August 7, 1978

DISTRIBUTION:
Docket(2)
NRC PDR(2)
L PDR(2)
ORB#4 Reading
VStello
BGrimes
RIngram
EConner
OELD
OI&E(6)
BJones(8)
BScharf(15)
JMcGough
BHarless
DEisenhut
ACRS(16)
OPA, CMiles
DRoss
TAbernathy
JBuchanan
RDiggs
Gray file(2)
4 Xtra

Dockets Nos. 50-317 ✓
and 50-318

Baltimore Gas and Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President - Supply
P. O. Box 1475
Baltimore, Maryland 21203

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 34 and 16 to Facility Operating Licenses Nos. DPR-53 and DPR-69 for the Calvert Cliffs Nuclear Power Plant (CCNPP) Units Nos. 1 and 2, respectively. The amendments are in accordance with your applications dated November 30, 1976, May 17, July 27, and September 19, 1977, and supplemental information dated February 6 and 23, November 30, and December 3, 1976, March 4 and 28, May 3, July 21, August 11 and 19, September 19 and 29, November 10, 1977 and March 9, 1978.

The Unit No. 2 amendment removes the satisfied license conditions for:

- installation of a permanent means of protection against reactor coolant system overpressurization during plant startup and shutdown, and

- installation of a permanent neutron streaming shield.

These amendments also modify the Appendix A Technical Specifications (TS) to incorporate changes proposed in the above-referenced letters by:

- correcting the control element assembly (CEA) drop time (CCNPP Unit No. 2 only);

- imposing a modified water hole peaking factor (CCNPP Unit No. 2 only);

Const. 1

GD

- including a resistance temperature detector (RTD) response time (CCNPP Unit No. 2 only);
- modifying the incore detector operability requirements to be more definitive and to remove unnecessary requirements;
- adding low temperature overpressure protection requirements;
- authorizing the removal from service of both pressurizer safety valves when in Mode 5 provided an adequate relief pathway is provided;
- specifying surveillance requirements for safety injection throttle valve positions;
- modifying the secondary water chemistry requirements; and
- clarifying the surveillance requirements for spent fuel pool ventilation system testing.

We have deferred the evaluation of your request to reduce the lift setting low tolerance on the steam line safety valves to allow plant startup of both units with inoperable CEA reed switch position indicators and to change the Unit No. 1 acceptance criteria for individual containment structure tendons (Items 4, 6 and 7 of your July 27, 1977 letter).

Some of your proposed TS changes have been modified to meet our requirements. These modifications have been discussed with and accepted by your staff.

Copies of the Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,
Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 34 to License No. DPR-53
2. Amendment No. 16 to License No. DPR-69
3. Safety Evaluation
4. Notice

STS	EB	RSB:DOR
JMcGough*	VNoonan*	PCheck*
7/[illegible]/78	7/ /78	7/ /78

cc w/enclosures: See next page *SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB#4:DOR	ORB#4:DOR	ORB#1:DOR	OELD	C-ORB#4:DOR	
SURNAME	RIngram*	MConner:dn*	GZech*	C Woodhead	RWReid	DEisenhut
DATE	7/ /78	7/ /78	7/ /78	7/11/78	8/ 7/78	7/28/78

- including a resistance temperature detector (RTD) response time (CCNPP Unit No. 2 only);
- modifying the incore detector operability requirements to be more definitive and to remove unnecessary requirements;
- adding low temperature overpressure protection requirements;
- authorizing the removal from service of both pressurizer safety valves when in Mode 5 provided an adequate relief pathway is provided;
- specifying surveillance requirements for safety injection throttle valve positions;
- modifying the secondary water chemistry requirements; and
- clarifying the surveillance requirements for spent fuel pool ventilation system testing.

We have deferred the evaluation of your request to reduce the lift setting low tolerance on the steam line safety valves to allow plant startup of both units with inoperable CEA reed switch position indicators and to change the Unit No. 1 acceptance criteria for individual containment structure tendons (Items 4, 6 and 7 of your July 27, 1977 letter).

Some of your proposed TS changes have been modified to meet our requirements. These modifications have been discussed with and accepted by your staff.

Copies of the Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. to License No. DPR-53
2. Amendment No. to License No. DPR-69
3. Safety Evaluation
4. Notice

STS JMcGough	EB	RSB:DOR
JMcGough*	VNoonan*	PCheck*
7/ /78	7/ /78	7/ /78

cc w/enclosures: See next page *SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE➤	ORB#4:DOR	ORB#4:DOR	ORB#1:DOR	OELD	C-ORB#4:DOR	
SURNAME➤	RIngram*	MConner:dn*	GZech*		RWReid	
DATE➤	7/ /78	7/ /78	7/ /78	7/ /78	7/ /78	

- including a resistance temperature detector (RTD) response time (CCNPP Unit No. 2 only);
- modifying the incore detector operability requirements to be more definitive;
- adding low temperature overpressure protection requirements;
- authorizing the removal from service of both pressurizer safety valves when in Mode 5;
- specifying surveillance requirements for safety injection throttle valve positions;
- reducing the lift setting low tolerance on the steam line safety valves;
- modifying the secondary water chemistry requirements; and
- clarifying the surveillance requirements for spent fuel pool ventilation system testing.

We have deferred the evaluation of your request to allow plant startup of both units with inoperable CEA reed switch position indicators and to change the Unit No. 1 acceptance criteria for individual containment structure tendons (Items 6 and 7 of your July 27, 1977 letter).

Some of your proposed TS changes have been modified to meet our requirements. These modifications have been discussed with and accepted by your staff.

Copies of the Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No.   to
   License No. DPR-53
2. Amendment No.   to
   License No. DPR-69
3. Safety Evaluation
4. Notice

STS	EB V. Noonan	RSB:DOR
JMcGough	~~LShao~~	~~RBaer~~ PCheck
6/ /78*	6/12/78	6/9/78

~~cc w/enclosures: See next page~~

OFFICE➤	ORB#4:DOR	ORB#4:DOR	ORB#1:DOR	OELD	C-ORB#4:DOR	
SURNAME➤	RIngram	for EConner:dn	GZech		RWReid	
DATE➤	6/12/78	6/5/78	6/21/78	6/11/78	6/ /78	

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

cc w/enclosures:
James A. Biddison, Jr.
General Counsel
G and E Building
Charles Center
Baltimore, Maryland 21203

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. R. C. L. Olson
Baltimore Gas and Electric Company
Room 922 - G and E Building
Post Office Box 1475
Baltimore, Maryland 21203

Mr. R. M. Douglass, Chief Engineer
Calvert Cliffs Nuclear Power Plant
Baltimore Gas and Electric Company
Lusby, Maryland 20657

Bechtel Power Corporation
ATTN: Mr. J. C. Judd
Chief Nuclear Engineer
15740 Shady Grove Road
Gaithersburg, Maryland 20760

Combustion Engineering, Inc.
ATTN: Mr. P. W. Kruse, Manager
Engineering Services
P. O. Box 500
Windsor, Connecticut 06095

Calvert County Library
Prince Frederick, Maryland 20678

Mr. Bernard Fowler
President, Board of County
Commissioners
Prince Frederick, Maryland 20768

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
Sixth & Walnut Streets
Philadelphia, Pennsylvania 19106

cc w/4 cys. enclosures and 1 cy
of BG&E filings dtd.:
5/17, 7/27 & 9/19/77
Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
Annapolis, Maryland 21401

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

BALTIMORE GAS & ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
License No. DPR-53

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

   A. The applications for amendment by Baltimore Gas & Electric Company (the licensee) dated November 30, 1976, May 17, 1977, July 27, 1977, and September 19, 1977, as supplemented, comply 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 I;

   B. The facility will operate in conformity with the applications, 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-53 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

V
3/4 3-29
3/4 3-30
3/4 4-2
3/4 4-3
3/4 4-26a (added)
3/4 4-26b (added)
3/4 5-5
3/4 5-5a (added)
3/4 5-6

3/4 7-10
3/4 7-11
3/4 7-12
3/4 9-15
B 3/4 4-1
B 3/4 4-2
B 3/4 4-11
B 3/4 5-2
B 3/4 7-3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AZIMUTHAL POWER TILT:

   At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

   1. At least 75% of all incore detector segments,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate:

   1. At least 75% of all incore detector locations,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detector location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

An OPERABLE quadrant symmetric incore detector segment group shall consist of a minimum of three OPERABLE rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the AZIMUTHAL POWER TILT,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the AZIMUTHAL POWER TILT.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6*.

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 80% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

b. With two reactor coolant pumps in opposite loops not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 51.1% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in opposite loops:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

c. With two reactor coolant pumps in the same loop not in operation, STARTUP and/or continued POWER OPERATION may proceed provided the water level in both steam generators is maintained above the Steam Generator Water Level-Low trip setpoint, the THERMAL POWER is restricted to ≤ 46.8% of RATED THERMAL POWER,

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in the same loop:

1. Power Level-High
2. Reactor Coolant Flow-Low
3. Thermal Margin/Low Pressure
4. Axial Flux Offset

MODES 3, 4#** and 5#**:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump.* The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour to accommodatetransition between shutdown cooling pump and reactor coolant pump operation, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

**A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures ≤ 275°F unless 1) the pressurizer water volume is less than 600 cubic feet or 2) the secondary water temperature of each steam generator is less than 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if switch is made while operating, or

b. Prior to reactor criticality if switch is made while shutdown.

#See Special Test Exception 3.10.5

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 At least one of the following pressurizer code safety valves shall be OPERABLE:*

Valve	Lift Settings (±1%)
RC-200	2500 psia
RC-201	2565 psia

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Both valves may be removed in MODE 5 provided at least one valve is replaced by a spool piece which allows the pressurizer to relieve directly to the quench tank.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 The following pressurizer code safety valves shall be OPERABLE:

Valve	Lift Settings (±1%)
RC-200	2500 psia
RC-201	2565 psia

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of $\leq$ 450 psig, or

b. A reactor coolant system vent of $\geq$ 1.3 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is $\leq$ 275°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a $\geq$ 1.3 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a $\geq$ 1.3 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

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

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

d. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 75 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 0.6 $\pm$ 0.1 lbs of TSP from a TSP storage basket is submerged, without agitation, in 80 $\pm$ 5 gallons of 77 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

g. By verifying the correct position of each electrical position stop for the following Emergency Core Cooling System throttle valves:

1. During each performance of valve cycling required by Specification 4.0.5 by observation of valve position on the control boards.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 4 hours following completion of maintenance on the valve or its operator by measurement of stem travel when the ECCS subsystems are required to be OPERABLE.

HPSI SYSTEM

Valve Number	Valve Number
MOV-616	MOV-617
MOV-626	MOV-627
MOV-636	MOV-637
MOV-646	MOV-646

h. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates:

HPSI System
Single Pump

170 ± 5 gpm to each injection leg.

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One[#] OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All high-pressure safety injection pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is ≤ 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

*With pressurizer pressure < 1750 psia.

#A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is ≤ 275°F.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN with the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 3.6 seconds when tested pursuant to Specification 4.0.5.

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-3.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the feedwater cation conductivity exceeding its limit, restore the conductivity to within the limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With the pH of the blowdown from any steam generator exceeding its limit, restore the pH to within its limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

c. With the total specific conductivity of the blowdown from any steam generator exceeding its limit, restore the conductivity to within its limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.7-3.

TABLE 3.7-3

SECONDARY WATER CHEMISTRY LIMITS

Water Sample Location	Total Cation Conductivity µmhos/cm² @ 25°C Limit	Total Specific Conductivity µmhos/cm² @ 25°C Limit	pH @ 25°C Limit
Feedwater	$\leq$ 0.5*	N.A.	N.A.
Steam Generator Blowdown	N.A.	$\leq$ 7**	7.5 $\leq$ pH $\leq$ 9.5***

*This limit may be exceeded for up to 96 hours provided the Total Cation Conductivity does not exceed 1.5 µmhos/cm² @ 25°F for more than 48 hours.

**This limit may be exceeded for up to 96 hours provided the Total Specific Conductivity does not exceed 20 µmhos/cm² @ 25°C for more than 48 hours.

***During startup from wet layup condition, the pH limit may be exceeded for up to 96 hours, provided this limit is not exceeded for more than 48 hours while in MODE 1.

TABLE 4.7-3

SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENTS

Water Sample Location	Parameters	Sample and Analysis Frequency
Feedwater	Total Cation Conductivity	At least once per 24 hours
Steam Generator Blowdown*	Total Specific Conductivity	At least once per 24 hours
Steam Generator Blowdown*	pH	At least once per 24 hours

* If blowdown from any one steam generator is secured (not in operation), a sample of the bulk water in the affected steam generator shall be analyzed for specific conductivity and pH at least once per 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that each exhaust fan maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The proviisons of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, Thermal Margin/Low Pressure and Axial Flux Offset trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs ≤ 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 7.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 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 on or more of the RCS cold legs are ≤ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for the ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Spectral Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 25%, 50%, 75% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

3/4.4.12 LETDOWN LINE EXCESS FLOW

This specification is provided to ensure that the bypass valve for the excess flow check valve in the letdown line will be maintained closed during plant operation. This bypass valve is required to be closed to ensure that the effects of a pipe rupture downstream of this valve will not exceed the accident analyses assumptions.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basesment is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

The secondary water chemistry program is designed to provide maximum protection to both the steam generator and secondary system internals. The most damaging chemical reactants enter the system via condenser cooling water ingress. Accumulation of these impurities in the steam generators may lead to loss of metallurgical integrity and/or eventual component failure. The limits presented in Table 3.7-3 are those prescribed by the NSSS supplier as "limited-operation" chemistry parameters and are consistent with the most recent industry standards. By routine monitoring of these parameters, plant personnel are able to rapidly detect and limit the duration of ingress of chemically detrimental species and thereby maintain steam generator tube integrity.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 SALT WATER SYSTEM

The OPERABILITY of the salt water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the ECCS pump room exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the

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

BALTIMORE GAS & ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. DPR-69

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

   A. The applications for amendment by Baltimore Gas & Electric Company (the licensee) dated November 30, 1976, May 17, 1977, July 27, 1977, and September 19, 1977, as supplemented, comply 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 I;

   B. The facility will operate in conformity with the applications, 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, Facility Operating License No. DPR-69 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   A. Revise paragraph 2.C.2. to read as follows:

      2. <u>Technical Specifications</u>

         The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   B. Delete in their entirety paragraphs 2.C.5.c.(1) and (2).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

V
3/4 1-23
3/4 2-2
3/4 2-8
3/4 3-6
3/4 3-29
3/4 3-30
3/4 4-2
3/4 4-3
3/4 4-27a (added)
3/4 4-27b (added)
3/4 5-5
3/4 5-5a (added)
3/4 5-6

3/4 7-10
3/4 7-11
3/4 7-12
3/4 9-15
B 3/4 2-2
B 3/4 4-1
B 3/4 4-2
B 3/4 4-11
B 3/4 5-2
B 3/4 7-3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 2.5 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the limits of Figure 3.2-2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the following expression:

SURVEILLANCE REQUIREMENTS (Continued)

$$\frac{L}{17.085} \times M \times N$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

3. N is the maximum allowable fraction of RATED THERMAL POWER as determined by Figure 3.2-3 of Specification 3.2.2.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.10,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion,

   5. A THERMAL POWER measurement uncertainty factor of 1.02, and

   6. A water hole peaking factor of 1.005.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F_{xy}^T is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F_{xy}^T is required and the value of T_q used to determine F_{xy}^T shall be the measured value of T_q.

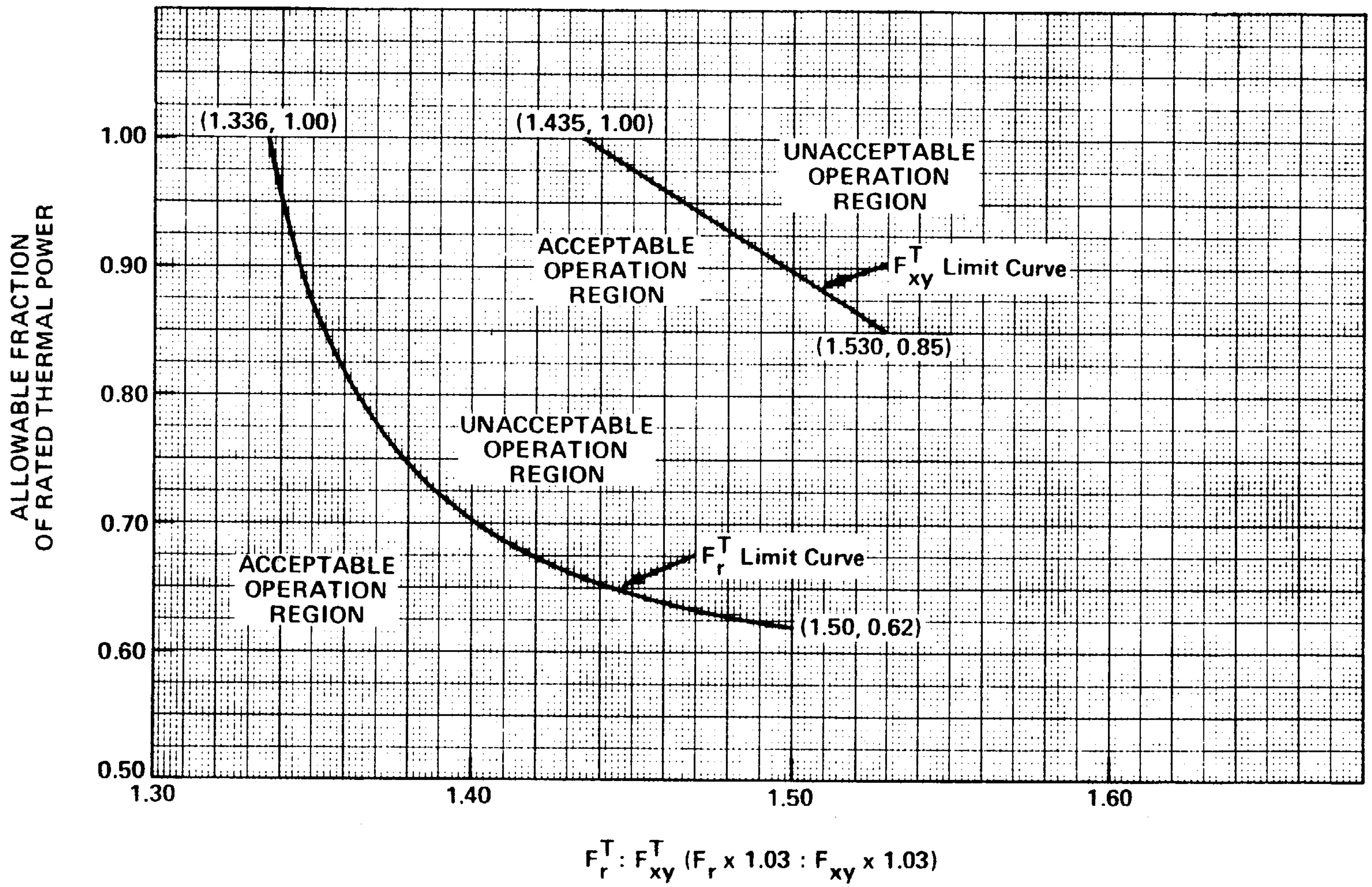

FIGURE 3.2-3

Total Radial Peaking Factor Versus Allowable Fraction of RATED THERMAL POWER

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	≤ 0.40 seconds*# and ≤ 5.0 seconds##
3. Reactor Coolant Flow - Low	< 0.65 seconds
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Containment Pressure - High	≤ 0.90 seconds
6. Steam Generator Pressure - Low	≤ 0.90 seconds
7. Steam Generator Water Level - Low	≤ 0.90 seconds
8. Axial Flux Offset	≤ 0.40 seconds*# and ≤ 5.0 seconds##
9. Thermal Margin/Low Pressure	≤ 0.90 seconds*# and ≤ 5.0 seconds##
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

#Response time does not include contribution of RTDs.

##RTD response time only. This value is equivalent to the time interval required for the RTDs output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AZIMUTHAL POWER TILT:

   At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

   1. At least 75% of all incore detector segments,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate:

   1. At least 75% of all incore detector locations,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detector location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

An OPERABLE quadrant symmetric incore detector segment group shall consist of a minimum of three OPERABLE rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the AZIMUTHAL POWER TILT,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the AZIMUTHAL POWER TILT.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6*.

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 80% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

b. With two reactor coolant pumps in opposite loops not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 51.1% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in opposite loops:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

c. With two reactor coolant pumps in the same loop not in operation, STARTUP and/or continued POWER OPERATION may proceed provided the water level in both steam generators is maintained above the Steam Generator Water Level-Low trip setpoint, the THERMAL POWER is restricted to ≤ 46.8% of RATED THERMAL POWER,

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in the same loop:

1. Power Level-High
2. Reactor Coolant Flow-Low
3. Thermal Margin/Low Pressure
4. Axial Flux Offset

MODES 3, 4#** and 5#**:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump.* The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour to accommodate transition between shutdown cooling pump and reactor coolant pump operation, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

**A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures $\leq$ 275°F unless 1) the pressurizer water volume is less than 600 cubic feet or 2) the secondary water temperature of each steam generator is less than 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if switch is made while operating, or

b. Prior to reactor criticality if switch is made while shutdown.

#See Special Test Exception 3.10.5.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 At least one of the following pressurizer code safety valves shall be OPERABLE:*

Valve	Lift Settings (±1%)
RC-200	2500 psia
RC-201	2565 psia

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Both valves may be removed in MODE 5 provided at least one valve is replaced by a spool piece which allows the pressurizer to relieve directly to the quench tank.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 The following pressurizer code safety valves shall be OPERABLE:

Valve	Lift Settings (±1%)
RC-200	2500 psia
RC-201	2565 psia

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 400°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of $\leq$ 450 psig, or

b. A reactor coolant system vent of $\geq$ 1.3 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is $\leq$ 275°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a $\geq$ 1.3 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a $\geq$ 1.3 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

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

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

d. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 75 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 0.6 $\pm$ 0.1 lbs of TSP from a TSP storage basket is submerged, without agitation, in 80 $\pm$ 5 gallons of 77 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

g. By verifying the correct position of each electrical position stop for the following Emergency Core Cooling System throttle valves:

1. During each performance of valve cycling required by Specification 4.0.5 by observation of valve position on the control boards.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 4 hours following completion of maintenance on the valve or its operator by measurement of stem travel when the ECCS subsystems are required to be OPERABLE.

HPSI SYSTEM

Valve Number	Valve Number
MOV-616	MOV-617
MOV-626	MOV-627
MOV-636	MOV-637
MOV-646	MOV-646

h. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates:

HPSI System
Single Pump

170 $\pm$ 5 gpm to each injection leg.

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One# OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All high-pressure safety injection pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is ≤ 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

*With pressurizer pressure < 1750 psia.

#A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is ≤ 275°F.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN with the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 3.6 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEM

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-3.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the feedwater cation conductivity exceeding its limit, restore the conductivity to within the limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With the pH of the blowdown from any steam generator exceeding its limit, restore the pH to within its limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

c. With the total specific conductivity of the blowdown from any steam generator exceeding its limit, restore the conductivity to within its limit within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.7-3.

TABLE 3.7-3

SECONDARY WATER CHEMISTRY LIMITS

Water Sample Location	Total Cation Conductivity μmhos/cm^2 @ 25°C	Total Specific Conductivity μmhos/cm^2 @ 25°C	pH @ 25°C
	Limit	Limit	Limit
Feedwater	$\leq$ 0.5*	N.A.	N.A.
Steam Generator Blowdown	N.A.	$\leq$ 7**	7.5 $\leq$ pH $\leq$ 9.5***

*This limit may be exceeded for up to 96 hours provided the Total Cation Conductivity does not exceed 1.5 μmhos/cm^2 @ 25°C for more than 48 hours.

**This limit may be exceeded for up to 96 hours provided the Total Specific Conductivity does not exceed 20 μmhos/cm^2 @ 25°C for more than 48 hours.

***During startup from wet layup condition, the pH limit may be exceeded for up to 96 hours, provided this limit is not exceeded for more than 48 hours while in MODE 1.

TABLE 4.7-3

SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENTS

Water Sample Location	Parameters	Sample and Analysis Frequency
Feedwater	Total Cation Conductivity	At least once per 24 hours
Steam Generator Blowdown*	Total Specific Conductivity	At least once per 24 hours
Steam Generator Blowdown*	pH	At least once per 24 hours

* If blowdown from any one steam generator is secured (not in operation), a sample of the bulk water in the affected steam generator shall be analyzed for specific conductivity and pH at least once per 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that each exhaust fan maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

REFUELING OPERATIONS

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The proviisons of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.3 are satisfied, and 4) the TOTAL RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.10, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal

Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F^T_{xy} = F_{xy} (1 + T_q)$ and $F^T_r = F_r (1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F^T_{xy}, F^T_r and T_q are within their limits provide assurance that the actual values of F^T_{xy}, F^T_r and T_q do not exceed the assumed values. Verifying F^T_{xy} and F^T_r after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.4 FUEL RESIDENCE TIME

The limitation on fuel burnup during the initial fuel cycle insures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur until well beyond the proposed first cycle of operation which is about 525 Effective Full Power Days. However, operation beyond the first cycle will require further analyses.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, Thermal Margin/Low Pressure and Axial Flux Offset trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq$ 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 46°F (34°F)when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 7.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to

BASES

limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 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 ≤ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for the ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Spectral Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 25%, 50%, 75% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

3/4.4.12 LETDOWN LINE EXCESS FLOW

This specification is provided to ensure that the bypass valve for the excess flow check valve in the letdown line will be maintained closed during plant operation. This bypass valve is required to be closed to ensure that the effects of a pipe rupture downstream of this valve will not exceed the accident analyses assumptions.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basesment is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

The secondary water chemistry program is designed to provide maximum protection to both steam generator and secondary system internals. The most damaging chemical reactants enter the system via condenser cooling water ingress. Accumulation of these impurities in the steam generators may lead to loss of metallurgical integrity and/or eventual component failure. The limits presented in Table 3.7-3 are those prescribed by the NSSS supplier as "limited-operation" chemistry parameters and are consistent with the most recent industry standards. By routine monitoring of these parameters, plant personnel are able to rapidly detect and limit the duration of ingress of chemically detrimental species and thereby maintain steam generator tube integrity.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 SALT WATER SYSTEM

The OPERABILITY of the salt water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the ECCS pump room exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 34 AND 16 TO

FACILITY OPERATING LICENSES NOS. DPR-53 AND DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-317 AND 50-318

1.0 Introduction

By applications dated November 30, 1976, May 17, July 27, and September 19, 1977, and supplemental information dated February 6 and 23, November 30 and December 3, 1976, March 4 and 28, May 3, July 21, August 11 and 19, September 19 and 29, November 10, 1977 and March 9, 1978, Baltimore Gas and Electric Company (BG&E or the licensee) complied with Unit No. 2 license conditions 2.C.5.c(1) and (2) and requested changes to the Technical Specifications (TS) for Calvert Cliffs Nuclear Power Plant (CCNPP) Units Nos. 1 and 2.

The CCNPP Unit No. 2 license conditions being evaluated are:

- installation of a permanent means of protection against reactor coolant system overpressurization during plant startup and shutdown; and
- installation of a permanent neutron streaming shield.

The proposed changes to the TS consist of:

- correcting the control element assembly (CEA) drop time (CCNPP Unit No. 2 only);
- imposing a modified water hole peaking factor (CCNPP Unit No. 2 only);
- including a resistance temperature detector (RTD) response time (CCNPP Unit No. 2 only);
- modifying the incore detector operability requirements to be more definitive and to remove unnecessary requirements;

- adding low temperature overpressure protection requirements;
- authorizing the removal from service of both pressurizer safety valves when in Mode 5 provided an adequate relief pathway is provided.
- specifying surveillance requirements for safety injection throttle valve positions;
- modifying the secondary water chemistry requirements; and
- clarifying the surveillance requirements for spent fuel pool ventilation system testing.

We have deferred the evaluation of BG&E's requests to reduce the lift setting low tolerance on the steam line safety valves to allow plant startup of both units with an inoperable CEA reed switch position indicator and to change the Unit No. 1 acceptance criteria for individual containment structure tendons (Items 4, 6 and 7 of the Reference 13 application).

2.0 Discussion and Evaluation

2.1 Neutron Shielding

2.1.1 Background

During the power ascension program for CCNPP Unit No. 1 in January 1975, BG&E extrapolated the low power level neutron radiation exposure readings to values greater than the design values for 100% power level operation. Following discussions with us, BG&E proposed to use containers of demineralized water for a temporary neutron shield for Unit No. 1 through the second scheduled refueling and for Unit No. 2 for the first cycle.[1,2] We accepted this proposal in April 1976;[3] and, as license condition 2.C.5.c(2) this commitment was documented in the CCNPP Unit No. 2 license issued in November, 1976.

In May 1977,[9] BG&E notified us that after actively pursuing the design of a permanent neutron shield, they had concluded that the installed temporary shield was adequate and should be considered permanent.

2.1.2 Structural Design Analysis

The proposed reactor cavity neutron shield consists of containers holding demineralized water. The containers are made from lightweight nylon-neoprene material and are designed to rupture when subjected to the pressure wave associated with a LOCA in the reactor vessel cavity.[19] The containers rest on grating which is supported by a structural steel frame consisting of beams and columns. This framed steel structure is anchored to the refueling pool floor. The support platform is designed to support the shielding material as well as resist uplift due to pressure generated by the postulated reactor cavity LOCA. The support structure is constructed of ASTM A-36 structural steel and ASTM A-304 stainless steel.[16]

The shield support structure is designed as a seismic Category I structure. The design, fabrication and erection of the structural steel is in compliance with the "Manual of Steel Construction" Seventh Edition, by the American Institute of Steel Construction (AISC) and AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," dated February 1969. The design has considered those loads which may act on the structure during its lifetime, such as dead and live loads, accident induced loads, including pressure and jet loads, and seismic loads. The loading combinations and acceptance criteria are consistent with those specified in Standard Review Plan 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments".

The shield support structure was dynamically analyzed for seismic and accident loads by a single degree of freedom system. The Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) seismic loadings were based on the appropriate CCNPP reactor building ground response spectrum curves. These curves were generated for two horizontal and for vertical directions, using various damping values.

The criteria used in the analysis, design, and construction of the temporary neutron shield structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff. The use of these criteria provide reasonable assurance that the temporary neutron shield structure will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

2.1.3 Performance Analysis

To determine the maximum pressure that could be generated in the reactor cavity during a reactor depressurization event, BG&E reanalyzed the design basis accidents which are described in Section 14.16.5 of the Final Safety Analysis Report (FSAR). BG&E stated that this reanalysis showed that this shield would increase the peak cavity differential pressure by only about one percent, i.e., to a value of 31.4 psid.

To prevent the shield bags from becoming missiles in the unlikely event that a LOCA occurs in the reactor cavity, the bags have been provided with tear strips tethered to the support structure. If the bags are lifted off of the grating and accelerated upward by a pressure differential or a jet, they will be cut open by the tear strip and the water will be released.

BG&E stated that after the shield water is released, some bags could fall through the operating floor. With respect to whether these damaged bags could disable certain essential cooling pumps, BG&E stated that for these bags to get to the emergency recirculation sump, the bags would have to find their way down two more floor levels and travel across a considerable distance on the containment floor. Because of the tortuous paths that damaged bags would have to travel, BG&E concludes that it is unlikely for any of the bags to reach the emergency recirculation sump. In the event that these bags do reach the sump, however, a steel-framed, grating-covered box will prevent the bags from entering the sump. BG&E further stated that because of the reactor cavity pressurization during a LOCA, it is unlikely that any of the bags will fall through the thirty inch annulus between the reactor vessel flange and the primary shield wall. Nevertheless, even if this did occur, and they blocked the four inch drain at the bottom of the reactor cavity, sufficient water could still flow to the sump for recirculation through the reactor coolant pipe penetrations in the primary shield wall.

For the design basis depressurization event, a 0.5 pound per square inch pressure differential will produce more than 1g (32.2 ft/sec^2) of upward acceleration on the one foot thick water shield. This acceleration will be sufficient to disperse the water bags so that air and steam will begin to flow through the shield area and not further enhance the magnitude of the pressure rise in the reactor cavity. In addition, BG&E stated that in this design the bottom of the shield structure is about four feet above the reactor vessel flange. This provides 173 square feet of vent area from the reactor cavity. On the basis of these considerations, we conclude that the additional shielding will not significantly increase the peak pressure in the reactor cavity during a design basis depressurization accident.

In regard to the generation of potentially destructive missiles, the nylon-neoprene material from which these bags are fabricated is lightweight. It weighs less than ten ounces per square yard. Consequently, after the water is released by the tear strips, we conclude that these bags will not have enough kinetic energy to form destructive missiles.

In regard to the effect of this shield on the boron concentration of water in the containment sump, BG&E stated that the entire amount of water in this shield, which is approximately 3700 gallons, is less than one percent of the inventory in the refueling water tank. Consequently, the dilution of the boric acid solution in the sump following a LOCA would not be significant. Also, because of the comparatively long and tortuous pathways from the shield grating to the emergency recirculation sump, we conclude that it is highly unlikely that any of these bags will get to the sump. Even if they did get to the sump, we find that because a significant portion of the large sump screen box is vertical, these bags will not significantly inhibit the flow of the recirculation cooling water.

The nylon-neoprene material has been analyzed and tested for fire, chemical, and radiation resistance. From the fire resistance tests, which were done in accordance with ASTM-D-1929, it was found that the material will begin to smoke when it is heated to a temperature of 375°F. The average ignition temperature was found to be 618°F, but this material has about five grams of atomic chlorine in it per square yard to retard flame propagation. Tests show this to be sufficient to classify it as a "flame resistant" and a "self-extinguishing" material. From the chemical tests, it was found that the nylon-neoprene material will not react with the basic (pH of 8.5 to 10.5) spray and sump fluid. From the radiation testing, it was found that the mechanical properties of the nylon and neoprene start to deteriorate after 10^7 rads of neutron irradiation. This means the bags may have to be replaced about every three years. BG&E intends to confirm the continued integrity of all of the bags each refueling.(19) This will be accomplished by draining the bags during disassembly of the shield, refilling them during reassembly and checking for leaks. Any bags that leak will be replaced. If, during reactor operation bags begin to leak, the water will be collected in the containment sump. This should be detected as unidentified reactor coolant system (RCS) leakage; and, if greater that the TS limit of 1 gallon per minute, reactor shutdown would be required.

2.1.4 Occupational Exposure Analysis

We have reviewed the licensee's proposal for shielding the neutrons streaming from the reactor annular gap with respect to occupational radiation exposure. Based on the shield design of one foot of water and relevant dosimetric measurements, the licensee measured a reduction of gamma and neutron radiation from levels as high as 25 rem/hr to a maximum of less than 2 rem/hr. Most areas in containment were reduced to less than 0.3 rem/hr. These dose equivalent rate values are based on neutron and gamma radiation surveys that were made during power escalation and extrapolated to 100% power.

We have reviewed these radiation levels by evaluating the licensee's instrumentation used for the measurement of neutron and gamma-ray dose equivalent rates, the neutron spectral distribution for typical Combustion Engineering (CE) reactors (in the absence of Calvert Cliffs spectral data), and the resultant factor of reduction of the neutron and gamma dose equivalent rates based on the licensee's measurements before and after the water shield installation. As a result of this review, we conclude that the licensee's measurement program and resultant neutron and gamma dose rates are acceptable. They are also supported by theoretical calculations of radiation levels made at another facility having proposed a similar shield.

We have also reviewed the annual man-rem burden estimated by the licensee from all operations inside containment when the reactor is at power and during refueling operations. This burden is estimated to be 5 man-rem/yr after the shield is installed. This value should be compared against the 60 man-rem exposure from the same operations and occupancy without the shield. As a result of this review, we conclude that the licensee's predicted 5 man-rem/yr burden with an installed shield is reasonable. It represents a small part (about 1%) of the annual man-rem burden expected for all plant operations of typical PWR's and is acceptable.

We conclude that the impact of this modification will be to effect a significant reduction in occupational exposure to personnel entering the containment for inspection and minor maintenance while the reactor is at power and that, based on the staff's Technical judgement, their exposure should be as low as is reasonably achievable and below the limits of 10 CFR 20.

2.1.5 Conclusion

We conclude that the shielding structural design meets the code requirements, that the shielding will perform satisfactorily not generating any destructive missiles or adversely effecting the operation of the containment sump, and that it will significantly reduce occupational radiation exposure. Therefore, licensee condition 2.C.5.c(2) has been satisfied and is removed from the CCNPP Unit No. 2 license.

2.2 Update Unit 2 TS

In the recently issued amendment for CCNPP Unit No. 1 [22], three TS changes were made that equally apply to Unit No. 2. A brief synopsis of the Reference 22 Safety Evaluation is presented for completeness.

2.2.1 CEA Drop Time

The CEA drop time should be reduced from 3.0 seconds to 2.5 seconds in TS 3.1.3.4. This will correct this specification since the 3.0 second value is for 100% insertion and the TS value should be the 90% insertion value of 2.5 seconds. This is an administrative correction only.

2.2.2 Water Hole Peaking

In December 1977, a 4.6% increase in peaking of power in fuel pins near water holes was identified which was not included in previous CE analyses. Ordinarily this 4.6% increase would be applied directly as a penalty to the Limiting Condition for Operation (LCO) and Limiting Safety System Settings (LSSS) limits. However, certain credits were identified which would mitigate the consequences of this additional water hole power peaking. We have not completed our review of either the original CE analytical methods, the 4.6% penalty, or the credits claimed. However, at the current state of review, we believe that the integrated package of the original analysis, the penalty, and the credits would be conservatively bounded if a 2.8% penalty were imposed on the current Departure from Nucleate Boiling Ratio (DNBR) LCO and LSSS limits for the total radial peaking factor (F_r^T) and a 0.5% penalty were imposed on the linear heat rate (KW/FT) LOCA limit. At our request, BG&E administratively impose these same limits since December 1977. We believe that this penalty is conservative relative to the penalty which will result at the completion of our review.

2.2.3 Reactor Protection System (RPS) Response Time Testing

BG&E proposed to add a footnote to TS Table 3.3-2 to state that Resistance Temperature Detectors (RTDs) response times are not included in the values for RPSs utilizing RTDs. In subsequent discussions with BG&E, we concluded that the RTD response time should be included in the table. BG&E has agreed. This is consistent with recent changes to other standard TS facilities.

The RPS for the CCNPP units has three trip functions which require an input of the reactor coolant temperature as determined by one or more RTDs. These RTDs, located in instrument wells in the primary coolant system, provide inputs to the "Power Level-High", "Axial Flux Offset", and "Thermal Margin/Low Pressure," trip functions of the RPS.

TS Surveillance Requirement 4.3.1.1.3 requires all RPS trip functions to undergo periodic tests to confirm that their response times* are within specified limits. TS Table 3.3-2 contains the limits for the RPS response time tests. The existing TS requires that the response time of both sensor and trip circuit be less than 1.0 second. The RTDs themselves have response times on the order of several seconds. The revised TS Table 3.3-2 will include an explicit RTD response time in addition to the previously stated channel response time which excluded the RTD.

Submittals dated June 1, 8 and 15, 1977, on Docket No. 50-336 provided a description of methods used by CE in determining the trip "setpoints" of the RPS. Although our review is not complete, we conclude from our review to date that the CE computer code CESEC, used to predict the consequences of various reactor transients as presented in the FSAR, represents the RTDs such that they exhibit a (maximum acceptable) response time of 5 seconds.

The change to the TS to incorporate an explicit RTD response time has the effect of making the required RPS response time tests consistent with the assumptions made in the plant safety analysis. Accordingly, the consequences of reactor transients previously analyzed by BG&E will not be more severe nor will there be any reduction in safety margin. Accordingly, it is appropriate to explicitly include the 5 second response time for RTDs in Table 3.3-2 for each RPS channel that utilizes RTDs.

In the process of our review, we discovered that the response time for the Reactor Coolant Flow-Low functional unit was incorrect. According to CE and BG&E, the response time value used in safety analyses has always been 0.65 seconds. This in an administrative correction only.

---

* Technical Specification 1.25 defines Reactor Trip System Response Time as "...the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism."

2.3 Incore Detector Operability

2.3.1 Background

Early in Cycle 2 operation,[10] BG&E proposed TS changes which would permit Unit No. 1 to operate at full power if the number of operable incore detectors fell below the TS operability requirements. In our review of these proposed changes, we determined that some of the changes would improve the core monitoring capability. Although the requested change was not needed because the detector failure rate was less than expected for the remainder of Cycle 2, BG&E is still requesting the TS change to authorize the improved monitoring.

The Incore Detector System is required for four types of monitoring in the CE reactors:

1. Recalibrations of the excore neutron flux detection system;

2. Monitoring the azimuthal power tilt;

3. Calibration of the power level neutron flux channels; and

4. Monitoring the linear heat rate.

The present TS specify one set of operability requirements to encompass all four functions. This set consists of two requirements:

- At least 75% of detector locations shall be OPERABLE.

- There shall be at least two sets of fourfold 90° symmetric OPERABLE detector locations.

An OPERABLE location was defined as a location in which at least three of the possible four segments are OPERABLE. Therefore, considerable usable data was discarded by this definition of OPERABLE.

In the proposed TS for Incore Detectors, the requirements for each function the incore system performs is treated separately, and thus unnecessary requirements can be relaxed. Conversely, for each function being performed by the incore system, the TS requirements are based on the actual mathematical model used to satisfy the requirement. The result is a more logical and consistent TS.

We have specified an absolute minimum requirement that at least one operable detector segment in each core quadrant on each of the four axial elevations be operable. The remainder of the operability requirements are as presented in the following discussions on each function of the Incore Detector System.

2.3.2 Monitoring Azimuthal Power Tilt

BG&E has proposed that since tilt values are computed on each detector level, the tilt computation on a given level does not require the operability of any detectors in that string except those on that level. BG&E requested that the OPERABILITY requirement based on locations be replaced by the requirement that sufficient detector segments be OPERABLE to compute at least two tilt values on each of the four detector segment levels. This change in Specification 3.3.3.2 for each unit greatly increases the number of tilt values that can be computed.

Presently, BG&E uses a computer code, INCA, to compute tilt values using sets of OPERABLE fourfold quadrant symmetric detector segments. The tilt equation has three parameters to fit, so the tilt equation could be fit using either sets of three segments or sets of four segments. BG&E proposed to modify INCA so that it will compute tilt values using either threefold or fourfold sets of OPERABLE quadrant symmetric segments. BG&E has studied the difference in predictions of three data point and four data point tilt calculations and determined that these differences are small in the context considered, and no significant loss of resolution results from computing tilts from three, rather than four, data points. We have evaluated the proposed change and conclude that it provides a comparable level of azimuthal power tilt resolution and is, therefore, acceptable.

2.3.3 Recalibration of the Excore Neutron Flux Detector System

Again the proposed TS places requirements on OPERABLE segments rather than OPERABLE locations. The proposed TS requires that at least 75% of the detector segments remain OPERABLE. In addition it requires a minimum of nine OPERABLE incore detector segments on each detector level and a minimum of two OPERABLE detector segments in the inner 109 fuel assemblies and two OPERABLE segments in the outer 108 fuel assemblies. These new requirements guarantee that adequate instrumentation in each region of the core is available to monitor core performance.

The change from requiring 75% of detector locations OPERABLE to requiring 75% of detector segments OPERABLE is acceptable. BG&E has presented data which substantiate that there is no loss of resolution in the excore detector recalibration if at least 75% of the detector segments remain OPERABLE. The study BG&E performed consisted of randomly failing up to 25% of the detector segments in a simulation study, and determining the difference in results computed from a full complement of detectors and the results computed with a percentage of detectors failed. Our review has not uncovered any means of non-random failure of the detectors. We conclude that random failure of the detectors will not preclude regions of the core becoming uninstrumented and on the basis of this study, we find this proposed TS acceptable.

2.3.4 Monitoring Peaking Factors and Linear Heat Rates

The present TS is based on having 75% of detector locations OPERABLE. The uncertainties applied to the setpoints are based on the concept of OPERABLE locations, rather than OPERABLE segments. Furthermore, the core follow program, performed by CE for BG&E, which assures that the incore detector measurement errors are within the 3.95% bound assumed in the INCA Topical Report (CENPD 145), is also based on operable locations. Although our review of this topical is not complete, it has progressed to the point that we consider the core follow program to be a valid technique for assuring that the 3.95% limit is not exceeded. Thus, this part of the proposed TS continues to require 75% of the detector locations to be operable. The same two requirements of the previous section are also required to guarantee that no large region of the core becomes uninstrumental for this monitoring. These proposed TS requirements give better assurance than the current TS that improved monitoring is used. On this basis, we conclude that the proposed TS are acceptable.

2.4 Low Temperature Overpressure Protection

BG&E proposed a low temperature overpressure protection system (OPS) for CCNPP Units Nos. 1 and 2.[5,7,12,18] The system incorporates a defense in depth concept for overpressure protection, utilizing operator training, administrative procedures, Technical Specifications, and hardware improvements to meet the criteria established by the NRC staff. The objective of the OPS is, first, to insure that pressure transients while operating at low Reactor Coolant System (RCS) temperatures become and remain unlikely events, and second, to mitigate the consequences of a pressure transient should one occur. The proposed mitigating system includes sensors, actuating mechanisms, and valves to prevent a RCS pressure transient from exceeding the pressure-temperature limits included in the CCNPP Technical Specifications as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50). These Appendix G limits are those established by using procedures defined in Appendix G to Section III of the ASME Code. Appendix G to 10 CFR 50 states that these ASME Code limits can be used for startup and shutdown when the reactor is not critical. For criticality, Appendix G to 10 CFR 50 requires more stringent rules than Appendix G to Section III of the ASME Code.

The history of the generic low temperature overpressure protection issue is described in NUREG-0138 (July 1976). Briefly, a series of over thirty incidents had occurred in pressurized water reactors (PWRs) since 1972 in which the Appendix G pressure-temperature limits had been exceeded at temperatures less than normal operating temperature..

These incidents consisted of two varieties of pressure transients: a mass input type from charging pumps, safety injection pumps, or safety injection accumulators, and an energy input type caused by thermal feedback when a reactor coolant pump (RCP) sweeps cooler primary system water through a steam generator with a hot secondary side. These incidents usually occurred in a water solid system during startup or shutdown operations.

Pressure transients which could occur at normal operating temperature, approximately 570°F, are mitigated in most plants by large code safety valves located on the pressurizer. These are mechanical valves which open against a spring pressure of about 2400 psia. The code safety valves are quite simple, having no electrical components, and as such are considered passive, failure free components. These code safety valves are tested in accordance with ASME Code, Section XI requirements.

Prior to the introduction of an OPS, pressure transients initiated while operating at lower temperatures were not protected against and there were no pressure relief devices in the reactor coolant system to prevent these transients from exceeding the Appendix G pressure-temperature limits. Nuclear reactors such as Calvert Cliffs, which have a pressure limit in excess of 2500 psia at 570°F, have a limit of only 700 psia at 200°F. The code safety valves with settings in the 2400 psia range would not be able to relieve a pressure transient at low RCS temperature without the Appendix G limits being violated by a large amount.

The Appendix G pressure limit drops off rapidly at lower temperatures because the reactor vessel material and welds have significantly less toughness at lower temperatures and are therefore more susceptible to flaw induced failure. In addition, factors such as copper content in welds and neutron fluence levels affect the material toughness and contribute to the reduction in safety margin to vessel failure at low temperature conditions. The Calvert Cliffs overpressure protection analysis was performed utilizing the Appendix G curves for 2 to 10 years of full power operation as the basis for maximum allowable pressure.

As a solution to the low temperature overpressurization problem, the licensee identified a set of power operated relief valves (PORV's) located on the pressurizer which are normally available for overpressure protection during normal plant operations. These usually have a single pressure setpoint just below the opening pressure of the mechanical code safety valves and are designed to relieve small pressure transients without requiring the code safety valves to lift. The licensee proposed to provide the PORV's with a low pressure setpoint to which they could be switched as the plant cooled down. If a pressure transient would occur at these lower temperatures and the lower setpoint had been selected, there would then be a pathway to relieve system pressure.

The PORV's are significantly more complicated than the code safety valves since the PORV's require electrical circuitry to sense pressure, transmit a signal to the valve, and actuate the solenoid to open the valve. Thus it is desirable to insure redundancy and separability in the circuitry to preclude a single failure from disabling the entire OPS system.

In a series of meetings and through correspondence with PWR vendors and licensees, we developed a set of criteria, which if adhered to, would produce an acceptable OPS. These criteria are:

(1) Operator Action: The licensee could not take credit for operator action for 10 minutes after the operator became aware of an ongoing transient.

(2) Single Failure: The system had to be designed to relieve overpressure transients assuming the worst case single failure in addition to the event which caused the transient.

(3) Testability: The system had to be testable on a periodic basis consistent with the system's employment.

(4) Seismic and IEEE 279 Design: Ideally the system should meet seismic Class I and IEEE 279 design requirements. The basic objective is that the system should not be vulnerable to a common failure mode which both initiated a pressure transient and caused a failure of equipment needed to terminate the transient.

In addition to the four formally stated criteria mentioned above, a number of additional criteria were established in the process of the staff review of generic submittals from the various vendors and in the exchange of information between the staff and the licensees.

Foremost among these was the requirement that the licensees show protection for the limiting mass addition transient regardless of the administrative procedures proposed to eliminate that potential scenario. Each licensee, therefore, was required to analyze the effects of the single pump start which would produce the most limiting mass addition transient and most severely challenge the Appendix G limits. For Calvert Cliffs a High Pressure Safety Injection (HPSI) pump start produces the most limiting pressure transient.

For the worst case energy addition transient it is acceptable to limit the severity of the transient in the analyses by assuming a maximum ΔT across the steam generator. By maximum ΔT it is meant the maximum difference in the temperature between the primary loop coolant and the secondary loop water in the steam generator. For this case and for other scenarios, the licensees were requested to develop Technical Specifications which delineated the actions required to limit the severity of these scenarios and also provide justification for their action.

Another criterion for the design of the OPS was that the electrical instrumentation and control system provide a variety of alarms to alert the operator to (1) properly enable the low temperature OPS at the proper temperature during cooldown, and (2) indicate if a pressure transient was occurring. Additionally the electrical system had to provide positive assurance that the isolation valve upstream of each PORV was open when the system was enabled by wiring its position into the enable alarm. The enable alarm would not be permitted to clear until the OPS mode selector switch for each PORV system was placed in the low pressure setpoint position <u>and</u> the isolation valve was opened.

BG&E submitted a generic overpressurization protection report prepared by CE.[5] This report was prepared for the CE Owner's Group comprising five utilities. The generic report provided information on RCS response to postulated pressure transients that occur at low temperatures during heatup and cooldown, and provided a general description of design modifications which could be used to prevent overpressurization of CE designed Nuclear Steam Supply Systems (NSSS). The NRC staff, in conjunction with its review of the CE generic report, requested that BG&E commit to a schedule for implementing a permanent or interim version of the OPS by December 31, 1977, and requested additional information related to the application of the generic aspect of the OPS as pertinent to CCNPP.[6] BG&E submitted additional information to the staff on equipment and procedural improvements as well as a schedule for implementation of the proposed system (7,8).

BG&E stated that an interim OPS for both units was installed during 1977. The system does not fully meet the staff criteria for OPS. The OPS for Unit No. 1 was upgraded to meet all staff criteria during the past refueling outage in April 1978. An upgraded system meeting all staff criteria will be installed on Unit No. 2 during its next refueling, currently scheduled for September 1978. We find this implementation schedule and the use of an interim OPS for Unit No. 2 acceptable.

BG&E submitted the CCNPP specific report in Reference 12 and additional plant specific data was supplied in Reference 18.

2.4.1 Technical Specifications and Operating Procedures

One cornerstone of the Calvert Cliffs OPS is the use of TS and operating procedures to limit the probability of initiating pressure transients at low temperatures (<275°F) and to insure the enabling, disabling, and proper functioning of the OPS.

The TS specify the conditions required for starting a RCP, the PORV OPERABILITY requirements and the PORV surveillance requirements. We conclude that these TS will provide assurance that pressure transients at low temperatures will be unlikely and that the system will function to prevent overpressure transients from exceeding Appendix G limits. We further conclude that the TS meet the criteria established by the staff and are acceptable.

The licensee will make extensive use of operating procedures to provide a large measure of the administrative protection against overpressure transients. Among these operating procedures for low temperature operating conditions are the following:

(1) When RCS temperature, pressure, and other operating conditions permit, a pressurizer steam volume of 60% of the pressurizer volume will be maintained.

(2) The TS require the maximum ΔT across the steam generator to be less than 34°F prior to starting a RCP at low temperature. This insures that functioning of one of the two PORV's will provide sufficient relief capacity such that Appendix G will not be violated in the event of an inadvertent RCP start.

(3) Steam generator temperature will be reduced to 220°F before placing the shutdown cooling system in operation. This will minimize the amount by which the primary side of the steam generator can be cooled down relative to the secondary side.

(4) Emergency Core Cooling System (ECCS) component testing will be conducted with a steam bubble or with the reactor vessel head removed. Operational testing of the Safety Injection and Chemical and Volume Control System (CVCS) components (i.e., pumps, valves, automatic signals, etc.) will be accomplished with a non-solid RCS.

We conclude that these operating procedures contribute measurably to plant protection from low temperature overpressure transients. We also conclude that the licensee's method of measuring steam generator ΔT is acceptable.

The steam generator ΔT of 34°F specified in the TS as the operating limit is a result of determining the minimum ΔT required to prevent an Appendix G violation and then factoring in uncertainties. In this particular case the maximum permissable ΔT, calculated to be 46°F, was reduced by uncertainties in instrument accuracy (9°F) and the difference in steam generator bulk fluid and shell side temperature (3°F).

2.4.2 Hardware

Acceptable performance of the OPS depends on the proper functioning and adequate relief capacity of the two PORV's located on the pressurizer. The NSSS vendor and the licensee demonstrated that with two PORV's functioning, all postulated mass and energy addition transients could be mitigated. If one PORV is assumed to fail, administrative procedures must be relied upon to limit the severity of the limiting transients in both the energy and mass addition cases to insure that Appendix G limits are not violated.

Administrative procedures, backed by TS, require that the maximum ΔT across the steam generator be less than 46°F (exclusive of measurement inaccuracies) to limit the severity of the RCP start energy addition transient. As previously noted, uncertainties due to instrument inaccuracy and the maximum temperature drop from the steam generator bulk fluid to the outer shell of the steam generator reduce the maximum measured ΔT to 34°F. This ΔT will be incorporated into the Calvert Cliffs TS.

Numerous assumptions were employed in the modeling of PORV relief to insure conservatism in the analysis of this design base energy addition transient:

- The RCS was assumed to be water solid.
- The RCS was assumed to be rigid during the transient (no expansion).
- A single PORV was assumed to fail.
- RCS letdown flow was assumed isolated.
- Heat absorption by the RCS metal mass was not considered.
- Conservatively high heat transfer coefficients across the steam generator were used.
- RCP start was assumed to be instantaneous.

With a PORV low pressure setpoint of 450 psig, the licensee showed that one PORV will provide sufficient relief capacity to limit the maximum RCS pressure to ~520 psia for an energy addition transient. This analysis assumed a ΔT of no more than 46°F across the steam generator.

We conclude that the licensee and vendor have demonstrated that the OPS can protect the RCS from exceeding Appendix G limits for an energy addition transient even with the additional single failure of a PORV.

Protection from the effects of the limiting mass addition transients was afforded by the licensee by assuring that components of the ECCS system would be disabled by procedure and Technical Specification during cooldown. This is accomplished at 310°F by placing one HPSI pump control switch in a pull-stop position and caution tagging it, by placing the second HPSI pump switch in the same configuration at 225°F, and finally, by placing the third HPSI pump in this configuration at 160°F. This provides assurance that Appendix G limits will not be violated should a single PORV fail prior to or during a mass addition transient. Conservatisms included in the limiting mass addition transient model, the inadvertent single HPSI pump start, were as follows:

- The RCS was assumed to be water solid.
- Letdown flow from the RCS was assumed isolated.
- The RCS was considered to be rigid during the transient (no expansion).
- Mass addition was assumed to occur with the highest fluid density that could occur. PORV relief was assumed to occur with with the lowest fluid density that could occur.
- A single PORV was assumed to fail.
- A conservative Bernoulli equation was utilized to model PORV relief.

Our guidance to the licensee for analyzing the mass addition transient was to show that Appendix G limits were not violated assuming that the safety injection pump which could produce the worst case transient inadvertently started, regardless of administrative procedures calling for disabling the pumps at various stages. For CCNPP, the worst pump start would be a HPSI pump. The licensee demonstrated that a single HPSI pump plus one charging pump mass input transient would produce a peak pressure of 450 psig, the PORV valve low pressure setpoint. The equilibrium pressure for the HPSI pump and one charging pump output balanced by single PORV relief is 430 psia. The quick opening time of the valve (∿3 milliseconds) results in the transient immediately being relieved, producing an equilibrium pressure of 430 psia. This corresponds to an Appendix G limit that would exist at temperatures well below the refueling temperature (∿130°F).

We conclude that the licensee has demonstrated that the OPS will prevent overpressurization of the RCS due to mass addition transients, assuming the single failure of a PORV.

2.4.3 Electrical, Instrumentation and Control

The basic design criteria that were applied in determining the acceptability of the electrical, instrumentation and control aspects of the low temperature overpressure protection system are:

- Operator Action - no credit can be taken for operator action until ten minutes after the operator is aware, through an action alarm, that a pressure transient is in progress.
- Single Failure Criteria - the low temperature overpressure protection system shall be designed to protect the reactor vessel given a single failure in addition to the failure that initiated the pressure transient.
- Testability - the system design shall include provisions for testing on a schedule consistent with the frequency that the system is relied upon for pressure protection.
- Seismic Design and IEEE - 279 Criteria - the pressure protection system shall meet both seismic Category I and IEEE - 279 Criteria.

In addition to complying with the above stated criteria, the licensee, at the request of NRC, has agreed to modify the electrical design of the PORV's actuation circuitry as follows:

- eliminate the common "Minimum Pressure Temperature (MPT) Enable" switch and replace it with separate "MPT Enable" switches (one each per channel);
- add an alarm to alert the operator of the need for enabling the low setpoint circuitry;
- add upstream isolation motor operated valve (MOV) position indication to the alarm circuitry; and
- eliminate two cross-connecting test switches designed to permit testing of the sensing and actuating portions of each train.

Presently installed PORV's meet seismic criteria that are consistent with the basic objective of preventing a potential LOCA pathway. The design of the additional electrical equipment added for overpressure mitigation is consistent with existing plant design criteria and satisfies the seismic and IEEE -279 criteria, i.e., a single failure which initiates an overpressurization event will not disable the overpressure mitigating system. Power is supplied to all electrical components from vital supplies designed to operate during a seismic event and following a loss of off-site power. Cable raceways for this equipment are supported to withstand a seismic event.

The sensing/actuating/relieving system consists of two redundant trains with independent vital power supplies. The trains are independent and separate through the elimination of the common "MPT Enable" switch and two cross-connecting switches that were designed to permit testing of the sensing and actuating portions of each train.

The overpressure mitigating system is provided with separate and independent pressure-temperature signals, bistables and power supplies to each PORV. The design of the circuitry is such that the PORV's do not cycle shut automatically after opening on a pressure signal. Operator action is required to shut the PORV or the upstream MOV before RCS pressure decreases to the point at which damage could result to reactor coolant pumps (RCP's).

Prior to cooling the RCS below 275°F, normal operating procedures require the manual actuation of the high pressure alarm, the resetting of the handswitches to the "MPT Enable" position, and verification that the handswitches for the MOVs which isolate the PORV's are in the "open" position and the valves indicate "open". When the PORV's are reset to the low pressure position, an annunciator window will light to indicate the low pressure PORV mode of operation. The annunciator window will remain lit until the PORV's are reset to the normal operating position. A variable setpoint high pressure alarm which monitors maximum permissible RCS pressure as a function of RCS temperature has been utilized. This alarm will be actuated when the RCS temperature is less than 275°F. Once the PORV's are reset to provide low temperature relief at 450 psig, plant cooldown can be resumed.

Normal operating procedures require that, during plant heatup when the RCS temperature exceeds 275°F, the PORV's are reset to the normal relief setpoint of 2,385 psig. At the same time, alarms will be manually deactivated by operator action, and the temperature interlock will activate. This will prevent inadvertent lifting of the PORV's at the low setpoint. After the PORV's are reset to the normal setpoint of 2,385 psig, normal plant heatup may be resumed.

We find the above electrical design features of the low temperature overpressure mitigating system acceptable.

A computer-generated variable setpoint high pressure alarm is procedurally set to alarm at an increasing pressure which is below the maximum allowable pressure at the existing RCS temperature. This alarm will be instituted when the RCS temperature is less than 275°F. The mode of annunciation is the audible alarm and typewritten printout associated with the computer. The sensors used are PT-103 and PT-103-1, which monitor pressurizer pressure. Procedural controls prevent the intentional removal from service of the computer when the RCS is in a water solid condition. We find this design to be acceptable.

In accordance with the NRC staff position requiring an isolation valve alarm, the licensee has agreed to add upstream isolation MOV position indication to the alarm circuitry. This design feature provides positive assurance that the isolation valve upstream of each PORV is open, when the system is enabled, by wiring its position into the enable alarm. We find this acceptable.

Any component capable of an energy or mass input which would result in a RCS overpressurization will be disabled when its operation is not essential to safe plant operations. The following operating procedures will be used:

(1) Reactor Coolant Pumps - When it is not practical to initiate and maintain a steam volume in the RCS, RCP operation will only be initiated when the secondary-to-primary temperature differential is 5°F or less.

(2) HPSI Pumps - The required number of HPSI pumps to be disabled to ensure overpressure protection will be accomplished by racking out the HPSI pumps breakers. In addition, the HPSI header isolation valves are locked shut when the RCS is water solid and cold (less than 200°F). Caution tags, to alert the operator, are used where operation of a HPSI pump or valve could result in RCS overpressurization.

(3) Pressurizer Heaters - During RCS water solid conditions, the pressurizer heaters will be controlled by disabling and marking with caution tags until pressurization commences.

(4) Charging Pumps - Charging pumps that are not required during RCS water solid conditions will be disabled and tagged. Generally, only one charging pump is required during cold shutdown conditions.

We find the above procedures for disabling non-essential components acceptable.

We find the electrical, instrument and control aspects of the proposed design acceptable on the basis that: (1) the proposed overpressure protection system complies with IEEE Std 279-1971, and seismic criteria; (2) the system is redundant and satisfies the single failure criterion; (3) the design requires no operator action prior to ten minutes after the operator receives an overpressure action alarm; (4) the system is testable on a periodic basis; and (5) the proposed changes to the Technical Specifications reduce the probability of overpressurization events to acceptable levels.

2.4.4 System Testing

The Calvert Cliffs OPS is designed to be capable of being tested to ensure that the input signal to the control system is correctly transmitted to the PORV operating solenoid. A channel functional test of the instrumentation and control hardware will be conducted once every 18 months with the PORV isolated. PORV valve testing and testing frequency will conform to the applicable requirements of ASME Code Section XI, Subsection IWV. This frequency of testing will ensure that the system is operable prior to any reliance on the OPS. We find the proposed test program acceptable.

2.4.5 Conclusions

The system presented by the Baltimore Gas & Electric Company to provide protection for the Calvert Cliffs plants from low temperature overpressure transients provides assurance that these transients will be unlikely events and that, should they occur, the plants will be protected.

We conclude therefore, that the Calvert Cliffs OPS meets the criteria established by the staff for overpressure protection and is acceptable as a low temperature overpressure protection system. We further conclude that the TS agreed upon by BG&E for Units 1 and 2 on the OPS design are in consonance with the OPS criteria and are therefore acceptable. License condition 2.C.5.c(1) has been satisfied and is therefore removed from the CCNPP Unit No. 2 license.

2.5 Pressurizer Safety Valves

BG&E proposed TS changes to allow both pressurizer safety valves to be removed for maintenance during cold shutdown in Mode 5 provided a spool piece is used to replace one of the valves, thus allowing the pressurizer to relieve directly to the quench tank[13]. This is item 3 of the referenced request. No hardware modifications are necessary to implement this change. This change will not degrade the overpressure protection capability of the plant and is therefore acceptable.

2.6 Safety Injection Throttle Valves

BG&E proposed a change to the TS for CCNPP Units Nos. 1 and 2 that would incorporate surveillance requirements for safety injection throttle valve position settings(17). This proposed change is in response to our letter of July 18, 1977, which included a sample specification(11). The proposed TS deviate from our sample TS in that, according to BG&E, the HPSI valves utilize a gear-driven rotary switch which is housed inside the operating mechanism and, therefore, it is not possible to physically measure the valve position after the valve has been stroked. They proposed the alternative of checking the valve position limit switch while the valve is open during the monthly test. We find the proposed TS, modified slightly to meet our requirements, to be acceptable in meeting our requirements.

2.7 Secondary Water Chemistry

BG&E proposed that the secondary water chemistry TS be revised for both units to reduce the number of licensee event reports (LER's) while still maintaining adequate protection of the secondary system(13). This request is item 1 of the July 27, 1977 submittal.

The secondary water chemistry specifications are designed to provide protection to both the steam generator and secondary system internals. The most damaging chemical reactants enter the system via condenser cooling water ingress. Accumulation of these impurities in the steam generators may lead to corrosion induced degradation and eventual component failure. The limits listed in Table 3.7-3 are slightly higher than the normal maximum reached during routine conditions and violations of these limits would generally indicate condenser cooling water leakage. Since the maximum cation conductivity, specific conductivity, and pH levels may be expected to increase during plant startup, provisions have been made to exceed the "steady state" limits for a period of time adequate to stabilize the water chemistry within the prescribed limits in the table.

The TS change that BG&E proposed was to eliminate the transient limits in Table 3.7-3. Both the normal and transient limits are based on the recommendations of the steam generator manufacturer, CE. We have determined that these limits should remain in the TS. However, by changing the format of this specification, the number of LER's can be greatly reduced and only the abnormal conditions, such as a condenser leak, will be reported. This can be accomplished by replacing the "normal" and "transient" headings in Table 3.7-3 with columns labeled "limit" and then adding footnotes specifying the time, the maximum value of the conductivity or pH, and special conditions when the limit may be exceeded. BG&E has agreed to the modified proposed specifications 3.7.1.6 and 4.7.1.6. We conclude that the proposed secondary water chemistry TS changes are acceptable.

2.8 Spent Fuel Ventilation System

In item 5 of Reference 13, BG&E proposed a minor change to TS paragraph 4.9.12.d.3 for both units. The change would revise the words "the ventilation system" to "each exhaust fan." This is simply a clarification of the TS requirements and is therefore acceptable.

3.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 7, 1978

REFERENCES

1. BG&E Proposal to Install Temporary Neutron Shielding, J. W. Gore to B. C. Rusche, 2/6/76.

2. BG&E Proposal to Install Temporary Neutron Shielding, J. W. Gore to B. C. Rusche, 2/23/76.

3. NRC Acceptance of Proposed Temporary Shielding at CCNPP Unit No. 2, R. C. DeYoung to J. W. Gore, 4/7/76.

4. BG&E Application Supplement for Cycle 2 Requesting Revised Incore Detector TS, V.R. Evans to B. C. Rusche, 11/30/76.

5. BG&E Submittal of Generic Report on Overpressure Protection for CE Plants, A.E. Lundvall to D. L. Ziemann, 12/3/76.

6. NRC Request for Additional Information and Implementation Schedule for Overpressure Protection, D.L. Ziemann to A. E. Lundvall, 1/10/77.

7. BG&E Response to NRC Request for Additional Information on Overpressure Protection of 1/10/77, A.E. Lundvall to D. L. Ziemann, 3/4/77.

8. BG&E Response to NRC Request for Additional Information on Overpressure Protection of 2/4/77, A.E. Lundvall to D. L. Ziemann, 3/28/77.

9. BG&E Proposal to Consider Temporary Neutron Shielding as Permanent, A. E. Lundvall to D. L. Ziemann, 5/3/77.

10. BG&E Application for Revised Incore Detector TS, A.E. Lundvall to E. G. Case, 5/17/77.

11. NRC Request for Safety Injection Throttle Valve Position Control, Generic, D. K. Davis to A. E. Lundvall, 7/18/77.

12. BG&E Specific Plant Report on Overpressure Protection, A. E. Lundvall to D. K. Davis, 7/21/77.

13. BG&E Application for Seven Miscellaneous TS Changes, A. E. Lundvall to E. G. Case, 7/27/77.

14. BG&E Response to NRC Questions on Incore Detector Analyses, W. J. Lippold to M. Conner, 8/11/77.

15. BG&E Response to NRC Questions on Incore Detector Analyses, W. J. Lippold to M. Conner, 8/19/77.

16. BG&E Response to NRC Questions on Neutron Shielding, R. F. Ash to D. K. Davis, 9/19/77.

17. BG&E Application on ECCS Valve Position Stops, A. E. Lundvall to E. G. Case, 9/19/77.

18. BG&E Response to NRC Request for Additional Information on Overpressure Protection of 8/26/77 Including Suggested Technical Specifications, A. E. Lundvall to D. K. Davis, 9/29/77.

19. BG&E Response to NRC Concerns on Neutron Shielding, A. E. Lundvall to D. K. Davis, 11/10/77.

20. BG&E Evaluation of NRC Modification to Incore Detector TS, P. S. Walsh to M. Conner, 11/10/77.

21. BG&E Response to NRC Questions on Seven Miscellaneous TS Changes, A. E. Lundvall to R. W. Reid, 3/9/78.

22. NRC Amendment No. 32 for CCNPP Unit No. 1, R. W. Reid to A. E. Lundvall, 3/31/78.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-317 AND 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 34 to Facility Operating License No. DPR-53 and Amendment No. 16 to Facility Operating License No. DPR-69, issued to Baltimore Gas & Electric Company (the licensee), which revised the Technical Specifications (TS) for operation of the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit No. 1 and deleted satisfied conditions of the license and revised its appended TS for CCNPP Unit No. 2. The facilities are located in Calvert County, Maryland. The amendments are effective as of the date of issuance.

The amendments (1) delete satisfied Unit No. 2 license conditions for installation of a permanent means of protection against reactor coolant system overpressurization during plant startup and shutdown, and installation of a permanent neutron streaming shield; and (2) revise the TS by correcting the control element assembly drop time (CCNPP Unit No. 2 only), imposing a modified water hole peaking factor (CCNPP Unit No. 2 only), including a resistance temperature detector response time (CCNPP Unit No. 2 only), modifying the incore detector operability requirements to be more definitive, adding low temperature overpressure protection requirements, authorizing the removal from service of both pressurizer safety valves when in Mode 5, specifying surveillance requirements for safety injection throttle valve positions, modifying the secondary water chemistry requirements, and clarifying the surveillance requirements for spent fuel pool ventilation system testing.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated November 30, 1976, May 17, July 27, and September 19, 1977, and supplemental information dated February 6 and 23, November 30 and December 6, 1976, March 4 and 28, May 3, July 21, August 11 and 19, September 19 and 29, November 10, 1977, and March 9, 1978, (2) Amendment No. 34 to License No. DPR-53, (3) Amendment No. 16 to License No. DPR-69, and (4) the Commission's related Safety Evaluation. All of these items are available for publ inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Calvert County Library, Prince Frederick, Maryland. A copy of items (2) through (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 7th day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors