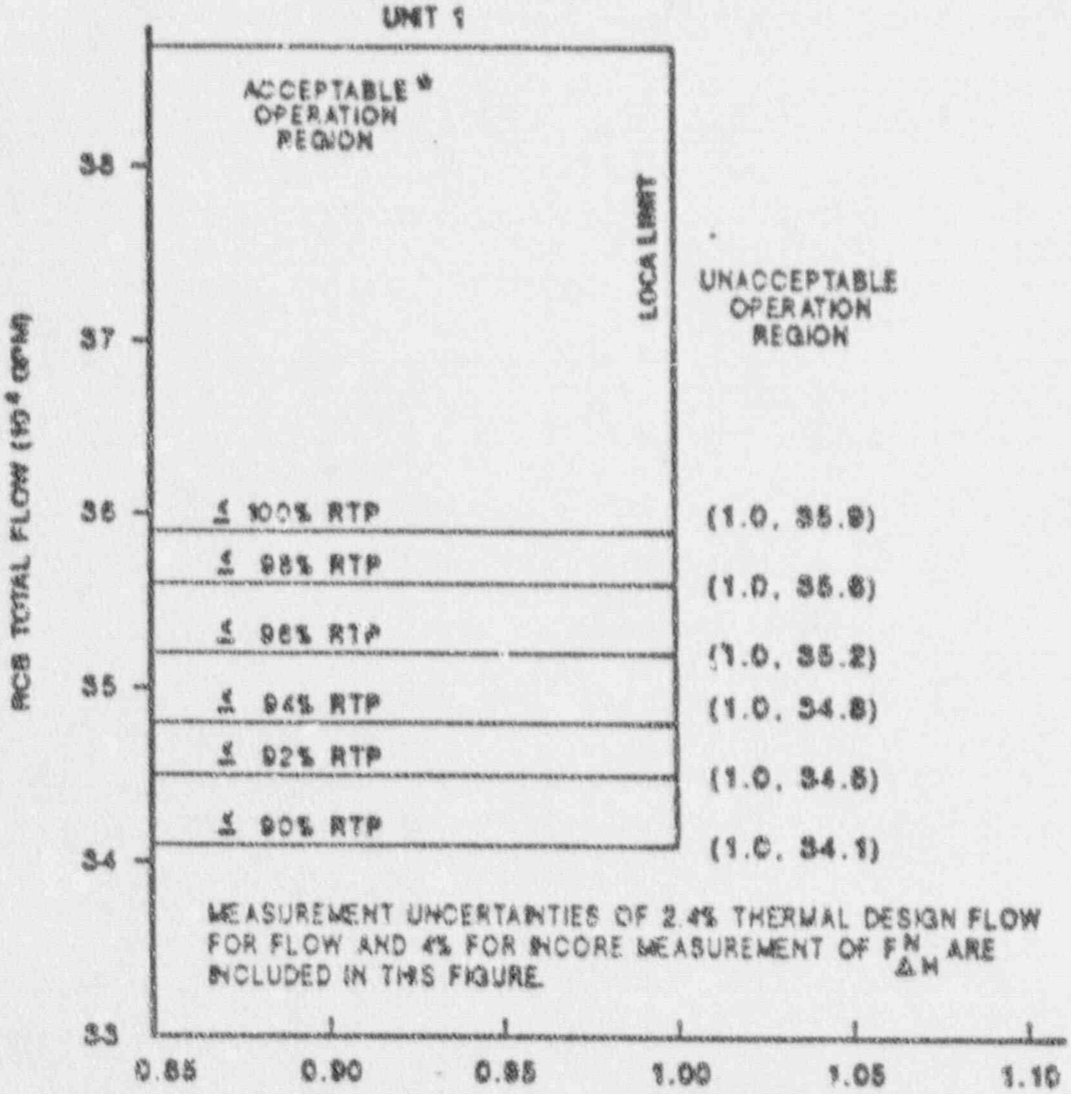
ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.



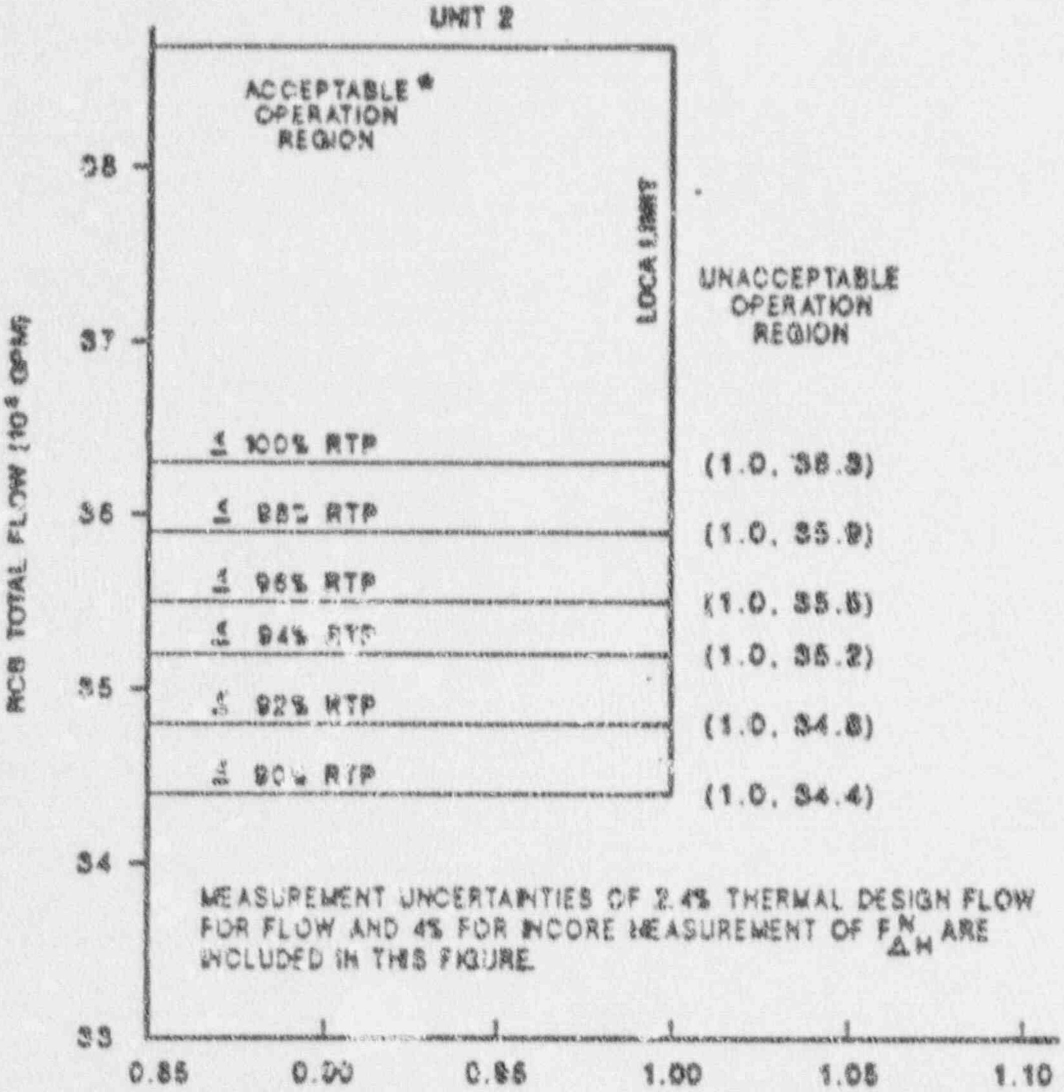
$R = F^N_{\Delta H} / F^{RTP}_{\Delta H}\ (1.0 + PF_{\Delta H}\ (1.0-P))$
LOPAR FUEL

$R = F^N_{\Delta H} / F^{RTP}_{\Delta H}\ (1.0 + PF_{\Delta H}\ (1.0-P))$
VANTAGE 5 FUEL

*WHEN OPERATING IN THE RESTRICTED POWER REGION, THE RESTRICTED POWER LEVEL SHALL BE CONSIDERED 100% RTP FOR FIGURE 2.1-1

FIGURE 3.2-3a

RCS TOTAL FLOWRATE VERSUS R(UNIT 1)



$R = F^N_{\Delta H} / F^{RTP}_{\Delta H} \, (1.0 + PF_{\Delta H} \, (1.0-P))$
LOPAR FUEL

$R = F^N_{\Delta H} / F^{RTP}_{\Delta H} \, (1.0 + PF_{\Delta H} \, (1.0-P))$
VANTAGE 5 FUEL

*WHEN OPERATING IN THE RESTRICTED POWER REGION, THE RESTRICTED POWER LEVEL SHALL BE CONSIDERED 100% RTP FOR FIGURE 2.1-1

FIGURE 3.2-3b

RCS TOTAL FLOWRATE VERSUS R(UNIT 2)

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

For Unit 1 and 2, Cycle 4:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

For Unit 1 and 2, Cycle 5 and after:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for the trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

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HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

1. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

2. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

3. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained, and

4. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions 1. through 4., above, are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated per Specification 3.2.3 and used in Figure 3.2-3a and Figure 3.2-3b accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limits specified in the COLR for LOPAR fuel and for VANTAGE 5 fuel. These values are the values used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus are the maximum "as measured" values allowed.

Margin between the safety analysis limit DNBRs (1.44 and 1.48 for the LOPAR fuel thimble and typical cells, respectively, and 1.68 and 1.71 for the VANTAGE 5 thimble and typical cells) and the design limit DNBRs (1.33 and 1.37 for the LOPAR fuel thimble and typical cells and 1.30 and 1.32 for the VANTAGE 5 fuel thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty of maximum 12.5 percent and the appropriate fuel rod bow DNBR penalty (less than 1.5 percent for both fuel types per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

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rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8 will be included in the annual report. The following information shall be included: (1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of specific activity above the steady-state level; and (5) the time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.5 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the RMCP and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

*This tabulation supplements the requirements of 10 CFR 20.407.

**A single submittal may be made for a multiple unit plant.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.6 The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the RMCP and PCP, (2) in conformance with 10 CFR 50.36a and Section IV.B.1 Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges and failures to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.8.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,
2. Control Rod Insertion Limits for Specification 3/4.1.3.6,
3. Axial Flux Difference for Specification 3/4.2.1,
4. Heat Flux Hot Channel Factor, K(Z) and K(Z) - $F_Q(z)$ (F_Q^{RTP} for Specification 3/4.2.2), and
5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$ ($F_{\Delta H}^{RTP}$ and $PF_{\Delta H}$ for Specification 3/4.2.3).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, June 1983 (Westinghouse Proprietary),
2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary),

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*A single submittal may be made for a multiple unit plant. The submittal should combine those sections that are common to all units at the plant; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary),

4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985. (Westinghouse Proprietary), and

5. WCAP-10266-P-A, Revision 2 with Addenda, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, December 14, 1987. (Westinghouse Proprietary).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and lots of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. ALL REPORTABLE EVENTS;

RECORD RETENTION (Continued)

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications;

e. Records of changes made to procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the PSRC and NSOC;