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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated July 22, 1993, as supplemented by letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   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. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 66 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented prior to startup from refueling outage 4.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Pages 3 and 5 of License*
2. Changes to the Technical Specifications

Date of Issuance: April 28, 1995

*Pages 3 and 5 are attached, for convenience, for the composite license to reflect this change.

ATTACHMENT 1 TO LICENSE AMENDMENT

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise the license as follows:

Remove Pages	Insert Pages
3	3
5	5

at the above designated location in Oswego County, New York, in accordance with the procedures and limitations set forth in this license;

(2) Rochester Gas and Electric Corporation, Central Hudson Gas & Electric Corporation, New York State Electric & Gas Corporation, and Long Island Lighting Company, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess the facility at the designated location in Oswego County, New York, in accordance with the procedures and limitations set forth in this license;

(3) Niagara Mohawk Power Corporation, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Niagara Mohawk Power Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Niagara Mohawk Power Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, in amounts as required, any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Niagara Mohawk Power Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Niagara Mohawk Power Corporation is authorized to operate the facility at reactor core power levels not in excess of 3467 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(6) Initial Startup Test Program (Section 14, SER, SSERs 4 and 5)

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(7) Operation with Reduced Feedwater Temperature (Section 15.1, SSER 4)

Niagara Mohawk Power Corporation shall not operate the facility with reduced feedwater temperature for the purpose of extending the normal fuel cycle. The facility shall not be operated with a feedwater heating capacity less than that required to produce a feedwater temperature of 405°F at rated steady-state conditions unless analyses supporting such operations are submitted by Niagara Mohawk Power Corporation and approved by the staff.

(8) Safety Parameter Display System (SPDS) (Section 18.2, SSERs 3 and 5)

Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall have operational an SPDS that includes the revisions described in their letter of November 19, 1985. Before declaring the SPDS operational, the licensee shall complete testing adequate to ensure that no safety concerns exist regarding the operation of the Nine Mile Point Nuclear Station, Unit No. 2 SPDS.

(9) Detailed Control Room Design Review (Section 18.1, SSERs 5 and 6)

(b) Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall provide the results of the reevaluation of normally lit and nuisance alarms for NRC review in accordance with its August 21, 1986 letter.

(c) Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall complete permanent zone banding of meters in accordance with its August 4, 1986 letter.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70.

ATTACHMENT 2 TO LICENSE AMENDMENT

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-6	1-6
2-3	2-3
B2-4	B2-4
3/4 1-20	3/4 1-20
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-17	3/4 3-17
3/4 3-48	3/4 3-48
3/4 3-52	3/4 3-52
3/4 4-5	3/4 4-5
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B3/4 2-4	B3/4 2-4
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B3/4 6-1	B3/4 6-1
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B3/4 6-3	B3/4 6-3

PRIMARY CONTAINMENT INTEGRITY

1.31 (Continued)

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

PROCESS CONTROL PROGRAM

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of radioactive wastes, based on demonstrated processing of actual or simulated wet or liquid wastes, will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 61, 10 CFR 71, and Federal and State regulations and other requirements governing the transport and disposal of radioactive waste.

PURGE - PURGING

1.33 PURGE and PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3467 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Intermediate Range Monitor, - Neutron Flux - High	≤120/125 divisions of full scale	≤122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux - Upscale, Setdown	≤15% of RATED THERMAL POWER	≤20% of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased 2) High-Flow-Clamped	≤0.58 (W-ΔW)[(a)] + 59%, with a maximum of ≤113.5% of RATED THERMAL POWER	≤0.58 (W-ΔW)[(a)] + 62%, with a maximum of ≤115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - Upscale	≤118% of RATED THERMAL POWER	≤120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1052 psig	≤1072 psig
4. Reactor Vessel Water Level - Low, Level 3	≥159.3 in. above instrument zero*	≥157.8 in. above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤8% closed	≤12% closed
6. Main Steam Line Radiation [(b)] - High	≤3.0 x full-power background	≤3.6 x full-power background
7. Drywell Pressure - High	≤1.68 psig	≤1.88 psig

* See Bases Figure B3/4 3-1.

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW=0 for two loop operation. ΔW=5% for single loop operation.

(b) See footnote (**) to Table 3.3.2-2 for trip setpoint during hydrogen addition test.

BASES TABLE B2.1.2-2

NOMINAL VALUES OF PARAMETERS* USED IN THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT**

PARAMETER	VALUE
THERMAL POWER	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1005 psig
Bundle Enrichment	3.0 Wt % U-235
R-Factor:	
0 - 10 GWD/ST	0.915
10 - 15 GWD/ST	0.954
> 15 GWD/ST	0.954

* The values in this table are for a representative plant.

** The Statistical analysis has been evaluated and shown to be valid at 3467 MW(t) with GE fuel (References: NEDC 31984P, "Generic Evaluations of GE BWR Power Uprate", Volume 1; NEDC-24011-P-A, GESTAR II; and NEDE-31152P, "GE Fuel Bundle Designs").

SURVEILLANCE REQUIREMENTS

4.1.5 (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining that the available weight of sodium pentaborate is greater than or equal to 5500 lb and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1235 psig is met.

d. At least once per 18 months during shutdown by:

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. Demonstrating that the pump relief valve setpoint is less than or equal to 1394** psig and verifying that the relief valve does not actuate during recirculation to the test tank.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

** Bench-tested setpoint value.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the Trip System in the tripped condition provided at least one OPERABLE channel in the same Trip System is monitoring that parameter.

(b) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, and the Refuel position one-rod-out interlock is OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 136.4** psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

* Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** To allow for instrument accuracy, calibration and drift, a setpoint of less than or equal to 125.8 psig turbine first stage pressure shall be used.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1	-	Be in at least HOT SHUTDOWN within 12 hours.
ACTION 2	-	Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 3	-	Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.
ACTION 4	-	Be in at least STARTUP within 6 hours.
ACTION 5	-	Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.
ACTION 6	-	Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than or equal to 136.4* psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.
ACTION 7	-	Verify all insertable control rods to be inserted within 1 hour.
ACTION 8	-	Lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 9	-	Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

* To allow for instrument accuracy, calibration, and drift, a setpoint of less than or equal to 125.8 psig turbine first-stage pressure shall be used.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Primary Containment Isolation Signals (Continued)		
a. Reactor Vessel Water Level*		
1) Low, Low, Low, Level 1	≥17.8 in.	≥10.8 in.
2) Low, Low, Level 2	≥108.8 in.	≥101.8 in.
3) Low, Level 3	≥159.3 in.	≥157.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. Main Steam Line		
1) Radiation - High**	≤3x Full Power Background	≤3.6x Full Power Background
2) Pressure - Low	≥766 psig	≥746 psig
3) Flow - High	≤121.5 psid	≤122.8 psid
d. Main Steam Line Tunnel		
1) Temperature - High	≤167.2°F	≤170.6°F
2) ΔTemperature - High	≤70.0°F	≤71.7°F
3) Temperature - High MSL Lead Enclosure	≤148.2°F	≤151.6°F
e. Condenser Vacuum Low	≥8.5 in Hg vacuum	≥7.6 in. Hg vacuum
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	≤135°F	≤144.5°F
g. Reactor Vessel Pressure - High (RHR Cut-in Permissive)	≤128 psig	≤148 psig
h. SGTS Exhaust - High Radiation	≤5.7x10^{-3} μCi/cc	≤1.0x10^{-2} μCi/cc

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥108.8 in.*	≥101.8 in.
2. Reactor Vessel Pressure - High	≤1065 psig	≤1080 psig

* See Bases Figure B3/4 3-1.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve - Fast Closure	2**

* A Trip System may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other Trip System is OPERABLE.

** This function shall be automatically bypassed when turbine first-stage pressure is less than or equal to 136.4 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of less than or equal to 125.8 psig shall be used.

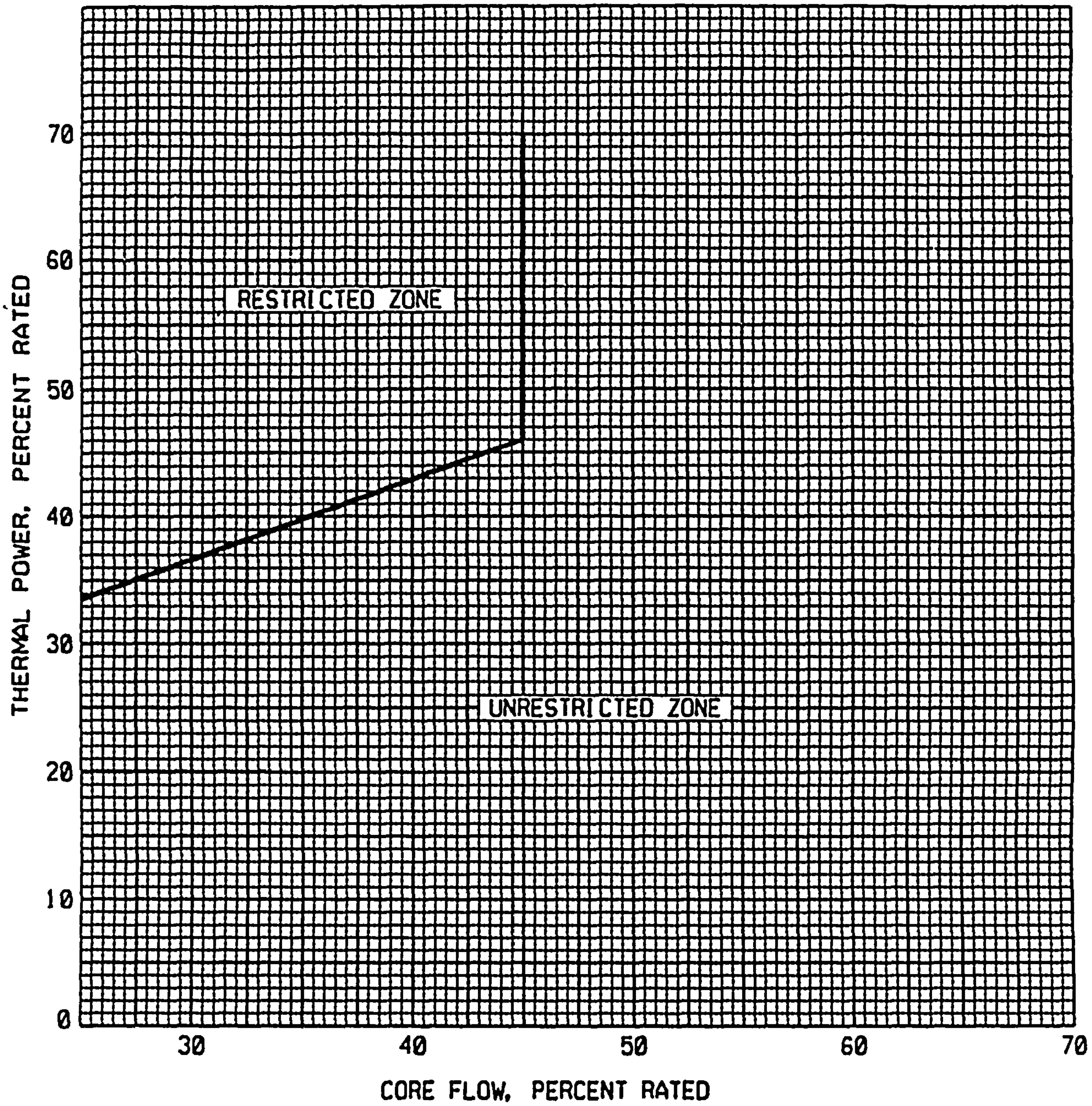

FIGURE 3.4.1.1-1 PERCENT OF RATED CORE THERMAL POWER VS. PERCENT OF RATED CORE FLOW

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITIONS FOR OPERATION

3.4.2 The safety valve function of at least 16 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings*; the acoustic monitor for each OPERABLE valve shall be OPERABLE:

2 safety/relief valves @ 1165 psig ±1%
4 safety/relief valves @ 1175 psig ±1%
4 safety/relief valves @ 1185 psig ±1%
4 safety/relief valves @ 1195 psig ±1%
4 safety/relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required 16 safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that the average water temperature in the suppression pool is less than 110°F, close the stuck-open safety/relief valve(s); if unable to close the open valve(s) within 5 minutes or if the average water temperature in the suppression pool is 110°F or more, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ 1/4 T	WITHDRAWAL TIME (EFPY)
1	3°	0.46	10
2	177°	0.46	20
3	183°	0.46	Spare

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

REACTOR STEAM DOME

LIMITING CONDITIONS FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1035 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1035 psig, reduce the pressure to less than 1035 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1035 psig at least once per 12 hours.

* Not applicable during anticipated transients.

PLANT SYSTEMS

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE, restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to 150 psig or less within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of 600 gpm or more in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1015 + 20, - 80 psig.*

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively, and a description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

CORE OPERATING LIMITS REPORT

6.9.1.9

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

    1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1.

    2) The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint for Specification 3.2.2.

    3) The K_f core flow adjustment factor for Specification 3.2.3.

    4) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

    5) The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.

    6) Control Rod Block Instrumentation Setpoint for the rod block monitor upscale trip setpoint and allowable value for Specification 3.3.6.

    and shall be documented in the CORE OPERATING LIMITS REPORT.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document.

CORE OPERATING LIMITS REPORT

6.9.1.9 (Continued)

1) The GESTR-LOCA and SAFER Models of the Evaluation of the Loss-of-Coolant Accident - SAFER/GESTR Application Methodology, NEDE-23785-1-PA, latest approved revision.

2) General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-US, latest approved revision.

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

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements 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 in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 RECORD RETENTION

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

6.10.1.1 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 logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety

c. All REPORTABLE EVENTS submitted to the Commission

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

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

f. Records of radioactive shipments

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

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

BASES TABLE B3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS*

PARAMETERS | **VALUE**

Plant:

1. Core THERMAL POWER .... 3536 MWt** which is 102% of rated power

2. Vessel Steam Output .... 15.35×10^6 lbm/hr which corresponds to 102.3% of rated steam flow

3. Vessel Steam Dome Pressure .... 1055 psia

4. Design Basis Recirculation Line Break Area for:

   a. Large Breaks .... 3.1 ft^2

   b. Small Breaks .... 0.09 ft^2

Fuel:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO†
Initial Core	8 x 8	13.4	1.4	1.20
Reload	8 x 8	14.4	1.4	1.20

* A more detailed listing of input of each model and its source is presented in Volume II of Reference 1 and subsection 6.3.3 of the USAR.

** This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

† For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 second after LOCA regardless of initial MCPR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

3/4.2.3 (Continued)

while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR, after initially determining that a LIMITING CONTROL ROD PATTERN exists, ensures MCPR will be known following a change in THERMAL POWER or power shape and therefore, operation while exceeding a thermal limit will be avoided.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the linear heat generation rate (LHGR) in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR, after initially determining a LIMITING CONTROL ROD PATTERN exists, ensures that LHGR will be known following a change in THERMAL POWER or power shape and therefore, operation while exceeding a thermal limit will be avoided.

References

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, latest approved revision.

2. The GESTR-LOCA and SAFER Models of the Evaluation at the Loss-of-Coolant Accident - SAFER/GESTR Application Methodology, NEDE 23785-1-PA, latest approved version as identified in COLR.

BASES

ECCS - OPERATING AND SHUTDOWN

3/4.5.1 & 3/4.5.2 (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 517/1550/6350 gpm at differential pressures of 1200/1130/200 psi, respectively. Initially, water from the condensate storage tank is used instead of water injected from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup water at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low-pressure core cooling systems.

The Surveillance Requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires the reactor to be shut down. The pump discharge piping is maintained full to prevent water hammer damage.

Upon failure of the HPCS system to function properly after a small-break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low-pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low-pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety/relief valves although the safety analysis only takes credit for five valves. It is, therefore, appropriate to permit two valves to be out of service for up to 14 days without materially reducing system reliability.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the control room and site boundary radiation doses to within the limits of General Design Criterion (GDC) 19 and 10 CFR 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at a pressure of 39.75 psig, Pa. Updated analysis results in a maximum expected pressure of less than 39.75 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 La during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening. Leak testing of valves in potential bypass leakage pathways is performed at a test pressure of 40.00 psig rather than Pa, 39.75 psig, for consistency with the accident analysis.

The leakage rates specified for the main steam line isolation valves, the main steam drain line isolation valves, and the postaccident sampling system gas sample and return line block valves are used to quantify the maximum amount of primary containment atmosphere that could bypass secondary containment and leak directly to the environment after a design-basis loss-of-coolant accident. These data are used to determine the radiological consequences of this accident and ensure that the resultant doses are within the limits of GDC 19 and 10 CFR 100.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

PRIMARY CONTAINMENT

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the design pressure of 45 psig in the event of a loss-of-coolant accident (LOCA). A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of less than 39.75 psig does not exceed the design pressure of 45.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 4.7 psi. The limit of 14.2 to 15.45 psia for initial positive containment pressure will limit the total pressure to 39.75 psig, which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

In addition, the maximum drywell average air temperature is also the limiting initial condition used to determine the maximum negative differential pressure acting on the drywell and suppression chamber following inadvertent actuation of the containment sprays.

3/4.6.1.7 PRIMARY CONTAINMENT PURGE SYSTEM

The 14-inch drywell and 12-inch suppression chamber supply and exhaust valves are limited to 90 hours of use per 365 days during purge or vent operations in OPERATIONAL CONDITIONS 1, 2, and 3 to meet the requirements of Branch Technical Position CSB 6-4 for valves greater than 8 inches in diameter. The requirement to limit the opening of 2CPS*AOV105, 2CPS*AOV107, 2CPS*AOV109, and 2CPS*AOV110 to 70 degrees, and 2CPS*AOV111 to 60 degrees ensures these valves will close during a LOCA or steam line break accident, and therefore, the site boundary dose guidelines of 10 CFR 100 would not be exceeded in the event of an accident during purging or venting operations.

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PURGE SYSTEM

3/4.6.1.7 (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The leakage limit shall not be exceeded when the leakage rates are determined to be less than or equal to 4.38 scf/hour per 14-inch valve and 3.75 scf/hour per 12-inch valve when pressurized to 39.75 or 40.0 psig, as applicable.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression pool water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression pool water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Because all of the gases in the drywell are purged into the suppression pool air space during a LOCA, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design-basis accident is less than 40 psig, which is below the design pressure of 45 psig. Maximum water volume of 154,794 cubic feet results in a downcomer submergence of 11 feet 0 inch, and the minimum volume of 145,495 cubic feet results in a submergence approximately 18 inches less. The majority of the Bodega Bay tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as detailed in Specification 3.5.3.