OCTOBER 2 1 1978

Docket No.: 50-318

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

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant Unit No. 2. The amendment consists of changes to the Technical Specifications in accordance with your request dated July 26, 1978, and supplemental information dated July 31, August 14, September 7 and October 6, 16 and 17, 1978.

The amendment authorizes operation with modified (sleeved and reduced flow) guide tubes for the Control Element Assemblies (CEA's) and revises the Appendix A Technical Specifications by:

(1) incorporating changes resulting from the analyses of Cycle 2 reload fuel, and

(2) authorizing the removal of all part length CEA's.

We agree with the statements made by your staff and by Combustion Engineering that the sleeving of CEA guide tubes is not considered a permanent solution to the problem. The performance of the CEA sleeved and reduced flow guide tubes will need to be evaluated at the end of Cycle 2. Your staff has agreed to provide such an evaluation program, including the planned inspections of the CEA guide tubes at least 90 days prior to CCNPP Unit No. 2 shutdown for the Cycle 3 reload outage.

Cont.!
GD

OFFICE➤						
SURNAME➤						
DATE➤						

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

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

Sincerely,

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

Enclosures:
1. Amendment No. 18 to DPR-69
2. Safety Evaluation
3. Notice

cc. w/enclosures: See next page

STSG:DOR
DBrinkman
10/19/78

OFFICE	ORB#4:DOR	ORB#4:DOR	OELD	C-EB:DOR	C-RSB:DOR	C-ORB#4:DOR
SURNAME	RIngram	MDonnelly	Woodhouse	VNoonan	PCheck	RReid
DATE	10/19/78	10/19/78	10/21/78	10/19/78	10/19/78	10/21/78

cc w/enclosure(s):
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
Post Office 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 and Walnut Streets
Philadelphia, Pennsylvania 19106

cc w/4 cys enclosures and 1 cy
of BG&E filings dtd: 7/26 & 31, 8/14,
9/7 & 10/6, 16 & 17/78
Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
Annapolis, Maryland 21401

Director, Department of State Planning
301 West Preston Street
Baltimore, Maryland 21201

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. 18
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated July 26, 1978, as supplemented July 31, August 14, September 7 and October 6, 16 and 17, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-69 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 18, 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: October 21, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 18

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.

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CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals.

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.24 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be 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.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.26 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.28 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

LOAD FOLLOW OPERATION

1.29 LOAD FOLLOW OPERATION shall involve daily power level changes of more than 10% RATED THERMAL POWER or daily insertion of control rods below the long term insertion limits.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
b. Three Reactor Coolant Pumps Operating	≤ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of ≤ 80% of RATED THERMAL POWER.	≤ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of ≤ 80% of RATED THERMAL POWER.
c. Two Reactor Coolant Pumps Operating - Same Loop	≤ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of ≤ 46.8% of RATED THERMAL POWER.	≤ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of ≤ 46.8% of RATED THERMAL POWER.
d. Two Reactor Coolant Pumps Operating - Opposite Loops	≤ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of ≤ 51.1% of RATED THERMAL POWER.	≤ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of ≤ 51.1% of RATED THERMAL POWER.

TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Reactor Coolant Flow - Low (1)		
a. Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating*	≥ 95% of design reactor coolant flow with 4 pumps operating*
b. Three Reactor Coolant Pumps Operating	≥ 72% of design reactor coolant flow with 4 pumps operating*	≥ 72% of design reactor coolant flow with 4 pumps operating*
c. Two Reactor Coolant Pumps Operating - Same Loop	≥ 47% of design reactor coolant flow with 4 pumps operating*	≥ 47% of design reactor coolant flow with 4 pumps operating*
d. Two reactor Coolant Pumps Operating - Opposite Loops	≥ 50% of design reactor coolant flow with 4 pumps operating*	≥ 50% of design reactor coolant flow with 4 pumps operating*

*Design reactor coolant flow with 4 pumps operating is 370,000 gpm.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.19. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the minimum DNBR is no less than 1.19 for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

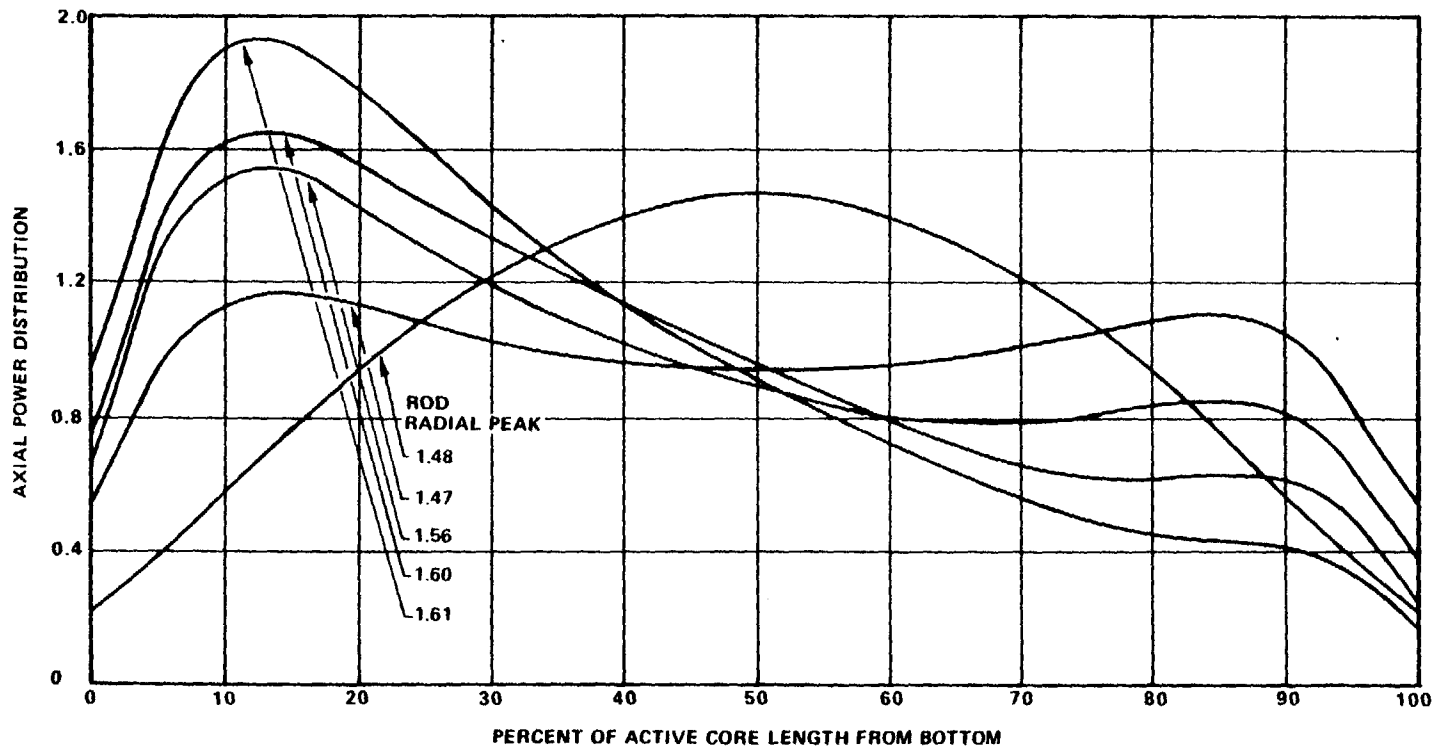

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

The conditions for the Thermal Margin Safety Limit curves in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 to be valid are shown on the figures.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a CE-1 calculated DNBR of less than 1.19 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I, 1969 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 30% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the CE-1 calculated DNBR above 1.19 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of CE-1 calculated DNBR from going below 1.19 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 13 minutes before auxiliary feedwater is required.

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a CE-1 calculated DNBR of less than 1.19 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the CE-1 calculated DNBR is less than 1.19.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1750 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time, measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 52 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 52 psia allowance is made up of a 22 psia pressure measurement allowance and a 30 psia time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 3.4% Δk/k.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN < 3.4% Δk/k, immediately initiate and continue boration at ≥ 40 gpm of 1720 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 3.4% Δk/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.3×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 900 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

LIMITING CONDITION FOR OPERATION

2. Declare the CEA inoperable. After declaring the CEA inoperable, POWER OPERATION may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

g. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in at least at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted shall be determined to be OPERABLE by inserting it at least 7.5 inches at least once per 31 days.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated OPERABLE at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

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POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within ± 2.25 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Deleted.

b. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 6 hours either:

1. Restore the inoperable position indicator channel to OPERABLE status, or

2. Be in at least HOT STANDBY, or

3. Reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

a) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or

LIMITING CONDITION FOR OPERATION

b) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted or fully withdrawn (as applicable) CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted or fully withdrawn (as applicable), and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With more than one pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 4.5 inches at least once per 12 hours except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

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 $\leq$ 3.0 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.

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*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

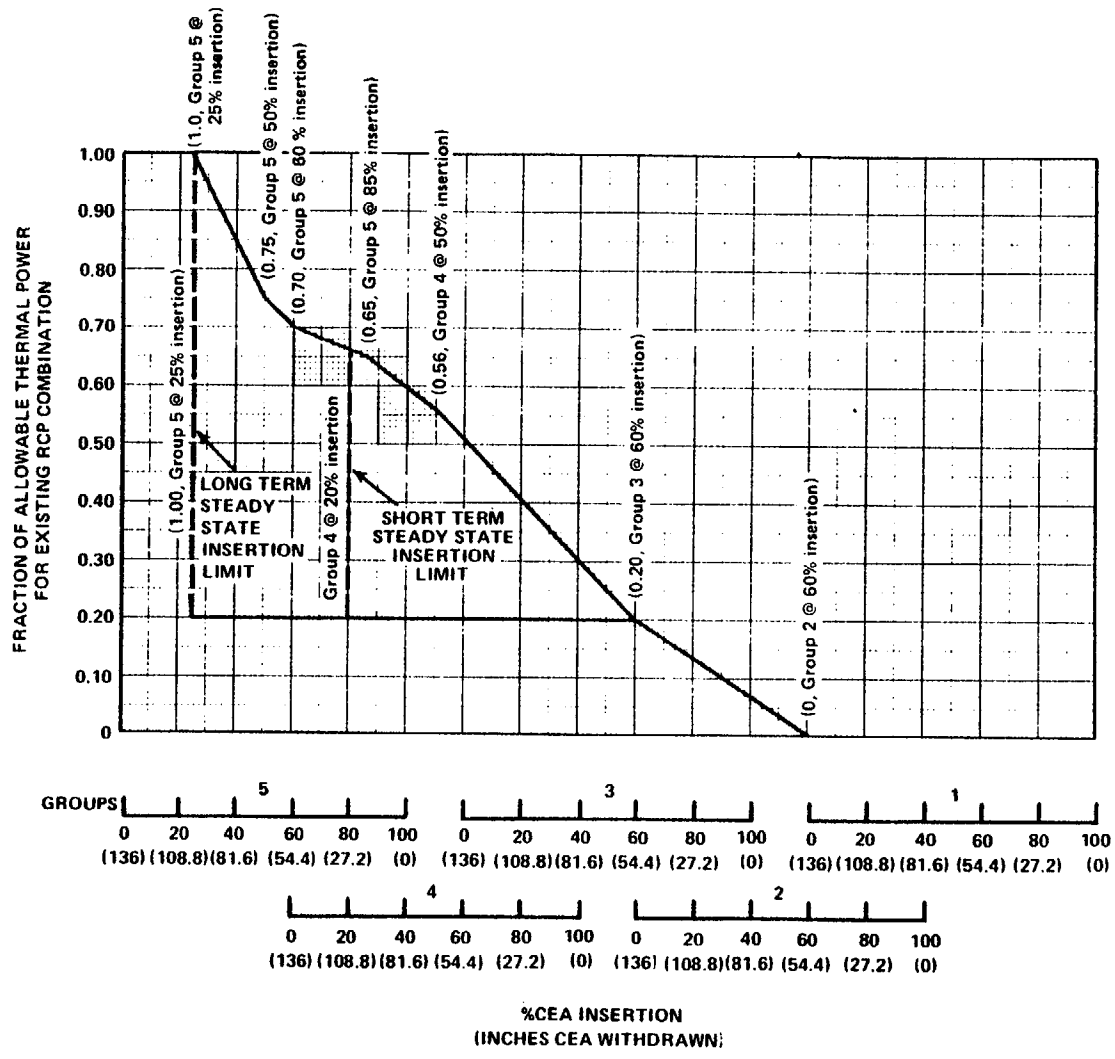

FIGURE 3.1-2 CEA Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination

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 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. 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:

$$M \times N$$

where:

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

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F^{I}_{xy} curve shown up 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 1.058*,

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, and

5. A THERMAL POWER measurement uncertainty factor of 1.02.

*An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION.

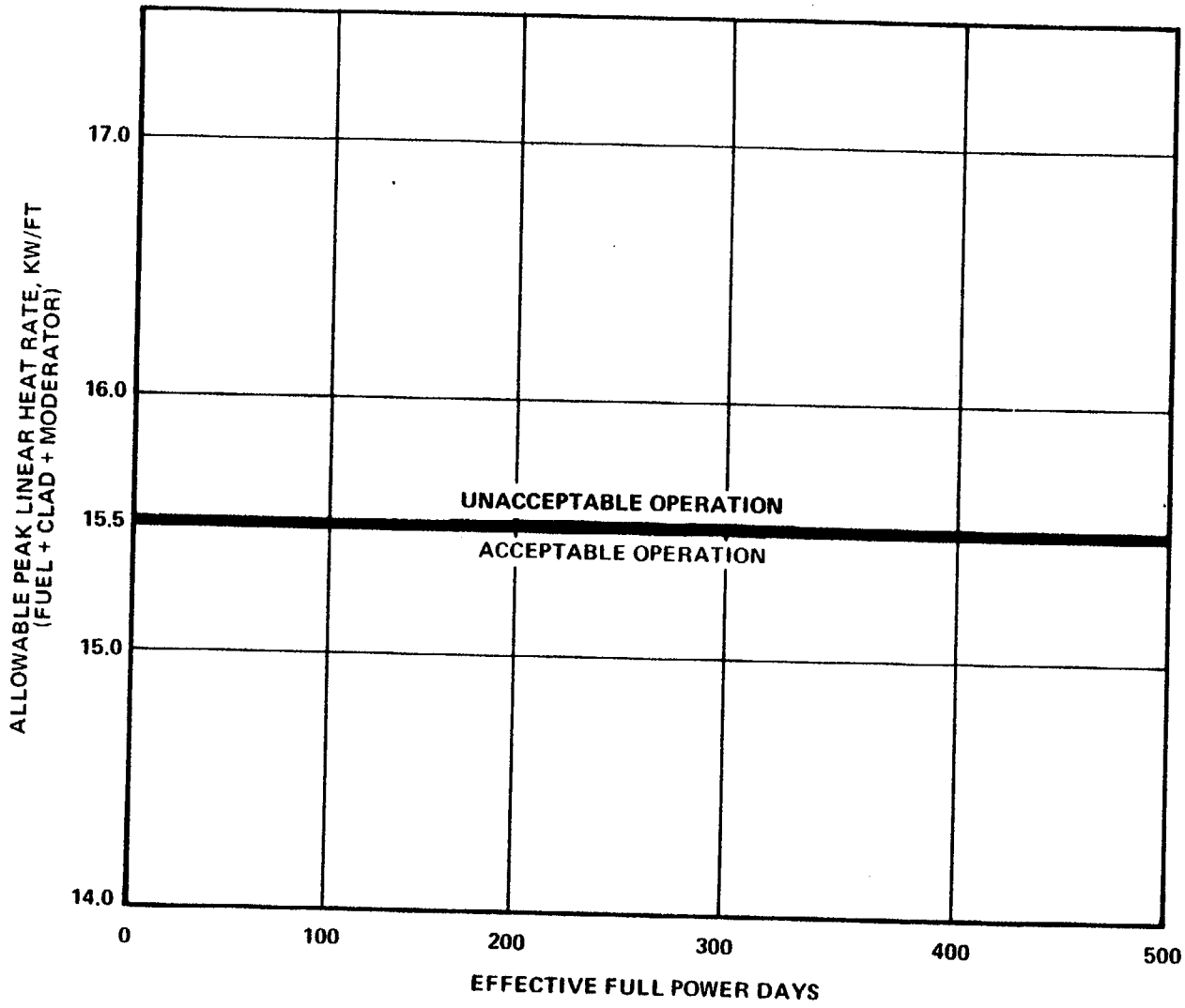

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

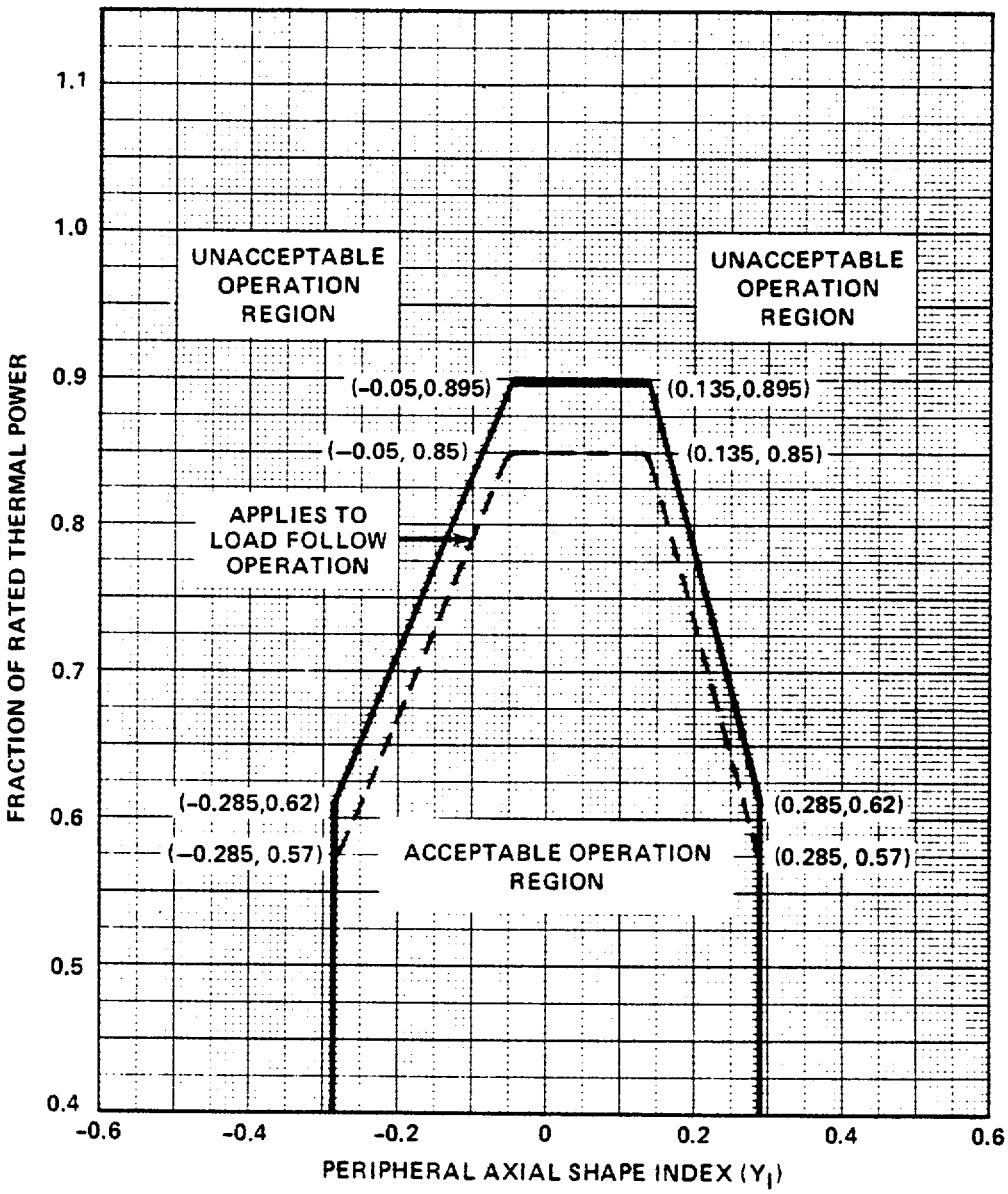

FIGURE 3.2-2

Linear Heat Rate

Axial Flux Offset Control Limits

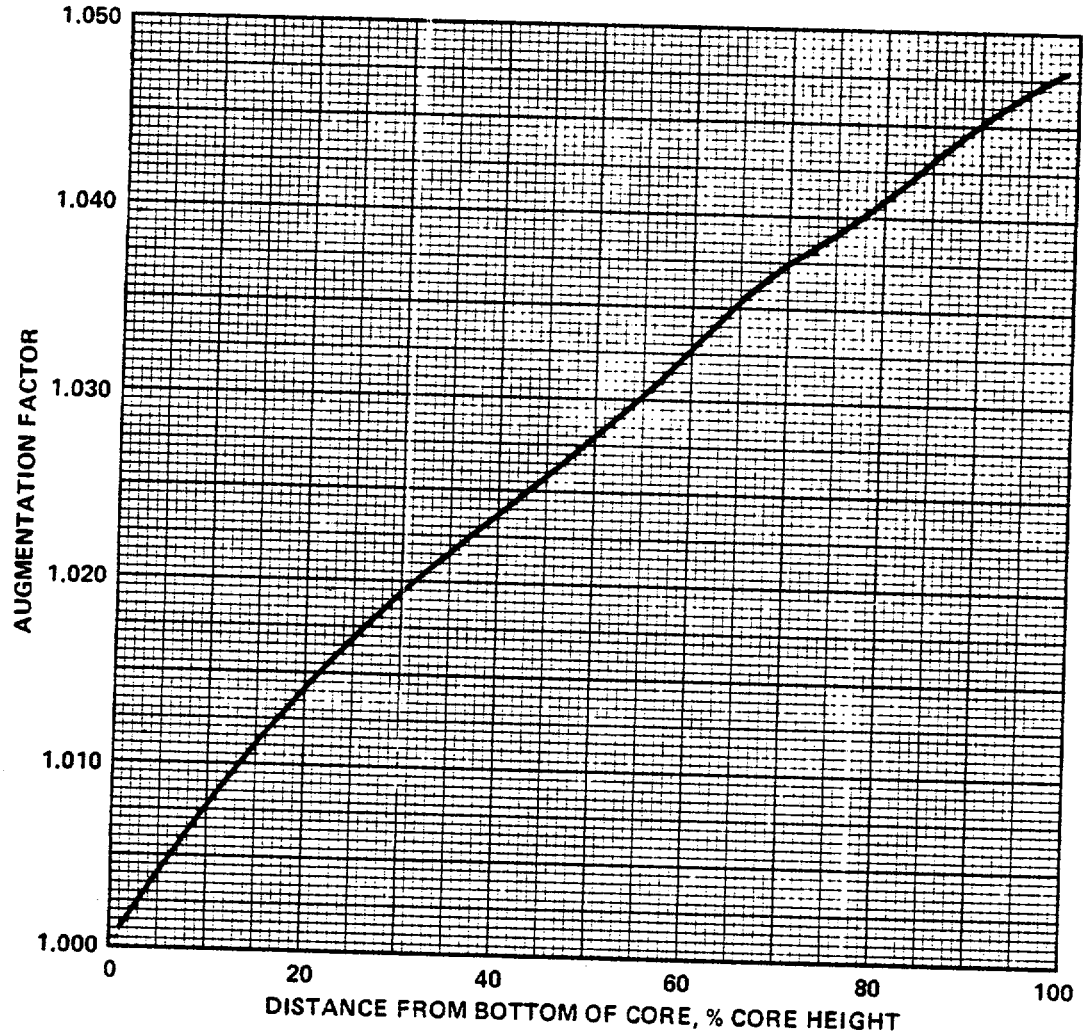

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.61.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.61, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+Tq)$ and F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map 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^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

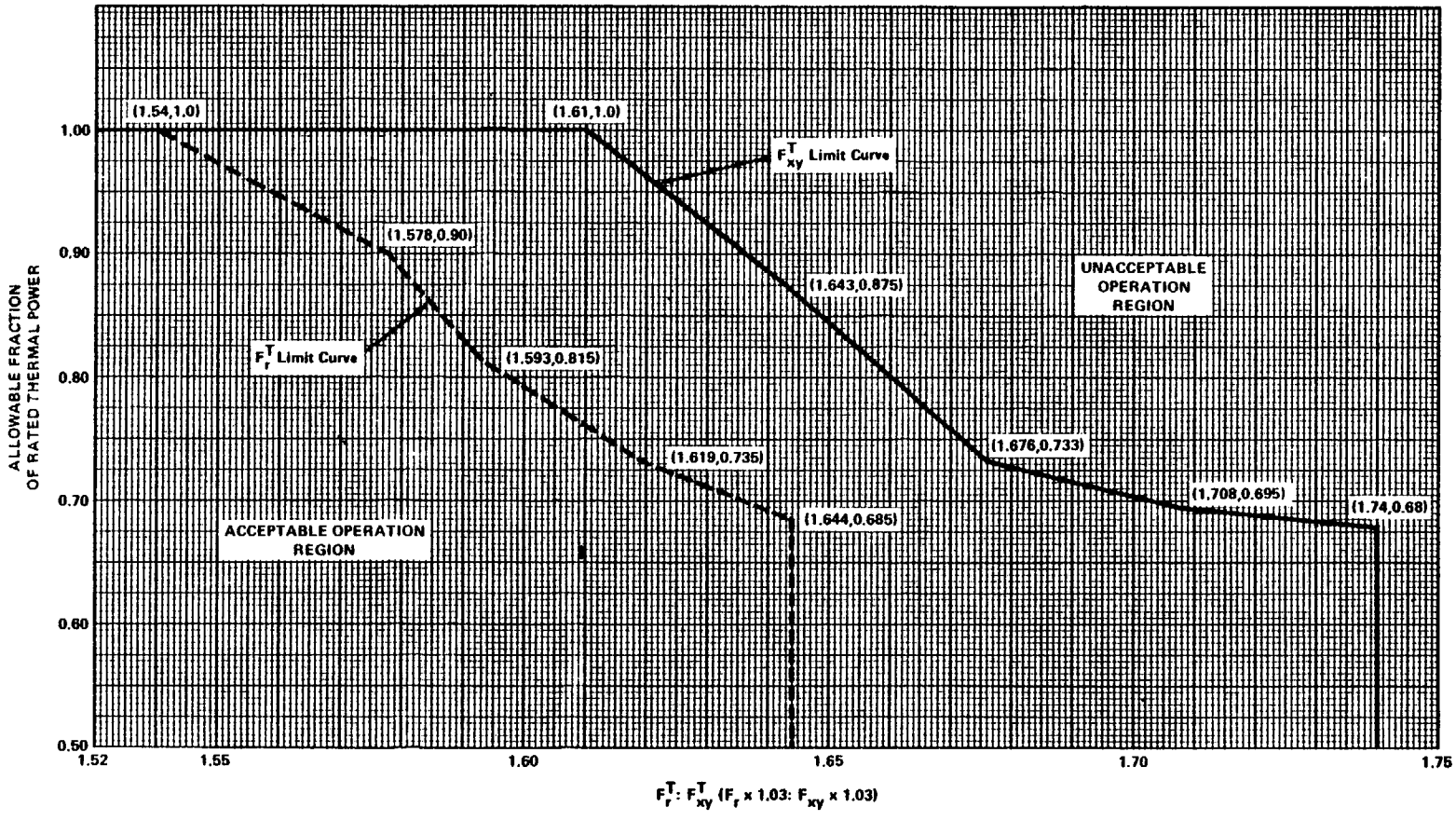

FIGURE 3.2-3
Total Radial Peaking Factor Versus Allowable Fraction of RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to $\leq$ 1.54.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.54, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

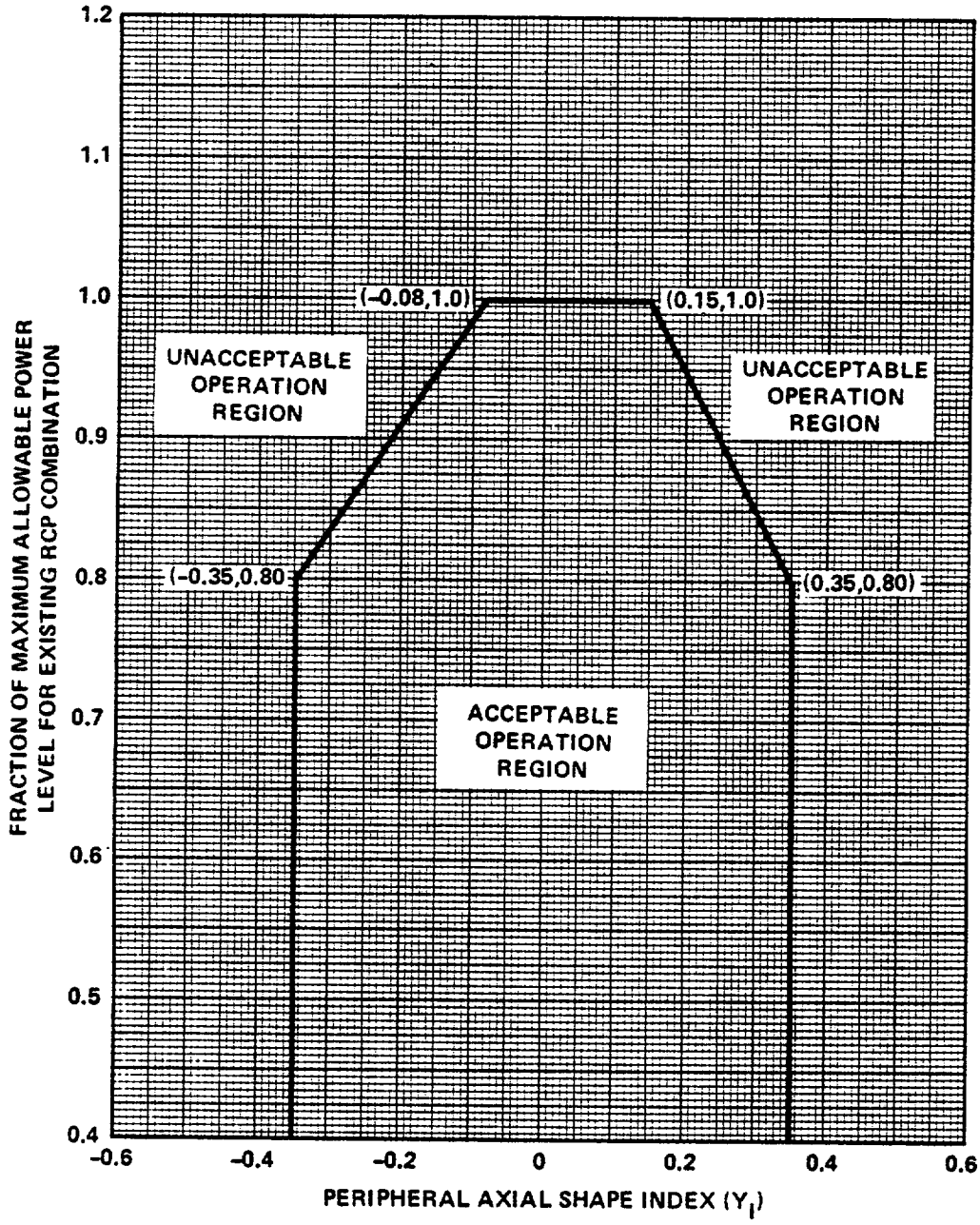

FIGURE 3.2-4

DNB Axial Flux Offset Control Limits

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.030.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 12 hours, and

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

Parameter	LIMITS			
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	≤ 548°F	**	**	**
Pressurizer Pressure	≥ 2225 psia*	**	**	**
Reactor Coolant System Total Flow Rate	≥ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	Figure 3.2-4	**	**	**

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

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	≤ 0.50 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.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

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

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, and 3.2.3 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2 and 3.2.3 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY wthin 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2 or 3.2.3 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2 or 3.2.3 are suspended.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.4% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 3813 gallons of 7.25% boric acid solution from the boric acid tanks or 47,204 gallons of 1720 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Transient Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

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 1.058, 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^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} 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^T_r 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 $F^T_{xy} = F_{xy}\ (1 + Tq)$ 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.

3/4.2.5 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 CE-1 calculated DNBR of 1.19 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.

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 CE-1 calculated DNBR above 1.19 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

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

LOW POPULATION ZONE

FIG. 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-318

1.0 Introduction

By application dated July 26, 1978, and supplemental information dated July 31, August 14, September 7 and October 6, 16 and 17, 1978, Baltimore Gas and Electric Company (BG&E or the licensee) requested an amendment to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2 (CCNPP-2). The amendment request consisted of:

(1) Approval to operate with modified (sleeved and reduced flow) Control Element Assembly (CEA) guide tubes;

(2) Technical Specification (TS) changes resulting from the analyses of Cycle 2 reload fuel; and

(3) TS changes authorizing the removal of part length control element assemblies (PLCEA's).

The associated specific TS changes are described in Section 4.0 of this Safety Evaluation (SE).

2.0 Background

In the original Cycle 2 reload application for CCNPP-2, BG&E proposed to replace 84 irradiated fuel assemblies with new fuel assemblies that have stainless steel sleeves installed in the CEA guide tubes, sleeve any fuel assemblies that have substantial guide tube wear or will be placed in CEA locations, and remove the PLCEA's from the Cycle 2 core.(1)

The CEA guide tube wear problem in Combustion Engineering (CE) designed facilities was first discovered at Northeast Nuclear Energy Company's Millstone Unit No. 2 in December 1977. Since that discovery, four of the facilities utilizing the CE design have performed refueling operations where worn guide tubes or guide tubes in new fuel assemblies to be placed in CEA locations have been sleeved. These facilities are Millstone Unit No. 2, Calvert Cliffs Unit No. 1, St. Lucie Unit No. 1 and Maine Yankee. All of these NSSS designs are very similar, especially CCNPP-1 which is identical, to CCNPP-2.

Our review of the CEA guide tube wear issue for the four reactors mentioned above has shown that sleeving is an acceptable means of mitigating guide tube wear for one cycle of operation(2,3). Since the extent of guide tube wear varies from reactor to reactor, we must review the extent of sleeving at a specific reactor. We must also review the licensee's stress analyses for sleeved and unsleeved guide tubes to determine if the stresses are still within allowable limits. This review is presented in Section 3.1 of this SE.

CE, the involved licensees, and the NRC staff have considered the sleeving of guide tubes as an "interim fix" since it was first proposed for CCNPP-1 in early 1978. CE has been involved in considerable experimental work to develop a "long term or permanent fix" since that time. In a meeting with the NRC staff on August 7, 1978, BG&E and CE presented the results of additional CEA guide tube flow tests supporting the earlier tentative conclusion that the wear stems from flow-induced vibration. The flow of primary importance is the guide tube flow. BG&E has proposed to install 16 fuel assemblies in CCNPP-2 Cycle 2 with reduced flow through the CEA guide tubes to reduce the CEA vibration and therefore migigate the guide tube wear. Maine Yankee has returned to operation with two essentially the same test fuel assemblies in service. Our SE of this modification found that no bulk boiling in the guide tube is anticipated since the available cooling exceeds the minimum required to preclude exceeding the saturation temperature at any axial position in the guide tube(3). The evaluation of the demonstration test, described by BG&E letter of September 7, 1978(4), is presented in Section 3.1.5 of this SE. The evaluation of the Cycle 2 fuel design is presented in Section 3.2 of this SE.

In a further effort to limit CEA guide tube wear, BG&E has proposed the removal of all PLCEA's[1]. The evaluation of the removal of all PLCEA's is presented in Section 3.3 of this SE.

The Emergency Core Cooling System (ECCS) performance analysis for Cycle 2 was provided in July 31, 1978[5]. Section 3.6 of this SE provides the Cycle 2 Loss of Coolant Accident (LOCA) analysis.

In the process of this review, we have requested and received additional information necessary for this evaluation[6 thru 11].

BG&E requested a new clarifying footnote for TS Table 3.2-1, Departure from Nucleate Boiling (DNB) parameters. We concluded a change in the basis will accomplish the same goal of clarification. BG&E agreed to our proposed change to TS Basis 3/4.2.5.

CCNPP-2 is licensed to operate at 2700 MWt. The rated power and all operating conditions remain the same for Cycle 2.

3.0 Evaluation

3.1 CEA Guide Tube Integrity

Indicationsof significant wear in the CEA guide tubes of fuel assemblies have been found during the fuel inspection program during refueling operations at four CE designed facilities. The guide tube wear has been observed at the location of the CEA tips during long periods of power operation. CCNPP-2 was operated with the CEA in the "full up" position for about two-thirds of Cycle 1 and in the "three inch inserted" position for the remainder of the cycle. Operation with the CEA inserted three inches was authorized by Amendment No. 13, to the CCNPP-2 license[12].

The guide tubes serve in a dual capacity as the primary structural members of the fuel assembly and as guiding channels for the control rods during insertion.

Considering the above findings, BG&E instituted an eddy current testing (ECT) program to quantify the extent of wear experienced during Cycle 1[4]. This program was developed to assess the thermal hydraulic performance and structural integrity of fuel assemblies with worn guide tubes for service in Cycle 2. The observations were incorporated in analyses to demonstrate the ability of the core to maintain its coolable geometry and the ability of the CEA's to scram, as required by the safety analyses. The licensee has concluded that fuel

assemblies with worn guide tubes can be operated safely. However, the licensee has decided to modify 131 of the 217 fuel assemblies in the core by an addition of stainless steel sleeves. The 131 sleeved assemblies include the following: 68 new assemblies with what has been the standard size and number of the flow holes; 14 assemblies which were in CEA locations in Cycle 1 but will be in non-CEA locations in Cycle 2 and 49 assemblies which were in non-CEA locations in Cycle 1 but will be in CEA locations in Cycle 2. The sleeves will restore lost structural margins to those CEA guide tubes which developed wear during Cycle 1 and will mitigate wear in those assemblies placed in CEA locations in Cycle 2.

CE developed the method of reinforcing worn guide tubes with thin stainless steel sleeves. The sleeves are inserted within the guide tubes, bridging the worn cross-sections, to provide a significant increase in strength and stiffness.

The sleeves are made of type 304 stainless steel, slightly cold-worked to provide a yield strength of over 60,000 psi. They are chromium plated on the ID and on the upper part of the OD to improve wear resistance. The sleeves extend from the top of the guide tube to several inches below the location of observed or expected CEA-induced wear. The sleeves are securely fastened in place by mechanically "bulging" both the sleeve and the guide tube at the lower end of the sleeve. This "bulge" extends for approximately one inch axially, and results in diametral expansions of the guide tubes of a few hundredths of an inch on new (unirradiated) guide tubes, and slightly less on used (both worn and unworn) irradiated tubes.

In addition to the guide tube expansion, the lower portion of the sleeves is expanded diametrally toward the guide tubes, so that the annular gap between the guide tube and the sleeve is approximately zero at room temperature. At operating temperature contact stresses develop from differential thermal expansion between the Zircaloy and the stainless steel. The gap in the upper portion of the assembly permits axial and radial differential thermal expansion of the sleeve without imposing significant loads on the assembly.

A series of slots and holes is provided in the sleeves to permit water flow in the annulus between the sleeve and the guide tube, minimizing the possibility of "steaming" caused by poor heat transfer between the sleeve and the guide tube.

The sleeving modification serves as an interim solution to mitigate the effects of guide tube wear but does not eliminate the source of the wear. Investigations are continuing by CE through out-of-reactor flow visualization tests in an effort to characterize the mechanism of flow-induced control rod vibration which causes the wear. Initial results indicate that the control rod vibration amplitude is sensitive to the magnitude of coolant flow through the guide tube. Prototypes of fuel assemblies, designed for either decreased guide tube flow or flow diversion, have been sucessfully tested in the CE out-of-reactor test facility. One of the prototypes, employing decreased flow in 16 fuel assemblies, will be incorporated on a test basis during Cycle 2. (See discussion of Demonstration Program in Section 3.1.5.)

All guide tubes in fuel bundles to be under CEA's during Cycle 2 will be unworn and sleeved with the exception of 12 of the demonstration assemblies.

3.1.1 Structural-Mechanical

The stainless steel sleeve provides reinforcement by adding strength and stiffness in the worn region. It is free to expand axially under heatup or cooldown. Consequently, because of its manner of installation, the sleeve does not provide axial support. However, it does significantly limit lateral deflection of the guide tube arising from both external moments and moments generated by the asymmetrical wear and thus reduces guide tube stresses.

The licensee has performed fuel assembly stress analyses using loadings for normal and accident conditions and the limiting amount of guide tube wear observed in CE cores, with and without sleeves. The resulting stresses were below allowable values. The various mechanical loads applied to the fuel assemblies included: fuel assembly holddown loads, fuel assembly handling loads, CEA scram deceleration loads, and seismic loads. The capability of the guide tubes to sustain these loads was determined by demonstrating that the lateral deflection of the guide tubes and the associated mechanical friction during scram were insufficient to prevent CEA insertion and that a coolable geometry was maintained by limiting permanent deformation of the fuel assembly.

The licensee has provided an analysis of the mechanical integrity of the core for a postulated LOCA and has concluded that the fuel remains in a coolable array. While these analyses did not include a treatment of asymmetric blowdown loads, the addition of sleeves would not change the overall core response to these loads. However, a review of the response of the core to this loading condition has been deferred pending resolution of the generic Category A-2 Task Action Plan, "Asymmetric Blowdown Loads on PWR Reactor Vessels." The targeted completion date of this program that includes a revised LOCA analysis for the largest credible break size is January 1980. The continued operation in the interim period of time is justified in view of the low probability of a large pipe break.

The conclusion that the probability of a pipe break severe enough to result in substantial transient loads on the vessel support system or other structures is acceptably small was derived from NRC staff short-term, interim criterion to determine if an acceptable level of safety exists for the reactor vessel supports of operating PWR's under conditions of a postulated pipe break. This interim criterion is based on a simplified probabilistic model that incorporates elastic fracture mechanics techniques to estimate the probability of a pipe break. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick walled pipe. Determination of the critical flaw size was based on an estimated fracture toughness value at a minimum temperature of 200°F and a uniform tensile stress equal to the consideration of various operating conditions producing elastically calculated stresses ranging in value from 1 to 3 times the material minimum yield strength.

Then using the calculated critical flaw size, the subcritical growth rate, and an estimated probability distribution of an undetected flaw in thick-walled pipe welds, the upper bound probability of pipe break was estimated to be acceptably low. This conclusion is in agreement with a recent publication by Dr. S. H. Bush, previously of the ACRS staff, which states that actual failure statistics confirm low rates in large pipes, with higher rates as the pipe size decreases.* The estimated pipe break probability is considered acceptably low to justify short-term operation of nuclear power plants.

In addition, other conservative factors exist which not only tend to mitigate the resulting loads of this postulated accident but further reduces the low probability of the occurrence of this event. These factors are: (1) that the break of primary concern must be very large, (2) that it must occur at a specific location, (3) that the break must occur essentially instantaneously, and (4) that these welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI. Therefore, we conclude that Cycle 2 operations of a CCNPP-2 can continue during the interim period of approximately two years while this matter is being resolved.

* Nuclear Safety, Volume 17, No. 5, September-October 1976, Article on Plant Safety Features, S. H. Bush.

A seismic analysis was completed for the effects of a postulated safe shutdown earthquake using the St. Lucie Unit No. 1 reactor vessel flange acceleration time history. The licensee has determined that the response to this time history envelopes the response at his facility. We conclude that this input conservatively defines seismic excitation of the core. The seismic analysis accounts for the interaction effects of adjacent fuel assemblies and the core shroud through the use of appropriate gap and impact elements. Therefore, we find the licensee's seismic analysis methods to be acceptable.

The licensee's analyses show that the stress during expected and postulated loading conditions in all guide tubes will remain below the unirradiated yield strength of the Zircaloy-4 material. In addition, the stainless steel sleeve stress intensity was calculated for the corresponding portion of the load that it carries and the stress was shown to be less than the material yield strength as given in Table 1-2.2, Appendix I, Section III of the ASME Pressure Vessel Code.

Interaction between the sleeve and guide tube creates substantial secondary stresses in addition to the before-mentioned primary stresses. Differential thermal expansion, differential irradiation induced growth, and creep have been considered and the resulting stresses have been determined.

Scram tests of a sleeved fuel assembly were also conducted to measure the 90% CEA insertion time. The tests were performed at operating temperatures and maximum flow conditions. Because of the changes to the guide tube bleed and cooling holes in the 16 modified bundles, the rod drop time in the 12 of those bundles which will be under CEA's in Cycle 2 is expected to increase. Therefore, the TS limit for 90% CEA insertion time will be increased from 2.5 to 3.0 seconds. This change in rod insertion time will be addressed in Section 3.4 of this SE. The measured insertion times for the sleeved fuel assemblies in other CE facilities fell within the normal TS limit of 2.5 seconds.

We have concluded that the licensee's calculated stress intensities are low enough to assure an adequate margin of safety. Furthermore, we have concluded that the licensee has demonstrated scramability and coolability as required by the General Design Criteria.

3.1.2 Control of Sleeving Procedure

The sleeving procedure* used was the same as that previously employed at other facilities where CE has performed sleeving modification. It includes qualification of the tooling before each operation, and replacement of those parts of the tooling subject to wear or deterioration

*CE Document No. 00000-ESS107, dated August 16, 1978.

before any deleterious effects on the process could occur. After sleeving, the following checks are made to ensure that the process was performed correctly.

(1) A pull test of 50 lbs. was performed on each sleeve.

(2) A visual inspection was performed to ensure that the sleeve is properly seated and that no debris is left in the area.

(3) Two separate gaging operations, using a single thimble gage, and a five-finger gage, were performed to ensure that there will be no interference with CEA operation.

3.1.3 Testing of Sleeved Guide Tubes

CE has performed a number of tests on sleeved guide tubes to verify the mechanical strength of the assembly, effect of sleeves on scram time, wear performance, and possible enhanced corrosion in the annulus between the sleeve and tube.

CE determined that the force necessary to pull out a sleeve from the guide tube is on the order of 800 pounds, and after 15 thermal cycles between room temperature and 625 F to simulate relaxation that would occur in service, the pull-out force was still greater than 400 pounds.

They also ran a loop test on a sleeved assembly with a CEA inserted at the nominal full-out position to simulate the condition causing the guide tube wear. The chromium plated sleeves showed no measurable wear after 464 hours, just a slight polishing or burnishing. The mating CEA finger tips also showed no wear, just a slight polish. Sleeved tubes were cut open and examined metallographically. No evidence of accelerated corrosion in the crevices (annuli) was found.

Scram tests were also run on sleeved assemblies to determine if the presence of the sleeves, or the reduction in clearance (reduced by about a factor of 2) between the CEA fingers and the inside of the tube would affect scram time. The results of these tests showed negligible effects on scram time.

3.1.4 Conclusion on Sleeving

We have evaluated the information submitted by the licensee and have concluded that the sleeved guide tubes will perform their function of reducing guide tube stresses to acceptably low values, and that

the mechanical design of the sleeved assembly is satisfactory for at least one fuel cycle. Any long term effects of relaxation of the mechanical "bulge" joint, including the possibility of radiation-enhanced relaxation, will have to be evaluated on selected assemblies at the next refueling outage.

BG&E will prepare a CEA guide tube evaluation program plan and submit it to the NRC at least 90 days prior to the scheduled refueling outage at the end of CCNPP-2 Cycle 2. The NRC will review the inspection program and the acceptance criteria, evaluating them with respect to fuel handling accident limits.

Some details of our evaluation are provided below.

3.1.4.1 Wear Resistance

Chromium plating of stainless steel and other similar alloys is commonly used in reactors, and has performed well. Chromium plate is extremely hard and wear resistant, often orders of magnitude better than materials such as Zircaloy and stainless steel. Further, the desirable frictional and anti-galling properties of chromium plate tend to reduce wear on mating softer materials. We conclude that chromium plated sleeves are not likely to be worn significantly during at least one fuel cycle.

3.1.4.2 Mechanical Properties

The mechanical joint between the sleeve and the guide tube is designed to be several inches below the area of excessive wear. The diametral expansion of the lower portion of the sleeve also is intended to be below the lowest wear area to prevent stressing of the worn region of the guide tube through thermal contact stresses between the sleeve and guide tube. There should be no prior cracks, notches, or severe hydriding where the stresses in the guide tube occur. The mechanical properties of the irradiated Zircaloy guide tube will be more than adequate to sustain the stresses involved.

3.1.4.3 Crevice Corrosion and Hydriding

The installation of a sleeve in a guide tube creates an annulus between the guide tube ID and the stainless steel sleeve OD which reduces to a crevice at the expanded region. In response to our questions, CE considered the possibility of enhanced corrosion and hydriding of the guide tubes in the crevice areas. They have stated that the crevice in the "bulge" area will be too small (and after short exposure will

be further closed up by corrosion product) to provide an entrance for the necessary water to cause extensive corrosion. They also argued that in the sleeve expansion region, this crevice will be closed at operating temperatures by the differential thermal expansion between the Zircaloy and the stainless steel, and the water will be squeezed out of the crevice, also limiting possible corrosion.

The crevice above the expanded region will be water filled. Holes and slots in the sleeve will allow some water circulation, minimizing the corrosion problems from stagnant water or acceleration of corrosion rate by the presence of steam phase.

We, too, have evaluated the possibility of detrimetral enhanced corrosion and hydriding in the sleeve-to-tube crevice. Factors considered by the NRC staff included:

(1) Similar crevices between stainless steel and Zircaloy are present in Westinghouse low parasitic fuel assemblies and operate successfully.

(2) The 464-hour tests described in Section 3.1.3 showed that there were no short-term problems under out-of-pile conditions.

(3) In sleeved assemblies, the portion of the guide tubes subjected to high loads, such as the "bulge" area, will not have wear-induced cracks or sharp notches. Under these conditions, some enhanced hydriding could be tolerated.

(4) In reviewing reactor experiences with crevices, no enhanced corrosion or hydriding has been noticed except in those cases where concentration of nonvolatile impurities such as lithium hydroxide has occurred. Since the lithium hydroxide concentration could be increased in the sleeve/tube crevice by boiling (even if intermittent), there is some possibility of accelerated corrosion, enhanced hydrogen pickup, or both. The long-range aspects of the problem, including study of the possibility of hydrogen migration to the bulge region, are still under active review by the NRC staff.

We have concluded that there is a likelihood of some enhanced corrosion but it should not be severe enough to compromise the mechanical integrity of the sleeved design. Operation with sleeved guide tubes is acceptable for Cycle 2.

3.1.5 Demonstration Program

Based on the favorable results of experimental work at CE, the licensee will insert 16 modified Batch D low enrichment fuel assemblies in Cycle 2. The modification consists of decreasing the number and size of the flow holes and the size of the bleed hole. Tests have indicated that the resulting decrease in guide tube flow was accompanied by less CEA flow-induced vibration and less guide tube wear. The modified assemblies will not have stainless steel sleeves installed in the guide tubes.

The 16 modified fuel bundles will be placed in a symmetric array throughout the core. Four will be put in non-CEA locations, four will go under single CEA's, and eight (as four pairs) will go under dual CEA's. Analyses performed by CE led to the conclusion that the reduced flow was still in excess of that necessary for CEA cooling. The decreased flow orificing is expected to increase the scram rod time to 90% insertion. Consequently, the TS CEA drop time limit will increase from 2.5 to 3.0 seconds. The increased insertion time was assumed in the reevaluation of the Design Basis Events resulting in no loss of conservatism, thus the safety analysis remains valid (see Sections 3.5 and 3.6).

Several factors played a role in determining the core locations of the 16 modified fuel assemblies. It was desired to include locations of relatively high CEA-induced wear in order that the new design would be put to a meaningful test but not by using only the highest wear locations. Based on preliminary ECT measurements of Cycle 1 fuel assemblies, the locations selected include a broad spectrum of wear, but not the very worst. Fuel management considerations dictated putting four bundles in non-CEA locations. It was desired to use some dual CEA locations. Modified fuel assemblies are hydraulically dissimilar from sleeved assemblies thus the two should not be mixed under dual CEAs. Since there were no experimental data showing how a dual CEA would perform under conditions of differential flow between the two assemblies involved, the modified assemblies were installed as pairs under dual CEA's.

The CE out-of-pile tests demonstrated that the modified fuel bundles will exhibit relatively little wear at the end of Cycle 2. BG&E will examine those demonstration assemblies at the end of Cycle 2 and report the results to the NRC. Since two assemblies of essentially the same design are operating in the current Maine Yankee cycle, albeit in locations of little anticipated wear, there will be some results in hand relative to the performance of the design which should disclose any major problems at CCNPP-2 prior to the end of Cycle 2 operation.

The CEA's which will be operating in the modified assemblies will represent two rod banks, Bank A and Bank 5. Bank A, a shut down bank, operates in the fully withdrawn position. This will allow a meaningful test of the design modification. Bank 5, a regulating bank, are repositioned during the cycle thus providing a different, presumably less severe, test of the design.

We conclude that the demonstration test of 16 reduced flow fuel assemblies is acceptable because the available wear data show that the guide tube wear in these CEA locations will be less than the acceptable wear observed in Cycle 1.

3.2 Cycle 2 Fuel Design

The 217 fuel assembly Cycle 2 core will consist of:

Batch Identification	Weight % (w/o) Enrichment	Number of Fuel Assemblies
D	3.03	48
D/	2.73	20
*D/	2.73	16
B	#	65
C	#	68

As a result of the CEA guide tube wear problem, 68 of the new fuel assemblies (Batch D and D/) will be sleeved. The 16 Batch *D/ fuel assemblies are a demonstration test which will not be sleeved but will have smaller bleed holes to reduce the cooling flow in the guide tubes. These 16 fuel assemblies will be placed in the core under dual CEA's, single CEA's and in unrodded locations in order to assess the performance of these assemblies under different installed configurations. The thermal hydraulics of the demonstration fuel assemblies will be evaluated in Section 3.2.3 of this SE.

BG&E has used the Cycle 3 reload analysis for CCNPP-1 as a "reference cycle" for the Cycle 2 reload analysis for CCNPP-2. Analyses outside the envelope of the reference cycle have been reanalyzed by CE for BG&E.

#irradiated fuel from Cycle 1

3.2.1 Mechanical

Other than the changes discussed above relating to the CEA guide tube wear, the mechanical design for the new reload fuel is identical to that of the Batch E fuel used in the reference cycle. The differences between the reload fuel and the original core loading are:

a) A fill gas pressure reduction of 40 psi.

b) A reduction in pellet dish depth by 0.002 inches, increasing the stack density to 10.46 grams/cc.

These changes are considered to be minor. The reduction in fill gas pressure results in a lower fuel temperature for the reload fuel and lowers internal pressure throughout the fuel cycle which would tend to decrease the predicted time to clad collapse. The 2 mil reduction in pellet dish depth acts to lessen the detrimental effects of pellet clad interaction.

An analytical prediction of the time of cladding creep collapse for all Cycle 2 fuel has been performed by CE using the CEPAN code which has been reviewed and approved by NRC. From this analysis it has been concluded by CE that the collapse resistance of all the fuel rods is sufficient to preclude cladding collapse during its design life time. The design lifetime of this fuel will not be exceeded during Cycle 2 operation. The Batch B fuel which is the most limiting with regard to clad collapse will have accumulated 22,700 Effective Full Power Hours (EFPH) by the end of Cycle 2. This is well below the predicted time to clad collapse which has been calculated to be greater than 28,400 EFPH. We find these design changes and analysis to be acceptable.

3.2.2 Nuclear

The licensee has evaluated the effect of the stainless steel CEA guide tube sleeve on the physics parameters of the core that are related to the safety and performance analysis to determine if these parameters were significantly affected by the insertion of guide tube sleeves in the upper region of the fuel. The licensee's evaluation was based on the conservative assumption that all CEA guide tubes are sleeved. The physics parameters considered were total core reactivity, radial pin peaking in the region of the sleeves, scram reactivity worth, and axial power shape. The evaluation showed that the scram reactivity decreased by less than 0.02% $\Delta\rho$; however, this is negligible considering the margin available (worth available less worth required) is greater than 1.5% $\Delta\rho$. The changes in total core reactivity, radial pin peaking in the region of the sleeves, and axial shape index are also insignificant.

The licensee has also reviewed the effect of the reduced diameter flow holes in the CEA guide tubes in the 16 demonstration Batch *D/fuel assemblies. The CEA guide tubes in these assemblies have a significantly reduced flow due to the smaller flow holes provided to alleviate guide tube wear. Consequently the cooling fluid attains a higher temperature and a correspondingly lower density than the water in the standard CEA guide tube. The results of the licensee's study show that no substantial change in axial and radial power distributions is anticipated as a result of the decreased flow in the modified CEA guide tubes.

The Batch D reload fuel is comprised of 3 sets of assemblies with 2 enrichments as previously described in Section 3.2 of this SE. Cycle 2 burnup is expected to be between 9800 Megawatt Days per Metric Ton Uranium (MWD/MTU) and 10,200 MWD/MTU. The licensee has examined the Cycle 2 performance characteristics for Cycle 1 termination point of between 16000 and 17000 MWD/MTU. The actual Cycle 1 burnup, as stated by the licensee, was 16182.4 MWD/MTU.

The Cycle 2 moderator temperature coefficient (MTC) is calculated to be $0.4 \times 10^{-4} \Delta\rho$/ F at the Beginning fo Cycle (BOC) and $-2.0 \times 10^{-4} \Delta\rho$/ F at the End of Cycle (EOC). These values for MTC are bounded by the values used in the reference cycle which are $0.4 \times 10^{-4} \Delta\rho$/ F at BOC and $-2.2 \times 10^{-4} \Delta\rho$/ F.[2] We find these values of MTC to be acceptable.

Doppler coefficients calculated for Cycle 2 are $-1.50 \times 10^{-5} \Delta\rho$/ F at BOC Hot Zero Power (HZP), $- 1.18 \times 10^{-5} \Delta\rho$/°F at BOC Hot Full Power (HFP), and $-1.25 \times 10^{-5} \Delta\rho$/ F at EOC HFP. These values are slightly more negative than the reference cycle at HFP at both the BOC and EOC conditions. Changes of this magnitude, 3% more negative at HFP EOC and 0.75% less negative at HFP EOC, have a minimal impact on the analysis of postulated Anticipated Operational Occurrences (AOO's) and accidents that result in a reactor cooldown. The slightly more negative values of the Doppler coefficient act to add additional conservatism to AOO's and accidents during which fuel temperature is tending to increase. We find the values of the Doppler coefficient calculated for Cycle 2 to be acceptable.

The total delayed neutron fraction for Cycle 2 has decreased slightly at EOC from that in the reference cycle. This would have a minor impact on the CEA ejection accident. The CEA ejection accident has been reanalyzed and is discussed in section 3.5 of this SE.

At EOC 2, the reactivity worth of all CEA's inserted, less the highest worth CEA stuck allowance is 8.1%$\Delta\rho$. The reactivity worth required to shutdown the plant including power defect HFP to HZP, shutdown margin and safeguards allowance required to control the steam line break incident at EOC 2 is 6.5%$\Delta\rho$. The margin available in negative reactivity is 1.6%$\Delta\rho$ which is more than adequate to account for any uncertainty in nuclear calculations. We find these shutdown margins to be acceptable.

3.2.3 Thermal Hydraulics

The licensee stated that the Departure from Nucleate Boiling Ratio (DNBR) analyses for Cycle 2 have been performed using the same methodology and design codes as were used in the reference cycle. The codes used in the DNBR analyses are COSMO and TORC which have been approved by the NRC staff. TORC was used in the generation of limiting conditions for operation on DNBR margin in the TS and was also used for all AOO's and postulated accidents which were reanalyzed for Cycle 2. Either TORC or COSMO was used in the analysis of those AOO's and postulated accidents not specifically reanalyzed.

The TORC thermal hydraulics computer code has been developed to replace COSMO-INTHERMIC.[13, 14] TORC employs the CE-1 DNBR correlation whereas COSMO-INTHERMIC employs the W-3 DNBR correlation. The TORC code has been approved for use in licensing and the CE-1 correlation has been approved with a 1.19 DNBR limit.[15]

Although the use of TORC/CE-1 involves a change in the DNBR safety limit from 1.30 to 1.19, there is no change in the acceptance criteria with respect to fuel damage. The change is a result of new experimental data and the CE-1 correlation derived from the data on file. The previous limit of 1.30 was for the W-3 correlation and the limit of 1.19 is for the CE-1 correlation. Either of these limits, when considered in conjunction with its DNBR correlation, corresponds to a 95% probability at a 95% confidence level that DNB will not occur. Therefore, the use of TORC impacts the DNBR Limiting Safety System Settings (LSSS) and Limiting Conditions for Operation (LCO) only. TORC/CE-1 produces results in better agreement with experimental data than COSMO-INTHERMIC/W-3. Based on these considerations, we find the use of TORC with CE-1 DNRB limit of 1.19 to be acceptable.

The licensee has stated that 101 fuel assemblies will exceed the NRC determined penalty threshold burnup of 24000 MWD/MTU during Cycle 2. At the end of Cycle 2 the maximum burnup attained by any of these assemblies will be 30600 MWD/MTU. The DNBR penalty at this burnup is 2.2 percent. The licensee has examined the power distributions for Cycle 2 and has found that the maximum radial peak at HFP in any of the assemblies that eventually exceed the 24000 MWD/MTU threshold is at least 7.3 percent less than the maximum radial peak in the entire core. This margin is considerably greater than the 2.2 percent penalty imposed on fuel assemblies exceeding the 24000 MWD/MTU burnup threshold in Cycle 2. Therefore, no power penalty for fuel rod bowing is required in Cycle 2.

The modifications to the fuel assemblies to alleviate the CEA guide tube wear problem have a small effect on their thermal hydraulic performance. As discussed previously in this SE, Cycle 2 will have two different modifications: 1) guide tube sleeving and 2) reduction in guide tube flow demonstration test.

The guide tube sleeving effects thermal hydraulic performance in three areas; core bypass flow, boiling in the guide tube sleeve annulus, and CEA cooling. As stated by the licensee, sleeving reduces the guide tube flow from 1400 lbm/hr to 700 lbm/hr. This change, however, compared to total core bypass flow is a minor effect which is in the conservative direction; i.e., it tends to increase the flow slightly through the core. Bypass flow must be maintained below 3.7% to preserve the design thermal margin. Sleeving improves this margin.

The second area of consideration is the potential for boiling in the guide tube-sleeve annulus. The licensee states that no boiling will occur in the region in which the sleeve is expanded into contact with the guide tube since the CEA linear heat rate of 3.68 KW/ft is below the boiling limit of 6.5 KW/ft. In the nonexpanded region, axial peaks can be maintained such that CEA linear heat rates are below the 1.2 KW/ft boiling limit. Therefore, boiling is unlikely in this region. If boiling does occur, slots and holes in the sleeve assure that any expansion due to boiling is relieved and no mechanical damage will be caused. It is our opinion that limited boiling in this region is acceptable.

The criteria for adequate CEA cooling is that there is no bulk boiling in the guide tube during operation. The licensee states that cooling flow of 388 lbm/hr is required to meet this criteria. The cooling flow of 700 lbm/hr exceeds the minimum by a substantial margin. We find this to be acceptable.

The 16 demonstration fuel assemblies will have reduced guide tube cooling flow due to the reduction in number and size of the flow holes. The CEA cooling flow for this design has been stated by the licensee to be 565 lbm/hr. This exceeds the bulk boiling criteria of 388 lbm/hr and has a minimal impact in the conservative direction on total core bypass flow. We find this to be acceptable.

3.3 Uncertainty in Nuclear Power Peaking Factors

3.3.1 Documentation of Uncertainties

Reference 16, which is still under review by the NRC staff, documents the assumed uncertainties in F_r^I and F_q^I of 5.1% and 5.8%, respectively, and Reference 17 documents the 4.6% water hole power peaking bias. At

the present stage of review, we have concluded that the 5.1% and 5.8% are nonconservative. In addition, the 4.6% should have an uncertainty associated with it which the licensee has not factored into their analysis. However, whatever uncertainty is inherent in the 4.6% could logically be applied to the 5.1% and 5.8%, and our present position is to accept the 4.6% fully and assign any uncertainties in the water hole peaking to the uncertainties in F_r^T and F_q^T.

3.3.2 Uncertainties in F_r^T and F_q^T Used in the Safety Analysis

In Reference 17, which documents the water hole peaking bias of 4.6%, certain computational conservatisms are cited which could act as credits to mitigate the effects of the additional 4.6% peaking. Based on these credits, we agreed with the CE licensees at a December 16, 1977 meeting that an additional penalty of 2.8% rather than 4.6% could be justified. However, with the newly adopted use of TORC CE-1, thermal margin/low pressure methodology and statistical methods, this reduction in penalty can no longer be justified and BG&E has used the full 4.6% peaking bias penalty in the safety analysis for Cycle 2 operation of CCMPP-2.

The 4.6% bias has been applied to the computed peak pin powers which are then used in the remainder of the safety analysis. This method accounts for the 4.6% bias completely, and it need not be incorporated in the uncertainty treatment. BG&E used uncertainties of 5.1% and 5.8% for F_r^T and F_q^T, respectively.

3.3.3 Status of NRC Staff Review of CE Uncertainties

We have submitted to BG&E an extensive list of questions concerning the justification for the 5.1% and 5.8%.[19] These questions were answered in part in Reference 20. However, a number of responses were not complete, and many of the questions of Reference 19 have not yet been addressed. From the available data, we conclude that uncertainties of 8% for F_r^T and 10% for F_q^T can be justified, and recent CE analyses have demonstrated sufficient credit to offset the difference between the 8% and 10% and the 5.1% and 5.8% used in the safety analysis.

BG&E performed the safety analysis for CCNPP-2 assuming that they would be able to reach 100% power level by identifying sufficient credit to offset the difference between the 8% and 10% and the 5.1% and 5.8% uncertainties used in the safety analysis for F_r^T and F_q^T, respectively. BG&E was able to identify sufficient credit for all but the LHR(F_q^T) LCO uncertainty of 10%, and BG&E proposed that this be changed. The licensee's basis for this change is that the 10% figure was determined by the NRC staff as a reasonable bound on the measurement uncertainty under the full range of anticipated operating conditions. In this cycle, BG&E proposes to reduce this uncertainty to 7% during equilibrium operation with the CEAs above the long term insertion limit. During load following operations, CCNPP-2 will be operated with the 10% LHR LCO uncertainty.

BG&E has stated that they can demonstrate that the measurement uncertainty during equilibrium operation is reduced to below 7% and proposes to use the 7% value as the uncertainty that should have been used in their safety analysis for the LHR LCO.

We estimate that the use of a 7% uncertainty in the incore instrument system measurements provides a 95/95 level of confidence/probability that the pin which is measured to be the hottest pin in the core will not violate the peak linear heat generation limiting condition for operation during steady state operation with control rods above their long term insertion limits. The 7% uncertainty provides an approximately 95/85 level of confidence/probability that the pin which is measured to be the hottest pin in the core will not violate the peak linear heat generation rate limiting condition for operation during power changes utilizing control rods inserted below their long term insertion limits. The TS specify the use of the 10% uncertainty udring load follow operation as defined in TS 1.29 to maintain the 95/95 confidence/probability level that the pin which is measured to be the hottest pin in the core will not violate the peak linear heat rate limits.

The TS limits the time that rods may be inserted beyond the long term insertion limits to four hours per 24 hour interval, five EFPD per 30 EFPD, and 14 EFPD per year. During this short term operation the 7% uncertainty provides an approximate 95/85 level of confidence that the pin which is measured to be the hottest pin in the core will not violate the peak linear heat rate limits. This TS time limit precludes the 95/85 tolerance limit from being in effect a very large fraction of the time. Operating for a short period with a 95/85 confidence/probability level has an insignificant impact on the overall safety of the plant. In addition, the insertion of the rods immediately reduces power which increases the margin to the peak linear heat generation limit.

3.3.4 Conservatisms in the Safety Analysis to Offset Uncertainties

As stated above, in lieu of justifying the **5.1% and 5.8%**, BG&E has cited a number of known conservatisms in the safety analysis methodology for CCNPP-2 for which they do not take credit[12]. These are enumerated in the following sections.

3.3.4.1 DNBR Conservatism

There are three areas in which conservatisms are known to exist in the DNBR safety analysis. In the first one of these, the degree of conservatism is known, since comparisons with a more exact model have been performed. In the latter two, the degree of conservatism has not been evaluated; however, it is clear that they are conservative. These three areas of conservatism are:

Conservatism in DNB Limits Due to Statistical Combinations

In the DNB limit analysis, the assumed uncertainties in various measured parameters are not combined in a single equation, but

are factored into functional relationships as biases at various points in the analysis[18]. This biasing of functional relationships throughout the analysis is equivalent to adding the absolute power uncertainties equivalent to the uncertainties in the various measured parameters and applying the total power uncertainty to the best estimate calculation. The specific uncertainties along with their equivalent power uncertainties are given below.

ASI	0.06 ASIU	$\geq$ 2.2%
Pressure	22 PSI	$\geq$ 0.8%
Temperature	2 F	$\geq$ 0.9%
Flow	4%	$\geq$ 5.0%
Power	5% (LSSS)	$\geq$ 3.5%
	2% (LCO)	$\geq$ 1.4%

In the BG&E analysis, the equilivant sum of these uncertainties is 12.4% for LSSS, 10.3% for LCO. Treating these measurement uncertainties as statistically independent, the proper method for combining them is Root Sum Squire (RSS). The RSS combination yields 6.6% for LSSS, 5.8% for LCO, giving a net conservatism in the analysis of 5.8% for LSSS, 4.5% for LCO.

Conservatism in Pseudo-Hot-Pin Synthesis

In computing F_r^T, the pseudo-hot-pin synthesis is used in the CE core monitoring program, INCA[16]. In this technique, the axially integrated hot pin power is computed by integrating the hottest pin in each axial region. If a single pin is the hottest in every axial region, then this technique produces correct results, and if the hot pin is different in different axial regions, as BG&E states is realistically expected, the results will be conservative. We have not allowed credit for this conservatism because it has not been quantified.

Conservatism in Axial Flux Shapes

In the multiplicity of the axial flux shapes used in the safety analysis, as computed by the QUIX code, more severe shapes are calculated than are expected to occur during actual operation[18]. This results in conservatism in the safety analysis. We have not allowed credit for this conservatism because it has not been quantified.

3.3.4.2 Conservatism in LHR LSSS

The TS Axial Shape Index (ASI) trip tent is constructed to lie within the safety analysis LSSS tent. The construction has been done so that

the TS ASI trip tent contains at least 5% conservatism compared with the safety analysis tent. A TS ASI trip tent identical to the safety analysis ASI LSSS tent would be acceptable to us. Therefore, we find it acceptable to use the 5% margin to offset nonconservatisms in the LHR LSSS.

3.3.4.3 Conservatism in LHR LCO

BG&E states that the ECCS analysis of record is conservative because the calculation of the peak clad temperature was made without using the PARCH code. The late reflood heat transfer benefit from the use of the PARCH calculated steam cooling heat transfer would have reduced the peak clad temperature (PCT).

This analysis predicts that if the peak linear heat generation rate prior to the accident is 15.5 KW/ft, then the PCT during the event will be less than or equal to 2123 F. A predicted value of 2200 F would be acceptable (10 CFR 50.46). Hence there exists margin to accommodate small changes in the assumed initial value of the peak linear heat generation rate of 15.5 KW/ft.

For a small change in the peak linear heat generation rate, the change in the PCT can be approximated by a derivative relationship, that is; a 1% increase in F_q^T corresponds to a 10 to 20 F increase in PCT. If F_q^T is raised by 1.2%, this corresponds to an approximate increase of from 12 F to 24 F in PCT. This 1.2% balances the deficit between 5.8% and 7.0%, and on this basis we find the use of 5.8% uncertainty in F_q^T for the LHR LCO acceptable.

It should be pointed out that by performing a revised ECCS analysis, BG&E could probably have demonstrated a 4.2% credit which would balance the deficit between 5.8 and the 10% previously used.

3.3.4.4 NRC Staff Evaluation of Credits

The required credits to offset the nonconservatism in the peaking factors are the following:

DNBR LCO & LSSS	LHR LSSS	LHR LCO
8.0 - 5.1 = 2.9%	10.0 - 5.8 = 4.2%	7% - 5.8% = 1.2%

Since we have just demonstrated credit in excess of these figures, we find it acceptable to use the identified margin to offset the nonconservatism in F_r^T and F_q^T.

3.4 Removal of Part Length CEA's

To preclude guide tube wear at locations occupied by part length CEA's (PLCEA's), the licensee is removing the PLCEA's and installing guide tube plugs for Cycle 2 operation. The guide tube plugs will have essentially the same configuration as the standard CEA spiders. The CEA guide tube plugs will have in place of the 5 standard control fingers, 5 solid 304 stainless steel fingers that extend approximately 5 inches into the top of the fuel assembly. Each finger has a leaf spring type attachment that positions the finger such that it will not vibrate against the wall of the upper end post fitting. Considering their shorter length and increased stiffness, the plugs will not be susceptible to the flow induced vibration that has produced guide tube wear. In any case, the CEA plugs do not extend into the Zircaloy portion of the fuel assembly. Therefore, any vibration that does occur will not result in CEA guide tube wear.

Mechanically the plug is held in place by a spring compressed against the upper guide structure with a downward force of 150 lbs plus the weight of the plug assembly. The hydraulic uplift force on the plug assembly is calculated to be 90 lbs so no uplift problem is anticipated with the plug assembly.

The licensee has stated that the CEA guide tube cooling flow rate increases by 22% when the PLCEA's are replaced by the plug assembly due to decreased hydraulic resistance caused by the difference in length of the PLCEA's as compared with the plugs. This increases the core bypass flow by 0.02% of the vessel flow rate. This increase in bypass flow is compensated for by the decrease in bypass caused by CEA guide tube sleeving and the reduction in the CEA guide tube flow holes. In any case, resulting total core bypass flow, 3.4% at the end of the second cycle, remains below the thermal design limit on bypass flow of 3.7% of the total vessel flow rate.

The fingers of the CEA guide tube plugs are several inches above the active fuel and a physics analysis performed by the licensee indicates that there will be no adverse effect on the core power distribution as a result of installing the plug assemblies.

The licensee assessed the impact of replacing the PLCEA's with plugs in the safety analysis for Cycle 2 and has determined that the probability of occurrence of design basis events is not increased and the consequences of those events remain within the previous analysis. We believe that based on acceptable bypass flow and minimal nuclear effects, operation with plugs is acceptable for Cycle 2.

3.5 Analyses of Anticipated Operational Occurrences (AOO's)

References 1 and 5 discuss the safety analyses of postulated AOO's for CCNPP-2 Cycle 2. The licensee classifies the list of postulated AOO's into two categories. The first category includes those AOO's for which the Reactor Protection System (RPS) LSSS's as specified in the plant TS assure that the Specified Acceptable Fuel Design Limits (SAFDL's) are not exceeded. The second category includes those AOO's for which initial steady state overpower margins are maintained by adherence to the LCO's specified by the TS for the plant. Adherence to the LCO's assure that SAFDL limits are not exceeded.

The loss of flow transient causes the most rapid change in DNBR and both a reactor trip and steady-state overpower margin is required to maintain the SAFDL's. The LCO's and LSSS's for Cycle 2 TS were calculated using the methods described in CENPD-199, "CE Setpoint Methodology." This topical report is currently under review by the NRC staff. The review has progressed sufficiently to conclude that its application for this purpose is acceptable.

The required AOO reanalyses were done using the computer code CESEC which is currently under review by the NRC staff. However, the review has progressed to the point that there is reasonable assurance that the results dependent on CESEC will not be appreciably altered by any revision resulting from our review.

The licensee stated in Reference 1 that the need for reanalysis of a particular AOO is determined by comparison of the key parameters for that AOO to those of the last cycle for which a complete analysis was performed. If the key parameters are within the envelope of the reference cycle data, no reanalysis is required. A reanalysis might also be performed in case it could lead to a significant relaxation of TS restrictions.

BG&E has proposed a change to the plant TS raising the high power level trip from 106.5% to 107.0% power. The safety analysis assumes a trip at 112% of rated power. A 5% power measurement uncertainty has always been applied in the process of generation LSSS limits. In the past, this uncertainty was applied in a multiplicative fashion (which yields the equivalent of a 5.5% of rated power uncertainty), but evaluations showed that application of the uncertainty in this fashion is conservative. In accordance with current methods (as described in CENPD-199-P), the power measurement uncertainty is now deducted algebraically. It is this difference in the manner in which the uncertainty is applied that leads to the 107% versus 106.5% LSSS limit. We have reviewed this change and find it to be acceptable.

The sleeving of the CEA guide tubes has a negligible effect on CEA rod drop times but the reduction of the CEA guide tube flow holes does impact on the rod drop times. As previously stated, the Cycle 2 reload will have 16 prototype fuel assemblies with reduced flow holes. The effect of these flow holes on rod drop times is to increase the time to 90% insertion from 2.5 to 3.0 seconds. BG&E has identified this as a proposed change to the plant TS. To assess the impact of this change in rod drop time, the licensee has examined all the design basis events which could require a trip to prevent exceeding SAFDL limits. An evaluation of these design basis events showed that only 5 events may be adversely affected by increased scram time. For these evaluations it was conservatively assumed that all the CEA's are inserted at the same insertion versus time characteristic curve as in the 16 fuel assemblies with the reduced guide tube flow. Those transients which were reanalyzed are discussed below.

3.5.1 CEA Withdrawal

The CEA Withdrawal event has been reanalyzed for Cycle 2 due to an increase in differential CEA worth relative to the reference cycle analysis and the increase in the CEA insertion time to 90% insertion from 2.5 to 3.0 seconds. The reanalysis is based on the consideration of the zero and full power cases and shows that for both cases minimum DNBR and peak linear heat rate limits are not violated. A reevaluation of the Thermal Margin/Low Pressure (TM/LP) trip bias factor verified that it is still conservative.

3.5.2 Reactor Coolant System (RCS) Depressurization

The licensee has proposed a change to plant TS that increases the pressure time delay from 14 to 30 psia in the TM/LP system. This pressure bias which accounts for lags in measurement of process variables is set by the RCS depressurization transient for Cycle 2. The reference cycle required a pressure bias of 30 psia to envelope future cycles. The licensee has analyzed this transient using the new scram rod insertion versus time curve and has determined that the 30 psia time allowance is conservative. The total pressure allowance in the TM/LP will be 52 psia which is comprised of a 22 psia pressure measurement allowance plus the 30 psia time delay allowance. We find this change in plant TS to be acceptable.

3.5.3 Loss of Coolant Flow

The loss of coolant flow transient was reanalyzed for Cycle 2 since the four reactor coolant pump coastdown is faster than the reference

cycle and the control rod drop time to 90% insertion has been increased by 0.5 seconds due to the reduction in the CEA guide tube cooling holes. The analysis shows that the minimum calculated DNBR is greater than or equal to 1.19 based on the CE-1 correlation. In evaluating the effect of increased CEA insertion time on the Loss of Coolant Flow and Seized Rotor events the licensee used a spectrum of axial power distributions to determine the most adverse power distribution at a given axial shape index. In reviewing these axial power distributions it has been determined by the licensee that the maximum CEA insertion needed to maintain the DNBR greater than or equal to 1.19 is 55% for the Loss of Coolant Flow and Seized Rotor events. The original CEA insertion versus time characteristic (i.e., for the unsleeved fuel assemblies) used in the original safety analysis is identical to that used in the revised safety analysis up to 55% rod insertion. Past the 55% insertion point, the insertion tends to slow down due to the smaller flow holes in the lower section of the guide tube that act to increase the damping by providing a greater dash pot effect. Therefore, for both the Loss of Coolant Flow and Seized Rotor events, the reduced flow holes in the CEA guide tubes which cause the rod drop times to change do not reduce the minimum DNBR below that previously calculated.

3.5.4 Conclusion

We have reviewed the licensee's analyses of AOO's for Cycle 2 operation of CCNPP-2 and conclude that they are acceptable.

3.6 Postulated Accidents Other Than LOCA

The licensee has reviewed the postulated accidents other than LOCA. References 1 and 5 discuss the safety analysis performed for this category of accident for CCNPP-2 Cycle 2. Postulated accidents other than the LOCA, as other plant events, need to be reanalyzed only if the key parameters influencing the event are not enveloped by the reference cycle data. Those accidents that were reanalyzed are discussed below.

3.6.1 Seized Rotor

The Seized Rotor event has been reevaluated to verify that the increase time to 90% CEA insertion of 0.5 seconds does not cause the DNBR to fall below 1.19 based on the CE-1 correlation. In reevaluating the event the licensee used the same methodology as previously discussed in this SE, Section 3.4.3, for the Loss of Coolant Flow. Since the CEA insertion versus time characteristic used in the analysis for the fuel assemblies with the modified flow holes is identical to that characteristic used in the reference cycle analysis for less than 55% insertion, and the maximum CEA insertion needed to turn the DNBR in the transient to keep it above 1.19 is approximately 55%, the licensee concluded that the results and conclusions reported previously for the Seized Rotor event remain valid.

3.6.2 CEA Ejection

The full power and zero power CEA ejection events were reanalyzed due to increases in the tilt allowance, K-factors, post-ejection radial peaking factors, time to 90% CEA insertion, and a decrease in the delayed neutron fraction for CCNPP-2 Cycle 2. The analysis was done using the most limiting parameters at any time during Cycle 2 to provide added conservatism. The results of the reanalysis show that the decreases in the ejected CEA worth and axial power peaks offset the increases in the tilt allowance, K-factors, post ejection radial peaking factors, time to 90% CEA insertion and the decrease in delayed neutron fraction. Consequently, the conclusions reached in the reference cycle analysis remain applicable for CCNPP-2 Cycle 2. The licensee's analysis shows that for both the zero power and full power cases the clad damage pellet enthalpy threshold of 200 cal/gm is not violated. Therefore no fuel rods are predicted to suffer clad damage.

3.6.3 Conclusions

We have reviewed the accident analyses for events other than LOCA for CCNPP-2 Cycle 2 and conclude that they are acceptable.

3.7 Cycle 2 LOCA Analysis

The licensee performed a LOCA analysis for CCNPP-2 Cycle 2 to reevaluate the limiting large break LOCA(5). The licensee has stated that the blowdown and refill-reflood hydraulic calculations performed for CCNPP-1 Cycle 2, as described in Reference 21, apply to CCNPP-2 Cycle 2 since the primary system and containment parameters are the same for both units. The only differences between CCNPP-1 and CCNPP-2, so far as ECCS is concerned, are the fuel stored energy parameters which have no effect on the blowdown and refill-reflood hydraulic calculations. Therefore the licensee performed clad temperature analysis using the STRIKIN-II code to account for the different fuel pin conditions, only.

Although the 1.0 double ended slot/pump discharge (DES/PD) break was the worst for Cycle 1, the 0.8 DES/PD is the worst for Cycle 2 since fuel rod failure is predicted to occur during blowdown.

In Cycle 2, for burnups greater than or equal to 27506 MWD/MTU, the fuel to clad gap gas pressure becomes high enough to cause clad rupture during blowdown. Clad rupture during blowdown leads to higher reflood clad temperatures because of increased Zirconium-steam reaction and decreased effectiveness of rod-to-rod thermal radiation. The earlier in the fuel cycle clad rupture during blowdown occurs and the earlier in the blowdown phase clad rupture occurs, the greater the stored energy in the fuel. This increase in fuel stored energy leads to higher PCT. The 0.8 DES/PD break was determined to have the highest clad temperature during blowdown. This break will then have blowdown rutpure occurring earliest in the fuel cycle and earliest in the accident and consequently the highest PCT. For burnups less than 27506 MTD/MTU, blowdown rupture is not predicted to occur and the 1.0 DES/PD reported as the worst break for Cycle 1 will continue to result in the highest clad temperature.

ECCS PERFORMANCE ANALYSIS RESULTS

BREAK	PEAK CLAD TEMPERATURE	LOCAL CLAD OXIDATION	HYDROGEN GENERATION
1.0 DES/PD	1991 °F	10.24%	<0.51%
0.8 DES/PD	2123 °F	15.53%	<0.63%

As indicated in the tabulation above, the predicted values of PCT, local clad oxidation and hydrogen generation are below their respective limits of 2200 F, 17% and 1% as specified in 10 CFR 50.46(b).

We conclude, as a result of our review, that the CCNPP-2 Cycle 2 ECCS performance is in conformance with the criteria specified in 10 CFR 50.46(b) and is, therefore, acceptable.

4.0 Technical Specifications

The TS changes proposed for this amendment are summarized in the following statements.

Page 1-3, 3/4 1-20, 3/4 1-21, 3/4 10-1 and 5-4

These changes result from the removal of the PLCEA's and, therefore, the reference to PLCEA's in TS 1.13, 3.1.3.2, 4.1.3.2, 3.1.3.3, 3.10.1, 4.10.1.3 and 5.3.2.

Page 1-6

The definition of Load Follow Operation will be introduced here to be used for LHR LCO uncertainties in Specification 4.2.1.4.

Page 2-7

The RPS trip setpoint and allowable value for maximum rated thermal power would be increased from 106.5 to 107.0%. This change would be in response to a new mathanatical method of combining the errors of measurement.

Page 3/4 1-1

The shutdown margin would be increased to 3.4% $\Delta k/k$ in three places. This change would be made to respond to the revised steamline break accident moderator cooldown analysis.

Page 3/4 1-5

The moderator temperature coefficient will be less negative, -2.3×10^{-4} $\Delta k/k/F$ for Cycle 2 verses -2.5×10^{-4} $\Delta k/k/F$ for Cycle 1. This less negative value will be bounded by the value for the reference cycle.

Page 3/4 1-23

The CEA drop time would be increased from 2.5 seconds to 3.0 seconds in TS 3.1.3.4. This increase in CEA drop time would result from the changed hydraulic characteristics of the 16 demonstration fuel assemblies.

Page 3/4 1-27

New CEA insertion limit (Figure 3.1-2) would be added based on revised physics calculations.

Pages 3/4 2-1 & 3/4 2-2

New surveillance requirement verifying CEA positions when excore detector monitoring would be necessary.

Pages 3/4 2-2 & 3/4 2-3

Reduced uncertainty factor would be balanced by reduced allowable linear heat rate when incore detector monitoring is used. A set of uncertainty values will be added depending on whether the reactor is in Load Following Operation or not.

Pages 3/4 2-4 & 3/4 2-5

New axial flux offset (Figure 3.2-2) and augmentation factors (Figure 4.2-1) would be added based on revised physics calculations. A dashed line applies when the reactor is in Load Following Operation.

Pages 3/4 2-6, 3/4 2-7, 3/4 2-8, 3/4 2-9 & 3/4 2-10

These power distribution limit changes would be made to accommodate increased peaking for Cycle 2 operation. They are based on revised physics calculations and application of the standard CE setpoint methodology. The reference to PLCEA's will also be removed from TS 4.2.2.3 and 4.2.3.3.

Page 3/4 2-11

Figure 3.2-4 would include the increase in allowable azimuthal tilt.

Page 3/4 2-13

The old TS 3.2.5 would be eliminated since the core can not achieve a core exposure that would result in clad collapse. The DNB parameters LCO would be moved unchanged to this page.

Page 3/4 2-14

The DNB parameters table (Table 3.2-1) would be moved from page 3/4 2-15 and the values for reduced RCP operation would be removed since such operation is not approved.

Page 3/4 3-6

The previously incorrect response time for the reactor coolant flow-low RPS trip would be corrected.

5.0 Physics Startup Testing

The physics startup tests for CCNPP-2 Cycle 2 will verify nuclear design, power distribution and control rod worth predictions. The proposed physics startup test program was included in Section 10 of the July 26, 1978 reload submittal[1]. This program includes:

Hot Functional Tests

CEDM Performance Verification
RCS Flow Verification

Initial Criticality and Lower Power Physics Tests

Initial Criticality
CEA Symmetry Check
Critical Boron Concentrations
- All Rods Out
- Groups 1 through 5 inserted
CEA Group Worths for Groups 1 through 5
Isothermal Temperature Coefficient

Power Ascension Tests

Critical Boron Concentration and Power Distribution Verifications for ARO with equilibrium Xenon (50% and 100%)
ITC and Power Coefficient Measurements (50% and 100%)

The program was discussed with the licensee and additional information regarding acceptance criteria and actions to be taken if the acceptance criteria are not met was supplied by Reference 9.

The results of this physics startup test program will be submitted to the NRC in the form of a summary report within 45 days of completion of the program. We conclude that this program is acceptable.

6.0 Conclusions

Based on our evaluation of the applications and available information and subject to the requirements set forth above, we conclude that it is acceptable for the licensee to proceed with Cycle 2 operation of CCNPP-2 in the manner proposed.

We have determined that the amendment does 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 amendment involves 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 this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 21, 1978

7.0 References

1. BG&E Application for Cycle 2 Reload for CCNPP-2, A. E. Lundvall to R. W. Reid, July 26, 1978.

2. NRC Amendment No. 32 for Cycle 3 Reload for CCNPP-1, R. W. Reid to A. E. Lundvall, March 31, 1978.

3. NRC Amendment No. 40 for Cycle 4 Reload for Maine Yankee, R. W. Reid to R. H. Groce, August 18, 1978.

4. BG&E Guide Tube Demonstration Test for Cycle 2 Reload, A. E. Lundvall to R. W. Reid, September 7, 1978.

5. BG&E Cycle 2 ECCS Performance Analysis Reload, A. E. Lundvall to R. W. Reid, July 31, 1978.

6. NRC Request for Additional Information on Guide Tube Inspection Program, R. W. Reid to A. E. Lundvall, June 6, 1978.

7. BG&E Details of Guide Tube Inspection Program, J. W. Gore to A. E. Lundvall, August 14, 1978.

8. NRC Request for Additional Information on Cycle 2 Reload, R. W. Reid to A. E. Lundvall, September 8, 1978.

9. BG&E Response to Cycle 2 Reload Questions, J. W. Gore to R. W. Reid, October 6, 1978.

10. BG&E Additional Information on Guide Tube Wear, A. E. Lundvall to R. W. Reid, October 16, 1978.

11. BG&E Revised Response on Cycle 2 Uncertainties, A. E. Lundvall to R. W. Reid, October 17, 1978.

12. NRC Amendment No. 13 for CCNPP-2 on CEA Insertion to Reduce Guide Tube Wear, D. K. Davis to A. E. Lundvall, January 6, 1978.

13. CENPD-161-P, TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core, Combustion Engineering, July 1975.

14. CENPD-206-P, TORC Code Verification and Simplified Modeling Methods, Combustion Engineering, January 1977.

15. Evaluation of Topical Report CENPD-161-P, letter from K. Kniel, NRC, to A. E. Scherer, CE, September 14, 1976.

16. CENPD-145, INCA: Method of Analyzing In-Core Detector Data in Power Reactor, Ober, Terney, and Marks, Combustion Engineering, April 1975.

17. CEN-85(B)-P, Solution to Increased Water Hole Peaking on Operating Reactors, Combustion Engineering, February 16, 1978.

18. CENPD-199, CE Setpoint Methodology, Combustion Engineering, April 1976.

19. Request for Additional Information, letter from R. W. Reid, NRC, to A. E. Lundvall, Jr., BG&E, April 4, 1978.

20. Response to Request for Additional Information on CEN-85(B)-P, letter from A. E. Lundvall, Jr., BG&E, to R. W. Reid, NRC, April 28, 1978.

21. BG&E Cycle 1 ECCS Performance Analysis for CCNPP-2, J. W. Gore to B. Rusche, November 5, 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-69, issued to Baltimore Gas & Electric Company (the licensee), which revised the Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit No. 1 (the facility) located in Calvert County, Maryland. The amendment is effective as of its date of issuance.

The amendment authorizes operation with modified (sleeved and reduced flow) guide tubes for the Control Element Assemblies (CEA's) and revises the Appendix A Technical Specifications by: (1) incorporating changes resulting from the analyses of Cycle 2 reload fuel, and (2) authorizing the removal of all part length CEA's.

The application for the amendment complies 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 had 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §50.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 26, 1978, as supplemented July 31, August 14, September 7 and October 6, 16 and 17, 1978, (2) Amendment No. 18 to License No. DPR-69, and (3) the Commission's related Safety Evaluation. All of these items are available for public 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 20678. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C., Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21st Day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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OCTOBER 16 1978

Docket Nos. 50-317
~~and 50-318~~

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

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendments Nos. 35 and 17 to Facility Operating Licenses Nos. DPR-53 and DPR-69 for the Calvert Cliffs Nuclear Power Plant Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated June 13, 1978 and staff discussions.

The amendments change the Technical Specifications to delete the requirements for ten hydraulic snubbers on the Unit No. 1 and four hydraulic snubbers on the Unit No. 2 support systems associated with the Containment Charcoal Filter Dousing System.

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

Sincerely,

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

Enclosures:
1. Amendment No. 35 to DPR-53
2. Amendment No. 17 to DPR-69
3. Safety Evaluation
4. Notice

cc w/enclosures:
see next page

Cont. 1
GD

OFFICE	ORB#4	ORB#4	OELD	ORB#4		
SURNAME	RIngram	MConner:acr		RReid		
DATE	10/ /78	10/ /78	10/ /78	10/ /78		

cc w/enclosure(s):
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
Post Office Box 500
Windsor, Connecticut 06095

Calvert County Library
Prince Frederick, Maryland 20678

Director, Department of State
Planning
301 West Preston Street
Baltimore, Maryland 21201

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 and Walnut Streets
Philadelphia, Pennsylvania 19106

cc w/4 cys enclosures and 1 cy
of BG&E filings dtd: 06/13/78

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. 35
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated June 13, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 35, 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: October 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 35

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

3/4 7-48
3/4 7-49

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-60-8	SERVICE WATER FROM CONTAINMENT COOLER #13 66'	I	Yes	No
1-60-8A	SERVICE WATER FROM CONTAINMENT COOLER #13 66'	I	Yes	No
1-60-9	SERVICE WATER TO CONTAINMENT COOLER #11 44'	I	Yes	No
1-60-9A	SERVICE WATER TO CONTAINMENT COOLER #11 44'	I	Yes	No
1-60-10	SERVICE WATER TO CONTAINMNET COOLER #11 44'	I	Yes	No
1-60-10A	SERVICE WATER TO CONTAINMENT COOLER #11 43'	I	Yes	No
1-60-11	SERVICE WATER FROM CONTAINMENT COOLER #11 43'	I	Yes	No
1-60-11A	SERVICE WATER FROM CONTAINMENT COOLER #11 43'	I	Yes	No
1-60-12	SERVICE WATER FROM CONTAINMENT COOLER #11 43'	I	Yes	No
1-60-12A	SERVICE WATER FROM CONTAINMENT COOLER #11 43'	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-60-25	SPRAY TO CONTAINMENT CHARCOAL FILTER #11 63'	I	Yes	No
1-60-26	SPRAY TO CONTAINMENT CHARCOAL FILTER #11 66'	I	Yes	No
1-60-27	SPRAY TO CONTAINMENT CHARCOAL FILTER #11 66'	I	Yes	No
1-60-28	SPRAY TO CONTAINMENT CHARCOAL FILTER #11 66'	I	Yes	No
1 61-1	CONTAINMENT SPRAY PUMP #12 DISCHARGE -15'	A	No	No

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. 17
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated June 13, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-69 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 17, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 17

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

3/4 7-41
3/4 7-43

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-60-7	SERVICE WATER OUTLET CONT. COOLER #23 65'	I	Yes	No
2-60-8	SERVICE WATER INLET CONT. COOLER #21 43'	I	Yes	No
2-60-9	SPRAY TO CONT. CHARCOAL FILTER #23 66'	I	Yes	No
2-60-10	SPRAY TO CONT. CHARCOAL FILTER #23 66'	I	Yes	No
2-60-12	SPRAY TO CONT. CHARCOAL FILTER #23 72'	I	Yes	No
2-60-14	SPRAY TO CONT. CHARCOAL FILTER #23 66'	I	Yes	No
2-60-15	SPRAY TO CONT. CHARCOAL FILTER #23 66'	I	Yes	No
2-60-18	SPRAY TO CONT. CHARCOAL FILTER #23 68'	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-60-28	SPRAY TO CONT. CHARCOAL FILTER #22 77'	I	Yes	No
2-61-1	HPSI PUMP #23 SUCT. FROM S/D COOLING H.X. #22 9'9"	A	No	No
2-61-2	HPSI PUMP #23 SUCT. FROM S/D COOLING H.X. #22 9'9"	A	No	No
2-61-3	CONT. SPRAY PUMP #22 DISCH 3'6"	A	No	No
2-61-4	CONT. SPRAY HDR PENETRATION PIPING 10'0"	A	No	No
2-61-5	CONT. SPRAY HDR DOWNSTREAM S/D COOL H.X. #22 17'5"	A	No	No
2-61-5A	CONT. SPRAY HDR DOWNSTREAM S/D COOL H.X. #22 17'5"	A	No	No

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 35 AND 17 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

Introduction

By application for license amendment dated June 13, 1978, Baltimore Gas & Electric Company (BG&E, or the licensee), requested changes to the Technical Specifications (TS) for Calvert Cliffs Nuclear Power Plant (CCNPP) Units Nos. 1 and 2. The proposed changes to the TS consist of revising Table 3.7-4 to permit deletion of certain hydraulic shock suppressors (snubbers) listed in that table as part of a redesign of the support systems of the Containment Charcoal Filter Dousing System for each unit.

Background

BG&E is in the process of systematically reviewing all snubber installations at CCNPP. The intention of this review is to eliminate as many snubbers as possible. The reliability of the piping system is improved by minimizing the number of hydraulic snubbers. Hydraulic snubbers have a potential for leakage and miscalibration. This potential makes them inherently less reliable than other types of rigid and mechanical supports. Also a snubber without hydraulic fluid provides essentially no piping support.

BG&E has proposed a change to Table 3.7-4 of the CCNPP Units Nos. 1 and 2 TS to delete the requirements for ten snubbers on the Unit No. 1 and four snubbers on the Unit No. 2 charcoal filter dousing systems. Four of the snubbers removed from the Unit No. 1 system will be replaced with rigid restraints. The remaining support system will employ six and nine snubbers for the Unit 1 and Unit 2 charcoal filter dousing system, respectively.

The containment charcoal filter dousing piping was modified under the provisions of 10 CFR 50.59 by the addition of a manual isolation valve. This valve facilitates the testing of the system by isolating the spray header and preventing the inadvertent wetting of the charcoal.

Evaluation

A reanalysis of the piping system was performed by Bechtel Power Corporation using their ME 632 computer code. The code utilizes a linear elastic modeling technique and performs thermal, dead weight, and seismic analyses. While not specifically stated in the Final Safety Analysis Report (FSAR), the dousing system piping was originally designed in accordance with ANSI B31.7 Class II. This reanalysis was performed in accordance with the original design criteria. The calculated stresses for the modified system with the snubbers removed are well below the code allowables for all design conditions. We find that this reanalysis, using a standard calculational method, provides assurance that the containment charcoal filter dousing system continues to meet the original design criteria.

The redesign of this support system reduces the surveillance burden, improves the overall system reliability and thus reduces the potential of a safety system piping failure due to a malfunction of the support system. Thus, the reliability of the containment charcoal filter dousing system and the facility as a whole will be increased by reducing the number of hydraulic snubbers necessary to satisfy the design criteria.

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 Section 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.

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.

Dated: October 16, 1978

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 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 application for amendment dated June 13, 1978, (2) Amendments Nos. 35 and 17 to Licenses Nos. DPR-53 and DPR-69, and (3) the Commission's related Safety Evaluation. All of these items are available for public 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) and (3) 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 16th day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

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