Docket File

REGULATORY DOCKET FILE COPY

Docket No. 50-336 OCT 06 1980 cP/1

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications dated August 10, 1979 and May 9, August 29 and September 30, 1980, as supplemented on numerous other dates.

This amendment authorizes Cycle 4 operation at 2700 MWt with:

- A mixed core with one-third Westinghouse (W) fuel and two-thirds Combustion Engineering (CE) fuel; and
- Modified (sleeved, reduced flow and insert) guide tubes for the control element assemblies.

The amendment revises the Appendix A Technical Specifications by:

- Incorporating changes resulting from the analysis of the Cycle 4 reload with Westinghouse fuel;
- Adding surveillance requirements for engineering safety features components leakage outside containment;
- Allowing continuation of low temperature operation for special tests;
- Correcting the shutdown margin for the Mode 5 boron dilution event; and
- Preventing containment purging in Modes 1, 2, 3, and 4.

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. By letter dated September 22, 1980, you identified an editorial correction to our Amendment No. 60. A corrected Table 3.9-1 is enclosed.

In addition, the enclosed safety evaluation supporting this amendment addresses our evaluation of: 6D

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SURNAME ▶	8010240094					
DATE ▶						

- Containment electrical penetration replacement;
- Steam generator tube and support plate inspection;
- Reactor cooling pump speed sensing proximity probe and transmitter qualification;
- Reactor cavity neutron shield dose reduction;
- Steam generator feedwater piping inspection;
- Stem mounted limit switch replacement; and
- Reactor coolant system vent installation.

In the process of our evaluation of your request, we find the following items need your attention as documented herein. For each item, your staff has agreed to supply the documentation indicated on the schedule shown.

1. Provide at least 90 days before shut down for the Cycle 5 reload outage:
   a. An evaluation program (including the planned inspections) to determine the amount of guide tube wear experienced in each type (sleeved CE, sleeved W, low flow CE and inserted W) of fuel; and
   b. the plans for continued surveillance of the steam generator tubes and support plates;
2. Provide an analysis, by April 1, 1981, of the asymmetric blowdown loads in a LOCA for a mixed (W and CE fueled) core;
3. Provide, within 90 days of receipt of this letter, justification of the measurement uncertainty values for axial shape index, pressure, temperature, flow, power (LCO) and power (LSSS);
4. Provide, with the Cycle 5 reload analysis, an analysis of the worst large break LOCA burnup for W fuel in a CE designed NSSS; and
5. Expedite the development and submittal for staff review of operating procedures and proposed Technical Specifications to control the newly installed reactor coolant vent systems. Staff guidance is available in our September 5, 1980 letter.

You will note our enclosed Safety Evaluation addresses each of the items left for your attention in our May 12, 1979 letter transmitting the Cycle 3 reload

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DATE						

authorization. We appreciate your timely response, in accordance with agreements between our respective staffs, that has resulted in resolution of each item for this cycle.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by:

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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DATE	10/6/80	10/6/80	10/6/80	10/6/80	10/6/80	10/6/80

HANDLED ON

SEP 16 1980

E. L. "MONTE" CONNER

DETERMINATION OF PROPOSED LICENSING AMENDMENT

Licensee: ~~Northeast Nuclear Energy Company~~

Request for: Millstone Unit No. 2

Cycle 4 Reload Analysis

Request Date: August 29, 1980

Proposed Noticing Action: ( ) Pre-Notice Recommended

(X) Post-Notice Recommended

( ) Determination delayed pending completion of Safety Evaluation

Basis for Decision: See attached sheet.

Proposed NEPA Action: ( ) Environmental Impact Statement (EIS) Required

( ) Negative Declaration (ND) and Environmental Impact Appraisal (EIA) Required

(X) No EIS, ND or EIA Required

( ) Determination delayed pending completion of EIA

Basis for Decision:

Concurrences:

E L Conner 9/2/80

1. E. Conner, PM/ORB 3

Robert A Clark

2. R. Clark, BC/ORB 3

T Speis for 9/5/80

3. T. Speis BC/RSB

Wm V Johnston 9/9/80

4. W. Johnston BC/CPB DAP/9/9/80

T M Novak 9/11/80

5. T. Novak, AD/DOR

J R Gray

6. J. Gray OELD 9/15/80 — Predetermination appears to be supported by attached sheet. Note reload SER should support in detail ~~any~~ the no significant hazards finding. My concurrence here is only on the predetermination — may be subject to change based on completion of staff review and the reload SER.

DETERMINATION OF PROPOSED LICENSING

AMENDMENT FOR MILLSTONE 2

Basis for Decision

Millstone 2 is currently refueling (Cycle 4) for the first time with 72 fuel assemblies designed and manufactured by Westinghouse (W). The original core loading and the fuel assemblies for the past two reloads were from Combustion Engineering (CE). The Cycle 4 reload analysis was also performed by W except for the Small Break LOCA analysis which Northeast Nuclear Energy Company (NNECO) says is bounded by the previously approved CE analysis.

NNECO provided design specifications to W to ensure that the fuel assembly is basically identical to the CE fuel assembly used in the Millstone 2 core. In the Basic Safety Report (BSR), submitted by letter dated March 6, 1980, W states that the reload fuel assembly is designed to be mechanically and hydraulically compatible with the current Millstone 2 core. We are reviewing the physical fuel assembly design, the hydraulic characteristics and the physics parameters given in the BSR and the Reload Safety Analysis (RSA) dated June 3, 1980. This review has progressed to the point that we find that the small differences between the CE and W fuel assemblies should not result in a significant hazards consideration.

In the BSR, RSA, and NNECO letters of June 2 and 11, 1980, the reanalysis of all FSAR transients and accidents including large and small break LOCA by W has been documented. The results from this W reanalysis parallels the previous results obtained for the previous reload cycles analyzed by CE. The resultant instrumentation setpoints from the W analysis can be confirmed by comparison with Cycle 3 setpoints established by the CE analysis. The LOCA limits will remain well within the requirements of 10 CFR 50.46 by maintaining the Cycle 4 peak linear heat rates at rated power levels below 15.6 Kw/ft (same limit as Cycle 3). Although our review of transient and accidents is not yet complete, all indications are that our review of the use of W fuel assemblies in the Millstone 2 core will result in a finding that no significant hazards consideration is involved.

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

DISTRIBUTION:
Docket File
ORB#3 Rdg
PMKreutzer

October 7, 1980

Docket No. 50-336

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 61
  Referenced documents have been provided PDR

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#3:DL					
SURNAME	PMKreutzer/JL					
DATE	10/7/80					

NRC FORM 102 (1-76)

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

October 6, 1980

Docket No. 50-336

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications dated August 10, 1979 and May 9, August 29 and September 30, 1980, as supplemented on numerous other dates.

This amendment authorizes Cycle 4 operation at 2700 MWt with:

° A mixed core with one-third Westinghouse (W) fuel and two-thirds Combustion Engineering (CE) fuel; and

° Modified (sleeved, reduced flow and insert) guide tubes for the control element assemblies.

The amendment revises the Appendix A Technical Specifications by:

° Incorporating changes resulting from the analysis of the Cycle 4 reload with Westinghouse fuel;

° Adding surveillance requirements for engineering safety features components leakage outside containment;

° Allowing continuation of low temperature operation for special tests;

° Correcting the shutdown margin for the Mode 5 boron dilution event; and

° Preventing containment purging in Modes 1, 2, 3, and 4.

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. By letter dated September 22, 1980, you identified an editorial correction to our Amendment No. 60. A corrected Table 3.9-1 is enclosed.

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- Containment electrical penetration replacement;
- Steam generator tube and support plate inspection;
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- Reactor cavity neutron shield dose reduction;
- Steam generator feedwater piping inspection;
- Stem mounted limit switch replacement; and
- Reactor coolant system vent installation.

In the process of our evaluation of your request, we find the following items need your attention as documented herein. For each item, your staff has agreed to supply the documentation indicated on the schedule shown.

1. Provide at least 90 days before shut down for the Cycle 5 reload outage:

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   b. the plans for continued surveillance of the steam generator tubes and support plates;

2. Provide an analysis, by April 1, 1981, of the asymmetric blowdown loads in a LOCA for a mixed (W and CE fueled) core;

3. Provide, within 90 days of receipt of this letter, justification of the measurement uncertainty values for axial shape index, pressure, temperature, flow, power (LCO) and power (LSSS);

4. Provide, with the Cycle 5 reload analysis, an analysis of the worst large break LOCA burnup for W fuel in a CE designed NSSS; and

5. Expedite the development and submittal for staff review of operating procedures and proposed Technical Specifications to control the newly installed reactor coolant vent systems. Staff guidance is available in our September 5, 1980 letter.

You will note our enclosed Safety Evaluation addresses each of the items left for your attention in our May 12, 1979 letter transmitting the Cycle 3 reload authorization. We appreciate your timely response, in accordance with agreements between our respective staffs, that has resulted in resolution of each item for this cycle.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

Northeast Nuclear Energy Company

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Day, Berry & Howard
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Anthony Z. Roisman
Natural Resources Defense Council
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Washington, D.C. 20005

Mr. Lawrence Bettencourt. First Selectman
Town of Waterford
Hall of Records - 200 Boston Post Road
Waterford, Connecticut 06385

Northeast Nuclear Energy Company
ATTN: Superintendent
Millstone Plant
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Waterford, Connecticut 06385

Director, Technical Assessment
Division
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(AW-459)
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Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region I Office
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John F. Kennedy Federal Building
Boston, Massachusetts 02203

Waterford Public Library
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Waterford, Connecticut 06385

Northeast Utilities Service Company
ATTN: Mr. James R. Himmelwright
Nuclear Engineering and Operations
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Hartford, Connecticut 06101

Mr. John Shedlosky
Resident Inspector/Millstone
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Niantic, CT 06357

Mr. Charles B. Brinkman
Manager - Washington Nuclear
Operations
C-E Power Systems
Combustion Engineering, Inc.
4853 Cordell Ave., Suite A-1
Bethesda, Maryland 20014

cc w/enclosure(s) and incoming
dtd.: 8/10/79; 5/9/80; 8/29/80;
9/30/80
Connecticut Energy Agency
ATTN: Assistant Director, Research
and Policy Development
Department of Planning and Energy
Policy
20 Grand Street
Hartford, Connecticut 06106

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE HARTFORD ELECTRIC LIGHT COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER COMPANY, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. DPR-65

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

   A. The applications for amendment by Northeast Nuclear Energy Company, et al. (the licensee), dated August 10, 1979, May 9, August 29, and September 30, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the 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.

8010240098

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-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61, 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 A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: October 6, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

1-7
2-2
2-4
2-5
B 2-1
B 2-2
B 2-3
B 2-4
B 2-5
B 2-6
B 2-8
3/4 1-1
3/4 1-3
3/4 1-5
3/4 2-3
3/4 2-5
3/4 5-5a (added)
3/4 5-6 (repaginated)
3/4 5-6a (repaginated)
3/4 6-13
3/4 6-18
3/4 6-19
3/4 9-1
3/4 9-15 (corrected)
3/4 10-1
B 3/4 1-1
B 3/4 1-3
B 3/4 5-1
B 3/4 6-3
B 3/4 7-1
B 3/4 7-2

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3. HOT STANDBY	< 0.99	0	$\geq 300°F$
4. HOT SHUTDOWN	< 0.99	0	$300°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.98	0	$\leq 200°F$
6. REFUELING**	≤ 0.90	0	$\leq 140°F$

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

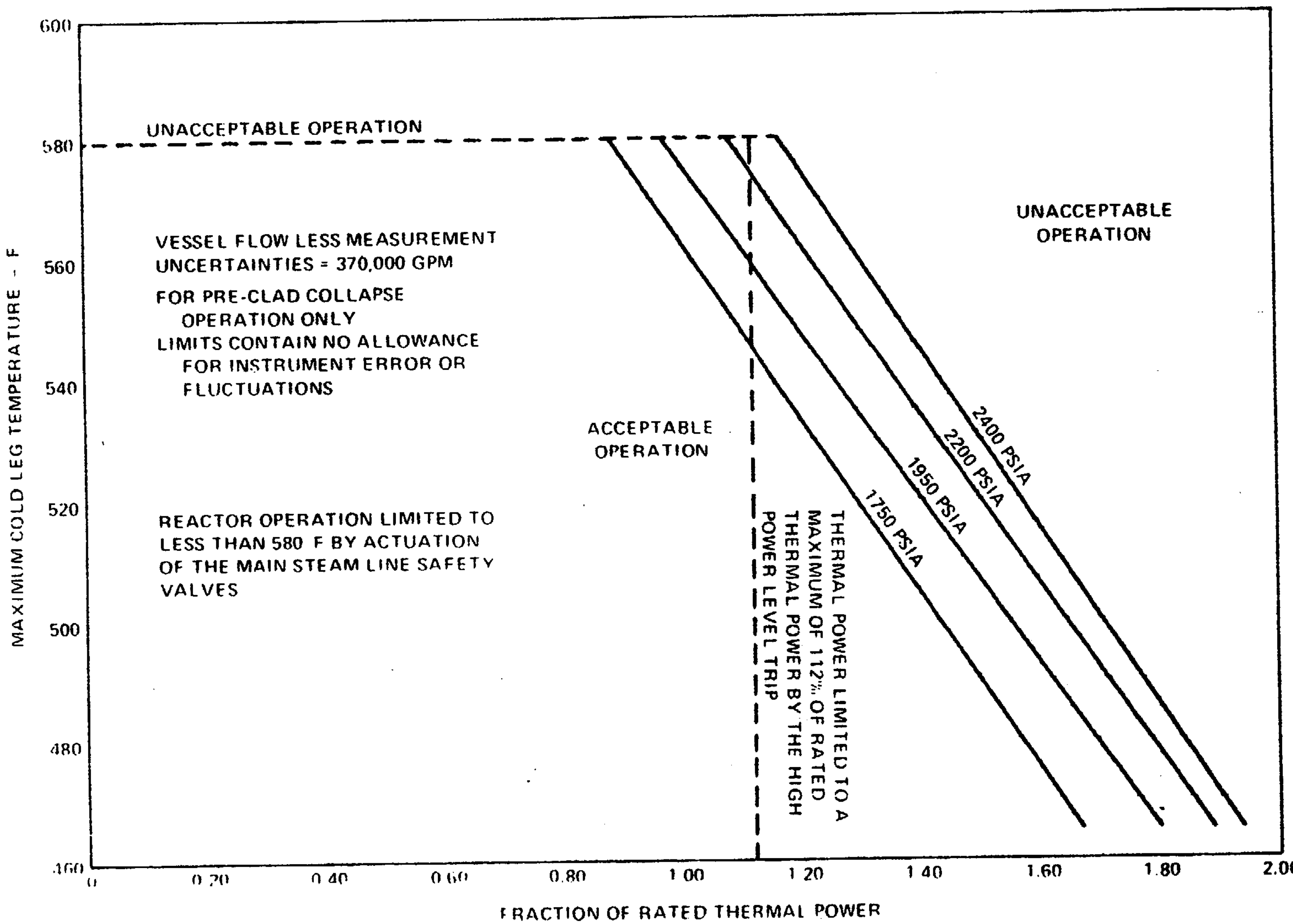

FIGURE 2.1-1 Reactor Core Thermal Margin Safety Limit - Four Reactor Coolant Pumps Operating

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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		
Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.1% of reactor coolant flow with 4 pumps operating*.
4. Reactor Coolant Pump Speed - Low	≥ 830 rpm	≥ 823 rpm
5. Pressurizer Pressure - High	≤ 2400 psia	≤ 2408 psia
6. Containment Pressure - High	≤ 4.75 psig	≤ 5.23 psig
7. Steam Generator Pressure - Low (2) (5)	≥ 500 psia	≥ 492 psia
8. Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 35.2% Water Level - each steam generator
9. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

* Design Reactor Coolant flow with 4 pumps operating is 370,000 gpm.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).
11. Loss of Turbine -- Hydraulic Fluid (3) Pressure - Low	$\geq$ 500 psig	$\geq$ 500 psig

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

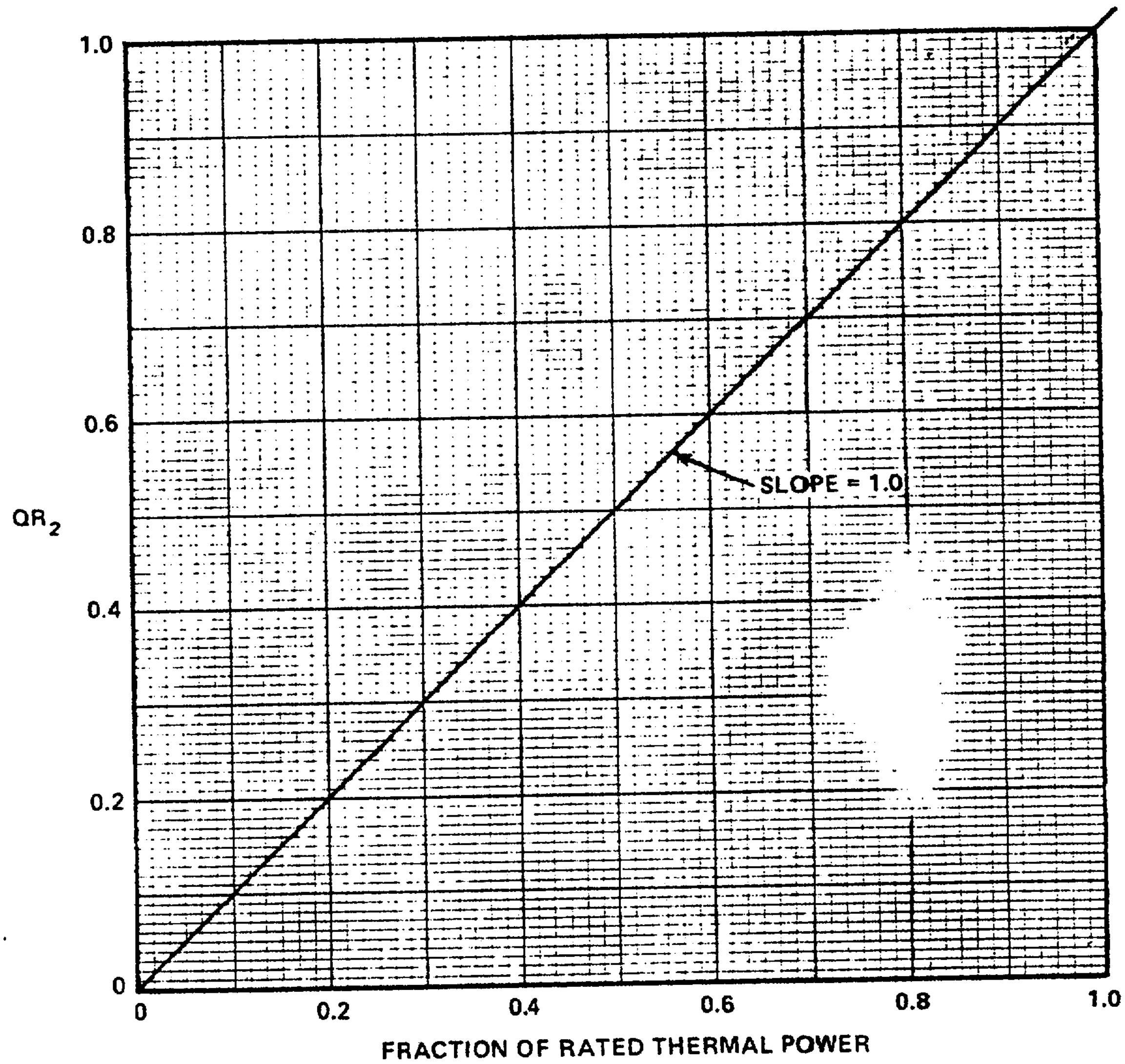

FIGURE 2.2-1

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

2.1 SAFETY LIMITS

BASES

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 W-3 correlation. The W-3 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 1.30. 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 Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.30. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperatures 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 2.2-1. The area of safe operation is below and to the left of these lines.

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The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

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 DNBR of less than 1.30 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 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 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 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SET POINTS

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 Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its 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 to occur for each trip used in the accident 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 9.6% 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 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% 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 accident 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

Reactor Coolant Flow-Low (Continued)

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 DNBR above 1.30 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 DNBR from going below 1.30 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 815 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 - Low

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 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the 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.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.30.

Thermal Margin/Low Pressure (Continued)

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, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. 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 67 psi to compensate for pressure measurement error, trip system processing 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 67 psi allowance is made up of a 22 psi pressure measurement allowance and a 45 psi 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 values. 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.

Underspeed - Reactor Coolant Pumps

The Underspeed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on all four reactor coolant pumps. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.30 following a 4 pump loss of flow event.

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 $\geq$ 3.20% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 3.20% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.20% $\Delta k/k$:

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN, required by Specification 3.1.1.1, shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

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. Prior to initial operation above 5% RATED THERMAL POWER after each refueling, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. 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 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. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.d, 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 refueling.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 2.0% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 2.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 2.0% $\Delta k/k$:

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN required by Specification 3.1.1.2 shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. At least once per 24 hours by consideration of the following factors:

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

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the core shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentation is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the core < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The reactor coolant flow rate through the core shall be determined to be $\geq$ 3000 gpm prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying $\geq$ 3000 gpm through the core.

MODERATOR TEMPERATURE COEFFICIENT (MTC)

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 -2.4×10^{-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 predicted values.

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

See Special Test Exemption 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 refueling.

b. At any THERMAL POWER, within 14 EFPD after each fuel loading at equilibrium boron concentration.

c. At any THERMAL POWER, within 14 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

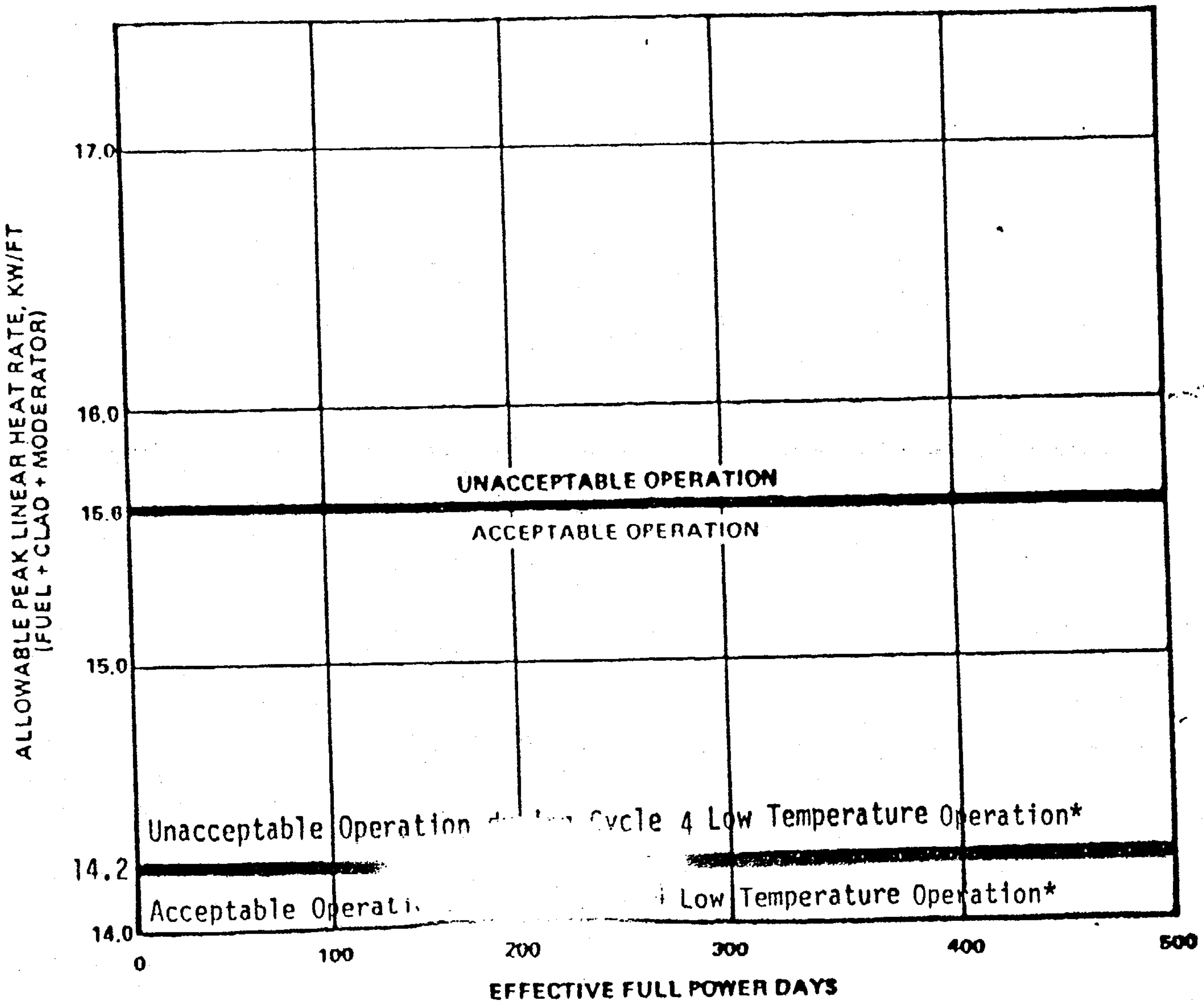

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

*During Cycle 4, low temperature operation is authorized, for periods not exceeding 24 hours with the inlet temperature greater than or equal to 537°F and without varying the programmed pressurizer level, for determining the moderator temperature coefficient or for performing turbine generator efficiency testing.

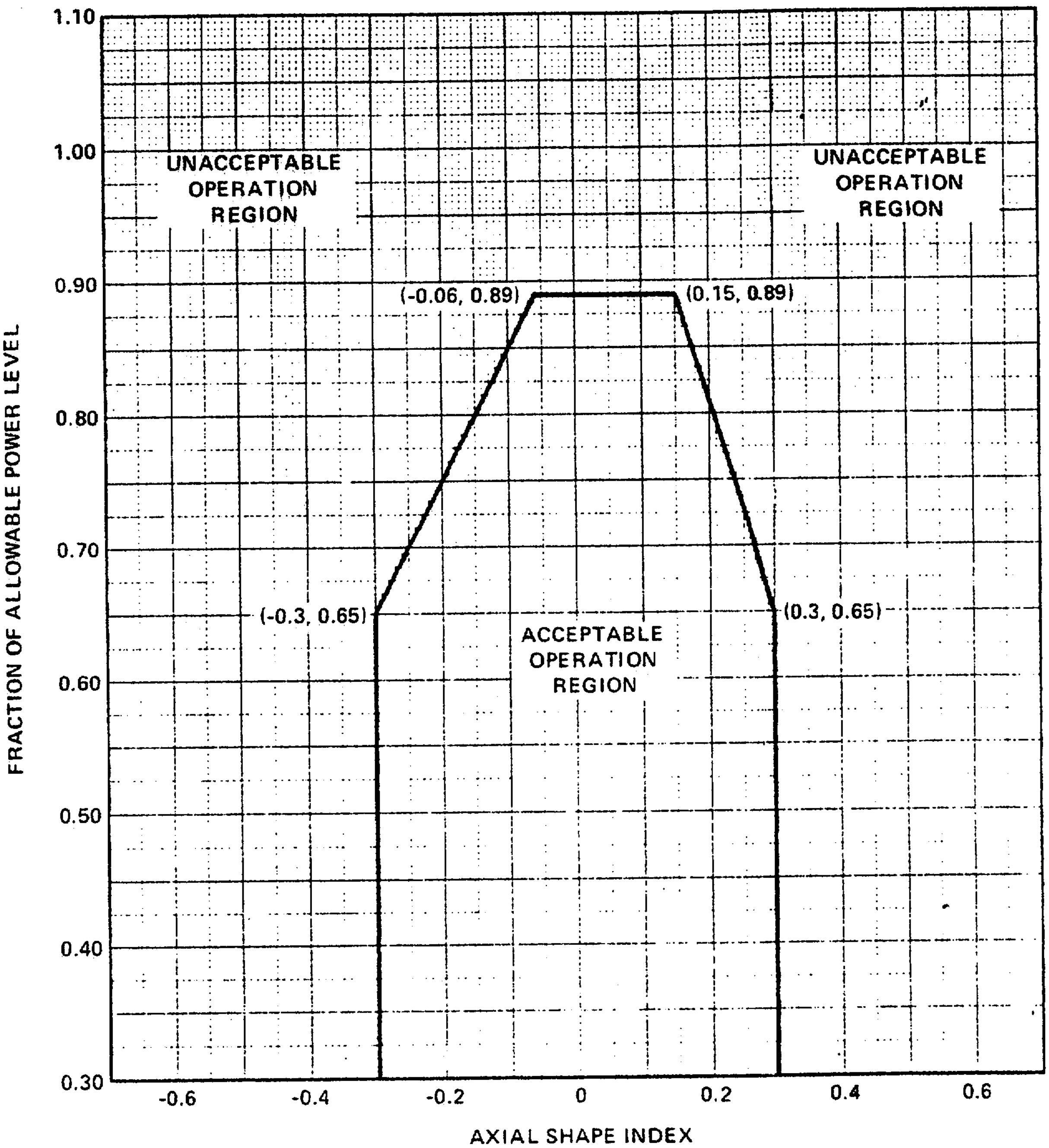

FIGURE 3.2-2 AXIAL SHAPE INDEX vs Fraction of Allowable Power Level per Specification 4.2.1.2c

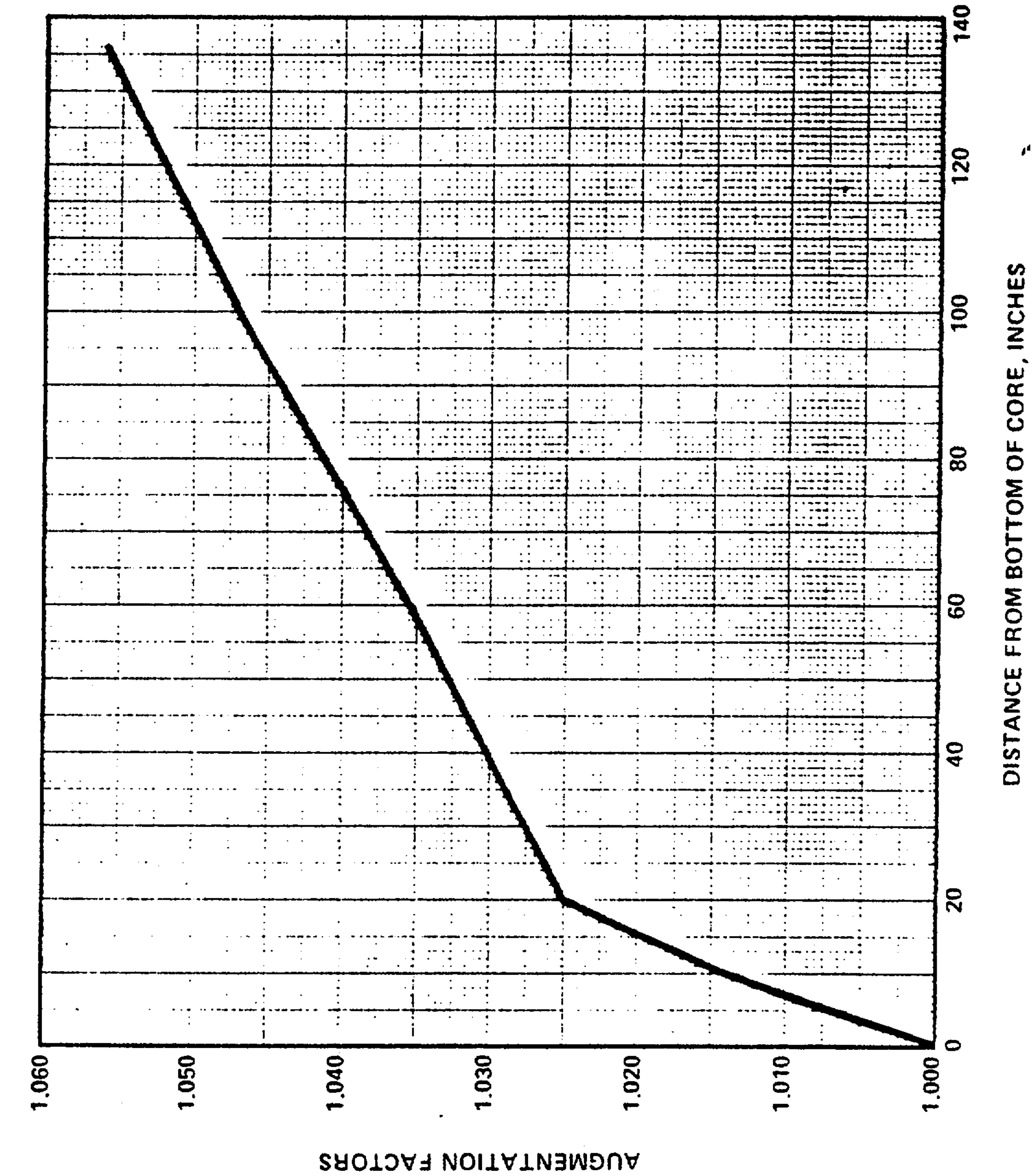

FIGURE 4.2-1 Augmentation Factors vs Distance From Bottom of Core

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 $\leq$ 1.615.

APPLICABILITY: MODE 1*.

ACTION:

With $F^T_{xy} > 1.615$, 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 Limit 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 $F^T_{xy} = F_{xy}(1+T_q)$ 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 > 0.02.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

10. Verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open
2-SI-659	SRAS Recirc.	Open*
2-SI-660	SRAS Recirc.	Open*
2-CH-434	Thermal Bypass	Closed**

b. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

c. At least once per 18 months by:

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

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

3. Verifying that a minimum total of 110 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained with the TSP storage baskets.

4. Verifying that when a representative sample of 0.35 $\pm$ 0.05 lbs of TSP from a TSP storage basket is submerged, without agitation, in 50 $\pm$ 5 gallons of 180 $\pm$ 10°F borated water from the RWST, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

*To be closed prior to recirculation following LOCA.
**2-CH-434, a manual valve, shall be locked closed.

5. Verifying a total leak rate less than or equal to 12 gallons per hour for the high pressure safety injection system in conjuction with the containment spray system (reference Specification 4.6.2.1.c) at:

   a) A high pressure safety injection pump discharge pressure of greater than or equal to 1125 psig on recirculation flow, for the parts of the system between the pump discharge and the header injection valves, including the pump seals.

   b) Greater than or equal to 22 psig at the pump suction for the piping from the containment sump check valve to the pump suction.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by cycling each power operated valve in the subsystem flow path not testable during plant operation through one complete cycle of full travel.

e. By a visual verification that each of the throttle valves in Table 4.5-1 will open to the correct position. This verification shall be performed:

1. Within 4 hours following the completion of each valve stroking operation,

2. Immediately prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator or its control circuit, or

3. At least once per 18 months.

f. By conducting a flow balance verification immediately prior to returning to service any portion of a subsystem after the completion of a modification that could alter system flow characteristics. The injection leg flow rate shall be as follows:

1. HPSI Headers - the sum of the three lowest injection flows must be $\geq$ 471 gpm. The sum of the four injection flows must be $\leq$ 675 gpm.

2. LPSI Header - the sum of the three lowest injection flows must be $\geq$ 2370 gpm. The sum of the four injection flows must be $\leq 4500 + \left[\frac{\text{RWST level (\%)} - 10(\%)}{90\%} \times 200 \right]$

g. At least once per 18 months, during shutdown, by verifying that on a Safety Injection Actuation test signal:

1. The valves in the boron injection flow path from the boric acid storage tank via the boric acid pump and charging pump actuate to their required positions, and

2. The charging pump and boric acid pump start automatically.

TABLE 4.5-1

ECCS THROTTLED VALVES

1.	2-SI-617	"A" HPSI Header - Loop 1A Injection
2.	2-SI-627	"A" HPSI Header - Loop 1B Injection
3.	2-SI-637	"A" HPSI Header - Loop 2A Injection
4.	2-SI-647	"A" HPSI Header - Loop 2B Injection
5.	2-SI-616	"B" HPSI Header - Loop 1A Injection
6.	2-SI-626	"B" HPSI Header - Loop 1B Injection
7.	2-SI-636	"B" HPSI Header - Loop 2A Injection
8.	2-SI-646	"B" HPSI Header - Loop 2B Injection
9.	2-SI-615	LPSI Header - Loop 1A Injection
10.	2-SI-625	LPSI Header - Loop 1B Injection
11.	2-SI-635	LPSI Header - Loop 2A Injection
12.	2-SI-645	LPSI Header - Loop 2B Injection

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump operates for at least 15 minutes,

4. Cycling each testable, automatically operated valve in each spray system flow path through at least one complete cycle,

5. Verifying that upon a sump recirculation actuation signal the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established, and

6. Verifying that all accessible manual valves not locked, sealed or otherwise secured in position and all remote or automatically operated valves in each spray system flow path are positioned to take suction from the RWST on a Containmenet Pressure--High-High signal.

b. At least once per 18 months, during shutdown, by cycling each power operated valve in the spray system flow path not testable during plant operation through at least one complete cycle of full travel.

c. At least once per 18 months by verifying a total leak rate less than or equal to 12 gallons per hour in conjunction with the high pressure safety injection system (reference Specification 4.5.2.c.5) at:

1) Dicharge pressure of greater than or equal to 254 psig on recirculation flow for those parts of the system between the pump discharge and the header isolation valve, including the pump seals.

2) Greater than or equal to 22 psig at the pump suction for the piping from the containment sump check valve to the pump suction.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT AIR RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four containment air recirculation and cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one containment air recirculation and cooling unit inoperable and both containment spray systems OPERABLE, restore the inoperable air recirculation and cooling unit to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

b. With one containment air recirculation and cooling unit inoperable and one containment spray system inoperable, restore either the inoperable air recirculation and cooling unit or the inoperable spray system to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

c. With two containment air recirculation and cooling units inoperable and both containment spray systems OPERABLE, restore at least one of the inoperable air recirculation and cooling units to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting, in low speed, each unit from the control room,

b. Verifying that each unit operates for at least 15 minutes, and

c. Verifying a cooling water flow rate of $\geq$ 500 gpm to each cooling unit.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	MAXIMUM ISOLATION TIME
A. CONTAINMENT ISOLATION VALVES			
2-PMW-43	Primary Makeup Water	Yes	5 seconds
2-CH-089	Reactor Coolant Letdown Line	No	5 seconds
2-CH-516	Reactor Coolant Letdown Line	No	5 seconds
2-SSP-16.1	Containment Sump to Aerated Waste Drain Tank	Yes	5 seconds
2-SSP-16.2	Containment Sump to Aerated Waste Drain Tank	Yes	5 seconds
2-RC-001	Reactor Coolant Sampling	Yes	5 seconds
2-RC-002	Reactor Coolant Sampling	Yes	5 seconds
2-RC-003	Reactor Coolant Sampling	Yes	5 seconds
2-RC-45	Reactor Coolant Sampling	Yes	5 seconds
2-LRR-61.1	Reactor Coolant Sampling	Yes	5 seconds
2-MS-220A	Steam Generator Blowdown	Yes	5 seconds
2-MS-220B	Steam Generator Blowdown	Yes	5 seconds
2-SI-312	Nitrogen Supply	Yes	5 seconds
2-LRR-43.1	Primary Drain Tank to Clean Radwaste System	Yes	5 seconds
2-LRR-43.2	Primary Drain Tank to Clean Radwaste System	Yes	5 seconds
2-CH-506	Reactor Coolant Pump Seal Controlled Bleedoff	No	5 seconds
2-CH-198	Reactor Coolant Pump Seal Controlled Bleedoff	No	5 seconds
2-CH-505	Reactor Coolant Pump Seal Controlled Bleedoff	No	5 seconds
2-GR-11.1	Waste Gas Header	Yes	5 seconds
2-GR-11.2	Waste Gas Header	Yes	5 seconds

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	MAXIMUM ISOLATION TIME
A. CONTAINMENT ISOLATION VALVES			
2-AC-12	Containment Air Sample	Yes	5 seconds
2-AC-15	Containment Air Sample	Yes	5 seconds
2-AC-20	Containment Air Sample	Yes	5 seconds
2-AC-47	Containment Air Sample	Yes	5 seconds
HV-8150	Containment Air Sample	Yes	5 seconds
HV-8151	Containment Air Sample	Yes	5 seconds
2-MS-191A	Steam Generator Sample	Yes	5 seconds
2-MS-191B	Steam Generator Sample	Yes	5 seconds
2-EB-91	Hydrogen Purge	Yes	5 seconds
2-EB-92	Hydrogen Purge	Yes	5 seconds
2-EB-99	Hydrogen Purge	Yes	5 seconds
2-EB-100	Hydrogen Purge	Yes	5 seconds
B. MANUAL			
2-SI-709*	Shutdown Cooling	Yes	Not Applicable
2-SI-463*	Safety Injection Tank Test Line	Yes	Not Applicable
2-SA-19*	Station Air	Yes	Not Applicable
2-RW-21*	Refueling Water Purification	Yes	Not Applicable
2-RW-63*	Refueling Water Purification	Yes	Not Applicable
2-RW-154*	Refueling Water Purification	Yes	Not Applicable
2-RW-232*	Refueling Water Purification	Yes	Not Applicable
2-AC-46*	Hydrogen Monitoring	Yes	Not Applicable
2-AC-51*	Hydrogen Monitoring	Yes	Not Applicable
C. OTHER - NOT APPLICABLE			

* May be opened on an intermittent basis under administrative control.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.2 The containment purge supply and exhaust isolation valves shall be locked closed and electrically deactivated.

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

ACTION:

With one containment purge supply and/or one exhaust isolation valve open and/or electrically activated, close the open valve(s) and electrically deactivate within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The containment purge supply and exhaust isolation valves shall be determined locked closed and electrically deactivated prior to each reactor startup.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two separate and independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a K_{eff} equivalent to no greater than 0.90 with all full length CEAs (shutdown and regulating) fully inserted.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend CORE ALTERATIONS and initiate and continue boration at $\geq$ 40 gpm until K_{eff} is reduced to $\leq$ 0.90. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1 The boron concentration of the refueling pool shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

TABLE 3.9-1

ACCESS DOORS TO SPENT FUEL POOL AREA

Door No.	Elevation	Location	Type	Area Serviced
291	14'6"	M.7 - 18.5	Double Door	SFP Skimmer System
292 or 207	14'6"	R/S - 18.9	Double Door or 8' Rollup Door	Solidification System
293	14'6"	Q/R - 18.0	Double Door	Maintenance Shop
208	14'6"	S - 18.9	16' Rollup Door	Railway Access
294	14'6"	Q - 20.7	Single Door	D/G Room
295	38'6"	F.8 - 18	8' Rollup Door	Aux. & R. W. HVAC
296	38'6"	F.8 - 18.5	Single Door	Aux. & R. W. HVAC
297	38'6"	F.8 - 18.5	Single Door	North Stairwell
---	38'6"	H.4 - 18.9	Double Sliding Door	Elevator
298	38'6"	M.4 - 18.9	Single Door	Penetration Room
299	38'6"	M.7 - 18.9	Double Door	Main Exh. Fan Room
247	38'6"	M.7 - 17.2	Single Door	South Stairwell
254	55'6"	S - 17.2	Single Door	Roof Above Storage Floor
253	55'6"	S - 18.9	Single Door	Roof Above F. O. Tanks

STORAGE POOL AREA VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.15 At least one Enclosure Building Filtration System shall be OPERABLE and capable of automatically initiating operation in the auxiliary exhaust mode and exhausting through HEPA filters and charcoal adsorbers on a storage pool area high radiation signal.

APPLICABILITY: WHENEVER IRRADIATED FUEL IS IN THE STORAGE POOL.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one spent fuel storage pool ventilation system is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.9.15 The above required Enclosure Building Filtration System shall be demonstrated OPERABLE:

a. At least once per 31 days, by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

b. At least once per 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

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: MODES 2 and 3.

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 $\geq$ 40 gpm of $\geq$ 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 $\geq$ 40 gpm of $\geq$ 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 AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 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 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, 3.2.2, 3.2.3 and 3.2.4 are suspended, immediately:

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

b. Be in HOT STANDBY within 2 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, 3.2.2, 3.2.3 or 3.2.4 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 THEMAL 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, 3.2.2, 3.2.3 or 3.2.4 are suspended.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

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 3.2% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 2% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

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 changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 10,060 + 700/-0 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for 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 assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating 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 4550 gallons of 6.25% boric acid solution from the boric acid tanks or 47,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The requirements for a minimum contained volume of 370,000 gallons of borated water in the refueling water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

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.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 2% $\Delta k/k$ SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn. This condition requires either 5,050 gallons of 6.25% boric acid solution from the boric acid tanks or 57,000 gallons of 1720 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section 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.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 20 steps) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a immovable or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 20 steps) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 20 steps) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time used in the accident analyses. Measurement with $T_{avg} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

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

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

The limit of one hour for operation with an inoperable safety injection tank minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

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

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

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The requirement to dissolve a representative sample of TSP in a sample of RWST water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures. The ECCS leak rate surveillance requirements assure that the leakage rates assumed for the system outside containment during the recirculation phase will not be exceeded.

The purpose of the ECCS throttle valve surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

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

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses. The leak rate surveillance requirements assure that the leakage assumed for the system outside containment during the recirculation phase will not be exceeded.

3/4.6.2.2 CONTAINMENT AIR RECIRCULATION SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment purge supply and exhaust isolation valves are required to be closed and electrically deactivated during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Such a demonstration would require justification of the mechanical operability of the purge valves and consideration of the appropriateness of the electrical override circuits. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

The post-incident recirculation systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psig) of its design pressure of 1000 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 12.7×10^6 lbs/hr which is 108 percent of the total secondary steam flow of 11.8×10^6 lbs/hr at 100% RATED THERMAL POWER.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.6$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

106.6 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 6.35×10^6 lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 7.94×10^5 lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of off-site power.

Either two motor driven pumps or the steam driven pump have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to $\leq$ 300°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 300°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

TABLE OF CONTENTS

1.0 Introduction

By applications dated August 10, 1979 and May 9, August 29 and September 30, 1980 and supplemental information as listed in the reference sections, Northeast Nuclear Energy Company (NNECO or the licensee) requested an amendment to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2 (Millstone-2 or the facility). The amendment request consists of:

- Appendix A (Safety) Technical Specifications (TS) changes resulting from the analyses of the Cycle 4 reload fuel;
- Continued approval to operate with modified (sleeved, reduced flow and insert) Control Element Assembly (CEA) guide tubes;
- Approval of Engineering Safety Features, (ESF) component leakage outside containment TS;
- Continued approval of low temperature operation for special tests; and
- Preventing containment purging in Modes 1, 2, 3 and 4.

The associated specific TS changes are described in Section 3.0 of the following Safety Evaluation (SE).

In addition, this SE addresses our evaluation of:

- Mode 5 boron dilution event review;
- Containment electrical penetrations replacement;
- Steam generator tube and support plate inspection;
- Reactor coolant pump (RCP) speed sensing proximity probe and transmitter qualification;
- Reactor cavity neutron shield dose reduction;
- Steam generator feedwater piping inspections;
- Replacement of stem mounted limit switches; and
- Reactor cooling system (RCS) vent installation.

In early 1977, NNECO indicated to the NRC staff their intention to change fuel assembly vendors from Combustion Engineering, Inc. (CE) to Westinghouse Electric Corporation (W). Meetings were held at least once per year to keep the staff aware of progress on this project (References 2, 18, 26 and 40). The proposed reload licensing schedule and official application were submitted in February and May 1980, respectively (References 22 and 33). In March 1980, NNECO submitted the

Basic Safety Report (BSR), (Reference 25) authorized by W for Millstone-2. On June 3, 1980, the Reload Safety Analysis (RSA), which gave the specific analysis results for Cycle 4 operation, was submitted (Reference 39). Other information is as listed in the letter references (Section 7.0 of this SE).

The basic approach taken by NNECO and W was to determine where the Cycle 3 analysis by CE is bounding the Cycle 4 analysis. This is logical since two-thirds of the Cycle 4 core remains CE fuel. In the majority of cases, such bounding is achieved according to the licensee. The staff review consists of confirming that the Cycle 4 analysis is indeed bounded by the Cycle 3 analysis of record and, where such condition does not exist, perform a complete review of the licensee analysis.

In our Reference 11 letter transmitting the Cycle 3 authorization for operation, a number of open issues were addressed. NNECO has provided the information necessary to evaluate these items, and we will evaluate each in Sections 2.9 through 2.14 of this SE. In addition, modifications performed to add a RCS vent system will need to be partially evaluated in Section 2.17.

2.0 Discussion and Evaluation

In this evaluation of the Cycle 4 reload using, for the first time, fuel assemblies designed and manufactured by Westinghouse in the Millstone-2 core, use is made of our generic review of the Reference 10 BSR and various other topical reports. Some of the topical reports have received formal NRC staff approval. In all cases where a topical report has not received such an approval, the report has been examined, its methods judged to be reasonable, and an appraisal has been made that a complete review will not reveal the methodology to be significantly in error. On this basis, all topicals referenced are judged to be acceptable for this reload of Millstone-2 and for operation at the licensed power level of 2700 MWt.

2.1 Physical Core Design

During the Cycle 4 refueling outage of Millstone-2, 4 Batch B and 68 Batch C fuel assemblies of the CE design will be discharged and replaced with 72 new Batch F fuel assemblies of the W design. The pertinent characteristics of the Cycle 4 core are:

Assembly Designation	Vendor	Number of Assemblies	Initial Enrichment w/o U235	BOC Burnup Average (MWD/MTU)	EOC Burnup Average (MWD/MTU)
B+	CE	1	2.336	17566	28616
D1	CE	24	2.7349	21363	32413
D2	CE	48	3.0207	19380	30430
E1	CE	24	2.730	12759	23809
E2	CE	48	3.235	8829	19879
F1	W	24	2.70	0	11050
F2	W	48	3.30	0	11050

2.1.1 Fuel Design

The objectives of the fuel system safety review are to provide assurance that: (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrencs; (b) fuel system damage is never so severe as to prevent control rod insertion when it is required; (c) the number of fuel rod failures is not underestimated for postulated accidents; and (d) coolability is always maintained. We have reviewed the information provided by the licensee in support of Millstone-2 Cycle 4 operation to insure these objectives are met.

The Millstone-2 Cycle 4 core will be comprised of: (a) 145 fuel assemblies that were manufactured by CE, the original NSSS vendor; and (b) 72 fuel assemblies supplied by W, the Cycle 4 reload fuel vendor.

The fuel management pattern was developed to accommodate a Cycle 3 endpoint exposure range of 9850 MWD/MTU to 10850 MWD/MTU. The actual core exposure achieved during Cycle 3 was 10391 MWD/MTU bringing the core average End of Cycle (EOC) exposure to 20833 MWD/MTU. After the core reload, the Beginning of Cycle (BOC) 4 core exposure will be 10381 MWD/MTU making the predicted EOC 4 average core exposure about 21431 MWD/MTU.

The W reload fuel was designed to be geometrically similar and compatible with the CE fuel presently in Millstone-2 (CE Reference). The following table provides a comparison of the fuel mechanical designs.

Design Parameters	CE Reference	W Reload
Fuel Assembly		
Fuel Rod Array	14x14	14x14
Number of Fuel Rods	176	176
Number of Spacer Grids	9	9
Number of Control Rod Guide Tubes	4	4
Number of Instrument Tubes	1	1
Assembly Pitch (inches)	8.180	8.180
Fuel Rod Pitch (inch)	0.580	0.580
Fuel Pellets		
Length (inch)	0.450	0.600
Column Height, cold (inches)	136.7	136.7
Theoretical Density (percent)	94.75 - 95.0	95.0
Diameter (inch)	0.3765	0.3805
Fuel Cladding		
Outer Diameter (inch)	0.440	0.440
Thickness (inch)	0.026 - 0.028	0.026
Control Rod Guide Tube		
Outer Diameter (inches)	1.035	1.035
Thickness (inch)	0.038	0.038
Instrument Tube		
Outer Diameter (inches)	1.035	1.035
Thickness (inch)	0.040	0.038

CE will not be supplying any of the fresh fuel assemblies for the Cycle 4 reload core. Therefore the safety evaluation of Cycle 4 operation with residual CE supplied fuel assemblies is mostly unchanged relative to that of the previous Cycle 3 reload safety evaluation report (Reference 11). However, our review has identified several issues related to the CE fuel that require judicious consideration. These issues arise and will be addressed mainly because of the higher exposures that residual CE fuel assemblies will achieve during Cycle 4 operation.

To ensure that the design bases of the Millstone-2 Cycle 4 reload fuel are met, W used their standard evaluation techniques including their fuel performance model. These are described in the Basic Safety Report (Reference 25). With exception of the fuel rod internal gas pressure design basis, the specific design bases are given in RESAR-414 (Reference a). The W reload fuel rods are designed such that the internal gas pressure will not exceed the nominal primary system coolant pressure during the design life of the fuel. This is an acceptable criterion according to the Standard Review Plan (Section 4.2) and it is more conservative than the criterion used in RESAR-414.

To establish the Reactor Protection System (RPS) setpoints, which determine Limiting Safety System Settings (LSSS) and the Limiting Conditions for Operation (LCOs), the W fuel is designed to conform to the following Specified Acceptable Fuel Design Limits (SAFDLs).

1. The peak linear heat rate must be below that which would cause incipient fuel centerline melting (4700°F).

2. The departure from nucleate boiling thermal limits must not be exceeded (W-3 DNBR $\geq$ 1.30).

These two SAFDLs are equivalent to the original SAFDLs used by CE, and they have been traditionally accepted by NRC.

2.1.2 Design Error

In accordance with the requirements of 10 CFR Part 21 ("Reporting of Defects and Noncompliance"), NNECO notified the NRC Office of Inspection and Enforcement of a fuel design error in the 72 W reload fuel assemblies to be used in Millstone-2 Cycle 4 operation (Reference 46).

The design error was discovered by W after shipment of fuel to the site. Specifically the design error was the result of a miscalculation in sizing the vertical dimension between the top of the guide thimble tube end plugs and the seating surface of the control element assembly (CEA). The revised calculations indicated that there would be inadequate clearance for CEA penetration during a scram at system operating temperature. W estimated (Reference 54) that the fuel assemblies would have been capable of sustaining such loading, but there was a potential for CEA damage due to impacting loads applied to the CEA bullet-shaped tips (Reference 54). Such damage could have consisted in loss of CEA cladding integrity and a corresponding loss of CEA poison inventory.

Consequently, the fresh Batch F fuel assemblies were shipped back to the W fuel fabrication facility in Columbia, South Carolina for design modifications. The modifications consisted of machine boring into the upper surface of the guide thimble tube end plugs to allow an additional 0.625 inch CEA penetration. This machining operation was accomplished by inserting custom-designed drill bits into each of the guide thimble tubes (via the top nozzle openings) and driving the bits by a shaft entering from the opposite end of the guide thimble tubes (via the existing threaded holes in the guide thimble tube end plugs).

W has reanalyzed the modified region where the guide thimble tube is welded to the end plug (Reference 58), and concluded that the boring operation did nnot degrade the fuel assembly load-carrying capability below that of the structural design criteria specified in the Millstone-2 FSAR. Additional verification of the weld integrity was provided by uniaxial tension tests performed on 8 qualified specimens. All specimens failed at a load in excess of the W minimum 7000 pound limit.

The NRR staff has reviewed and witnessed the design modification process. We conclude that the licensee and the vendor actions were prudent, the resultant modifications satisfactory, and no additional concerns remain for the Millstone-2 Cycle 4 reload fuel in this regard.

2.1.3 Cladding Collapse

CE has written a computer code that calculates time-to-collapse of Zircaloy cladding in a pressurized water reactor environment (Reference b). We have reviewed this code and found it acceptable as described in our safety evaluation, which is bound into Reference b. For Cycle 3 operation, CE performed time-to-cladding-collapse calculations using the CEPAN code and the worst-case combination of material properties and component dimensions including the allowable manufacturing tolerances. The results of this analysis showed that the minimum time-to-collapse is in excess of the design batch-average discharge lifetime of the CE fuel.

A topical report describing the details of a W cladding collapse model (Reference c), which, for a given fuel region, predicts initial collapse time and the collapsed rod frequency for pressurized rods containing relatively stable fuel, was reviewed by the staff. This revised analysis was based on the results of TV examinations of irradiated fuel rods, and the results indicated that the original collapse model significantly underpredicted the time and frequency of collapse (Reference d). The COLLAP computer code is used to perform these calculations. The revised model was accepted for use in safety analysis related to licensing subject to provisions specified in our safety evaluation report (Reference e), which required that no alterations to the specified curves used as input to the model be made. We find the model has been applied in the approved manner and, therefore, the cladding collapse calculations have been performed acceptably.

All W reload fuel rods are internally prepressurized with helium during final welding to reduce cladding compressive stresses during service. The combination of the level of prepressurization, 95% theoretical density fuel pellets, and cladding wall load-carrying capacity have been designed to preclude cladding collapse during the projected Cycle 4 lifetime in Millstone-2.

We conclude that the fuel rod cladding in Millstone-2 Cycle 4 core will not collapse and is, therefore, acceptable.

2.1.4 Fuel Rod Bowing

NNECO evaluated the fuel rod bowing effects on DNBR margin for Millstone-2 Cycle 4 CE fuel (Reference 60). Within the range of Cycle 3 termination points and predicted Cycle 4 lifetimes, no more than 73 assemblies will exceed the DNB reduction or penalty threshold burnup of 24,000 MWD/MTU. At EOC 4, the maximum burnup attained by any of these assemblies will be 35,800 MWD/MTU. The corresponding DNB penalty for 35,800 MWD/MTU is less than 4.4 percent.

The licensee has performed an examination of the power distributions that shows the maximum radial peak at HFP in any of the assemblies that eventually exceed 24,000 MWD/MTU is at least 15 percent less than the maximum radial peak in the entire core (Reference 60). Since the percent increase in DNBR should not be less than the percent decrease in radial peak, there exists at least 15 percent DNB margin for assemblies exceeding 24,000 MWD/MTU relative to the DNB limits established by other assemblies in the core. We, therefore, conclude that there is no need for a Cycle 4 rod bowing penalty on the CE fuel.

In the BSR (Reference 25), W has used a formula from the unapproved topical report WCAP-8691 that projects anticipated rod bow magnitudes due solely to geometrical changes in the fuel rod thickness and diameter and spacer grid span length (Reference f). This formula has been somewhat controversial and has in the past been rejected by NRC. Therefore, we have required that the degree of rod bowing in the W reload fuel be calculated with the existing approved method, which is relatively more conservative.

W has subsequently recalculated the degree of rod bowing with the approved method (Reference 44). This recalculation shows that the average burnup at which time a gap closure of 50% is attained is 32,000 MWD/MTU. (The value of 50% corresponds to the gap closure at which a possible DNBR penalty is required on W fuel designs.) Consequently, W has concluded that there is no need for a DNBR penalty on the W designed fuel assemblies.

We, therefore, conclude that there is no need for a Cycle 4 rod bowing penalty on the W fuel.

2.1.5 Thermal Performance

The engineering methods used by W to analyze the densification effects on fuel thermal performance have been previously submitted to the staff and approved for use in licensing (Reference g). The methods include testing, mechanical analyses, thermal and hydraulic analyses, and accident analyses. The results of our review are reported in a technical report on the densification of W PWR fuel (Reference h), and additional information on densification methods can be found in "The Analysis of Fuel Densification," NUREG-0085 (Reference i).

The improved W fuel thermal performance code as described in WCAP-8720 was used for the Millstone-2 safety analysis (Reference j). This code contains a revision of an earlier fission gas release model and revised models for helium solubility, fuel swelling, and fuel densification.

The new W code was approved with four restrictions as described in our safety evaluation of February 9, 1979 (Reference k). Three of those restrictions deal with numerical limits and have been complied with by the licensee. The fourth restriction relates to the use of the PAD-3.3 code for the analysis of fission gas release from uranium dioxide (UO_2) for power increasing conditions during normal operation. This restriction applies to the safety analysis of Millstone-2. However, W has stated that this restriction does not adversely affect the results of the safety analyses performed for Millstone-2. Although we believe that this is essentially correct for the planned operation of Millstone-2, W has prepared and submitted a detailed evaluation of this restriction (Reference l).

At this time, we have not completed our review of the W evaluation of this restriction. However, our review has progressed to the point where the following conclusions can be made:

- The W evaluation of our restriction on the use of the PAD-3.3 code supports their earlier statement that the restriction does not adversely affect the results of the safety analyses performed for Millstone-2.

- We continue to believe that this result is essentially correct and anticipate some additional information from W to confirm this conclusion.

- Because the restriction pertains to the release of fission gases from the fuel, any change in our conclusions would not have significant impact at low burnup, when the fission gas inventory in the fuel is low.

At this time, we can therefore state that for Cycle 4 operation at full power, the restriction for PAD-3.3 is not significant and the analyses as presently docketed are acceptable.

2.1.6 Fuel Rod and Spacer Grid Fretting Wear

The W reload fuel for Millstone-2 employs a spacer grid/fuel rod support (i.e., springs and dimples) design similar to that in standard W fuel assemblies. Therefore, W has not seen a need to conduct long-duration flow tests to investigate the grid/cladding fretting wear potential of the new W supplied fuel for Millstone-2 Cycle 4 reload. For their standard fuel, W has found acceptable experience in: (a) 1000-hour duration flow tests for several spacer grid/fuel rod configurations and; (b) post-irradiation examinations of spent fuel assemblies, which have not shown evidence of appreciable wear. We agree that the W design and experience are sufficient to conclude that the W reload fuel will have an acceptable resistance to fretting wear.

2.1.7 Swelling and Rupture During LOCA

The NRC staff has been generically evaluating three materials models that are used in ECCS evaluation models. Those models are cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have: (a) met and discussed our review with industry representatives (Reference m); (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis, (Reference n) and; (c) required fuel vendors and LWR licensees using Zircaloy cladding to confirm that their plants would continue to be in conformance with the ECCS criteria of 10 CFR 50.46 if the materials models of NUREG-0630 were substituted for those models of their ECCS evaluation models (Reference 16).

NNECO has responded to our request for information concerning the new fuel cladding materials models described in NUREG-0630 (References 20 and 21). NNECO has reviewed all of the subject information supplied by CE and is in agreement with the results. Those results are that the calculated peak fuel cladding temperature will be lowered with the use of the NUREG-0630 ramp-rate-dependent strain and flow blockage models, provided that offsetting margins are allowed for the use of the new CE revised thermal-hydraulic analyses, which has been previously submitted to NRC for review (Reference 4). The information provided did not address what impact the use of the NUREG-0630 rupture temperature model would have on the Millstone-2 LOCA analysis. In the stress region of application to the Millstone-2 analysis, the NUREG-0630 rupture temperature model underpredicts (i.e., is more conservative) than the CE rupture temperature model. However, we believe that the impact of this omission is adequately bounded by CE's conservative use of only peak strain and flow blockage values that are given in NUREG-0630, irrespective of the specific Millstone-2 cladding failure stress and temperature conditions. We therefore conclude that NNECO has provided an acceptable justification that the original CE fuel in Millstone-2 will remain in compliance with 10 CFR 50.46 criteria.

The W materials models for large-break LOCA analysis are described in WCAP-9528 (Reference 12). These models are virtually the same as those used in prior ECCS evaluation models by W and they were evaluated in NUREG-0630. Small differences are attributable to modifications that were made to reflect the geometrical differences in fuel designs for the Millstone-2 plant. We have also required plant analyses performed with the ECCS evaluation model as described in WCAP-9528 to be accompanied by supplemental analyses to be performed with the materials models of NUREG-0630.

Those supplemental calculations for the large-break LOCA analysis have been provided by NNECO (Reference 53). Also addressed was a recently identified non-conservatism of the W 1978 ECCS evaluation model. The new concern was discovered by W who formally notified the staff in November 1979 (Reference 17).

Specifically, W had discovered that the February 1978 ECCS evaluation model was, in part, based on cladding burst tests which were conducted at relatively fast temperature-ramp rates; whereas the LOCA analyses of actual plant heat-up rates were at relatively slow temperature-ramp rates.

The NNECO Reference 53 submittal assessed the impact of this calculational error and the NUREG-0630 models to be worth 855°F peak cladding temperature over that presently analyzed. Subsequently W calculated a required reduction in total peaking factor (F_Q) of 0.0269 which would offset the 855°F increment in peak cladding temperature.

However, $\underline{W}$ had identified a margin in F_Q available through the use of Millstone-2 Cycle 4 specific fuel input parameters rather than using the previous conservative input parameters. This margin was identified as being worth 0.0271 in F_Q. Thus no F_Q reduction is required.

Based on the above, we find that the concerns related to LOCA-induced cladding swelling and rupture are satisfied for Cycle 4.

2.1.8 Seismic and LOCA Mechanical Response

One of the NRC's generic unresolved safety issues deals with asymmetric blowdown loads in a LOCA (References 3, 5 and o). For the fuel assemblies, the asymmetric blowdown loads and the loads from the design-basis earthquake are used to determine if fuel assembly components meet certain acceptance criteria.

These analyses have been submitted by NNECO (References 23, 43 and 47). The asymmetric blowdown loads for a whole core of CE fuel have shown that grid deformation occurs in fuel assemblies adjacent to the core barrel, although these deformations were shown not to have an effect on the limiting LOCA analysis. However, the analysis was not done for the mixed (CE and $\underline{W}$) core, and the different mechanical properties (Inconel grid vs. Zircaloy grid) and design differences could have an adverse effect (Reference 57). The comparative statement that the $\underline{W}$ grid is stronger than the CE grid is not adequate, and a complete analysis (seismic plus LOCA) for the mixed core in Cycle 4 and future Cycles is, therefore, required.

The Action Plan (Reference o) for dealing with asymmetric blowdown loads provides a period of time to achieve resolution of this issue and gives a basis for continued plant operation within this period. Since the review of this issue for Millstone-2 is still active and will not be completed for about a year, resolution is not required at this time. In order for the fuel-related issue to reach resolution on the same schedule as the generic issue, the fuel assembly analysis for a mixed core will need to be submitted in about 6 months. NNECO has agreed to provide such an analysis by April 1, 1981. On the basis of NNECO's commitment to perform this analysis and the grace period allowed in the unresolved safety issue Task Action Plan, this issue is adequately resolved for the initiation of Cycle 4 operation.

2.2 Nuclear Analyses

The nuclear design model used for the analysis of the Millstone-2 Cycle 4 core using $\underline{W}$ reload fuel consists of design procedures, computer codes, and nuclear data libraries previously used by $\underline{W}$ for the analysis of $\underline{W}$ cores.

Because of some differences between the Millstone-2 (CE type core) and W reactor cores, some slight changes to the geometry descriptions in the computer codes were required. In addition to verifying these W standard nuclear design methods by application to critical experiments and W operating reactor data, the methods have been further verified by analysis of measured data from previous Millstone-2 cycles.

For Cycle 4, the following W computer codes were used: (1) fuel and non-fuel neutron cross sections were obtained with LEOPARD and CINDER, while cross sections for CEAs were calculated by HAMMER and AIM; (2) the TURTLE code was used for two- and three-dimensional diffusion depletion calculations; and (3) PANDA was used for axial diffusion depletion calculations. The PALADON nodal analysis code was used for core design and safety analysis calculations which require full core descriptions.

Since these codes have all either been reviewed and approved by the staff or are industry-wide accepted codes, we find their use acceptable for this reload.

2.2.1 Nuclear Parameters

Comparisons between measured and predicted startup physics data from Cycles 1, 2 and 3 are presented in the BSR for CEA worth, critical boron concentration, isothermal temperature coefficient, and radial power distribution. The agreement, in general, is good and is consistent with that obtained by other vendors with currently approved design methods and is, therefore, acceptable. A summary of core physics characteristics for Cycle 3 and those predicted for Cycle 4 is as follows:

	Units	Cycle 3	Cycle 4
Dissolved Boron			
Critical Boron Concentration (CEAs withdrawn)			
Hot, Full Power, Equilibrium Xenon, BOC	PPM	830	1000
Boron Worth			
Full Power, BOC	PPM/%Δρ	93	98
Full Power, EOC	PPM/%Δρ	82	82

	Units	Cycle 3	Cycle 4
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients			
Hot, Full Power, Equilibrium Xenon, BOC	10^{-4} Δρ/F	- .2	- .42
Hot, Full Power, EOC	10^{-4} Δρ/F	-1.8	-2.36
Doppler Coefficient			
Hot, BOC, Zero Power	10^{-5} Δρ/F	-1.44	-1.80
Hot, BOC, Full Power	10^{-5} Δρ/F	-1.13	-1.20
Hot, EOC, Full Power	10^{-5} Δρ/F	-1.22	-1.31
Total Delayed Neutron Fraction, β_{eff}			
BOC		.00624	.00584
EOC		.00524	.00508
Neutron Generation Time, ℓ			
BOC	10^{-6} sec	27.2	18.1
EOC	10^{-6} sec	31.8	19.7

At EOC 4, the reactivity worth with all CEAs inserted assuming the highest worth CEA is stuck out of the core is 6.32% Δρ assuming a 10% uncertainty reduction. The reactivity worth required for shutdown, including the contribution required to control the steam line rupture incident at EOC 4, is 6.18% Δρ. Therefore, sufficient CEA worth is available to accommodate the reactivity effects of the steam line break at the worst time in core life allowing for the most reactive CEA stuck in the fully withdrawn position and also allowing for calculational uncertainties. We have reviewed the calculated CEA worths and the uncertainties in these worths based upon appropriate comparison of calculations with experiments. On the basis of our review, we have concluded that the NNECO's assessment of reactivity control is suitably conservative, and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive CEA is stuck in the fully withdrawn position.

The augmentation factor (used to account for the power density spikes due to axial gaps caused by fuel densification) was included in the determination of F_Q for all accident analyses performed for Cycle 4. The TS limits on local power density, LOCA peak liner heat rate, and LOCA allowable power level also account for the augmentation factor. The Cycle 4 maximum augmentation factor of 1.056 is approximately the same as the Cycle 3 value of 1.054.

At a meeting held on June 4, 1980 between the NRC staff and representatives of NNECO and W (Reference 40), W provided analyses which showed that Cycle 4 peaking factors are within 0.5% of the Cycle 3 values. Since these peaking factors are reflected in the safety analyses and are less than the TS values, we find them acceptable.

Since the fuel rod support grid for the W supplied fuel assemblies will be Inconel-718 whereas the CE supplied fuel assemblies will have Zircaloy-4 grids, the effects of the nuclear and thermal expansion properties of both materials were considered in the evaluation of the physics parameters for Cycle 4. Calculations of $F_Q(Z)$ include a multiplicative factor, applied to the axial peaking factors, to account for axial inhomogeneities introduced by assembly grids. The inclusion of the grid multiplicative factor bounds the inhomogeneities due to either Zircaloy or Inconel grids and is, therefore, acceptable to the staff.

Comparisons of power peaking in fuel pins adjacent to CEA water holes using TURTLE (diffusion theory) and KENO (Monte Carlo) have shown an underprediction by diffusion theory, as expected. The maximum underprediction by TURTLE occurs diagonally next to a water hole. Due to the unavailability of experimental results on water hole peaking factors, the maximum bias was confirmed by comparisons of TURTLE and INCA results for Cycles 1, 2 and 3 (Reference 45). We find this water hole peaking correction to be acceptable.

The power distribution control philosophy to be used in Cycle 4 is Relaxed Axial Offset Control (RAOC) which is similar to the procedure used for Cycle 3 in most respects. One difference is that the new method relies on diffusion theory exclusively, whereas nodal methods were used previously in several areas.

Also, the method utilized in the xenon shapes library is different. However, the entire range of xenon and rod insertion limits are covered. Based on the information presented in the BSR and additional discussions with NNECO and W, we find the RAOC procedure acceptable for providing power distribution control limits for Cycle 4 operation.

2.3 Thermal Hydraulic Design

The thermal hydraulic design for Millstone-2 Cycle 4 is presented in the Basic Safety Report (Reference 25).

2.3.1 Hydraulic Compatibility

The W Cycle 4 reload fuel assembly for Millstone-2 is designed to be dimensionally and hydraulically compatible with the CE Cycle 3 reference fuel assembly. As shown in the following table, the fuel rod diameter (0.440 inch), fuel rod pitch (0.580 inch), and fuel assembly pitch (8.18 inch) are the same for both types of assemblies. Therefore, the rod bundle axial and lateral flow areas, the axial frictional pressure drop, and the lateral flow (crossflow) resistance will be the same for both designs. The hydraulic effects of the different configurations used by W Cycle 4 and the Millstone-2 reference Cycle 3 in the upper nozzle, lower nozzle and the grids have been minimized since the W components have, as closely as possible, the same blockage as the Millstone-2 reference cycle design. The pressure drop through these components consists primarily of form (expansions and contractions) rather than frictional losses. Therefore, matching the blocked area results in matching pressure drop.

	Cycle 4 Westinghouse	Millstone-2 Reference Cycle
Assembly Envelope, inch	8.19	8.19
Assembly Pitch, inch	8.18	8.18
Lower Nozzle Blocked Area, %	64	64
Rod Array	14x14	14x14
Thimble O.D., inch	1.11	1.115
Rod O.D., inch	.440	.440
Rod Pitch, inch	.580	.580
Assembly: fl/De	3.90	3.90
Number of grids	9	9
Grid Blocked Area, %	20	22
Upper Nozzle Blocked Area, %	56	57

The W fuel assembly was tested in the Fuel Assembly Test System (FATS) hydraulic loop to confirm that the resistance was the same where physical differences exist. The two areas of physical dissimilarity are:

- The Grid--The W and Millstone-2 reference cycles have different hold-down spring and dimple arrangements.

- Location of fuel rods off bottom--The rods for W design are from 0.17 to 0.20 inches above the top of the bottom nozzle. The Millstone-2 reference cycle fuel rods touch the bottom nozzle.

The results of the FATS test analysis show that the grids can be treated as having identical resistance and that the effects on pressure drop of the differences between the fuel rods on and off the bottom nozzle are negligible.

The similarities in dimensions and blockage area and the test results, showing insignificant differences in resistances, indicate that the W and Millstone-2 reference cycle fuel assemblies can be treated as being hydraulically identical. This hydraulic compatibility is assumed by the W BSR, and we find this assumption acceptable.

2.3.2 DNBR Review

A comparison of the thermal-hydraulic design conditions for Millstone-2, Cycles 2, 3 and 4 is as follows:

Parameter	Units	Cycle 2 Values (CE)	Cycle 3 Values (CE)	Cycle 4 Values (W)
Power Level	MWT	2611	2754	2754
Maximum Steady State Core Inlet Temperature	°F	554	551	551
Minimum Steady State RCS Pressure	psia	2200	2200	2200
Minimum Reactor Coolant Core Flow (2200 psia, 551°F)	E6 lb/hr	134.9	133.7	133.7
Maximum Allowed Initial Peak Linear Heat Rate (DBEs Other Than LOCA)	kw/ft	16.0	16.0	16.0
Steady State Linear Heat Rate to Fuel Centerline Melt	kw/ft	21.0	21.0	21.0

Parameter	Units	Cycle 2 Values (CE)	Cycle 3 Values (CE)	Cycle 4 Values (W)
Total Planar Radial Peaking Factors				
For DNB Margin Analyses (Fr)				
Unrodded Region		1.440	1.598	1.59
Bank 7 Inserted		1.550	1.806	1.74
For kw/ft Limit Analyses (Fxy)				
Unrodded Region		1.540	1.584	1.60
Bank 7 Inserted		1.660	1.822	1.74
CHF Correlation		W-3	CE-1	W-3
Minimum Acceptable DNBR		1.30	1.19	1.30

The design power level for Millstone-2 Cycle 4 remains 2700 MWt (the same as for Cycle 3). The safety analysis uses a power level of 2754 MWt (102% power) to allow for measurement uncertainties. A summary of our evaluation follows.

2.3.2.1 Critical Heat Flux

The steady state DNB analysis for Cycle 4 was performed using the THINC-I code in conjunction with the W-3 correlation (References r, s and t). For the W-3 correlation, the 95/95 confidence/probability limit for not suffering departure from nucleate boiling is a DNBR greater than 1.30. In the analysis, uncertainties in various measured parameters were factored in as biases for LCO and LSSS setpoints. This biasing of the measurement uncertainties in the analysis is equivalent to adding the absolute power 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 for Cycle 4, as determined with the THINC-I code in conjunction with the W-3 correlation (grid spacer correction = 1.0), and for Cycle 3, as determined with the TORC thermal hydraulic code in conjunction with the CE-1 correlation, are shown below.

Percent Uncertainties

Measured Parameter	Measured Parameter Uncertainty	Equivalent Power Uncertainty - % Cycle 3	Cycle 4
Axial Shape Index (ASI)	0.06 ASIU	2.2%	3.0%
Pressure	22 psi	0.8	0.5
Temperature	2 F	0.9	1.0
Flow	4%	5.0	2.0
Power (LCO)	2%	1.4	2.0
Power (LSSS)	5%	3.5	5.0

NOTE: Cycle 3 determined with TORC code in conjunction with CE-1 correlation. Cycle 4 determined with THINC-I code in conjunction with W-3 correlation.

LCO = Limiting Conditions for Operation
LSSS = Limiting Safety Systems Settings

The uncertainties in measured parameters were additively and statistically combined, as shown below to arrive at values for the Limiting Conditions for Operation (LCO) and Limiting Safety Systems Settings (LSSS).

Combined Uncertainties

	(1) Equivalent Sum	(2) Root Sum of Squares	(3) Difference (1) - (2)	(4) % Credit Taken	Net Uncertainty (1) - (4)
LCO					
Cycle 3	10.3%	5.8%	4.5%	3.0%	7.3%
Cycle 4	8.5	4.3	4.2	3.0	5.5
LSSS					
Cycle 3	12.4	6.6	5.8	3.0	9.4
Cycle 4	11.5	6.3	5.2	3.0	8.5

NOTE: Cycle 3 determined with TORC code in conjunction with CE-1 correlation. Cycle 4 determined with THINC-I code in conjunction with W-3 correlation.

LCO = Limiting Conditions for Operation
LSSS = Limiting Safety Systems Settings

For Cycle 4, the equivalent sum of these uncertainties is 8.5% for LCO and 11.5% for LSSS. These uncertainties were also treated as statistically independent and combined using the Root Sum Square (RSS) method. This combination resulted in RSS uncertainties of 4.3% for LCO and 6.3% for LSSS. Instead of taking full credit for statistical combination of the uncertainties using the RSS method, NNECO has taken partial credit for only 3% uncertainty for both the LCO and LSSS. For Cycle 4 this results in net uncertainties of 5.5% for LCO and 8.5% for LSSS. For Cycle 3, the same partial credit of 3% uncertainty was also applied and resulted in net uncertainties of 7.3% for LCO and 9.4% for LSSS. The following parameters related to LCO and LSSS are the same for Cycles 3 and 4 as shown before: power level (2754 MWt), maximum steady state core inlet temperature (551°F), minimum reactor coolant flow (133.7×10^6 lb/hr), and steady state linear heat rate to fuel centerline melt (16.0 kw/ft).

NNECO has agreed to provide justification for the measurement uncertainty values [Axial Shape Index (ASI), Pressure, Temperature, Flow, Power (LCO) and Power (LSSS)] for further review of the Cycle 4 power uncertainties. While our review of measurement uncertainties continues, LCO and LSSS limits will be maintained at the values used for Cycle 3 based on the more conservative Cycle 3 analysis results. This will have the effect of limiting the partial credit for statistical combination of uncertainties to 1.2% on LCO and 2.1% on LSSS compared to the 3% shown. We find this acceptable.

2.3.2.2 Reactor Coolant Flow

The design flow for the Cycle 4 analysis is 370,000 gpm (133.7×10^6 lb/hr at 2200 psi and 551°F) and is the same as the low flow limit included in the Technical Specifications and analysis for Cycle 3. The actual flow rate from measurements at Millstone-2 is 392,644 gpm, a value about 6% above that used in the analysis. We find the RCS flow input acceptable.

2.3.2.3 Rod Bowing

As discussed in Section 2.1.4, NNECO states that the effect of rod bowing for Cycle 4 CE fuel assemblies has been evaluated and at the end of Cycle 4 the maximum burnup will be 35,800 MWD/MTU for which the corresponding penalty is less than 4.4 percent. However, the reduction in DNBR due to rod bowing is offset by a credit for low radial peaking in the critical assemblies and no power penalty for rod bowing is required for Cycle 4 CE fuel assemblies. The Millstone-2 Cycle 4 exposure to W supplied Batch F fuel assemblies is predicted to be 11,050 MWD/MTU for which the corresponding gap closure will be less than that at which the CE DNBR correlation would require a reduction in DNB. Therefore, no rod bow penalty is required for Cycle 4 W fuel assemblies.

2.3.2.4 Peaking Factor

The total planar peaking factors for DNB margin analyses (Fr) and for kw/ft limit analyses (Fxy) are shown for Cycle 4 as well as for Cycles 2 and 3 and are relatively close in value for Cycles 3 and 4. Also, the table on pages 16 and 17 for Cycles 3 and 4, the maximum allowed initial peak linear heat rates (for DBEs other than LOCA) are identical (16.0 kw/ft) and, therefore, acceptable.

2.3.3 Peaking Factor Uncertainties

NNECO has submitted an Addendum to the BSR (Reference 35) which describes the power peaking factor uncertainty analysis used in the nuclear design of the reload fuel for Millstone-2 beginning with Cycle 4 operation. The analysis uses measured data from the first 3 cycles. Measured rhodium detector signals were combined with INCA coefficients, recalculated by W. The analysis, therefore, relied exclusively on W nuclear input data and measured signals. The uncertainty analysis accounts for the error in the

Fourier fit for the axial power shape used by INCA as well as a correction for three-dimensional effects on the power distribution. For Cycle 4 operation, NNECO has shown that the measurement uncertainties of 6% for Fr and 7% for F_Q used for Cycle 3 are adequate.

2.4 Accident and Transient Review

The licensee's analysis of accidents was provided in the Reference 25 BSR and the Reference 39 RSA. The proposed Cycle 4 TS were submitted by Reference 55. The RSA reanalyzed the boron dilution transient and the ejected CEA accident since, subsequent to the issuance of the BSR, some key input parameters for these events have been found to be nonconservative relative to those assumed in the BSR.

Since the BSR has not been fully reviewed and accepted by the NRC as a referenceable document, a parametric review for all the accidents and transients was conducted. This parametric review involved the assessment of the thermal-hydraulic and physics parameters calculated for Cycle 4 in light of the Cycle 3 methodologies.

In the parametric review, the input parameters and system and component behavior assumptions throughout the transient are compared for both cycles. When Cycle 4 parameters are equal to or bounded by their counterparts of Cycle 3, the Cycle 4 event was considered bounded by the Cycle 3 analysis. When an input parameter is not bounded by Cycle 3 values, the effect of such change on Cycle 4 operation is delineated as appropriate. The discussion below is separated into two categories, the anticipated operational occurrences (AOOs) and the postulated accidents.

2.4.1 Anticipated Operational Occurrences

Five of the AOOs have not been analyzed in the BSR for the following reasons:

- Startup of an Inactive RCP--Operation with less than 4 RCP running is precluded by the TS 3.4.1.
- Excess Load/Excess Heat Removal due to Feedwater Malfunction--The limiting cooldown transient is the Steam Line Break (SLB) whose analysis bounds these two AOOs. It is recognized that Excess Load and Excess Heat Removal are AOOs with moderate frequency of occurrence while the SLB is a postulated accident that is not expected to occur. However, the SLB analysis shows that no fuel experiences DNB, i.e., no fuel failure occurs as a result of this severe cooldown accident. Therefore, no fuel is expected to fail as a result of the less severe cooldown AOOs.
- Part Length CEA Drop/Part Length CEA Malpositioning--The part length CEAs have been removed from the Millstone-2 core. Therefore, no analysis is required.

The remainder of the AOOs are discussed below.

2.4.1.1 Boron Dilution

The boron dilution event has been analyzed for all operating modes using the following assumptions: (a) the RCS has the minimum initial boron concentration; (b) the RCS has the maximum critical concentration based on all the CEA out; (c) all the three charging pumps are delivering a maximum total of 132 gpm of unborated demineralized water into the reactor coolant system; (d) the RCS volume is at its minimum; and (e) the boron concentration is homogeneous. The time to criticality is calculated using the equation:

$$T = \frac{V}{Q} \ln \left(\frac{Ci}{Cc}\right)$$

where V = active RCS volume to be diluted (ft^3)
Q = maximum charging flow (ft^3/sec)
Ci = initial boron concentration (ppm)
Cc = critical boron concentration (ppm)

The NRC criteria requires a minimum time allowance of 30 minutes for operator intervention to terminate the transient during the refueling mode and of 15 minutes during any other mode of operation. The limiting dilution event for the Cycle 4 operation is for the refueling mode with a calculated time to criticality of 34 minutes which is more than the required 30 minutes. Therefore, we find this analysis and its results acceptable for all cases when the reactor is subcritical.

While the NRC criteria require that it be demonstrated that sufficient time is available for operator intervention during a Boron Dilution Event when the reactor is critical, no credit is given for operator intervention, and the analysis demonstrates that the consequences of a Boron Dilution Event without operator intervention are acceptable. Without operator intervention a Boron Dilution Event from power operation is terminated by the variable high power trip, the local power density trip, or the TM/LP trip. NNECO states that the most severe Boron Dilution Event would be less severe than the CEA Withdrawal Event because of the significantly slower reactivity insertion rate in the Boron Dilution Event analysis and a separate analysis is not required for the Boron Dilution Event. We concur with this statement.

2.4.1.2 Loss of Load/Loss of Feedwater

These two AOOs are both undercooling transients. The NRC requirements for these transients are that the DNBR and the overpressure criteria are not violated.

A comparison between the input parameters for Cycle 3 and those for Cycle 4 in the BSR revealed that there have been no changes to the reactor core or the reactor systems that would necessitate a non-conservative change in Cycle 4 analyses. Therefore, Cycle 3 analyses for the above AOOs are bounding to Cycle 4 operation, and the DNBR and the overpressure criteria are met.

2.4.1.3 Loss of Forced RCS Flow

This AOO is an undercooling transient. Similar to the Loss of Load/Loss of Feedwater transients above, there will be a probability of 95% with a confidence level of 95% that DNB will not occur, and the pressure will not exceed 110% of the code design value.

A comparison between the input parameters for Cycles 3 and 4 reveals no significant differences. We conclude that Cycle 3 analysis of this AOO is bounding to Cycle 4 operation and, therefore, is acceptable.

2.4.1.4 Malfunction of One Steam Generator

Out of a variation of malfunctions that could occur to one of the two steam generators, the licensee has determined that the loss of load to one steam generator is the most limiting asymmetric transient. The NRC requirements for this transient are that the DNBR and the overpressure criteria should be met.

Since the comparison between Cycles 3 and 4 parameters reveals no differences, we conclude that Cycle 3 analysis is bounding and the NRC acceptance criteria for this AOO are satisfied.

2.4.1.5 CEA Withdrawal

The CEA Withdrawal Event was reanalyzed from both the hot zero power condition and the full power initial condition. For the zero power case, two computer programs were used. WIT-6 was used to calculate the nuclear power (reactivity) transient and FACTRAN was then used to obtain the thermal heat flux transient and the fuel and clad temperatures. The reactor trips on the Variable High Power Trip at 25% power and the nuclear power does not overshoot the full power nominal value. The core and the RCS are not adversely affected since the combination of thermal power and the coolant temperature result in a DNBR greater than the limiting value at 1.30. For the full power case, the LOFTRAN computer program is used. The thermal margin/low pressure trip provides protection for this case and terminates the transient before the DNBR falls below 1.30. We have reviewed the initial conditions, the reactivity coefficients, and the CEA trip insertion characteristics and find the CEA withdrawal analyses and consequences acceptable.

2.4.1.6 CEA Drop

The CEA drop event was reanalyzed using standard W nuclear design methods to compute steady state power distributions. The peaking factors were then used in the THINC code to calculate the DNBR. LOFTRAN was used for the transient analysis. The results indicate that following the drop of the worst CEA, the reactor may return to full power without exceeding the core thermal limits. We have reviewed the assumptions used for initial system conditions as well as the reactivity feedback coefficients and dropped CEA worths used and find them to be acceptable.

2.4.2 Postulated Accidents

2.4.2.1 Main Steam Line Break (MSLB)

The MSLB is an overcooling transient. The NRC requirements for this accident are that the DNB criterion be met and that the radiological consequences be acceptable per 10 CFR 100. A comparison of the Cycle 3 key input parameters and the calculated values of those parameters for Cycle 4 indicates that the only significant difference is a decrease of the Moderator Temperature Coefficient (MTC) from -2.2×10^{-4} ΔK/K/°F to -2.4×10^{-4} ΔK/K/°F. The MTC decrease causes faster power rise if the accident occurs while the reactor is at full power. The faster power increase would cause an earlier reactor trip.

We conclude that the change in the Cycle 4 MTC would result in insignificant deviation from the conclusions reached in the Cycle 3 evaluation. Therefore, we find the NRC acceptance criteria for the MSLB accident to be met for Cycle 4.

2.4.2.2 Steam Generator Tube Rupture (SGTR)

The SGTR accident during Cycle 4 operation is bounded by the Cycle 3 analysis because the reactor power level, the inlet temperature, and the RCS pressure are the same for both cycles. Since the power level is the same, the radio-activity present in the reactor coolant and available for transfer to the secondary system is the same for both cycles. And since the initial pressure and temperature before the transient have not changed, the depressurization of the RCS is expected to be similar for both cycles.

The NRC acceptance criteria will be met for Cycle 4 operation.

2.4.2.3 RCP Seized Rotor

For the seized rotor accident, a comparison between the calculated Cycle 4 parameters and those parameters assumed in Cycle 3 analysis reveals no differences. Therefore, the NRC acceptance criteria will be met for Cycle 4 operation.

2.4.2.4 CEA Ejection

The CEA ejection accident was reanalyzed for both full power and zero power initial conditions at BOC and EOC using the TWINKLE code in one-dimension (axial) for the average core channel calculation and the FACTRAN code for the hot fuel rod transient heat transfer calculation. The analysis performed for the more limiting HFP case predicted a maximum fuel stored energy of 172 cal/gm which is well within the Regulatory Guide 1.77 limiting criterion of 280 cal/gm. We have reviewed the analysis assumptions including the Doppler and moderator temperature coefficients, delayed neutron fractions, initial fuel temperatures, ejected rod worths, hot channel factors and trip reactivity insertion and find the analysis to be conservative and the predicted consequences acceptable.

2.5 Loss of Coolant Accident Review

2.5.1 Large Break

The large break loss-of-coolant accident (LOCA) was analyzed by the licensee using a new model developed by W for reloads of CE NSSS (References 41 and 1). This model, which is applicable to CE 4 x 2 plants, uses as a basis Appendix K models previously approved for original models. First, there were changes needed to reflect the actual loop arrangement and core design of the CE NSSS. This includes the fact that fuel fabricated for Millstone -2 is dimensionally different from that used in W reactors using W fuel. The second type of change was to incorporate some improved analytical techniques approved for reactors equipped with upper head injection (UHI). The techniques adopted for CE NSSS reloads were the use of a split downcomer nodalization and improved drift flux slip flow modeling. We believe that these modifications meet the intent of Appendix K and this model is acceptable for the ECCS analysis of this reload.

Sensitivity studies documented in Reference 12 showed the limiting large break to be the double ended cold leg guillotine (DECLG). The analysis for CE fuel submitted for Cycle 3 confirms this finding (Reference 9). Therefore, the licensee needed only to submit DECLG analyses with appropriate discharge coefficients for the large break for this reload (Reference 41).

The following table presents the important results of three calculations:

Fuel Type Analyzed	W BOC 4	CE BOC 3	CE EOC 3
Total Power (MW)	2754	2754	2754
MLHGR (kw/ft)	15.6	15.6	15.6
Burnup MWD/MTU	∿500	6600	50000
Break	.6 DECLG	.6 DECLG	.8 DECLG
PCT (°F)	2110	1948	2081
Hot rod burst time (sec)	31.6	32.17	9.64
Hot rod burst location (ft)	7.5	8.44	∿6.00
End-of-bypass (sec)	21.65	22.0	19.8
Beginning of reflood (sec)	34.6	36.1	33.9
S.I. start (sec)	15.7	18.9	16.8
S.I. tank empty (sec)	66.8	63.5	61.3

The first calculation is for the limiting break with W fuel. The calculation was performed at early burnup with maximum densification and stored energy. This has traditionally been the worst case for W fuel in W NSSS. The second case was a CE calculation for the same discharge coefficient for the limiting CE fuel in Cycle 3. This calculation was also performed at the burnup which maximizes stored energy and peak cladding temperature (PCT). It was not, however, the worst case analyzed by CE. The third case (the limiting case for CE) used a slightly larger break size. CE also determined by direct analysis that high burnup (50,000 MWD/MTU) was the most limiting time in core life. Since high burnups were considered, the CE analyses would be applicable to CE fuel remaining in the reactor. For the W fuel, no information has been presented to determine the highest degree of burnup for their fuel in a CE NSSS. NNECO has agreed to provide this information with their Cycle 5 reload analysis. We believe that fission gas release effects which could cause high burnup fuel to be limiting would not be a factor in this first cycle (Cycle 4) with fresh W fuel. Therefore, the justification is not required until the next cycle.

The above table and References 12 and 41 show that the requirements of 10 CFR 50.46 are met for the cases analyzed. That is the peak cladding temperatures are all less than 2200°F, the local oxidations are all less than 17%, and the core wide oxidations are less than 1%.

2.5.2 Cladding Swelling and Rupture

The NRC staff has been generically evaluating three materials models that are used in ECCS evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have: (a) discussed our evaluation with vendors and other industry representatives (Reference a); (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference b); and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (References c and d).

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we have required that the ECCS analyses be accompanied by supplemental calculations to be performed with the materials models of NUREG-0630. For these supplemental calculations only, we have accepted other compensatory model changes allowed for the confirmatory operating reactor calculations mentioned above.

Those supplemental calculations have been provided by the licensee (Reference 53). Reference 53 also addressed a recently identified non-conservatism of the W 1978 ECCS evaluation model. The new concern was discovered by W who formally notified the staff in November 1979 (Reference 43).

Specifically, W had discovered that the February 1978 ECCS evaluation model was, in part, based on cladding burst tests which were conducted at relatively fast temperature-ramp rates; whereas the LOCA analyses of actual plant heatup rates (including those of Millstone-2) were at relatively slow temperature-ramp rates.

The Reference 53 submittal assessed the impact of this calculational error and the NUREG-0630 models to be offset by a corresponding reduction in F_Q of .0269. However, the licensee identified a margin in F_Q available through the use of a reduction in pellet temperature uncertainty (see Reference y for approval). This margin was worth 0.0271 in F_Q. Thus no F_Q reduction was required.

We find that this is a satisfactory accounting of this issue for large breaks for Cycle 4 and therefore, conclude that the licensee has satisfied our concerns related to the swelling and rupture issue.

2.5.3 Small Breaks

At our request, the licensee provided justification that the Cycle 3 small break LOCA analysis would remain valid for Cycle 4 operation (Reference 38). As noted in the reference, the phenomena affecting small break performance are primarily related to system variables. The fuel parameters affecting small break performance are power density, cladding thickness, and cladding diameter. These variables are identical or nearly identical for the W fuel and the limiting Cycle 3 fuel as shown by a preceding table in Section 2.1.1. To demonstrate the comparability of the W and CE analyses, the licensee provided a W calculation for the most limiting small break (Reference 38). Comparison of this analysis with the CE calculation for Cycle 3 (Reference g) shows reasonable agreement of most of the important results as shown in the following table:

Small Break (0.1 ft^2) Results

Parameter	Cycle 3 (CE)	Cycle 4 (W)
Peak Cladding Temp (°F)	1971	1978
PCt Elevation (ft)	9.7	11.2
Uncovery time (sec)	700	500
Recovery time (sec)	2200	1320
PCT time (sec)	1400	1312
Uncovery depth (ft)	6	6-1/2
Accumulator actuation (sec)	none	1313

The pressure transient, peak cladding temperature, and depth of uncovery are in very good agreement. There are some questions related to accumulator injection for this size break which need to be resolved before the W analysis would be acceptable. Also W proposed a model change (Reference 12) which results in an increase in core steam flow which requires further justification before approval. However, the licensee submittal (Reference 38) provides sufficient assurance that the requirements of 50.46 would be met for small breaks with all fuel present during Cycle 4.

2.5.4 LOCA Conclusions

The W large break model used for Cycle 4 is acceptable and meets the requirements of Appendix K. Large break spectrum requirements have been met and the large breaks analyzed comply with 10 CFR 50.46. An appropriate burnup sensitivity is required prior to Cycle 5 operation. Supplemental calculations supplied for swelling and rupture assessment are acceptable for Cycle 4. Cycle 3 small-break analyses are valid for Cycle 4 operation. W small-break model issues need resolution prior to Cycle 5 operation.

2.6 Radiological Consequences of Postulated Accidents

We have reviewed the BSR, RSA and the other submittals supporting Cycle 4 operation and find the potential radiological consequencs of design basis accidents to be appropriately bounded by the original May 10, 1974 Safety Evaluation or by the Cycle 3 Reload Safety Evaluation. Since the guidelines of 10 CFR Part 100 continue to be met, we find the potential consequences acceptable.

2.7 Low Temperature Operation

By application dated September 30, 1980 (Reference 59), NNECO requested that low temperature operation for short periods of time, as authorized for Cycle 3 by Amendment No. 55 (Reference 31), be allowed during Cycle 4. This previous authorization was for the performance of turbine generator efficiency testing; however, such testing was not completed during Cycle 3 because of an unexpected plant shutdown shortly after our approval.

In addition, the subject application identifies a concern with operating when the minimum RCS inlet temperature is below 549°F (value used in safety-analysis) to perform the moderator temperature coefficient determination test as required by the NRC. Proposed TS Table 3.2-1 specifies that the inlet temperature is to be greater than or equal to 537°F to validate the DNB margin analysis. The allowed peak linear heat rate should again be limited to 14.2 kw/ft instead of the normal 15.6 kw/ft value. We find that extension of the low temperature operation, for Cycle 4 is acceptable, as authorized by Reference 55 for Cycle 3.

2.8 Mode 5 Boron Dilution Event

By LER 80-05, dated March 21, 1980, NNECO notified us of a problem with the safety analysis for the Mode 5 (cold shutdown) boron dilution event. Their corrective action was to increase the shutdown margin from 1% to 2% when it was planned to drain down the RCS for any reason.

In Reference 55, NNECO shows that the analysis of record for this event uses 9500 ft^3 for the RCS volume. However, the RCS volume with the system drained to the centerline of the hot legs is only 4828 ft^3 . They conclude that the larger volume should be used when the RCS is full or all CEAs are inserted to yield a 1% shutdown margin.

In discussion with NNECO, we pointed out that it is never conservative to use the 9500 ft^3 volume for calculating the boron dilution event. The reason is that the borated coolant contained in the steam generators and pressurizer plus related piping is not really helpful in lengthening the time to the critical condition in the event unless good mixing is occurring (i.e., a RCP per loop operating). Confronted with this position, NNECO has agreed to a 2% shutdown margin requirement under all Mode 5 conditions.

We find that requiring a 2% shutdown margin in Mode 5 meets the acceptance criteria of SRP 15.4.6 and is, therefore, acceptable.

2.9 ESF Component Leakage Outside Containment

In response to Agreement No. 1 of our letter dated May 12, 1979 (Reference 11) NNECO proposed to include TS surveillance requirements to assure that leakage from emergency core cooling system (ECCS) and containment spray system (CSS) components outside containment are acceptable (Reference 13).

Our Reference 11 Safety Evaluation (SE) contained an analysis of the potential radiological consequences of leakage from engineered safety feature components outside containment following a postulated loss-of-coolant accident (LOCA). The SE concluded that the incremental doses when added to the LOCA doses are within the guidelines of 10 CFR Part 100 and are acceptable. We further stated that if the licensee proposed specifications limiting the amount of leakage to values equal to or less than those assumed in the SE, no further action will be required to assure acceptability of the radiological consequences of post-accident leakage from ESFs.

In Reference 13, NNECO proposed to limit the total maximum operational leakage rate from both ECCS and CSS to 12 gallons per hour. Standard Review Plan 15.6.5, Appendix B, states that the evaluation should be based upon twice the maximum operational leakage rate. The Reference 11 SE reviewed the radiological consequences due to leakage assuming a total leakage of 24 gallons per hour from the ECCS and CSS systems. Since the SE has evaluated the consequences at twice the maximum operational leakage rate proposed by the licensee, the consequences calculated in the SE are not changed and the conclusion reached in the SE remains valid. Therefore, we find the proposed additions to surveillance requirements, TS 4.5.2.C.5 and 4.6.2.1.c, acceptable.

2.10 Containment Electrical Penetrations

Agreement No. 2a of Reference 11 was for NNECO to propose a permanent type repair of the containment electrical penetrations. In response to this commitment, NNECO proposed to replace most of the 32 penetration modules which have experienced insulation resistance degradation with new modules qualified to IEEE 317-1976 requirements (Reference 34). In more recent correspondence (Reference 60), NNECO states that all 18 electrical penetration modules which had experienced serious degradation will be replaced. Approximately another 8 will be replaced as outage time permits. This would leave about 6 modules to be replaced next refueling outage. These remaining original electrical penetrations have not indicated insulation resistance degradation and NNECO finds that they are capable of performing their intended function.

We find that the licensee's containment electrical penetration replacement program is prudent and should be continued until all questionable penetrations have been replaced. Reference 60 indicates that the remaining penetrations have insulation resistance values greater than 100 megohms. This is the same acceptance criteria used for safety related circuits in past evaluations (see Section 2.6 of Reference 11). Therefore, we find that leaving about 6 electrical penetration modules with insulation resistance values greater than 100 megohms until future outages is acceptable.

2.11 CEA Guide Tube Integrity

A fretting wear has been observed (References u, v, w and x) in irradiated fuel assemblies taken from operating reactors with NSSS designed by CE. These observations revealed an unexpected degradation of guide tubes that are under control element assemblies (CEAs). It was subsequently concluded that coolant turbulence was responsible for inducing vibratory motions in the normally fully withdrawn control rods and, when these vibrating rods were in contact with the inner surface of the guide tubes, a wearing of the guide tube wall has taken place. Significant wear has been found to be limited to the relatively soft Zircaloy-4 guide tube because the Inconel-625 cladding on the control rods provides a relatively hard wear surface. The extent of the observed wear has appeared to be plant dependent and has in some cases extended completely through the tube wall.

The following table lists the Millstone-2 Cycle 4 fuel assemblies designated according to the supplying vendor and the design methods employed for mitigating guide tube wear.

Millstone-2 Cycle 4 Fuel Assemblies

Batch	Vendor	Design	Number Under CEAs	Total Number
B	CE	sleeved	1	1
D	CE	sleeved	32	56
D	CE	unsleeved	0	16
E	CE	sleeved	20	68
E	CE	reduced-flow	4	4
F	W	sleeved	12	68
F	W	inset	4	4

2.11.1 Combustion Engineering Supplied Fuel

Following a scheduled fuel assembly examination of the Millstone-2 core after Cycle 1 operation, NNECO and CE reported severe guide tube wear in some fuel assemblies (Reference u). As an interim fix, NNECO had CE installed stainless steel sleeves in nearly all fuel assembly guide tubes previously damaged or to be used in CEA positions.

Our review of the sleeving programs has been documented in previous safety evaluations (for example see the Millstone-2 Cycle 3 reload safety evaluation in Reference 11). Our prior safety evaluations concluded that guide tube sleeves will perform their function of reducing guide tube stresses to acceptably low values in worn assemblies and that sleeves are satisfactory for mitigating further fretting wear in irradiated or fresh fuel assemblies. Cycle 3 approval also permitted operation with four reduced-flow fuel assemblies. These fuel assemblies were placed in CEA positions and were the only Cycle 3 rodded assemblies which were unsleeved. The modifications that had been made to these four fuel assemblies consisted of decreasing the number and size of guide tube flow holes. CE out-of-pile flow tests (see Reference 11 for previous approval) have indicated that the resulting decrease in guide tube flow is accompanied by less CEA flow-induced vibration and, therefore, less guide tube wear. These reduced-flow fuel assemblies will remain in rodded positions during Cycle 4 operation.

In response to Agreement No. 2b of Reference 11, NNECO performed eddy current tests and visual inspections on previously rodded assemblies which were either sleeved or of the modified design during the Cycle 4 outage. The Cycle 4 outage surveillance revealed two anomalies. The first anomlaly was observed when a twice-burned CE fuel assembly, which had previously been in an instrument location but unsleeved, was sleeved during the Cycle 4 outage. Subsequent eddy current testing (ECT) on the center guide tube of this assembly revealed an axial crack about 3" long in the guide tube wall across the crimp area. For an unknown reason this crack occurred at a hoop strain far less than that which an early CE test in a hot cell had demonstrated was possible. Consequently this guide tube was recrimped. The second crimp was performed at a higher location which was supported by the top nozzle guide post rather than near the bottom of the guide tube.

The second anomaly was observed in a CE fuel assembly that had been rodded during Cycle 3. After removal of the CEA from this assembly, in preparation for fuel movement, it was observed that a guide tube sleeve was protruding about 5 to 6 inches above the guide tube nozzle. During Cycle 3, CE had informed NNECO that the ECT reading on this guide tube indicated a possibly poor crimp. Consequently, this guide tube sleeve was also recrimped and tested.

Except for these two anomalous assemblies, which we find have been adequately handled, the report on these examinations supports previous examination results and analyses that sleeves and reduced guide tube flow are acceptable methods of mitigating the consequences of guide tube wear (Reference 58). We, therefore, conclude that: (a) the guide tubes in the CE sleeved fuel assemblies will continue to meet their design functions; and (b) the guide tubes in the CE reduced-flow fuel assemblies should be acceptably resistant to wear. However, if future inspections after Cycle 4 operation reveals any failure to perform as extrapolated from Cycle 3 performance, the overall degradation to the core is restricted to a total of four fuel assemblies. Therefore, the use of the CE supplied fuel assemblies has been appropriately justified for continued operation.

2.11.2 Westinghouse Supplied Fuel

Sleeves are also used in the W supplied fuel assemblies to alleviate guide tube wear. The W sleeve design is similar to that of the CE design inasmuch as both designs are similarly dimensioned stainless-steel sleeves that are partially chrome plated and have series of slots and holes. (The chrome plate provides a bearing surface for CEA vibration and the slots and holes preclude coolant entrapment between the guide tube and the sleeve.) Major differences, however, do exist in the design of the upper end of the sleeves and the method of sleeve attachments.

On the CE sleeve design, the upper ends of the sleeves are conically shaped to fit the contour of the upper end fitting posts. Because the conical section is not connected to the post, free movement under heatup, cooldown, and differential irradiation growth exists between the guide tube and sleeve. The sleeves extend from the top of the upper end fitting posts to several inches below the area where the ends of the control rods reside when in the fully withdrawn position. The sleeves are securely fastened in place by mechanically "bulging" both the sleeve and the guide tube near the lower end of the sleeve.

The W sleeve design is completely cylindrical with no conically shaped end. And the mechanical attachment of the sleeve is accomplished by outwardly deforming the sleeve into two swage grooves, which are located in the top nozzle extension. For this W method of attachment, free movement of the sleeves is accommodated inversely to that of the CE method.

Of the 72 W supplied fuel assemblies to be used during Cycle 4, 68 assemblies will be sleeved and the remaining 4 will be demonstration assemblies. The demonstration assemblies are part of a longer-range effort to provide information on an alternate method of mitigating guide tube wear through the use of guide tube insets. Specifically, each guide tube in a demonstration assembly has two insets (i.e., rectangular deformations that locally reduce the original guide tube diameter) at two axial elevations in the upper end of each guide tube. Based on W out-of-pile vibration tests (Reference 56), it is expected that the insets will aid in centering the CEAs and reduce the amplitude of vibration, thus lessening the resulting wear to the guide tube wall.

It is our conclusion that: (a) the guide tubes in the W sleeved fuel assemblies will meet their intended design functions; and (b) the guide tubes in the W demonstration assemblies should be acceptably resistant to guide tube wear throughout Cycle 4. However, if future inspection after Cycle 4 operation reveals any failure to perform as predicted, the overall degradation to the core is restricted to a total of four fuel assemblies. Therefore, the use of the W supplied fuel assemblies has been appropriately justified for Cycle 4 operation.

2.11.3 Inspection Program

Because of the guide tube wear problem in Millstone-2 and other CE reactors, poolside inspections are being performed at each refueling outage. The surveillance program at the end of Cycle 4 should thus determine the adequacy of the W sleeve and insert design for use beyond Cycle 4 operation. As discussed in Reference 14, this surveillance program may include boroscopic examinations, also other inspection methods such as eddy current tests and mechanical pull testing may be required. NNECO has agreed to supply the specific details of the surveillance program for staff review at least 90 days prior to the Millstone-2 shutdown for the Cycle 5 reload outage.

2.12 Steam Generator Tube and Support Plate Inspections

References 28 and 32 provide steam generator tube test data and the inspection program to be completed during the Cycle 4 refueling outage in response to Agreement No. 2c of our Reference 11. More recently, NNECO has provided the preliminary results of this inspection program (Reference 60). They are:

(1) No tube defects or degradation exceeding 20% of tube-wall thickness were detected.

(2) One tube (line 85/row 91) was blocked to the 0.540 inch diameter probe at the tenth tube support plate on the hot-leg side of Steam Generator No. 2.

(3) The fraction of "egg-crate" tubes exhibiting a dent signal by eddy-current testing is increased from the previous inspection.

(4) The average dent size remained essentially unchanged for "egg-crate" tubes, approximately 1 mil, and exhibited slight increases for tube sheets and tube-support-plate regions, up to 1.5 mils. However, equipment accuracy is ± 2 mils.

As a result of the eddy-current inspection results described above, the tube blocked to the 0.540 inch probe was plugged. No other corrective actions were required.

In addition to the eddy-current inspection, NNECO performed a profilometer inspection of approximately 300 tubes and a visual examination of the secondary side were conducted. A preliminary evaluation of the profilometer results showed variable "denting" effects associated with the egg-crate supports, as indicated by a tube ovalization. This effect was smallest in the upper egg-crate evaluation. We conclude that data evaluated to date confirm the integrity of steam generator tubes and essential supports, and assure the continued applicability of current design basis analyses.

The visual inspections performed by NNECO confirmed to their satisfaction that the general condition remained essentially unchanged, as compared to the condition observed during the March 1979 Cycle 3 refueling outage.

Based upon the information provided, we find that the Millstone-2 steam generators are acceptable for Cycle 4 operation.

2.13 RCP Speed Sensing Proximity Probe and Transmitter

Agreement No. 3 of our Reference 11 letter was for performance of a multiple-frequency and multiaxis test in accordance with IEEE 344-1975 on the proximity probe and transmitter used in the RCS speed sensing system prior to the startup from the Cycle 4 refueling outage. NNECO has informed us that this testing was performed with satisfactory results (Reference 60). We, therefore, consider this issue resolved.

2.14 Reactor Cavity Neutron Shield Dose Reduction

NNECO agreed to provide an assessment of the neutron dose rate reduction and actual man-rem exposure savings experienced during Cycle 3 operation with the neutron shield installed in the area of the reactor cavity. The NNECO letter of November 9, 1979 provides this assessment in response to Agreement No. 4 of our Reference 11.

The results are reported in dose rate reduction factors such as:

Type of Radiation	Location	Dose Rate Reduction
neutron	38'6" level	25-150
	14'6" level	50-90
	-3'6" level	7-30
gamma	38'6" level	5-30
	14'6" level	2-15
	-3'6" level	1-8

The operating floor (38'6" level) measured reduction factors are slightly in excess of the factor of 40 which was the designed reduction factor. The total dose rate for areas of the containment which are occupied as required by operating personnel are now in general less than 100 mrem per hour, according to NNECO.

We find the Millstone-2 neutron shield installed during the Cycle 3 refueling outage has assisted NNECO in reducing their employees' radiation exposure in accordance with our as-low-as-reasonably-achieveable requirements.

2.15 Steam Generator Feedwater Piping

As a result of steam generator piping inspections required by I&E Bulletin No. 79-13, NNECO made pipe weld repairs to remove crack indications in the nozzle-to-pipe welds and the piping-to-first-elbow welds of both steam generators in November 1979. In Reference 30, NNECO committed to reinspect these same welds (AC-G-1, AC-G-2, BC-G-1 and BC-G-2) during the Cycle 4 refueling outage. NNECO has reported that radiographic inspection of these four welds has shown no cracking nor any other unacceptable code discontinuities.

We find the steam generator feedwater piping from the nozzle to the first support to be free from crack indications and, therefore, acceptable.

2.16 Replacement of Stem Mounted Limit Switches

Our Reference 11 Safety Evaluation, Section 2.15, documented NNECO's agreement to replace the stem mounted limit switches (SMLS) on valves SI-614, 624, 634 and 644 during the first unscheduled cold shutdown after September 15, 1979 (when replacement SMLS are estimated to be available). This action item resulted from I&E Bulletin 79-01. NNECO's letter of November 9, 1979 documents NNECO's replacement of these four SMLS with environmentally qualified switches.

2.17 RCS Vent Installation

One of the modifications to be made at all PWRs as a result of the Three Mile Island (TMI) accident is the installation of RCS vents. Guidance was provided on this Lessons Learned Item No. 2.1.9 in our letters of September 13 and October 30, 1979. Additional preliminary guidance has recently been given in our September 5, 1980 letter under Action Plan Item No. II.B.1.

NNECO provided their conceptual design in their letter of December 31, 1979. Since the RCS vents could only be installed during an outage, the licensee elected to install two vent manifolds to vent the domes of the reactor vessel and the pressurizer during the Cycle 4 reload. Because the operational procedures have not been developed by NNECO and the staff review is not completed, we find it necessary to review only the portions of the vent design dealing with inadvertent operation for the interim period until the entire vent review is completed.

NNECO states that the hardware modifications include the installation of two 0.612 ID (0.002 ft^2) vent manifolds, one located on the reactor vessel head and the other at the top of the pressurizer. Both manifolds are installed to existing penetrations of the reactor vessel and the pressurizer heads. The manifolds will discharge to a common sparger in the containment outer annulus adjacent to the Containment Air Recirculation units. Each manifold arrangement will consist of parallel redundant piping trains comprised of two solenoid operated globe valves per train (4 total) to provide RCS pressure boundary integrity. The first series valve in each train will provide the block valve function while the second valve will function as the vent valve.

Each valve will have remote-manual control capability from the control room with open and closed position indication. Power will be removed, remotely at the motor control center, from the valves during normal plant operation to preclude inadvertent operation of these valves.

In the case of a rupture of the reactor 0.002 ft^3 vessel head vent, NNECO finds that this hot leg break would result in peak clad temperature less than the 1971°F value calculated from the limiting cold leg break of 0.1 ft^3. There would be no core uncovery. NNECO also states that the rupture of the 0.002 ft^3 pressurizer vent is bounded by the analysis for the opening of a 0.0075 ft^2 power operating relief valve (PORV).

We find that since: (1) previously existing penetrations of the reactor vessel and pressurizer heads are utilized in this modification; (2) each manifold contains four valves in a parallel-series pair arrangement to insure operability and isolation ability; (3) all valves will be remotely disabled by removing the operating power during plant normal operation; and (4) since rupture of either manifold is bounded by a more serious accident scenario with acceptable consequences, this modification is acceptable for return to reactor operation until the entire vent system review is completed.

The review of the operating criteria and our other requirements for this system will be completed at a later time. We believe, however, that since this modification is completely installed at Millstone-2, the licensee should expedite the development and submittal for NRC review of operating procedures and TS. NNECO has agreed to do this.

2.18 <u>Containment Purge Valve Operability</u>

By application dated April 27, 1979, NNECO proposed to keep the containment purge valves locked closed in Modes 1 through 4. This is in response to our generic request of November 29, 1978. In our December 11, 1979 letter, we reiterated our concern regarding the design of the containment purge circuits, we requested NNECO to: (1) electrically disconnect and/or remove any bypass/override circuitry that does not satisfy our provided criteria; and (2) modify the basis for proposed Technical Specification (Section 3/4 6.1.7) to make explicitly clear that the purge isolation valves are required to be closed for two reasons - mechanical operability and electrical override considerations.

NNECO has notified us that the containment purge valves were locked closed and electrically disconnected during the Cycle 4 refueling outage and that the proposed TS change is necessary before restart. The necessary changes are: (1) to remove the containment purge valves from TS Table 3.6-3; (2) insert a new LCO 3.6.3.2 to require the containment purge valves be locked closed and electrically disconnected; (3) add a surveillance requirement to insure this status; and (4) clarify the basis as specified in our position.

Since these changes are in response to our requirements and insure containment in Modes 1 through 4, we find the modified proposed TS changes acceptable.

3.0 Technical Specification Changes

NNECO provided the proposed TS changes by Reference 55. As stated in earlier portions of this SE, the majority of the Cycle 4 analysis using W fuel is bounded by the Cycle 3 analysis where CE fuel was reloaded. This is as expected since 2/3 of the core remains CE fuel. The TS changes necessary are as follows.

3.1 Shutdown Margin

The shutdown margin was evaluated for a boron dilution event during the cold shutdown. It was concluded that a 2% $\Delta K/K$ shutdown margin is required so that at least 15 minutes would be available to the operator in order to terminate the deboration transient. We find this TS change to be acceptable. The pages affected are 1-7, 3/4 1-3, B 3/4 1-1, and B 3/4 1-3.

3.2 Axial Power Distribution

In the CE analysis for Cycle 3, a curve of axial power distribution values was used for the thermal margin safety limits. However, for the W Cycle 4 analysis, the axial power distribution methodology is utilized directly to produce the local power density-high trip setpoint. Therefore, TS Figure B2.1-1 is no longer required. We find this methodology acceptable.

The TS pages affected are 2-2 and B2-2.

3.3 RPS Trip Setpoint Limits

The reactor protection system (RPS) setpoints listed in TS Table 2.2-1 will be updated with the allowable values to include the maximum expected drift assumed to occur (between surveillance intervals) for each trip used in the safety analysis. The Cycle 4 power level-high setpoints are reduced slightly from the Cycle 3 values to meet the bounding criteria. The RCP speed sensing allowable value is reduced from 829 to 823 as a result of the Cycle 3 speed versus RCS flow data. NNECO states that the actual RCP speed sensing setpoint will remain at 845 rpm. We find these changes acceptable. The pages affected are 2-4, 2-5, B2-4, B3/4 7-1 and B3/4 7-2.

3.4 Thermal-Hydraulic

The staff has reviewed the Safety Limit Bases on pages B2-1 and B2-3 and the Limiting Safety System Settings Bases on pages B2-5, B2-6,and B2-8. These have been revised to change from the Cycle 3 method of analysis using the TORC thermal-hydraulic code and the CE-1 DNBR correlation to the acceptable Cycle 4 method of analysis using the THINC code and W-3 correlation. This results in an increase in the DNBR from 1.19 to 1.30. We conclude that these changes are acceptable.

3.5 Credit for Charging Pump Flow

In Reference 24, NNECO corrected the value of charging pump flow to be used in the small break LOCA analysis. We had approved one-half of one pump in Reference 11, however, the TS values were 44 gpm per pump. Since then inservice testing criteria have indicated that 41.4 gpm is the acceptable flow rate. The Cycle 4 analysis has used the value of 40 gpm (actually only 20 gpm is used as one-half is assumed to be lost). We find that this slight reduction is acceptable. The TS pages affected are 3/4 1-1, 3/4 1-3, 3/4 9-1, and 3/4 10-1.

3.6 Moderator Temperature Coefficient

The Cycle 4 MTC will be slightly more negative at -2.4×10^{-4} ΔK/K/°F at rated thermal power level. The Cycle 3 value was -2.2×10^{-4} ΔK/K/°F. We find this change supported by the safety analysis and, therefore, acceptable. The only TS page affected is 3/4 1-5.

3.7 Low Temperature Operation

TS Page 3/4 2-3 would be changed to allow low temperature operation during Cycle 4, with conditions as specified, for determining the MTC or for performing turbine generator efficiency testing. We find this change consistent with the approval given in Reference 31 and, therefore, acceptable.

3.8 Augmentation Factor

TS Figure 4.2-1 (page 3/4 2-5) will need revision to reflect the augmentation factor which applies to the W Batch F fuel. The proposed augmentation factors bound the CE fuel remaining in Cycle 4. We find this new curve acceptable.

3.9 ESF Component Leakage Outside Containment

As documented by Section 2.9 of this SE, the TS will need to specify the total ESF leakage value of 12 gph. Based on this SE, we find the proposed TS change acceptable. The TS pages to be changed are 3/4 5-5a, 3/4 6-13, B3/4 5-1 and B3/4 6-3.

3.10 Containment Purge Valves

In accordance with Section 3.18 of this SE, the TS will need to be modified to keep the containment purge valves locked closed and electrically disconnected in Modes 1 through 4. We find this modified proposed change acceptable. The TS pages to be changed are 3/4 6-18, 3/4 6-19, B3/4 6-2.

4.0 Physics Testing

The physics startup test program for Millstone-2 Cycle 3 was described in Reference 6. This entire program, including the tests, the acceptance criteria and the actions have been reviewed and approved by the NRC staff. The Cycle 4 startup test program will be identical to the program conducted for Cycle 3 with the exception of the power coefficient measurement. A revised test procedure for the power coefficient measurement may or may not be used during Cycle 4 startup testing. Since the power coefficient test is not mandatory, this is acceptable to the staff.

5.0 Environmental Consideration

We have determined that this 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.

6.0 Conclusion

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.

Date: October 6, 1980

7.0 References

1. W provides proprietary report on Millstone-2 fuel design testing, C. Eicheldinger to V. Stello, February 14, 1977.

2. NRC February 14, 1977 meeting summary on W design and testing program for Millstone-2, M. Mendonca to R. Baer, April 19, 1977.

3. NRC questions on asymmetric LOCA loads, V. Stello to all PWR licensees, January 25, 1978.

4. CE provides alternate ECCS Evaluation Model, A. Scherer to D. Ross, September 18, 1978.

5. NRC questions on asymmetric LOCA loads, B. Grimes to all PWR operating reactor licensees, January 16, 1979.

6. NNECO applies for Cycle 4 power increase, W. Counsil to R. Reid, February 12, 1979.

7. NNECO provides proprietary viewgraphs used in January 25, 1979 meeting on Millstone-2, Cycle 4 reload, W. Counsil to R. Reid, March 21, 1979.

8. NNECO provides non-QA version of small break LOCA analysis for Cycle 3, W. Counsil to R. Reid, March 22, 1979.

9. NNECO provides non-QA version of large break LOCA analysis for Cycle 3, W. Counsil to R. Reid, March 30, 1979.

10. Deleted

11. NRC issued Amendment 52 for Cycle 3 operation of Millstone-2, R. Reid to W. Counsil, May 12, 1979.

12. W submitted proprietary ECCS evaluation for reloads of CE NSSS (WCAP-9528), T. Anderson to J. STolz, June 11, 1979.

13. NNECO applies for ESF component leakage outside containment TS changes, W. Counsil to R. Reid, August 10, 1979.

14. NNECO provides proposed sleeve design to reduce CEA guide tube wear, W. Counsil to R. Reid, October 9, 1979.

15. W submittal of proprietary addendum to ECCS evaluation model for reloads of CE NSSS (addendum to WCAP-9528), T. Anderson to J. Stolz, October 15, 1979.

16. NRC questions on cladding swelling and rupture models for LOCA analysis, D. Eisenhut to all operating LWRs, November 9, 1979.

17. W provides information on non-conservatism discovered in 1978 ECCS evaluation model, T. Anderson to D. Eisenhut, November 16, 1979.

18. NRC memorandum on meeting on cladding rupture temperature, cladding strain and assembly flow blockage, R. Denise to R. Mattson, November 20, 1979.

19. NRC memorandum on potential deficiencies in ECCS evaluation models, H. Denton to Commissioners, November 26, 1979.

20. NNECO provides information on fuel cladding LOCA models, W. Counsil to D. Eisenhut, December 27, 1979.

21. NNECO provides information on fuel cladding LOCA models, W. Counsil to D. Eisenhut, February 8, 1980.

22. NNECO provides proposed Cycle 4 reload licensing schedule, W. Counsil to R. Reid, February 14, 1980.

23. CE Owner's Group provides information on asymmetric LOCA loads, A. Lundvall to H. Denton, February 15, 1980.

24. NNECO corrects the charging pump flow used in the small break LOCA analysis, W. Counsil to R. Reid, February 29, 1980.

25. NNECO submittal of proprietary Cycle 4 refueling basic safety report (BSR), W. Counsil to R. Reid, March 6, 1980.

26. NRC March 18, 1980 meeting summary on Millstone-2, Cycle 4 reload with W fuel, E. Conner to Docket 50-336, June 18, 1980.

27. NNECO provides extra copies of proprietary and non-proprietary versions of the BSR, W. Counsil to R. Reid, March 26, 1980.

28. NNECO provides preliminary steam generator tube eddy-curent test data and proposes inspection program, W. Counsil to R. Reid, March 31, 1980.

29. NNECO provides proposed CEA guide tube wear inspection program, W. Counsil to R. Reid, April 15, 1980.

30. NNECO provides proposed feedwater system piping inspection program, W. Counsil to R. Reid, April 24, 1980.

31. NRC issues Amendment 55 for low temperature testing, M. Fairtile to W. Counsil, April 29, 1980.

32. NNECO modifies the steam generator tube eddy-current inspection program, W. Counsil to R. Clark, May 7, 1980.

33. NNECO applies for Cycle 4 operation and requests increased maximum enrichment of fuel assemblies to be stored in the spent fuel pool, W. Counsil to R. Clark, May 9,1980.

34. NNECO provides proposed containment electrical penetration replacement program. W. Counsil to R. Clark, May 13, 1980.

35. NNECO submittal of a proprietary addendum to the BSR on nuclear uncertainties, W. Counsil to R. Clark, May 28, 1980.

36. NRC issues Amendment 58 authorizing an increase in the maximum enrichment of fuel to be stored in the spent fuel pool, R. Clark to W. Counsil, May 29, 1980.

37. NNECO submittal of the non-proprietary version of the addendum to the BSR on nuclear uncertainties, W. Counsil to R. Clark, May 30, 1980.

38. NNECO submittal of small break LOCA/ECCS performance results, W. Counsil to R. Clark, June 2, 1980.

39. NNECO submittal of the Cycle 4 refueling reload safety analysis (RSA), W. Counsil to R. Clark, June 3, 1980.

40. NRC June 4, 1980 meeting summary on Cycle 4 reload with W fuel, E. Conner to Docket 50-336, June 16, 1980.

41. NNECO submittal of large break LOCA/ECCS performance results, W. Counsil to R. Clark, June 11, 1980.

42. NRC questions on BSR fuel design and physics calculations, R. Clark to W. Counsil, June 20, 1980.

43. CE Owner's Group provides information on asymmetric LOCA loads, A. Lundvall to D. Eisenhut, June 30, 1980.

44. NNECO answers questions on fuel design and physics calculations, W. Counsil to R. Clark, July 7, 1980.

45. NNECO answers question 10 on power peaking in fuel pins, W. Counsil to R. Clark, July 22, 1980.

46. NNECO submittal on calculational error of guide tube length, W. Counsil to R. Clark, July 23, 1980.

47. CE Owner's Group provides information on asymmetrtic LOCA loads, A. Lundvall to D. Eisenhut, July 31, 1980.

48. NRC questions on BSR thermal-hydraulics and transient and accident analysis, and RSA reactor fuels and physics aspects, T. Novak to W. Counsil, August 6, 1980.

49. NNECO answers questions regarding the reference cycle transient and accident analyses, W. Counsil to R. Clark, August 7, 1980.

50. NNECO answers questions on the BSR thermal-hydraulics and transients and accident analysis, W. Counsil to R. Clark, August 7, 1980.

51. NNECO modifies the CEA guide tube wear inspection program, W. Counsil to R. Clark, August 14, 1980.

52. NNECO submittal of non-proprietary versions of their July 7 and 22, 1980 answers on the BSR, W. Counsil to R. Clark, August 14, 1980.

53. NNECO answers questions on BSR, RSA and LOCA performance results, W. Counsil to R. Clark, August 25, 1980.

54. NNECO response to verbal requests on design modifications to the fuel assemblies, W. Counsil to R. Clark, August 27, 1980.

55. NNECO submittal of proposed Technical Specifications for Cycle 4, W. Counsil to R. Clark, August 29, 1980.

56. NNECO provides information on mitigating guide tube wear, W. Counsil to R. Clark, September 18, 1980.

57. EG&G provides information on seismic and LOCA loading analysis of Millstone-2, J. Dearien to R. Tiller, September 19, 1980.

58. NNECO provides Cycle 4 outage guide tube wear surveillance, W. Fee to R. Clark, September 26, 1980.

59. NNECO applies for low temperature test during Cycle 5, W. Counsil to R. Clark, September 30, 1980.

60. NNECO provides information on electrical penetrations, guide tube wear, steam generator and feedwater line inspections, RCP speed sensing probe test, LOCA burst data and rod bowing analysis, W. Counsil to R. Clark, September 30, 1980.

8.0 Topical Reports

a. Docket Number 50-572, "RESAR-414, Reference Safety Analysis Report for the Westinghouse 3820 MWt NSSS," October 8, 1976.

b. CEND-187-A, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," March 1976.

c. WCAP-8377, "Revised Clad Flattening Model," July 1974.

d. WCAP-7982, "Fuel Densification Penalty," October 1972.

e. NRC issues safety evaluation of WCAP-8377, V. Stello to R. DeYoung, January 14, 1975.

f. WCAP-8691 Revision 1, "Fuel Rod Bow Evaluation," July 1979.

g. WCAP-8218, "Fuel Densification Experimental Results and Model for Reactor Application," October 1973.

h. USAEC Regulatory Staff Report, "Technical Report on Densification of Westinghouse PWR Fuel," May 14, 1976.

i. NUREG-0085, "The Analysis of Fuel Densification," July 1976.

j. WCAP-8720, "Improved Analytical Methods Used in Westinghouse Fuel Rod Design Computations," October 1976.

k. NRC issues safety evaluation on WCAP-8720, J. Stolz to T. Anderson, February 9, 1979.

l. WCAP-8720, Addendum 1, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations - Application for Transient Analysis," September 1979.

m. NRC November 1, 1979 meeting summary on cladding rupture temperature, cladding strain, and assembly flow blockage, R. Denise to R. Mattson, November 20, 1979.

n. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," April 1980.

o. NUREG-0371, "Task Action Plans for Generic Activities, Category A," November 1978.

p. NRC Federal Register Notice on Proposed Appendix to Section 4.2 of the Standard Review Plan (NUREG-75/087), J. Felton to C. Stephens, February 20, 1980.

q. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," March 1978.

r. WCAP-7015, Revision 1, "Subchannel Thermal Analysis of Rod Bundle Cores," January 1969.

s. WCAP-7838, "Application of the THINC Program to PWR Design," January 1972.

t. USAEC Report, "Boiling Crisis and Critical Heat Flux," 1972.

u. PNO-77-221, preliminary notification of event of unusual occurrence of guide tube wear, December 14, 1977.

v. CE provides information on guide tube wear, A. Scherer to V. Stello, December 23, 1977.

w. WYAPC provides information on guide tube wear, W. Johnson to V. Stello, February 14, 1978.

x. BG&E provides information on guide tube wear, A. Lundvall to V. Stello, Feburary 17, 1978.

y. Letter from J. F. Stolz, NRC, to T. M. Anderson, Westinghouse Electric Corporation, Review of WCAP-8720, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," dated March 27, 1980.

z. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980. Available from the NRC Division of Technical Information and Document Control.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 61 to Facility Operating License No. DPR-65, issued to Northeast Nuclear Energy Company, the Connecticut Light and Power Company, the Hartford Electric Light Company, and the Western Massachusetts Electric Company (the licensee), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2, (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to authorize Cycle 4 operation at 2700 MWt with a mixed core with one-third Westinghouse fuel and two-thirds Combustion Engineering fuel and modified guide tubes for the control element assemblies. The amendment also incorporates changes resulting from the analysis of the Cycle 4 reload with Westinghouse fuel, adds surveillance requirements for engineered safety features components leakage outside containment, allows continuation of low temperature operation for special tests, corrects the shutdown margin for Mode 5, and prevents containment purging in Modes 1, 2, 3 and 4.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations

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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 §51.5(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 applications for amendment dated August 10, 1979, May 9, August 29 and September 30, 1980, (2) Amendment No. 61 to License No. DPR-65, 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 Waterford Public Library, Rope Ferry Road, Waterford, Connecticut. 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 Licensing.

Dated at Bethesda, Maryland, this 6th day of October, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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Division of Licensing