DEC 03 1974

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Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 12 to Facility License No. DPR-28. This amendment includes Change No. 23 to the Technical Specifications and is in response to Vermont Yankee's request dated May 21, 1974, and supplements dated July 26 and August 23, 1974.

This amendment incorporates (1) a change in the limiting safety system settings of the total peaking factor for the reload core, (2) a deletion of the limiting safety system setting relative to the delay time for reactor scram upon actuation of the turbine control valve fast closure signal, (3) a change in the upscale trip setting on the rod block monitor, (4) a reduction in the allowable average scram insertion time for 90 percent insertion of all operable control rods and three fastest control rods of all groups of four control rods in a two by two array, (5) changes related to the effects of fuel densification in the 8 x 8 and 7 x 7 fuel assemblies on the linear heat generation rate (LHGR), (6) LHGR limits related to the Interim Acceptance Criteria (IAC) and ECCS modifications for the 8 x 8 fuel assemblies and related to ECCS modifications for the 7 x 7 fuel assemblies, and (7) deletion of the restriction for operation with 8 x 8 fuel.

Proposed changes to the Technical Specifications of Facility License No. DPR-28 were submitted with your evaluation of the reload core dated October 31, 1974, as required by Appendix K to 10 CFR Part 50 and the IAC for Emergency Core Cooling Systems for Light Water Power Reactors. Regulations require that the Vermont Yankee reactor be operated within the limits determined from both the IAC and Appendix K analysis until we have reviewed and issued approved Technical Specifications in accordance with an approved method of analysis for meeting Appendix K criteria.

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DATE➤						

 DEC 03 1974

Copies of the related Safety Evaluation, the Federal Register Notice, the Atomic Safety and Licensing Board's Order dated October 22, 1974, and the Atomic Safety and Licensing Appeal Board's Decision of November 27, 1974, also are enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Enclosures:
1. Amendment No. 12 w/Change No. 23
2. Safety Evaluation
3. Federal Register Notice
4. Order
5. Decision

cc w/enclosures:
Mr. James E. Griffin, President
Vermont Yankee Nuclear Power Corporation
77 Grove Street
Rutland, Vermont 05701

Mr. Donald E. Vandenburgh, Vice President
Vermont Yankee Nuclear Power Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

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Gregor I. McGregor, Esquire
Assistant Attorney General
Department of the Attorney General
State House, Room 370
Boston, Massachusetts 02133

Richard E. Ayres, Esquire
Natural Resources Defense Council
1710 N Street, N. W.
Washington, D. C. 20036

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RMDiggs
FDAnderson
DLZiemann
SKari
WOMiller
BScharf (15)
TJCarter
PCollins
SVarga
CHebron
RSchemel
ACRS (16)
HJMcAlduff, ORO
JRBuchanan, ORNL
TBAbernathy, DTIE
C. Miles

SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	L:ORB #2	L:ORB #2 DLZ	OGC	TR	L:AD/OR KRG	L:DD/RP
SURNAME	FAnderson:aw RMDiggs	DLZiemann		VStello	KRGoller	AGiambusso
DATE	11/ /74	11/ /74	11/ /74	11/ /74	12/2/74	12/3/74

Copies of the related Safety Evaluation, the Federal Register Notice, ~~and~~ the Atomic Safety and Licensing Board's Order dated October 22, 1974, also are enclosed.
and the Atomic Safety and Licensing Appeal Board's Decision of November 27, 1974

~~The changes to the Technical Specifications set forth herein should not be construed as authorizing operation of the Vermont Yankee reactor in a manner inconsistent with controlling decisions of the Atomic Safety and Licensing Appeal Board and the Commission.~~

delete

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Enclosures:
1. Amendment No. 12
   w/Change No. 23
2. Safety Evaluation
3. Federal Register Notice
4. Order

cc w/enclosures:
Mr. James E. Griffin, President
Vermont Yankee Nuclear Power Corporation
77 Grove Street
Rutland, Vermont 05701

Mr. Donald E. Vandenburgh, Vice President
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FDAnderson
DLZiemann
SKari
WOMiller
BScharf (15)
TJCarter
PCollins
SVarga
CHebron
RSchemel
ACRS (16)
HJMcAlduff, ORO
JRBuchanan, ORNL
TBAbernathy, DTIE

as changed (see yellow)

OFFICE ➤	L:ORB #2	L:ORB #2	L:ORB #2	OGC	L:OR	TR
SURNAME ➤	FDAnderson:aw	RMDiggs	DLZiemann	REK Kinsey	KRGoller	
DATE ➤	11/21/74	11/21/74	11/22/74	12/2/74	11/ /74	11/22/74

DEC 03 1974

cc w/enclosures:
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Chairman, Vermont Public
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Montpelier, Vermont 05602

John W. Stevens, Director
Conservation Society of Southern
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Mr. David M. Scott
Radiation Health Engineer
Agency of Human Services
Division of Occupational Health
P. O. Box 607
Barre, Vermont 05641

New England Coalition on Nuclear
Pollution
Hill and Dale Farm
West Hill - Faraway Road
Putney, Vermont 05346

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

Mr. Raymond H. Puffer
Chairman
Board of Selectman
Vernon, Vermont 05354

cc w/[illegible] filings dtd. [illegible] & 12/2/74.
Mr. Richard V. DeGrasse
State of Vermont
Public Service Board
7 School Street
Montpelier, Vermont 05602

Mr. Wallace Stickney
Environmental Protection Agency
JFK Federal Building
Boston, Massachusetts 02203

OFFICE ➤						
SURNAME ➤						
DATE ➤						

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. DPR-28

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated May 21, 1974, as supplemented by filings dated July 26 and August 23, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. Prior public notice of this amendment was given on June 28, 1974 (39 F.R. 24046), and an Atomic Safety and Licensing Board was appointed to rule on a petition seeking intervention. The petition was denied by the Atomic Safety and Licensing Board Order dated October 22, 1974. Although, on appeal, denial of intervention was vacated by the Atomic Safety and Licensing Appeal Board and remanded for further proceedings, the requested stay of facility operations was denied.

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DATE➤						

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Facility License No. DPR-28 is hereby amended by deleting Paragraph 3.F and by changing Paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 23.

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

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Change No. 23 to the
Technical Specifications

Date of Issuance:

DEC 03 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 12

CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

Delete pages 5, 6, 7, 8, 10, 14, 47, 72, 76, 96, 97 and 98 from the Appendix A Technical Specifications and insert the attached replacement pages bearing the same numbers. The changes on the revised pages are shown by a marginal line.

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DATE➤						

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is greater than 600 psig the reactor thermal power at any value of reactor core flow shall not exceed the safety limit shown in Figure 1.1.1.

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

The limiting safety system setting shall be as specified below:

A.1 The average power range monitor (APRM) flux scram setting shall be

$$S \leq 0.66\,W + 54,$$

where W is the percent of design driving loop flow. In the event of operation with a total peaking factor (T.P.F.) greater than the design value of 2.44 the setting shall be modified as follows:

23

$$S \leq \left[0.66\,W + 54\right]\left[\frac{2.44}{T.P.F.}\right]$$

1.1 SAFETY LIMIT

B. When the reactor pressure is less than 600 psig or reactor core flow is less than 5% of design, the reactor thermal power shall not exceed 269.2 MW(t)

C. 1. If an IRM or APRM scram condition exists for greater than 2.0 secs, a safety limit violation is assumed.

2.1 LIMITING SAFETY SYSTEM SETTING

where T.P.F. is the value of the actual total peaking factor.

This trip setting shall not exceed 90% of rated power within 30 seconds after initiation of a generator load rejection from initial generation powers of 164 MW(e) or more.

2. When the reactor mode switch is in the refuel or startup position, intermediate range monitor (IRM) scram shall be set at less than or equal to 15% of rated neutron flux. The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. The APRM rod block setting shall be

$$S_{RB} \leq 0.66\ W + 42,$$

where W is the percent of design driving loop flow. In the event of operation with a total peaking factor greater than the design value
23 of 2.44 the setting shall be modified as follows:

$$S_{RB} \leq [0.66\ W + 42] \left[\frac{2.44}{T.P.F.}\right]$$

where T.P.F. is the value of the actual total peaking factor.

C. Reactor low water level scram setting shall be at least 127 inches above the top of the active fuel.

2. When the process computer or another accurate time accounting device is unavailable, a safety limit violation is assumed if the scram condition exists and a control rod scram does not occur.

D. When the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the active fuel when it is seated in the core.

D. Reactor low low water level emergency core cooling system (ECCS) initiation shall be at least 82.5 inches above the top of the active fuel.

E. Turbine stop valve scram shall be less than or equal to 10% valve closure from full open.

F. Turbine control valve fast closure scram shall, when operating at greater than 30% of full power, trip upon actuation of the turbine control valve fast closure
23 relay.

G. Main steamline isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steamline low pressure initiation of main steamline isolation valve closure shall be at least 850 psig.

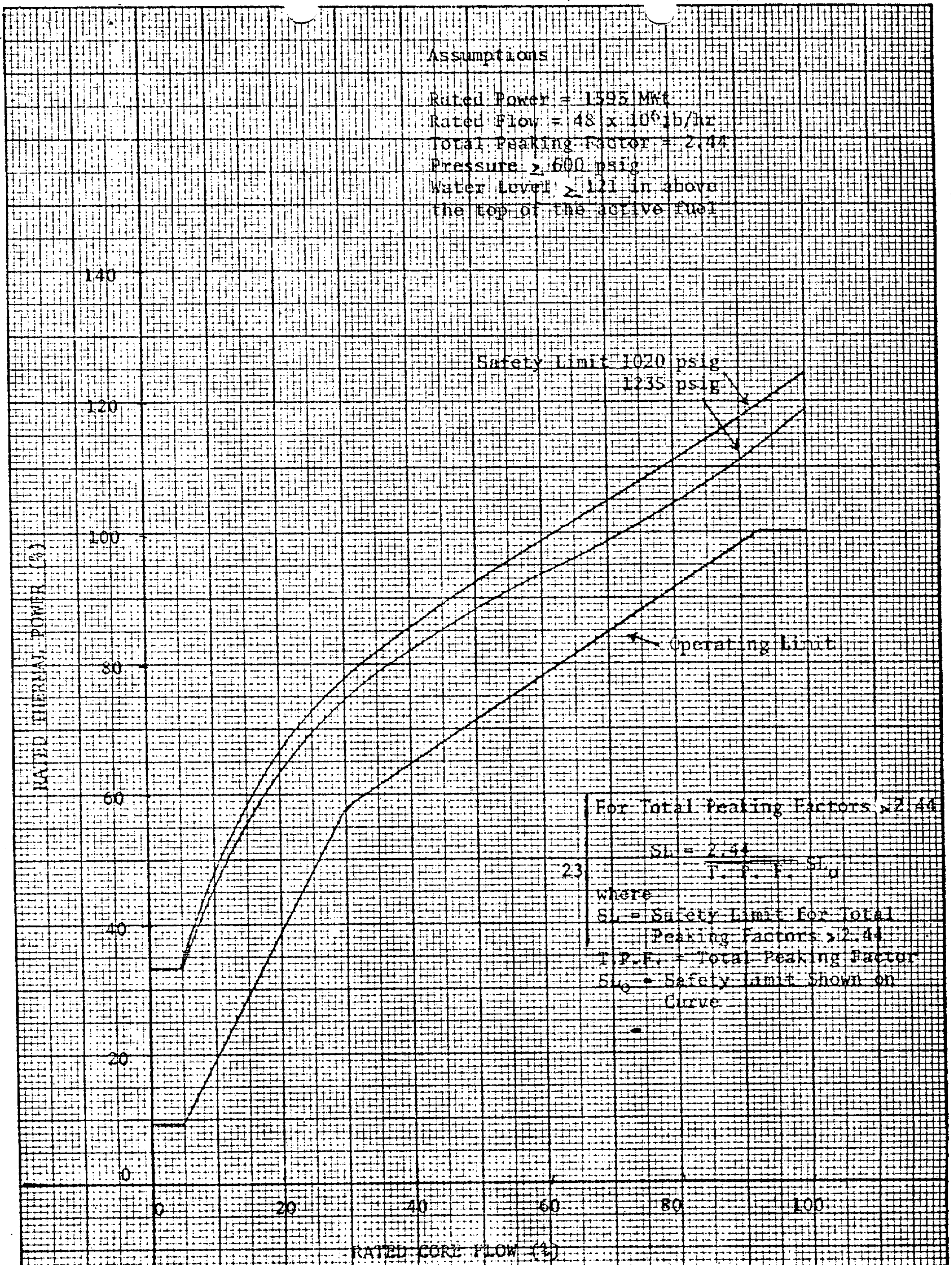

Figure 1.1.1 Fuel Cladding Integrity Safety Limit

was very conservatively drawn below all the available data. Since the correlation line was drawn below the data, there is a very high probability that operation at the calculated safety limit would not result in a critical heat flux occurrence. In addition, if a critical heat flux were to occur, clad performance would not necessarily be expected. Cladding temperature would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similiar in design to Vermont Yankee fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

Curves are presented for two different pressures in Figure 1.1.1. The upper curve is based on a nominal operating pressure of 1020 psig. The lower curve is based on a pressure of 1235 psig.

In no case is reactor pressure ever expected to exceed 1235 psig, and therefore, the curves will cover all operating conditions with interpolation. For pressure between 600 psig, which is the lowest pressure used in the critical heat flux data, and 1020 psig, the upper curve is applicable with increased margin.

The power shape assumed in the calculation of these curves was based on design limits and results in a
23 total peaking factor of 2.44 (1). For any peaking of smaller magnitude, the curves are conservative. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the Local Power Range Monitor System (LPRM). To maintain applicability of the safety limit curve, the safety limit will be lowered according to the correction factor given on Figure 1.1.1 in
23 the rare event of power operation with a total peaking factor in excess of 2.44.

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows which is 376°F for rated thermal power. For any lower feedwater temperature, subcooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt. This point is below the water level scram set point. As long as the water level is above this point the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased and therefore the core inlet enthalpy and subcooling would not be influenced. The values of the parameters involved in Figure 1.1.1 can be determined from information available in the main control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent power.

23 (1) NEDO-20558, Supplement 2 to Proposed Change No. 20, August 1974.

2.1 (cont'd)

transients result in violation of the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-referenced scram provides even additional margin.

The thermal hydraulic safety limit of Specification 1.1 was based on a total peaking factor of 2.44. A
23 correction factor has been included on Figure 1.1.1 to adjust the safety limit in the event the peaking
factor exceeds 2.44. Likewise, the scram setting should also be adjusted to assure MCHFR does not be-
come less than 1.0 in this degraded situation. If a peaking factor greater than 2.44 exists, the APRM
scram setting is adjusted downward by the equation given in the specification. The scram setting as
given by the equation will prevent MCHFR from becoming less than 1.0 for the worst expected transients.
If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done
as indicated in Specification 2.1.A.1.

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 33.8% of rated. The margin is adequate to accommodate anticipated maneuvers associated with station startup. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered the use of either one or two recirculation pumps.

B. APRM Control Rod Block Trips - Reactor power level may be varied by moving control rods or by varying
the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal
beyond a given point at constant recirculation flow rate, and thus to protect against the conditinn of
a MCHFR less than unity. As with the APRM scram setting, the APRM rod block setting is adjusted down-
23 ward if peaking factors greater than 2.44 exist. This assures a rod block will occur before MCHFR
becomes less than 1.0 even for this degraded case.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will prevent reactor operation with the steam separators uncovered, thus limiting carry-under to the recirculation loops. In addition, the safety limit is based on a water level below the scram point and therefore this setting is provided.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System (Note 1)	Trip Function	Refuel	Startup	Run	Trip Setting
23	Start up Range Monitor				
2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
2	b. Detector not Fully Inserted	X	X		
	Intermediate Range Monitor				
2	a. Upscale	X	X		$\leq$ 108/125 full scale
2	b. Downscale (Note 4)	X	X		$\geq$ 5/125 full scale
2	c. Detector not Fully Inserted	X	X		
	Average Power Range Monitor				
2	a. Upscale (Flow Bias)			X	$\leq$ 0.66W + 42 (Note 5)
2	b. Downscale			X	$\geq$ 2/125 full scale
	Rod Block Monitor (Note 6)				
23 1	a. Upscale (Flow Bias) (Note 7)			X	$\leq$ 0.66W + 41 (Note 5)
1	b. Downscale (Note 7)			X	$\geq$ 2/125 full scale
1	Trip System Logic	X	X	X	
1	Scram Discharge Volume	X	X	X	$\leq$ 12 gallons

3.3 LIMITING CONDITIONS FOR OPERATION

C. Scram Insertion Times

1. The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.00
23 | 90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
23 | 90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. Following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% of the control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the semiannual operating report to the AEC.

3.3 (cont'd)

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, can not occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 10% power the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. The control rod insertion and withdrawal sequences are established to assure that the maximum in sequence individual control rod or control rod segments which are withdrawn could not be worth enough to caused the core to be more than 0.013 ΔK supercritical if they were to drop out of the core in the manner defined for the rod drop accident. The rod drop accident that is applicable to Vermont Yankee is discussed in reference (1). The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified delta k limit on in-sequence control rod or control rod segment worths. Each core reload will be analyzed to show conformance to the limiting parameters.

23 | (1) NEDO-20558, Supplement 1 to Proposed Change No. 20, July 1974.

3.5 LIMITING CONDITIONS FOR OPERATION

J. Average Planar LHGR

23 During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

K. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left(1 - (\Delta P/P\ max)(L/LT)\right)$$

23 $LHGR_d$ = Design LHGR = 18.5 KW/ft for 7 x 7
= 13.4 KW/ft for 8 x 8

$\Delta P/P$ max = Maximum power spiking penalty = 0.026 for 7 x 7
= 0.021 for 8 x 8

LT = Total core length = 12 ft.

L = Axial position above bottom of the core

23 If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

4.5 SURVEILLANCE REQUIREMENTS

J. Average Planar LHGR

23 The APLHGR as a function of average planar exposure shall be checked daily during reactor operation at $\geq$ 25% of rated thermal power.

K. Local LHGR

23 The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% of rated thermal power.

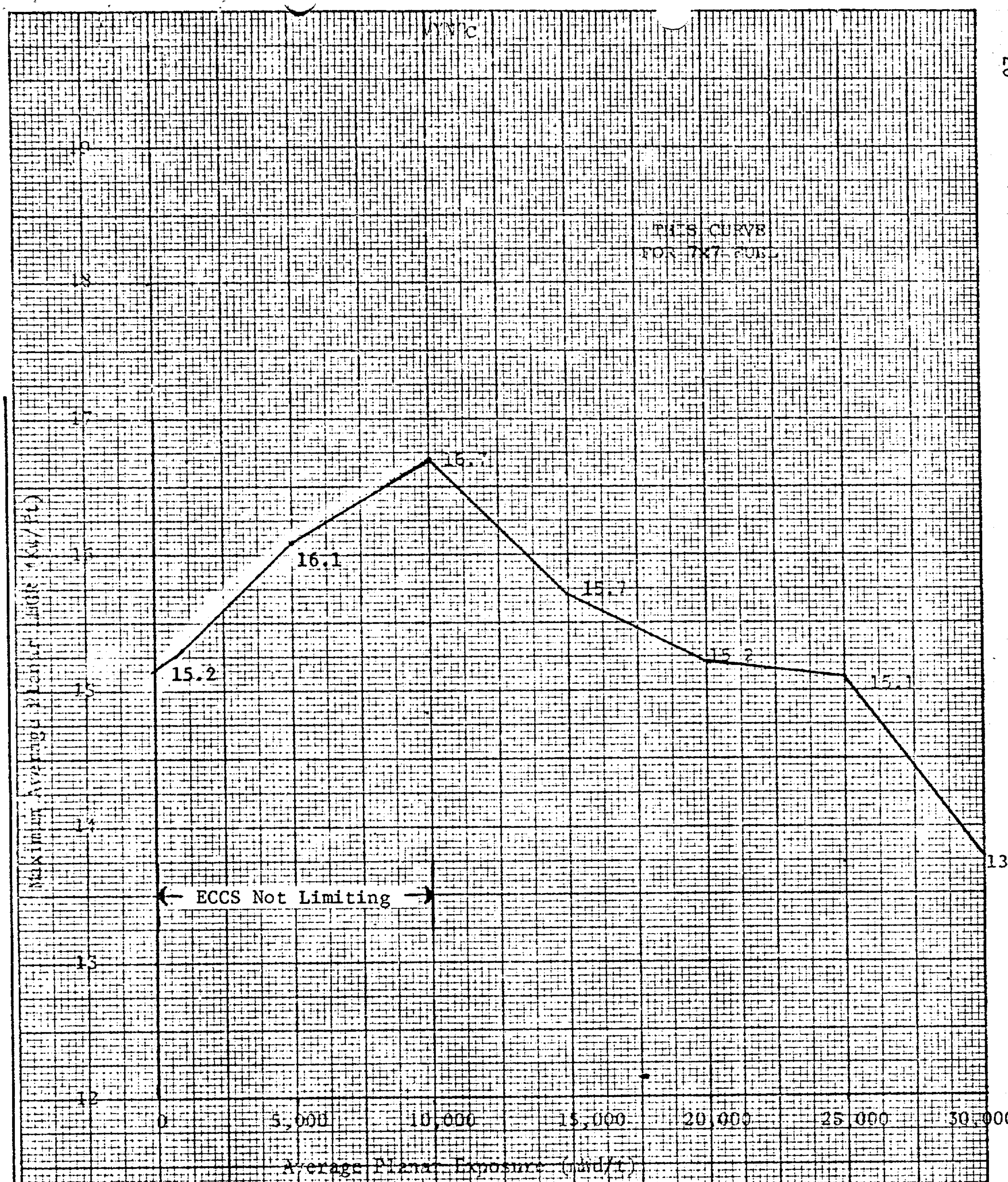

Figure 3.5.1.A Maximum Allowable Planar LHGR

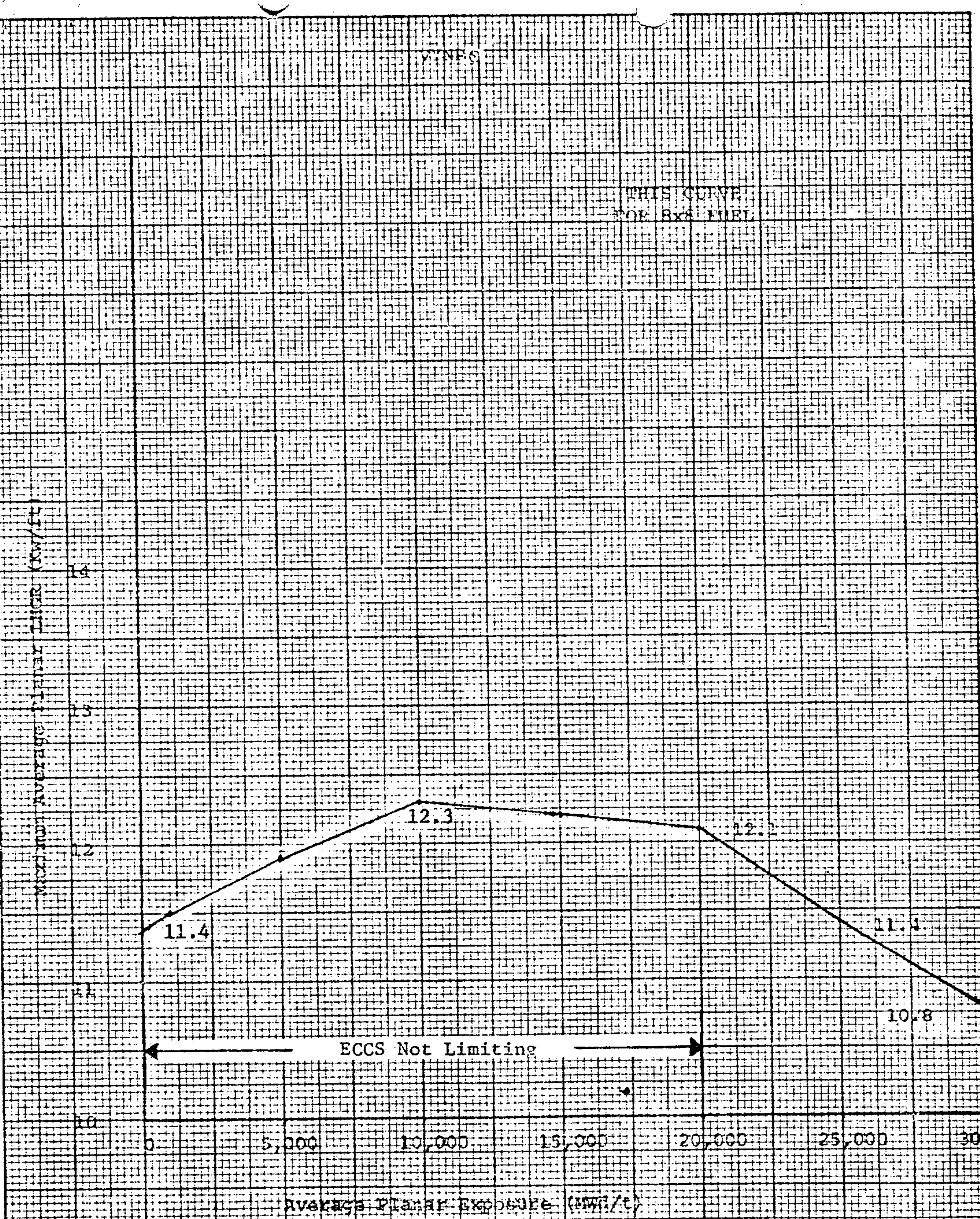

Figure 3.5.1.B Maximum Allowable Planar LHGR

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. DPR-28

(CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS)

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

By letter dated May 21, 1974, the Vermont Yankee Nuclear Power Corporation (VYNPC) submitted the General Electric Licensing Report NEDO-20103, "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring 1974." This report provides the technical evaluation on a BWR generic basis for Vermont Yankee Reload 2 fuel and constitutes the initial licensing submittal for use of this fuel in the Vermont Yankee Nuclear Power Station (VYNPS). Additional information providing specific evaluation of operating parameters relating to the use of this fuel in VYNPS was submitted by letter dated July 26, 1974, in the General Electric Licensing Report NEDO-20558, "Vermont Yankee Nuclear Power Station - Reload Application For 8 x 8 Fuel." Supplemental information providing errata sheets to NEDO-20558 and Sections on the Thermal Hydraulic Stability Analysis and the Transient and Core Dynamics for NEDO-20558 was submitted by letter dated August 23, 1974. Also included in the July 26, 1974 submittal of NEDO-20558 was proposed Technical Specifications which would permit VYNPC to load the Reload 2 fuel and operate VYNPS with the Reload 2 fuel. Some of these proposed Technical Specifications were revised by the submittal dated August 23, 1974, due to the information added to NEDO-20558.

The proposed changes to the Technical Specifications would:

1. Reduce the total peaking factor for the reload core;

2. Remove the delay time for reactor scram upon actuation of the turbine control valve fast closure relay signal;

3. Reduce the trip setting on the Rod Block Monitor - Upscale; and

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DATE ➤						

4. Reduce the allowable average scram insertion time for 90 percent insertion of all operable control rods and three fastest control rods of all groups of four control rods in a two by two array;

5. Change the linear heat generation rate (LHGR) limits related to the effects of fuel densification in the 8 x 8 and 7 x 7 fuel assemblies; and

6. Add a maximum average planar LHGR curve related to the IAC and ECCS modifications for the 8 x 8 fuel assemblies and change the maximum average planar LHGR curve related to the ECCS modifications for the 7 x 7 fuel assemblies.

The changes to the Technical Specifications associated with the evaluation of the loss-of-coolant accident for the reload core as required by Appendix K to 10 CFR Part 50 and the Interim Acceptance Criteria (IAC) for Emergency Core Cooling Systems for Light Water Power Reactors were submitted by VYNPC on October 31, 1974. Regulations require that the VYNPS be operated within the limits determined from both the IAC and Appendix K analysis until the AEC staff has reviewed and approved the method of analysis in accordance with Appendix K criteria.

The neutronic, thermal-hydraulic, and mechanical acceptability of the 8 x 8 fuel assembly design during normal operation, operational transients, and postulated accidents was evaluated by the Regulatory staff in a separate report(1). This staff report includes an evaluation of the safety of up to a full core loading of 8 x 8 fuel assemblies as compared with a core loading of 7 x 7 fuel assemblies. The use of 8 x 8 fuel for reload cores was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974.

EVALUATION

The 8 x 8 reload fuel has been designed to be compatible with and closely match the mechanical, nuclear, and thermal-hydraulic characteristics of the VYNPS initial core and the previous reload 7 x 7 fuel. The reference core is based on a pattern of reloading the 8 x 8 fuel as shown by Figure 1. No significant fuel loading asymmetries will exist. Only 40 Reload - 1 fuel assemblies (7 x 7 fuel) will remain in the core, as shown in Figure 1, with the 328 Reload - 2 fuel assemblies (8 x 8). The location of the remaining Reload - 1 fuel assemblies on the periphery of the core

(1) Technical Report on the General Electric Company 8 x 8 Fuel Assembly, February 5, 1974, Regulatory Staff, U. S. Atomic Energy Commission.

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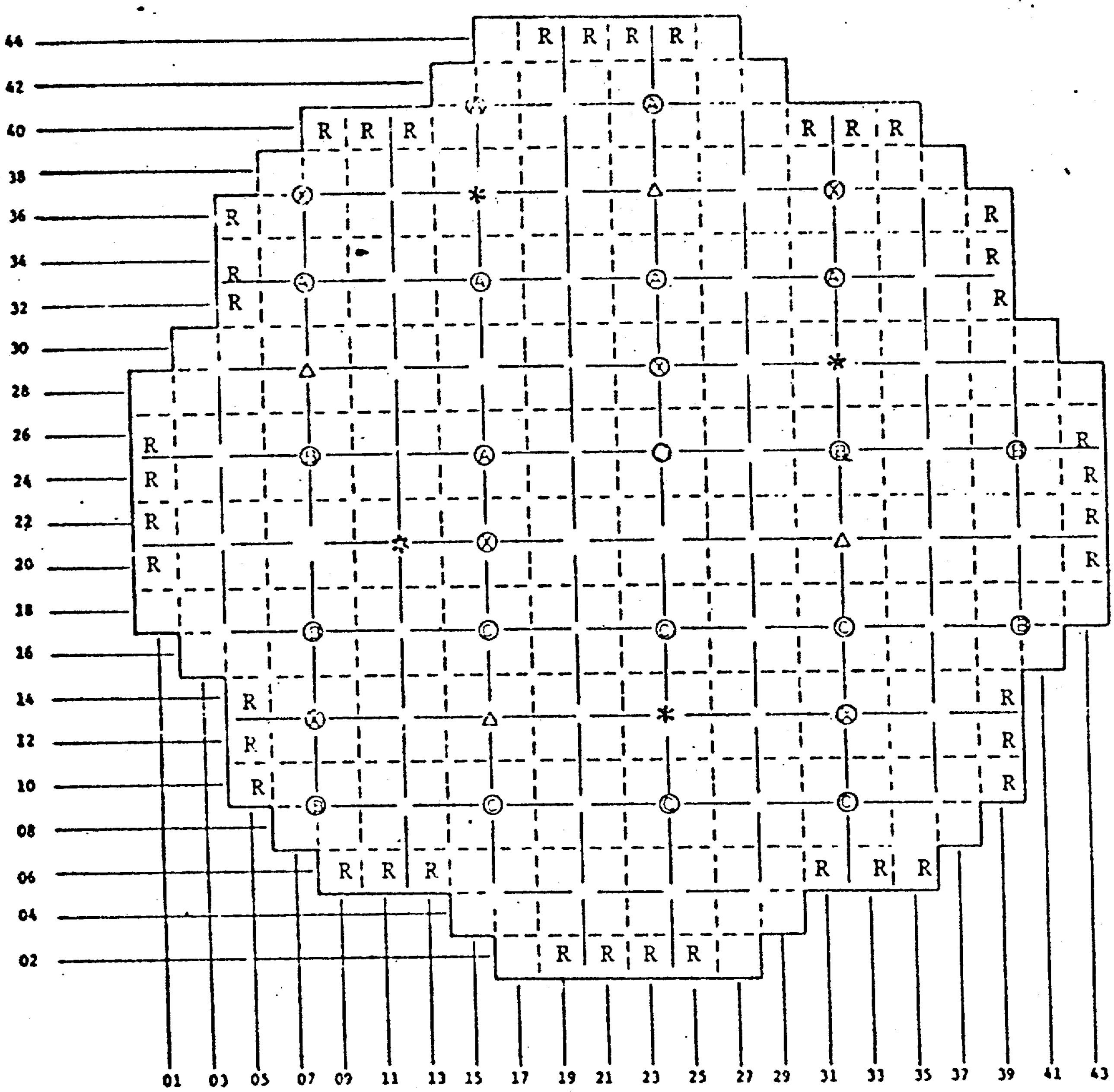

- LPRM LOCATION (COMMON LOCATION FOR ALL TIP MACHINES)
- LPRM LOCATION (LETTER INDICATES TIP MACHINE)
- IRM LOCATION
- SRM LOCATION
- * SOURCE LOCATION

R - 7x7 Bundle

BLANK = 8x8 Bundle

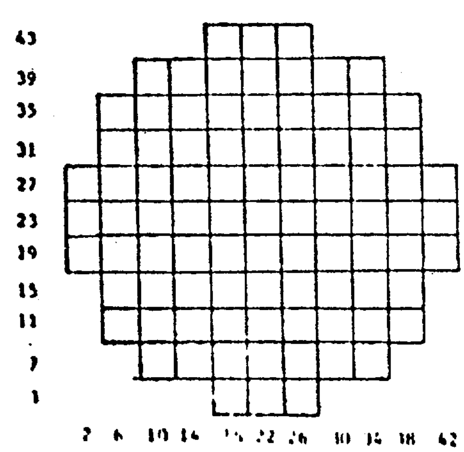

Figure 1 Reference Core Loading Pattern

assures that these fuel assemblies will operate at low power (significantly less than the design limit of 18.5 kw/ft) and therefore not restrict reactor operations. The thermal-hydraulic limiting conditions of operation and the response of the coolant circulation system is consistent with that used in our report[1]. The methods of analysis used by the licensee are identical to the methods approved by the staff. Therefore, the evaluations and conclusions of our report with respect to normal operations, abnormal operational transients, and accidents are fully applicable to VYNPS.

Our review[1] of the mechanical design of the 8 x 8 reload fuel concludes that the background of experience compiled by the General Electric Company is sufficient to enable GE to design fuel rods of new design with confidence in their durability. The VYNPS 8 x 8 fuel assemblies are of similar design and material as the 7 x 7 fuel assemblies which have successfully been operated at VYNPS. Both the 8 x 8 and 7 x 7 assemblies will operate at the same pressure and temperature and the fluid velocity and quality will be nearly identical and, therefore, the 8 x 8 fuel assemblies are expected to exhibit the same operational characteristics as the previously operated 7 x 7 assemblies.

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 assemblies using the same methods. The limiting accident loads results from a steam line break. The pressure differences following a steam line break are less than 10% greater than normal operating pressure differences. The loads following a steam line break are well below the allowable loads.

Based upon the above, we conclude that the mechanical design of the VYNPS 8 x 8 reload fuel is adequate to assure the mechanical integrity of the fuel assemblies. Additional assurance of acceptable fuel performance of the new fuel design is provided by the radiological surveillance maintained on the reactor primary coolant and off-gas to provide an early indication of incipient fuel failure caused by mechanical deterioration of the fuel assemblies.

We have also reviewed the nuclear characteristics of the 8 x 8 reload fuel. Based on our evaluation as reported[1] we conclude that a mixed 8 x 8 and 7 x 7 core will be similar neutronically to a 7 x 7 core and that the nuclear design is acceptable.

Our evaluation[1] of the expected thermal-hydraulic performance uses identical fuel damage limits and thermal-hydraulic criteria to evaluate both the 8 x 8 and 7 x 7 assemblies. The results of this evaluation show that the 8 x 8 assembly minimum critical heat flux ratio (MCHFR) is expected to be 11 percent greater than the MCHFR for a 7 x 7 assembly operating under similar conditions of flux peaking. Additionally, the 8 x 8 fuel assemblies operating at their

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design value provide 20% greater margin to the 1% cladding strain criteria than the 7 x 7 assemblies and the margin of design linear heat generation rate to pellet center line melting is 17% higher for 8 x 8 assemblies than for 7 x 7 assemblies. We have reviewed the thermal-hydraulic differences between the 7 x 7 and 8 x 8 assemblies involving a modified flow geometry and the introduction of an unfueled rod. The modified flow geometry will provide a more balanced subchannel flow in the 8 x 8 assembly than in the 7 x 7 bundle and, therefore, we conclude that the thermal performance is improved. The effect of the unheated rod has been previously reviewed[2] and we concluded that the effect of the unheated rod is not significant.

Based upon the above considerations we conclude that the thermal-hydraulic performance of the VYNPS 8 x 8 reload fuel is acceptable and will provide an increased margin of safety as compared with the previously operated 7 x 7 assemblies.

A. Proposed Changes to Technical Specifications

Although the performance characteristics of the 8 x 8 reload fuel are similar to previously authorized loadings, certain changes to the Technical Specifications are necessary to accommodate this fuel. These changes consist of:

1. Specifying a total peaking factor of 2.44 for the 8 x 8 fuel (as compared with a factor of 2.60 for the 7 x 7 fuel),

2. Reducing the delay time for reactor scram upon actuation of the turbine control valve fast closure from 300 milliseconds to zero,

3. Reducing the upscale set point of the Rod Block Monitor (RBM) from the present 108 percent to 107 percent of rated power,

4. Reducing the average scram time for 90 percent insertion of all operable control rods from the present 4 seconds to 3.5 seconds, and reducing the average scram time for 90 percent insertion of the three fastest control rods of all groups of four control rods in a two by two array from the present 4.25 seconds to 3.8 seconds,

(2) Change No. 17 for Oyster Creek, Docket No. 50-219, License DPR-16 Letter from D. J. Skovholt to Ivan Finfrock, Jersey Central Power and Light Company, dated November 16, 1973.

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5. Changing the LHGR limits related to the effects of fuel densification in the 8 x 8 and 7 x 7 fuel assemblies, and

6. Adding a maximum average planar LHGR curve related to the IAC and ECCS modifications for the 8 x 8 fuel assemblies and changing the maximum average planar LHGR curve related to the ECCS modifications for the 7 x 7 fuel assemblies.

1. The proposed change in total peaking factor (TPF) recognizes that different TPF are used for the 7 x 7 and 8 x 8 fuel assembly designs. The difference in TPF results from the change in the axial peaking factor of 1.40 for the 8 x 8 fuel design compared with 1.50 for the 7 x 7 fuel. Our report[1] notes the importance of the local peaking factor on nuclear design. However, the design local peaking factor does not change for the Vermont Yankee 8 x 8 fuel. The reduction of the TPF will result in a reduction of the linear heat generation rate from a maximum of 18.5 kw/ft for the 7 x 7 fueled core to a maximum of 13.4 kw/ft for the 8 x 8 fuel and 18.5 kw/ft for the 7 x 7 fuel in the reload core. On the basis of the staff report[1] and the above considerations, we conclude that the change in peaking factor is acceptable for VYNPS.

2. The proposed change in the delay time for reactor scram upon actuation of the turbine control valve fast closure would reduce the pressure transient resulting from generator trip or generator load rejection to less than that experienced from turbine trip. All turbine trip transients result in acceptable system pressures. and acceptable critical heat flux ratios of greater than 1.0. On the basis of these results, we conclude that the change in delay time is acceptable for VYNPS operation.

3. The RBM provides an operational guide and aid only and is not needed for rod withdrawal. The RBM trip function may be bypassed for reactor powers less than 30 percent of rated power. At present, the upscale trip set point of the RBM is specified to be 108 percent of the rated power which is the same as the APRM upscale trip set point. The proposed trip setting reduction to a more restrictive value of 107 percent provides some margin between the trip set points for the RBM and APRM from the same parameter (flow bias) and is therefore acceptable.

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4. The proposed changes in average scram times for 90 percent insertion of the control rods would reduce the allowable average scram time and result in less severe transients. Reactor operating experience has shown that actual measured control rod scram times are significantly less than the performance limits that had been previously established and that the average rod scram time for 90 percent insertion can be reduced from 4.00 to 3.50 seconds. Also, the time for the three fastest control rods of all groups of four control in two by two arrays can be reduced to "no greater than 3.80 seconds at 90 percent of the rod length inserted instead of 4.25 seconds". The reduced control rod scram times, based on reactor operating experience to date, are sufficiently longer than measured control rod scram times to allow for normal changes in control rod performance without reaching the technical specification limits. The original scram time limits were conservatively specified to allow for uncertainties related to control rod scram time deterioration. There are now sufficient control rod scram time measurements from operating reactors to reduce the allowance for uncertainties. Calculated scram reactivity is based in part on the technical specification scram time limits and using the new more stringent technical specification scram times specified, the calculated scram reactivity for the last half of rod insertion is somewhat faster and the magnitude of the power mismatch and pressure increase when the normal heat sink is unavailable is reduced; i.e., the calculated consequences of abnormal transients are less severe because control rods are assumed to scram at technical specification values. Therefore, the reduced scram times are acceptable for VYNPS operation.

5. The local LHGR limits have been changed to incorporate the effects of fuel densification on the operation of the 8 x 8 and 7 x 7 fuel assemblies. The LHGR design limits for the 8 x 8 fuel assemblies have been added to reflect the addition of the 8 x 8 fuel to the VYNPS core. Other changes relating to the surveillance of the local LHGR have been made for clarification of intent in respect to action to be taken if the limits should be exceeded and above what power levels the surveillance of local LHGR is required. At low power levels (at or below 25 percent rated), changes in power distribution due to fuel burnup or control rod motion are slow and daily surveillance is not required.

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6. The maximum average planar LHGR curve related to the IAC for the 8 x 8 fuel assemblies has been added in accordance with Paragraph 50.46 of 10 CFR Part 50 as an interim measure until we have reviewed and approved the method of analysis in accordance with Appendix K criteria. VYNPC has submitted the required ECCS analysis by letter dated October 31, 1974 which is being reviewed. The maximum average planar LHGR curve for the 7 x 7 fuel assemblies has been changed to take into account the modification of the ECCS approved by our letter dated November 1, 1974. Other changes relating to the surveillance of the average planar LHGR have been made for the same reason stated above for the local LHGR surveillance.

B. Abnormal Operational Transients

Abnormal operational transients were discussed in our report for 8 x 8 reload[1] and it was concluded that the reload fuel met the applicable criteria. As previously discussed, the mechanical, nuclear, and thermal-hydraulic characteristics of the 7 x 7 and 8 x 8 fuel are similar and will respond to transients similarly.

Our report[1] also concluded that the replacement of the 7 x 7 assemblies with 8 x 8 assemblies will not result in exceeding fuel damage limits during anticipated transients. The licensee has analyzed the events which have limiting MCHFRs including trip of recirculation pumps, a one pump seizure, a continuous withdrawal of a control rod, and misorientation of a reloaded fuel assembly. The results of these analyses show that the fuel damage limits, i.e., a MCHFR of unity and a cladding strain of one percent, are not reached during these transients. On the basis of the above, we have concluded that the VYNPS reactor, when reloaded with the proposed fuel and operated in accordance with the technical specifications, satisfies the fuel damage criteria for the abnormal operational transients.

C. Accident Analysis

The generic reevaluation of accidents to account for the effects of 8 x 8 fuel was discussed in our evaluation[2] and is applicable to VYNPS. That evaluation noted that the plant specific aspects of the review, such as compliance with the Interim Accpetance Criteria for Emergency Core Cooling, including the effects of densification, any necessary revisions to Technical Specifications requirements, and radiological consequences of postulated accidents would be addressed in a separate evaluation for the specific plant.

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We have reviewed the analysis of the loss-of-coolant accident (LOCA) on a generic basis and have concluded that the General Electric Evaluation Model (NEDO-10329), as modified by GE in NEDE-10801 to account for differences in geometry and subsequently modified by the staff to account for the effects of fuel densification, is applicable to the evaluation of the Emergency Core Cooling performance with 8 x 8 fuel assemblies in a General Electric boiling water reactor which has jet pumps. A separate analysis of the LOCA with proposed changes to the Technical Specifications for compliance with Appendix K of 10 CFR Part 50 has been submitted for our review by VYNPC dated October 31, 1974.

The radiological consequence of the postulated accidents is a function of the fission product release, including any change in fission product release because of the use of 8 x 8 fuel. The radiological consequences of a steam line break, fuel handling, control rod drop, and loss-of-coolant accidents were considered. As noted in our report on 8 x 8 fuel(1), the steam line break accident is almost entirely dependent on the limits placed on concentration of radioactivity in the primary coolant. These limits are not being modified and, therefore, the radiological consequences remain essentially unchanged. The resulting radiological doses will remain under 10 CFR Part 100 guidelines.

The fuel handling accident is dependent on the damage resulting from dropping an irradiated fuel element on other fuel elements. Since an 8 x 8 fuel assembly is the same size and approximately the same weight as a 7 x 7 assembly, it would impart the same energy to the same number of fuel assemblies as a dropped 7 x 7. Since the 8 x 8 fuel assembly design and fission product inventory are similar to the 7 x 7, the radiological consequences of dropping an assembly onto an 8 x 8 assembly will not be significantly different. The doses from a refueling accident are calculated to be less than 10 CFR Part 100 guidelines. Analyses of the control rod drop accident demonstrate that the dropping of an in-sequence control rod of maximum reactivity worth will not result in a peak fuel pellet enthalpy which exceeds the limit of 280 calories/gram. The number of 8 x 8 rods in the core which would perforate as a result of such an energy deposition is estimated to be higher than the number of 7 x 7 rods which would perforate as a result of a rod drop accident. However, the radiological consequences would be nearly the same because rod power is lower in the 8 x 8 fuel and the fission product inventory no greater than in a 7 x 7 assembly. The design basis loss-of-coolant accident doses are based on a conservatively large fission product inventory release which is independent of the number of perforations which would occur during a LOCA. Therefore, the radiological consequences of the design basis loss-of-coolant accident would also remain unchanged by the use of 8 x 8 fuel assemblies.

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CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

/s/

Fredric D. Anderson
Operating Reactors Branch #2
Directorate of Licensing

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: DEC 0 3 1974

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The Atomic Energy Commission (the Commission) issued on June 25, 1974, and published in the Federal Register on June 28, 1974 (39 F.R. 24046), a notice of consideration of proposed changes in the Technical Specifications of Facility Operating License No. DPR-28 issued to the Vermont Yankee Nuclear Power Corporation to permit operation of the Vermont Yankee Nuclear Power Station (located near Vernon, Vermont), using 8 x 8 fuel assemblies and to authorize changes in the limiting safety system settings and the limiting conditions for operation associated with the 8 x 8 fuel assemblies.

On July 29, 1974, the New England Coalition on Nuclear Pollution (NECNP) filed a timely petition for leave to intervene pursuant to 10 CFR 2.714 of the Commission's Rules of Practice. On October 22, 1974, The Atomic Safety and Licensing Board issued its Order Denying Petition Seeking Intervention by NECNP. On appeal by NECNP, the Atomic Safety and Licensing Appeal Board vacated the denial of intervention and remanded the matter to the Atomic Safety and Licensing Board for further proceedings. However, the Appeal Board denied NECNP's request for a stay of facility operations. Accordingly, the Commission has issued Amendment No. 12 incorporating Change No. 23 to the Technical Specifications of Facility Operating License No. DPR-28 to the Vermont Yankee Nuclear Power Corporation (the licensee). This change, effective immediately, authorizes operation

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of Vermont Yankee with the 8 x 8 fuel and changes the limiting safety system settings and the limiting conditions for operation associated with the 8 x 8 fuel assemblies.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated May 21, 1974 as supplemented on July 26 and August 23, 1974, (2) the Board's Order dated October 22, 1974, (3) Amendment No. 12 to License No. DPR-28, with Change No. 23, (4) the Commission's concurrently issued related Safety Evaluation, (5) the "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" dated February 5, 1974, (6) the Report of the Advisory Committee on Reactor Safeguards dated February 12, 1974, and (7) the Appeal Board's Decision (ALAB-245) of November 27, 1974. 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 Brooks Memorial Library at 224 Main Street, Brattleboro,

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Vermont 05301. A single copy of items (3) and (4) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 3rd day of December 1974

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

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UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

IN THE MATTER OF)
)
VERMONT YANKEE NUCLEAR POWER CORPORATION)
(Vermont Yankee Nuclear Power Station))

DOCKET NO. 50-271-OL
(Proposed Change to
Technical Specifications)

ORDER DENYING PETITION SEEKING INTERVENTION

This Atomic Safety and Licensing Board appointed to rule upon petitions to intervene in this proceeding has considered the one request which has been filed and which seeks intervention and an evidentiary hearing.

By notice given in the Federal Register on June 25, 1974, the Director of Regulation proposed to issue a modification of the Technical Specifications applicable to the Vermont Yankee Nuclear Power Corporation (Applicant) nuclear facility. The proposal would authorize principally a change to permit the loading of so-called 8 x 8 fuel rod assemblies rather than the presently authorized 7 x 7 arrangement. In the request for intervention by the New England Coalition on Nuclear Pollution (NECNP) is recognition that this change is desirable and will result in releases of lesser amounts of gaseous radionuclides. The contention for intervention is stated as follows:

> "With the improved 8 x 8 fuel and installation of an augmented off-gas system, Vermont Yankee is now able to perform at a far lower level of releases than previously.
>
> "... NECNP proposes that with the loading of the 8 x 8 fuel Vermont Yankee's Technical Specifications be amended

to lower gaseous releases to the levels to be reasonably expected from the 8 x 8 fuel in order to assure that the plant is operated at levels which are as low as practicable."

Both the Applicant and the Regulatory Staff oppose the NECNP petition upon several grounds, one of which is that the Technical Specifications sought to be changed would permit using 8 x 8 fuel assemblies (containing U-235) and also would revise the provisions in the Technical Specifications relating to the limiting conditions for operation associated with fuel densification for the 8 x 8 fuel assemblies. Applicant and the Staff emphasize that changes in the gaseous effluent specification are not sought. The argument made is that the selection of the fuel assembly specifications limits the review of the effect of the proposed change. One of the purposes here of course in changing the fuel assemblies is to reduce the level of radioactive gaseous effluents. The relationship of one proposed change in specifications to another is not to be disregarded if the <u>nexus</u> is reasonable. Applicant and the Staff contend that another procedure is available to one who seeks to have a Technical Specification changed and that that procedure should be adopted, although to do so would ignore the fact that the specification sought to be changed may be reasonably related to the specific one selected for public notice. The result of that contention would limit Atomic Safety and Licensing Boards and in effect require Boards to disregard obvious and known effects and interactions. This Board does not accept that contention for the reason that in the endeavor by the Commission to prevent "over-judicialization", the pleadings and issues presented for determination are not guided, like the original common law pleadings, solely by how carefully a selection has been made of a single subject for consideration. To this extent, the effect of a proposed change in one specification must be considered in connection with other reasonably related specifications.

That view, however, is not dispositive of the petition seeking intervention. The subject of gaseous releases from Applicant's facility has been reviewed by both the Atomic Safety and Licensing Board in its Initial Decision issued in February 1973 and in one of the Appeal Board Decisions issued in February 1974. Both decisions were predicated upon the Commission's regulation that an Applicant must

> "... make every reasonable effort to maintain radiation exposures ... as far below the limits specified in this part as practicable. The term ... means as low as is practicably achievable ...."

The Appeal Board cited 10 CFR 50.36a(b) in reference to whether the performance pursuant to the Technical Specifications "... will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in Par. 20.106. ..." The Appeal Board stated:

> "... average annual releases 'at small percentages' of Part 20 limits are achievable under normal operating conditions ... a radwaste system which can be shown to

> achieve 'small percentages' of Part 20 limits is <u>prima facie</u> in compliance with the regulations ... the 'small percentages' language ... cannot be equated with a fixed numerical standard which would effectively supersede the qualitative standards set forth in 10 CFR 20.1(c) and 50.34(a)."

To be considered also is the pendency of a proceeding in reference to proposed Appendix I to the Commission's regulations which would, if adopted, specify certain numerical standards. Until final action has been taken in reference to proposed Appendix I, this Board concludes that the small percentages ruling governs this case which, in some phases, has been in litigation for some time. The present record establishes that the proposed change in Technical Specifications to permit loading of the 8 x 8 fuel assemblies is an improvement by way of reduction in the level of radioactive gaseous effluents from the 7 x 7 assemblies, the limitation on which were found adequate and within and in compliance with the existing Commission regulations, and <u>a fortiori</u>, the effluent limitations for the 8 x 8 assemblies comply with Commission regulations.

Upon a consideration of the posture of this proceeding, and the rulings in the case, this Atomic Safety and Licensing Board concludes that adequate basis has not been established nor have valid contentions been presented to grant the petition to intervene in the proceeding.

WHEREFORE, IT IS ORDERED, in accordance with the Atomic Energy Act, as amended, and the Rules of Practice of the Commission, that the petition by the New England Coalition on Nuclear Pollution seeking intervention in this proceeding is denied.

ATOMIC SAFETY AND LICENSING BOARD
Established to Rule on Petitions

Samuel W. Jensch

By Samuel W. Jensch, Chairman

Issued:
October 22, 1974
Germantown, Maryland

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Michael C. Farrar, Chairman
Dr. John H. Buck, Member
Dr. Lawrence R. Quarles, Member

In the Matter of VERMONT YANKEE NUCLEAR POWER CORPORATION (Vermont Yankee Nuclear Power Station)	Docket No. 50-271

Mr. Anthony Z. Roisman, Washington, D. C., for the appellant, New England Coalition on Nuclear Pollution.

Messrs. John A. Ritsher and Thomas G. Dignan, Jr., Boston, Massachussetts, for the licensee, Vermont Yankee Nuclear Power Corporation.

Mr. David E. Kartalia for the AEC Regulatory Staff.

DECISION

November 27, 1974

(ALAB-245)

The lengthy proceeding involving the issuance of the basic operating license for the Vermont Yankee facility was recently concluded.[1/] We now have before us a

1/ In ALAB-179, we disposed of all but one of the issues in that proceeding. RAI-74-2 159, 185 (February 28, 1974). We later issued several more opinions dealing with the remaining "inerting" issue. Ultimately, the Commission determined that that issue should be resolved by rulemaking. Memorandum and Order dated November 7, 1974, RAI-74-11 ______.

sequel to that proceeding, involving a proposed amendment to the operating license.

At the licensee's request, the staff proposed to change the operating technical specifications to permit the use of different fuel assemblies. The New England Coalition on Nuclear Pollution (NECNP), which had participated in the operating license proceeding, sought to intervene. NECNP did not oppose the proposed change -- indeed, it recognized that the change is a beneficial one. Instead, it wished to intervene for the purpose of obtaining a hearing on an issue which it said is naturally interrelated with the staff proposal.

The Licensing Board denied intervention, holding that there was not a cognizable nexus between the staff's proposal and NECNP's contention. NECNP appealed. Additionally, NECNP requested that we stay the resumption of facility operations (the facility was shut down in anticipation of the fuel reloading).

It is not possible at this time to ascertain the extent of the relationship between NECNP's contention and the staff proposal. Such a determination turns on the outcome of a critical factual dispute, which we are unable to resolve on the record before us. Accordingly, we are vacating the denial of intervention and remanding the matter for the further proceedings described herein. There is, however, no warrant for precluding the resumption of operations with the new fuel. Thus, the request for a stay is being denied.

1. On June 28, 1974, the staff issued a notice of its proposal to amend the technical specifications to permit the use of different fuel assemblies. See 39 F.R. 24046. NECNP seeks to intervene and obtain a hearing, not on the staff proposal as such, but rather on what it claims is a related issue. Specifically, NECNP points out that the new fuel is expected to have a significantly lower leakage rate than the fuel which is being replaced. For that reason, argues NECNP, it should now be "practicable," within the meaning of the Commission's "as low as practicable" requirement,[2/] to reduce the level of routine radioactive emissions from the facility. Thus, NECNP says, the technical specifications limiting releases should be reduced accordingly. As may be seen, this argument hinges on NECNP's belief that the existing "as low as practicable" technical specifications made

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2/ See 10 C.F.R. 20.1(c); see also ALAB-179 (supra, fn. 1), RAI-74-2 at 164, fn. 13.

allowance for the use of fuel having a high leakage rate.

The staff and licensee oppose intervention on two grounds. In the first place, they argue, the opportunity for hearing existed only with respect to the precise technical specifications proposed to be changed by the staff (i.e., the specifications dealing directly with the fuel assembly). From this and the requirement that intervention be limited to the "subject matter of the proceeding"[3/] it follows, so their thesis goes, that NECNP cannot intervene in order to obtain a change in a different technical specification.

The second line of argument[4/] is that, in any event, the change in fuel should have no effect on the "as low as practicable" technical specifications. In explanation, the licensee claims that the existing specifications made no allowance for leaking fuel. If that is the case, it says, there would be no need to make an adjustment to take account of the change in fuel.

2. In an order issued on October 22, 1974,[5/] the Licensing Board established to rule on the intervention

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3/ See 10 C.F.R. 2.714(a).

4/ The staff's papers reveal that, while the first argument was stressed below, primary reliance is now being placed on the second argument.

5/ RAI-74-10 715.

petition rejected the first line of argument, refusing to hold that the grounds for intervention are limited to those which involve a challenge to the proposed amendment itself. It held, in effect, that intervention could also properly be founded on a claim that the adoption of a proposed amendment would warrant a change in a related technical specification. The crucial question, the Board opined, was whether a sufficient nexus existed between the change proposed by the staff and the allegedly related changes proposed by the potential intervenors. The Board answered that question by holding that the nexus existing here could not justify the granting of intervention.

3. We are in basic agreement with the Board's general approach to the novel question before it. We adopt its holding that the right to intervene is not limited to those who oppose a proposed change itself, but instead extends to those who raise related claims involving matters arising directly from the proposed change. We can discern no reason for denying to those who are affected by a proposal the opportunity to be heard on contentions which arise as a direct consequence or necessary implication of the proposal.

4. Ordinarily, that holding would not avail NECNP. In most circumstances, we would reject a claim that the technical specifications governing releases are related to, and should be altered to take account of, a proposal to use new fuel. For generally no such relationship exists and no alteration is appropriate.

We reach this conclusion for the following reasons. The Commission's "as low as practicable" regulations require that a number of factors, including any hazard to public health, be considered in ascertaining the appropriate limitations on emissions. Some degree of fuel leakage must always be anticipated.[6/] Of necessity, then, a factor representing fuel leakage must be considered in determining what is "as low as practicable." The only sensible approach consistent with the purpose of the regulations is to take into account the reasonable, general expectation for the rate of fuel leakage. In our

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6/ The record of the operating license hearing reveals that this is the case. See e.g., one Board member's summary of the evidence on this point (Tr. 1829) and the Final Environmental Statement, §III.D.2.a, p. III-23. (See also paragraph 3 of the June 1, 1973 Grube affidavit, relating to the predicted 1% failure rate due to hydriding.) Other materials, not in the record, confirm that this approach is followed generally. See "The Safety of Nuclear Power Reactors * * *," WASH-1250, §4.1.2, indicating that the typical levels of activity in BWR coolant are the equivalent of the levels resulting from the continuous release of the activity in the fuel-to-cladding gap in 0.1-0.2% of the rods (with 0.5% representing the maximum experienced).

opinion, the limitations thus established should not, in the absence of extraordinary circumstances, be varied to take account either of fuel which proves to have an excessively high leakage rate or of fuel which proves to be of exceptionally good quality.

This being the case, neither a change in fuel, nor a discovery that existing fuel rods are performing either well above or well below expectations, should provide a basis for a change in the applicable limitations on releases.[7/] Accordingly, even though the new fuel proposed to be used here may prove to have a significantly lower leakage rate than the fuel being replaced, we would ordinarily be able to say that there is no connection between the staff proposal to change the fuel assemblies and NECNP's contention that the technical specifications governing release rates should be reduced.[8/]

5. This case, however, is not so simple. For it may be -- and we cannot tell on the record before us --

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7/ On the other hand, information developed over the long term as to what can be reasonably expected in terms of fuel performance could affect what is "practicable" to achieve. In other words, what is relevant is the standard expected to be met, rather than the actual performance of one lot of fuel.

8/ Similarly, if the fuel developed excessively high leakage, the limitations on releases should not be raised to accommodate the leakage; rather, operation would have to be adjusted to meet the established limits.

that the existing release limits made allowance (albeit improperly) for the high leakage rate of the existing fuel. If that is so, it would establish the necessary connection between NECNP's contention and the proposal to change fuel.

The relevant facts are as follows. In the operating license proceeding, we rejected NECNP's claim that the expected releases of liquid and gaseous emissions of 1 and 2% of Part 20 limits, respectively, were impermissible. ALAB-179 (*supra*, fn. 1), RAI-74-2 at 164-167.[9/] In so ruling, we declined to express an opinion on the by-then-moot question as to the validity of a technical specification which had governed interim operations and which had permitted gaseous releases of up to 10% of Part 20 limits. RAI-74-2 at 167.

Contrary to the belief of the intervention board, those rulings do not dispose of the matter. For on April 10, 1974, without notice of opportunity for hearing,[10/]

9/ In that connection, the staff and applicant had indicated at the hearing that the interim limit on gaseous releases was expected to be reduced to the 2% level after a new off-gas system was installed.

10/ The staff justified the absence of notice on its determination that no "significant hazards consideration" was involved. We have previously expressed a view as to whether such a determination could be justified in certain circumstances (see ALAB-167, RAI-74-12 1151, 1153). We have some doubt as to its legitimacy here. In any event, NECNP was afforded no opportunity to challenge the revised technical specifications prior to their becoming effective.

the staff issued new technical specifications which set limits less stringent than the 1% and 2% levels on which we had ruled just six weeks earlier. Nothing in those revised technical specifications, the papers which accompanied them, or the papers before us, indicates the justification for adopting the less-restrictive limitations. While there might well have been other grounds for that action (such as a failure of the augmented off-gas system to perform according to expectations), the possiblity exists that the limits were raised to take account of, and make allowance for, the excessively leaking fuel then being utilized.

As should be clear from what we have said in Section 4 above, if the staff action was based on such a consideration, it was quite likely improper. In any event, if it was so based, it would provide the needed nexus between NECNP's contention and the proposal to change the fuel.

The record does not provide any information on the basis for the staff's action in April. But NECNP's right to intervene depends on what was done then. Accordingly, the denial of intervention must be vacated and the matter remanded so that the Board can obtain, for the record, an explanation as to what factors played a part in the April, 1974 decision to adopt less stringent limitations on emissions than we had been led to believe would be in effect. In particular, the staff must indicate whether the excessively high leakage rate in the existing fuel was a factor in that decision. If it was, intervention must be granted; if the Board becomes satisfied that it was not, intervention should be denied.

We might add that we recognize that the staff has continuing regulatory responsibility during a reactor's lifetime, and that changed circumstances may call for action different from that decided on at a hearing. At the same time, however, we believe that due regard for the adjudicatory system established by the Commission requires that, whenever the staff takes action that may be inconsistent with the result reached in the adjudicatory process, it set out fully the changed circumstances which it believes justifies that action.11/ It should have done so here; if it had, this matter could have been more readily resolved.

6. While it thus remains to be seen whether NECNP's contention should be admitted into controversy, and, if so, whether the existing limitations on radioactive emissions ought to be reduced, there is no reason now apparent to us why operation of the reactor with the new fuel should

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11/ See the Commission's Memorandum and Order of October 3, 1974, in Northern Indiana Public Service Co. (Bailly Nuclear-1), RAI-74-10 631-32.

be precluded pending the determination of those questions.[12/] There is no opposition to the underlying proposal to use the new fuel assemblies, and there is no claim that their use will create any increased risk to the public. The claim is only that the "as low as practicable" limits should be reexamined. The reactor has been operating for some time under the existing "as low as practicable" specifications. Releases with the new fuel will not be permitted to be in excess of those same specifications; if the new fuel performs as expected, actual releases will be lower than those which have occurred previously.

During a reactor's operating lifetime, developments may occur over the long term which could affect earlier beliefs as to what is "as low as practicable." Some developments may warrant investigation to determine

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12/ NECNP asked for a stay only during the pendency of the appeal. No stay was needed since we were able to decide the appeal prior to the scheduled resumption of operation. NECNP appeared to assume, however, that if we permitted intervention, no resumption of operations would be permitted until the entire proceeding was concluded. To avoid further debate and appeals on that score, we are expressing our view at this time.

whether emission limits should be reduced. But, in the absence of any new evidence that harm is being caused by releases at the level of the existing limits, operation should continue while such an investigation proceeds. There is no reason why resumption of Vermont Yankee's operations should be precluded by the pendency of this proceeding.

The denial of intervention is vacated and the matter is remanded to the intervention board for further proceedings not inconsistent with this opinion. The request for a stay of operations is denied.

It is so ORDERED.

FOR THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

Romayne M. Skrutski

Romayne M. Skrutski
Secretary to the Appeal Board

Concurring opinion of Mr. Farrar:

In my view, we are not required to reach the question dealt with in Section 4 of the foregoing opinion. As is clear from the remainder of the opinion, at this stage of this proceeding it matters not whether there would ordinarily be a cognizable relationship between a change of the type proposed by the staff and a contention such as NECNP's. For, owing to the uncertainty concerning the nature of the factors which led the staff to adopt the existing specifications after we had ruled on a different proposal, the denial of intervention here must be vacated in any event.

In short, the discussion in Section 4 is unnecessary to our decision. This being so, and also because the matter was not addressed directly in the parties' briefs, I would leave to another day any decision as to what factors may affect the ordinary "as low as practicable" determination. Although my own preliminary analysis of the subject would lead me in the majority's general direction, it seems to me that we should await committing ourselves until such time as the issue need be squarely faced and there has been a greater focus upon it by the litigants who may be then before us.

In all other respects, I am in agreement with the views expressed in the opinion.

Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

Pursuant to the Atomic Safety and Licensing Board's Order dated October 22, 1974, the Commission has issued the enclosed Amendment No. 12 to Facility License No. DPR-28. This amendment includes Change No. 23 to the Technical Specifications and is in response to Vermont Yankee's request dated May 21, 1974, and supplements dated July 26, and August 23, 1974.

This amendment incorporates: (1) a change in the limiting safety system settings of the total peaking factor for the reload core, (2) a deletion of the limiting safety system setting relative to the delay time for reactor scram upon actuation of the turbine control valve fast closure signal, (3) a change in the upscale trip setting on the rod block monitor, (4) a reduction in the allowable average scram insertion time for 90 percent insertion of all operable control rods and three fastest control rods of all groups of four control rods in a two by two array, and (7) deletion of the restriction for operation with 8 x 8 fuel.

Copies of the related Safety Evaluation, the Federal Register Notice, and the Atomic Safety and Licensing Board's Order dated October 22, 1974, also are enclosed. – last ¶

Proposed changes to the Technical Specifications of Facility License No. DPR-28 were submitted with your evaluation of the reload core dated October 31, 1974, as required by Appendix K to 10 CFR Part 50

on the linear heat generation rate (LHGR)

(5) changes related to ~~the effects of fuel densification in the 8x8 and 7x7 fuel assemblies~~, (6) LHGR limits related to the Interim Acceptance Criteria (IAC) and ECCS modifications for the 8x8 fuel assemblies and related to ECCS modifications for the 7x7 fuel assemblies

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and the IAC ~~Interim Acceptance Criteria (IAC)~~ for Emergency Core Cooling Systems for Light Water Power Reactors. Regulations require that the Vermont Yankee reactor be operated within the limits determined from both the IAC and Appendix K analysis until we have reviewed and issued approved Technical Specifications in accordance with an approved method of analysis for meeting Appendix K criteria.

Sincerely,

Last ¶ on other page

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Enclosures:
1. Amendment No. 12
   w/Change No. 23
2. Safety Evaluation
3. Federal Register Notice
4. Order

cc w/enclosures:
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Mr. Donald E. Vandenburgh, Vice President
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DATE ➤			11/ /74	11/ /74	11/ /74	11/ /74

ATTACHMENT TO LICENSE AMENDMENT NO. 12

CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

Delete pages 5, 6, 7, 8, 10, 14, 47, 72, ~~and~~ 76, 96, 97 and 98 from the Appendix A Technical Specifications and insert the attached replacement pages bearing the same numbers. The changes on the revised pages are shown by a marginal line.

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The Atomic Energy Commission (the Commission) issued on June 25, 1974, and published in the Federal Register on June 28, 1974 (39 F.R. 24046), a notice of consideration of proposed changes in the Technical Specifications of Facility Operating License No. DPR-28 issued to the Vermont Yankee Nuclear Power Corporation to permit operation of the Vermont Yankee Nuclear Power Station (located near Vernon, Vermont), using 8 x 8 fuel assemblies and to authorize changes in the limiting safety system settings and the limiting conditions for operation associated with the 8 x 8 fuel assemblies.

On July 29, 1974, the New England Coalition on Nuclear Pollution (NECNP) filed a timely petition for leave to intervene pursuant to 10 CFR 2.714 of the Commission's Rules of Practice. On October 22, 1974, The Atomic Safety and Licensing Board issued its Order Denying Petition Seeking Intervention by NECNP. Accordingly, the Commission has issued Amendment No. 12 incorporating Change No. 23 to the Technical Specifications of Facility Operating License No. DPR-28 to the Vermont Yankee Nuclear Power Corporation (the licensee). This change, effective immediately, authorizes operation of Vermont Yankee with the 8 x 8 fuel and changes the limiting safety system settings and the limiting conditions for

On appeal by NECNP, the Atomic Safety and Licensing Appeal Board vacated the denial of intervention and remanded the matter to the Atomic Safety and Licensing Board for further proceedings. However, the Appeal Board denied NECNP's request for a stay of facility operations.

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operation associated with the 8 x 8 fuel assemblies.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated May 21, 1974 as supplemented on July 26 and August 23, 1974, (2) the Board's Order dated October 22, 1974, (3) Amendment No. 12 to License No. DPR-28, with Change No. 23, (4) the Commission's concurrently issued related Safety Evaluation, (5) the "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" dated February 5, 1974, ~~and~~ (6) the Report of the Advisory Committee on Reactor Safeguards dated February 12, 1974 and (7) the Appeal Board's Decision (ALAB-245) of November 27, 1974. 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 Brooks Memorial Library at 224 Main Street, Brattleboro, Vermont 05301. A single copy of items (3) and (4) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this

FOR THE ATOMIC ENERGY COMMISSION

OFFICE ➤			Dennis L. Ziemann, Chief		
SURNAME ➤			Operating Reactors Branch #2		
DATE ➤			Directorate of Licensing		