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

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Am-18 DPR
Ch-29

Docket No. 50-271

NOV 12 1975

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 18 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment includes Change No. 29 to the Technical Specifications, and is in response to your requests dated May 28, 1975 and July 30, 1975, as supplemented September 15 and 22, 1975.

The amendment revises the provisions in the facility Technical Specifications to permit operation of the facility (1) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (2) using modified operating limits based upon an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR § 50.46. The amendment modifies various limits established in accordance with the Commission's Interim Acceptance Criteria, and with respect to Vermont Yankee, terminates the further restrictions imposed by the Commission's December 27, 1974 Order For Modification of License, and imposes instead, limitations established in accordance with the Commission's Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR § 50.46.

We have evaluated the potential for environmental impact associated with operation of the facility in the proposed manner. From this evaluation, we have determined that there will be no change in effluent types or total amounts, no increase in authorized power level, and no significant environmental impact attributable to the proposed action. Having made this determination, we have further concluded pursuant to 10 CFR Part 51, § 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Yankee Atomic Electric Company

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

Sincerely,

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

Enclosures:
1. Amendment No. 18
2. Negative Declaration
3. Environmental Impact Appraisal
4. Safety Evaluation
5. Federal Register Notice

cc w/encls:
See next page

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

John A. Ritsher, Esquire
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

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

Honorable M. Jerome Diamond
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

John A. Calhoun
Assistant Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

John R. Stanton, Director
Radiation Control Agency
Hazen Drive
Concord, New Hampshire 03301

John W. Stevens
Conservation Society of Southern Vermont
P. O. Box 256
Townshend, Vermont 05353

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

Chairman, Vermont Public Service Board
120 State Street
Montpelier, Vermont 05602

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

cc w/enclosures and cy of VY's filings dtd. 7/30, 9/15 &9/[illegible]
Mr. Richard V. DeGrasse
Public Service Board
7 School Street
Montpelier, Vermont 05602

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-28

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

   A. The applications for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated May 28, 1975 and July 30, 1975, as supplemented September 15 and 22, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-28 is hereby amended 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. 29."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 29 to the
Technical Specifications

Date of Issuance:
November 12, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 18

CHANGE NO. 29 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A as follows:

Remove Pages	Insert Pages
iii - iv	iii - iv
2	2
5 - 19	5 - 19
21	21
31	31
47 - 48	47 - 48
64 - 65	64 - 65
71	71
77 - 78	77 - 78
85	85
92R	92
96 - 98	96 - 98
101 - 102	101 - 102
104	104
108	108
110	110
122	122 - 122-a
124	124
	180-a - 180-r
181 - 187	181 - 187

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TABLE OF CONTENTS (CONT)

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G. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor, to verify the proper instrument channel response, alarm, and/or initiating action.

H. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

I. Minimum Critical Power Ratio - The Minimum Critical Power Ratio is defined as the ratio of that power in a fuel assembly which is
29 calculated to cause some point in that assembly to experience boiling transition as calculated by application of the GEXL correlation to the actual assembly operating power. (Reference NEDO-10958)

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

K. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

L. Operating - Operating means that a system or component is performing its intended functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Peaking Factor - The ratio of the fuel rod heat flux to the heat flux of an average rod in an
29 identical geometry bundle operating at the average core power.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals

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1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

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

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

29

When the reactor pressure is >800 psia and core flow is >10% of rated, the existence of a Minimum Critical Power Ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTING

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:

A. Trip Settings

The limiting safety system trip settings shall be as specified below: 29

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

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

where:

S = Setting in percent of rated thermal power (1593 MWt)

W = percent rated drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow.

1.1 Safety Limit

B. Core Thermal Power Limit (Reactor Pressure $\leq$800 psia or Core Flow $\leq$10% of Rated)

29 When the reactor pressure is $\leq$800 psia or core flow $\leq$10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

2.1 LIMITING SAFETY SYSTEM SETTING

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%) \frac{A}{MTPF}$$

where:

A = 2.62 for 7 x 7 fuel
= 2.44 for 8 x 8 fuel

MTPF = The value of the existing maximum total peaking factor. 29

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, the 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 $\leq$120/125 of scale.

B. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be as shown in Figure 2.1.1 and shall be:

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

1.1 SAFETY LIMIT

Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded
29 when scram is accomplished by a means
other than the expected scram signal.

2.1 LIMITING SAFETY SYSTEM SETTING

where:

S_{RB} = Rod block setting in percent of rated thermal power 1593 MWt

W = percent rated drive flow where 100% rated drive flow is that flow equivalent to 48×10^6 lbs/hr core flow.

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\%)\ \frac{A}{MTPF}$$ 29

where:

A = 2.62 for 7 x 7 fuel
= 2.44 for 8 x 8 fuel

MTPF = The value of the existing maximum total peaking factor.

1.1 SAFETY LIMIT

29 C. Whenever 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.

2.1 LIMITING SAFETY SYSTEM SETTING

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

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 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.

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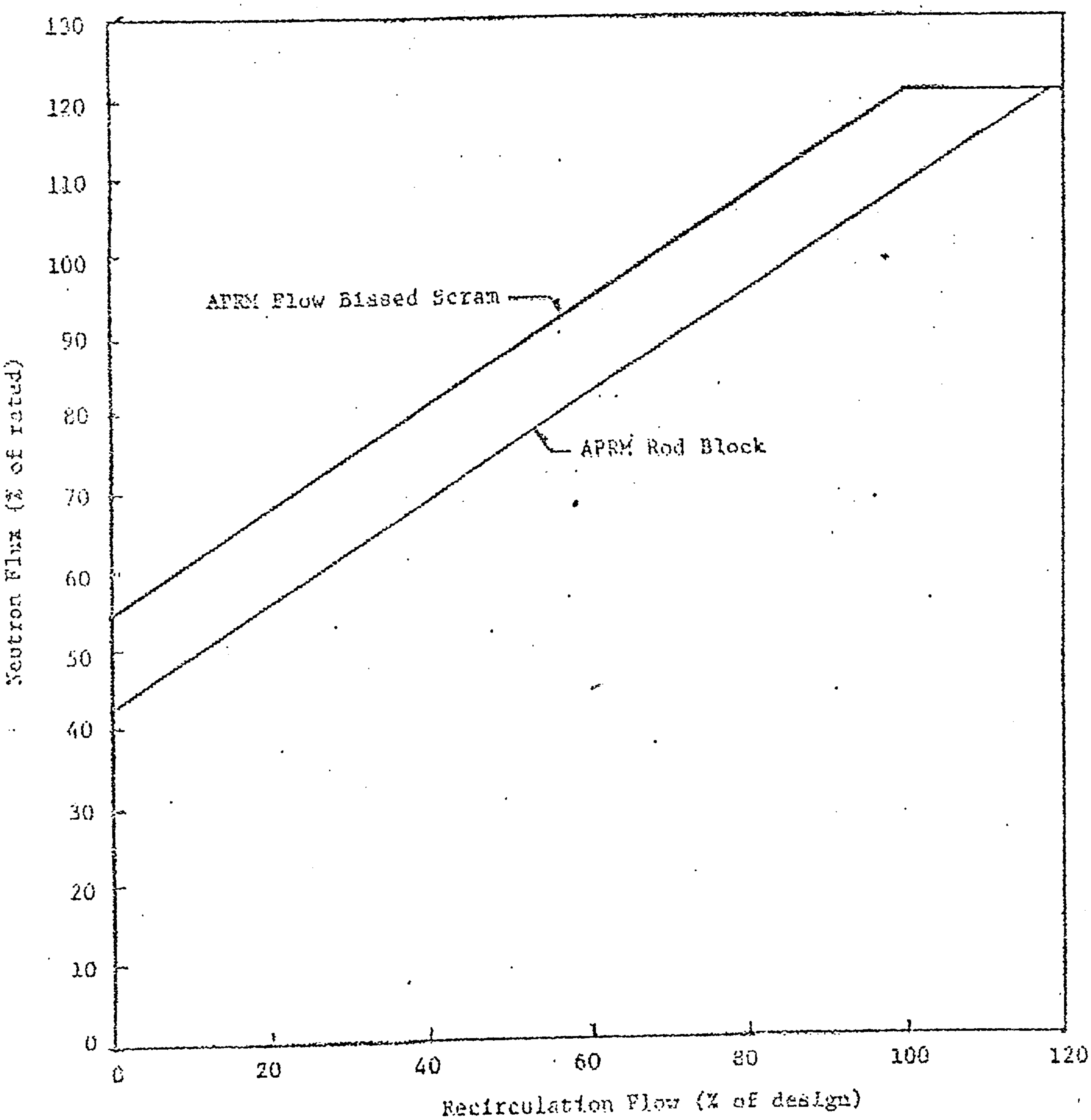
Figure 2.1-1

APRM Flow Reference Scram
and
APRM Rod Block Settings



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Bases:

1.1 Fuel Cladding Integrity

A. Fuel Cladding Integrity Limit at Reactor Pressure > 800 psia and Core Flow > 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily 29
result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB [1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

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The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass flux:	0.1 to 1.25 10^6 lb/hr	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Man/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	16, 64 Rods in an 8 x 8 array	
	49 Rods in a 7 x 7 array	

29

The required input to the statistical model are the uncertainties listed on Table 2.1-1, the nominal values of the core parameters listed in Table 2.1-2, and the relative assembly power distribution shown in Table 2.1-3. Table 2.1-4 shows the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

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The basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Vermont Yankee during any fuel cycle would not be as severe as the distribution used in the analysis.

B. <u>Core Thermal Power Limit (Reactor Pressure $\leq$800 psia or Core Flow$\leq$10% of Rated)</u> 29

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. <u>Power Transient</u>

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Vermont Yankee has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.1C.2 will be relied on to determine if a Safety Limit has been violated. 29

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

1.1 (cont.)

References

1. General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June, 1974 (NEDO-20340). 29

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)	
Feedwater Flow	1.76	29
Feedwater Temperature	0.76	
Reactor Pressure	0.5	
Core Inlet Temperature	0.2	
Core Total Flow	2.5	
Channel Flow Area	3.0	
Friction Factor Multiplier	10.0	
Channel Friction Factor Multiplier	5.0	
TIP Readings	8.7	
Bypass Void effect on TIP	3.95 - 4.53	
R Factor	1.6	
Critical Power	3.6	

Table 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	1665 MWt
Core Flow	48.0 Mlb/hr
Dome Pressure	1021 psig
Channel Flow Area	(7 x 7) .1069 ft^2 (8 x 8) .10665 ft^2
R-Factor	(7 x 7) 1.10 (8 x 8) 1.084

29

Table 1.1-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375 to 1.425	6.6
1.325 to 1.375	3.2
1.275 to 1.325	15.6
1.225 to 1.275	10.8
1.175 to 1.225	6.6
1.125 to 1.175	4.9
1.075 to 1.125	9.0
1.025 to 1.075	4.0
0.175 to 1.025	39.3
	Sum = 100

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Table 1.J-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7x7 Rod Array

R-Factor	Rod Sequence No.
1.098	1
1.083	2
1.075	3
1.062	4
1.052	5
1.042	6
1.042	7
≤ 1.027	8 thru 49

8x8 Rod Array

R-Factor	Rod Sequence No.	
1.100	1	
1.100	2	29
1.095	3	
1.095	4	
1.093	5	
1.093	6	
1.092	7	
≤ 1.077	8 thru 63	

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the VYNPS Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1665 MWt. The analyses were based upon plant operation in accordance with the operating map given in the FSAR. In addition, 1593 MWt. is the licensed maximum power level of VYNPS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operations as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition. 29

For analyses of the thermal consequences of the transients a MCPR of 1.28 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

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2.1 FUEL CLADDING INTEGRITY (Continued)

In summary:

i. The abnormal operational transients were analyzed to a power level of 1665 MWt.

ii. The licensed maximum power level is 1593 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

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APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.a, when the maximum total peaking factor is greater than 2.62 for 7x7 fuel and 2.44 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR >1.05 when the transient is initiated from MCPR $\geq$ 1.28.

b. Flux Scram Trip Setting (Refuel or Startup & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained 29
to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent IRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 850 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provided adequate protection against the the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.06. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence. 29

B. APRM Rod Block Trip Setting

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 condition of a MCPR less than 1.06. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.62 for 7x7 fuel and 2.44 for 8x8 fuel, thus preserving the APRM rod block safety margin. 29

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.

2.1 (Continued)

D. Reactor Low Water Level ECCS Initiation Trip Point

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature, and to limit clad metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram set point, and the ECCS initiation set point. To lower the ECCS initiation set point would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure. 29

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists. 29

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram set point at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not

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2.1 (Continued)

occur. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

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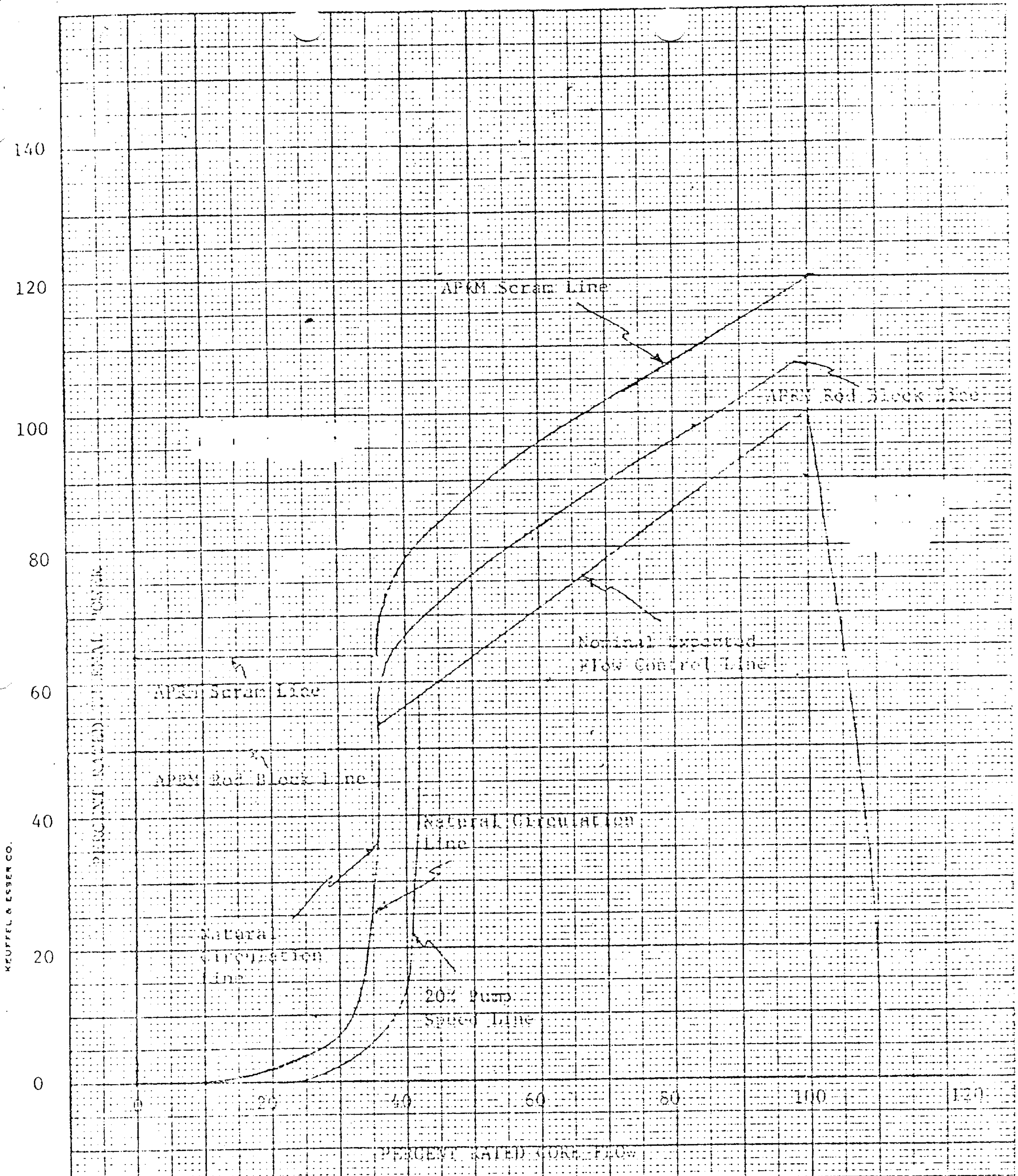

FIGURE 2.1-2
APRM FLOW BIAS SCRAM AND ROD BLOCK SETTINGS
SUPERIMPOSED ON OPERATING MAP

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the principle process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve settings shall be as follows: 29

1 valve at $\leq$1080 psig
2 valves at $\leq$1090 psig
1 valve at $\leq$1100 psig
2 valves at $\leq$1240 psig (safety valves)

Bases:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The respective design pressures are 1250 psig at 575°F and 1148 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to prevent exceeding the pressure vessel code limit for the worst-case isolation (pressurization) event (MSIV closure) assuming indirect (neutron flux) scram. 29

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 100 milliseconds.

B. During operation with a maximum total peaking factor (MTPF) greater than the design value (A) either:

a. The APRM scram and rod block settings shall not exceed the values determined by the equations given in Technical Specifications 2.1.A.1 and 2.1.B or

29

b. The power distribution shall be changed to reduce the maximum total peaking factor (MTPF) to or less than the design value (A).

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Once a day during reactor power operation the peak heat flux and total peaking factor shall be determined and the APRM scram and rod block settings, as given by the equations in Tables 3.1.1 and 3.2.5 and Technical Specifications 2.1.A and 2.1.B shall be calculated and instruments adjusted as necessary. 29

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must be Operating Refuel(1)	Startup	Run	Min. No. Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied(3)
1. Mode switch in shutdown		X	X	X	1	A
2. Manual scram		X	X	X	1	A
3. IRM High Flux	≤120/125	X	X	X(11)	2	A
Inop		X	X	X(11)	2	A
4. APRM						
29 High Flux (flow bias)	≤0.66W + 54% (4)			X	2	A or B
Inop				X	2(5)	A or B
Downscale	≥2/125			X	2	A or B
5. High Reactor Pressure	≤1055 psig	X	X	X	2	A
6. High Drywell Pressure	≤2 psig	X	X	X	2	A
7. Reactor Low water level	≥1.0 inch(6)	X	X	X	2	A
8. Scram discharge volume high level	≤24 gallons	X	X	X	2	A

TABLE 3.1.1 NOTES

1. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a) mode switch in shutdown
   b) manual scram
   c) high flux IRM or high flux SRM in coincidence
   d) scram discharge volume high water level.

2. There shall be two operable or tripped trip systems for each function.

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   B. Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   C. Reduce turbine load and close main steamline isolation valves within eight hours.
   D. Reduce reactor power to less than 30% of rated within eight hours.

4. "W" is percent rated drive flow where 100% rated drive flow is that flow equivalent to 48×10^6 lbs/hr core flow. 29

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. 1 inch on the water level instrumentation is 127 above the top of the active fuel.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure. 29

10. A turbine stop valve closure and generator load rejection bypass is permitted when the first stage turbine pressure is less than 30 percent of normal (220 psia).

11. The IRM scram is bypassed when the APRMs are on scale and the mode switch is in the run position.

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Bases:

4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B, and C. Sensors that make up Group A are of the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier and bi-stable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bi-stable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The peak heat flux and total peaking factor shall be checked once per day to determine if the APRM scram setpoint requires adjustment. This will normally be done by checking LPRM readings. Because few control rod movements or power changes occur, checking these parameters daily is adequate. 29

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System (Note 1)	Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
	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) 29
2	b. Downscale			X	$\geq$2/125 full scale
	Rod Block Monitor (Note 6)				
1	a. Upscale (Flow Bias) (Note 7)			X	$\leq 0.66W + 40\%$ (Note 5) 29
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

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$ 100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. 29

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30 day period.

7. The trip may be bypassed when the reactor power is $\leq$30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

High radiation monitors in the mean steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 limits are not exceeded for the control rod drop accident and 10 CFR 20 limits are not exceeded for gross fuel failure during reactor operations. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when reactor pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulatory malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1295°F; thus, there is no release of fission products other than those in the reactor water.

Low condenser vacuum has been added as a trip of the Group 1 isolation valves to prevent release of radioactive gases from the primary coolant through condenser. The set point of 12 inches of mercury absolute was selected to provide sufficient margin to assure retention capability in the condenser when gas flow is stopped and sufficient margin below normal operating values.

The HPCI and/or RCIC high flow, steam supply pressure, and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 6 valves. The trip settings are such that core uncovering is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual channel system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. Permanently installed circuits and equipment may be used to trip instrument channels. In the non-fail safe systems which require energizing the circuitry, tripping an instrument channel may take the form of providing the required relay function by use of permanently installed circuits. This is accomplished in some cases by closing logic circuits with the aid of the permanently installed test jacks or other circuitry which would be installed for this purpose.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease below 1.06. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRMs, six IRMS or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal. 29

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06. 29 29

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches 1.06. 29

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors provide isolation capability on the air ejector suction line. Isolation is initiated when either instrument reaches its upscale trip point. The immediate trip (within 1 minute) set point of 1.5 Ci/sec (30 minute decay) is based upon limiting the whole body dose at the site boundary to less than 5 Rem in the unlikely event of a boundary failure in the off-gas system concurrent with a spike release of radioactivity from the fuel. The assumption has been made that the rate of radioactivity increase within the 1 minute valve closure time period would be less than a factor of 5 based upon actual experience with such events. The delayed trip (within 15 minutes) set point of 0.3 Ci/sec (30 minute decay) is based upon limiting the whole body dose at the site boundary to less than 5 Rem in the event of off-gas system boundary failure concurrent with an off-gas release from the fuel of a lower value than considered above.

Two radiation monitors provide an isolation capbility on the off-gas line at the plant. Stack Isolation is initiated when either instrument reaches its upscale trip point. The trip point of 0.07 Ci/sec has been derived from the release limit of $0.08/\bar{E}\gamma$ assuming minimum holdup and corresponding maximum average disintegration energy and an isotopic mix corresponding to power operation. An energy shift is concurrent with plant shutdown, and consequently, the trip point may be adjusted to accommodate the change in mix yet remain below $0.08/\bar{E}\gamma$. The limit, $0.08/\bar{E}\gamma$, is established to prevent an off site annual whole body dose of 500 mRem (the 10CFR20 limit). The time delays are established based upon the flow path (e.g. 30 minutes if the carbon beds are in service and 2 minutes if they are bypassed).

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3.3 LIMITING CONDITIONS FOR OPERATION

(b) when the reactor is above 10% power the maximum worth of any control rod even presuming a single error by an operator shall be less than 2.0% delta k.

5. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate greater than or equal to three counts per second.

6. During ~~operation~~ with limiting control rod patterns either:

(a) Both RBM channels shall be operable; or

(b) Control rod withdrawal shall be blocked; or

29 (c) The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENTS

5. Prior to control rod withdrawal for startup or during refueling, verification shall be made that at least two source range channels have an observed count rate of at least three counts per second.

6. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

3.2 (Continued)

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and

3 (Continued)

a. A startup inter-assembly local power peaking factor of 1.30 or less.

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit. Above 10% power the consequence of a rod drop are less severe and the worths of rods in normal patterns are much less, therefore limiting rods worths to 2.0% delta k at power levels above 10% is conservative.

5. The Source Range Monitor (SRM) system has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality therefore two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR less than 1.06 29
During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

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3.3 (Continued)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage. The limiting power transient is that resulting from a turbine stop valve closure with a failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains 29
greater than 1.06.

The scram times for all control rods shall be determined during each operating cycle. The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram. The frequency of exercising the control rods under the conditions of two or more control rods valved out of service provides even further assurance of the reliability of the remaining control rods.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod (3x3) square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} <1.0. Other repeating rod sequences with more rods withdrawn resulted in k_{eff} >1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operation base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1%Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3.5 LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and Low Pressure Coolant Injection

1. Except as specified in Specifications 3.5.A.2 through 3.5.A.4 below and 3.5.H.3 and 3.5.H.4, both core spray and the LPCI subsystems shall be operable whenever irradiated fuel is in the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applied to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Coolant Injection

Surveillance of the core spray subsystems and LPCI shall be performed as follows:

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each refueling outage
b. Flow Rate Test - Core spray pumps shall deliver at least 3000 gpm (torus to torus) against a system head of 120 psig. Each LPCI pump shall deliver 8686±50 gpm (vessel to vessel) set by throttling loop injection valves 27A, and 27B.	Each refueling outage

29

LIMITING CONDITION FOR OPERATION

F. Automatic Depressurization System

1. Except as specified in Specification 3.5.F.2 below, the entire automatic depressurization relief system shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the automatic depressurization subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding seven days unless such a valve is sooner made operable, provided that during such seven days both the remaining
29 automatic relief system valves and the HPCI system are operable.

4.5 SURVEILLANCE REQUIREMENT

F. Automatic Depressurization System

Surveillance of the automatic depressurization system shall be performed as follows:

1. During each operating cycle each relief valve shall be manually opened with the reactor at low pressure until the thermocouples downstream of the valve indicates fluid is flowing from the valve.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

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3.5 (Continued)

G. Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. The RCIC meets this requirement. Reference Section 14.5.4.4 FSAR. The HPCIS provides an incidental backup to the RCIC system such that in the event the RCIC should be inoperable no loss of function would occur if the HPCIS is operable.

H. Minimum Core and Containment Cooling System Availability

The core cooling and the containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that adequate cooling capacity is available by precluding any combination of inoperable components from fulfilling the core and containment cooling function. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is cold and shutdown and there is no potential for draining the reactor vessel. However, if refueling operations are in progress, one coolant injection system, one diesel and a residual of at least 300,000 gallons is required to assure core flooding capability.

I. Maintenance of Filled Discharge Pipe

Full discharge lines are required when the core spray subsystems, HPCI and RCIC are required to be operable to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start.

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4.5 (Continued)

The pump operability check will be performed by starting the turbine manually, valves will also be stroked by manual actuation of the operators.

G. Reactor Core Isolation Cooling System

Frequency of testing of the RCIC system is the same as the HPCIs and demonstrates that the system is operable if needed.

H. Minimum Core and Containment Cooling System Availability

Immediate testing followed by daily tests of all low pressure core cooling subsystems and containment cooling service water systems including the operable standby diesel generator upon determination of one inoperable diesel generator adequately demonstrates the availability of core and containment cooling subsystems. This testing frequency is reduced to monthly during a refueling outage to permit various surveillance inspections on equipment. However, at least one diesel is maintained fully operable and tested weekly.

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent monthly assures that the core cooling subsystems will not experience water hammer damage when any of the pumps are started. Core spray subsystems and LPCI subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

29

3.6 LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

2. Both the sump and air sampling systems shall be operable during power operation. From and after the date that one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days.

3. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced
29 to and maintained at or below 95% of rated power.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

Reactor coolant system leakage shall be checked and logged at least once per day.

D. Safety and Relief Valves

1. A minimum of 1/2 of all safety valves shall be bench-checked or replaced with a bench-checked valve each refueling outage. Both valves shall be checked or replaced every two refueling outages. The lift point of the safety valves shall be set as specified in Specification 2.2.B.

2. A minimum of 1/2 of all relief valves shall be bench-checked or replaced with a bench-checked valve each refueling outage. All four valves shall be checked or replaced every two refueling outages. The set pressures shall be as specified in Specification 2.2.B.

3.6 LIMITING CONDITION FOR OPERATION

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle.

G. Single Loop Operation

1. Operation with a single recirculation loop is permitted for 24 hours unless the recirculation loop is sooner made operable. If the loop cannot be made operable, the reactor shall be in cold shutdown within 24 hours.

29

H. Recirculation System

1. Valves in the equalizer piping between the recirculation loops shall be closed during reactor operation.

3.6 & 4.6 (Continued)

greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The removal capacity from the drywell floor drain sump and the equipment drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purposes of this limiting condition a relief valve that is 29
unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ± 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

E. Structural Integrity

A pre-service inspection of the components listed in Table 4.2-4 of the FSAR will be conducted after site erection to assure freedom from defects greater than code allowance; in addition, this will serve as a reference base for further inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout plant life. The inspection program given in Table 4.2-4 was based on the proposed ASME code for in-service inspection which was followed except where accessibility for inspection was not provided. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

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3.6 & 4.6 (Continued)

Extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate or growth of defects from fatigue studies sponsored by the AEC. These studies show that is requires thousands of stress cycles beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

3.6 & 4.6 (cont'd)

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

G. Single Loop Operation

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, reactor operation for more than 24 hours under such conditions will not be authorized until the necessary analyses have been performed, evaluated, and deemed acceptable. 29

H. Recirculation System

The largest recirculation break area assumed in the ECCS evaluation was 4.43 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, reactor operation is prohibited unless the main equalizer valves in the equalizer line are closed.

LIMITING CONDITONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the
29 performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

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 3.11-1. 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 be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within prescribed limits within two (2) hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

29

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be appplied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

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LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (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:

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

$LHGR_d$ = Design LHGR = 18.5 KW/ft. (7 x 7)
13.4 KW/ft. (8 x 8)

29 $(\Delta P/P)_{max}$ = Maximum power spiking penalty
= 0.038 (7 x 7)
= 0.022 (8 x 8)

LT = Total core length = 12 feet
L = Axial position above bottom of core (in feet)

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 be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power. 29

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at ≥25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6 29

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LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, the Operating MCPR Limit shall be $\geq$ 1.28. at rated power and flow. For core flows other than rated the Operating MCPR Limit shall be the above value multiplied by K_f, where K_f is given
29 by Figure 3.11-2. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

D. Reporting Requirements

If any of the limiting values identified in Specs 3.11A, B or C are exceeded, a reportable occurrence report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

Bases:

3.11 Fuel Rods

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

29

This specifications assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20^\circ$F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperature are within the 10CFR 50, Appendix K limit. The limiting value for APLHGR is shown in Figures 3.11.1A and 3.11.1B of the Vermont Yankee Technical Specifications.

The calculational procedure used to establish the APLHGR shown on Figures 3.11.1A and 3.11.1B is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assume a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.11.1A and 3.11.1B. (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations. 29

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1.

REFERENCES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), submitted August 1974.

29

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

Table 1

SIGNIFICANT INPUT PARAMETERS TO THE VYNPS
LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	1665 MWt which corresponds to 105% of rated steam flow.
Vessel Steam Output	6.74×10^6 Lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1021 psig
Design Basis Recirculation Line Break Area	4.43 ft^2
Recirculation Line Break Area for Small Breaks	1.0 and 0.05 ft^2

29

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
Generic B (Reload 1)	7 x 7	18.5	1.5	1.18
8D219 (Reload 2)	8 x 8	13.4	1.5	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

Bases:

3.11 B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in Reference 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

Nov.

Bases:

3.11C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1. 29

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

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The limiting transient which determines the required steady state MCPR limit is the turbine trip without bypass transient. This transient yields the largest ΔMCPR. When added to the Safety Limit MCPR of 1.06 the required minimum operating limit MCPR of specification 3.11C is obtained.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, a more limiting Technical Specification limits operation of the reactor to a MCPR of 1.28 for 8 x 8 fuel based on consideration of a turbine trip transient with failure of a bypass valve. The MCPR valve used in the ECCS performance evaluation has been appropriately considered.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360[2] and on core parameters shown in Table 4-5 thru 4-7 (pages 4-8 and 4-9) of NEDO-20940.[1] 29

The evaluation of a given transient begins with the system initial parameters shown in Table 6-1 (page 6-12) of NEDO-20940[1] that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermal hydraulic SCAT code described in NEDE-20566[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

MCPR Limits for Core Flows Other than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed-up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.28 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated 29
for the same postulated transient event.

The K_f factor curves shown in Figure 3.11.2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow. 29

The K_f factors shown in Figure 3.11.2, are conservative for the Vermont Yankee NPS operation because the operating limit MCPR of 1.28 is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

3.11 FUEL RODS (Continued)

D. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specification 3.11.A, B, and C that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded, action is then initiated within fifteen minutes to restore operation to within the prescribed limits. Each event involving steady state operation beyond a specified limit shall be reported as a reportable occurrance. However, if the corrective action is taken as described, a thirty day written report will meet the requirement of this specification.

29

E. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. Vermont Yankee Nuclear Power Station Loss of Coolant Analyses Conformance with Appendix K to 10CFR50, May 1975.

5. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, November, 1973.

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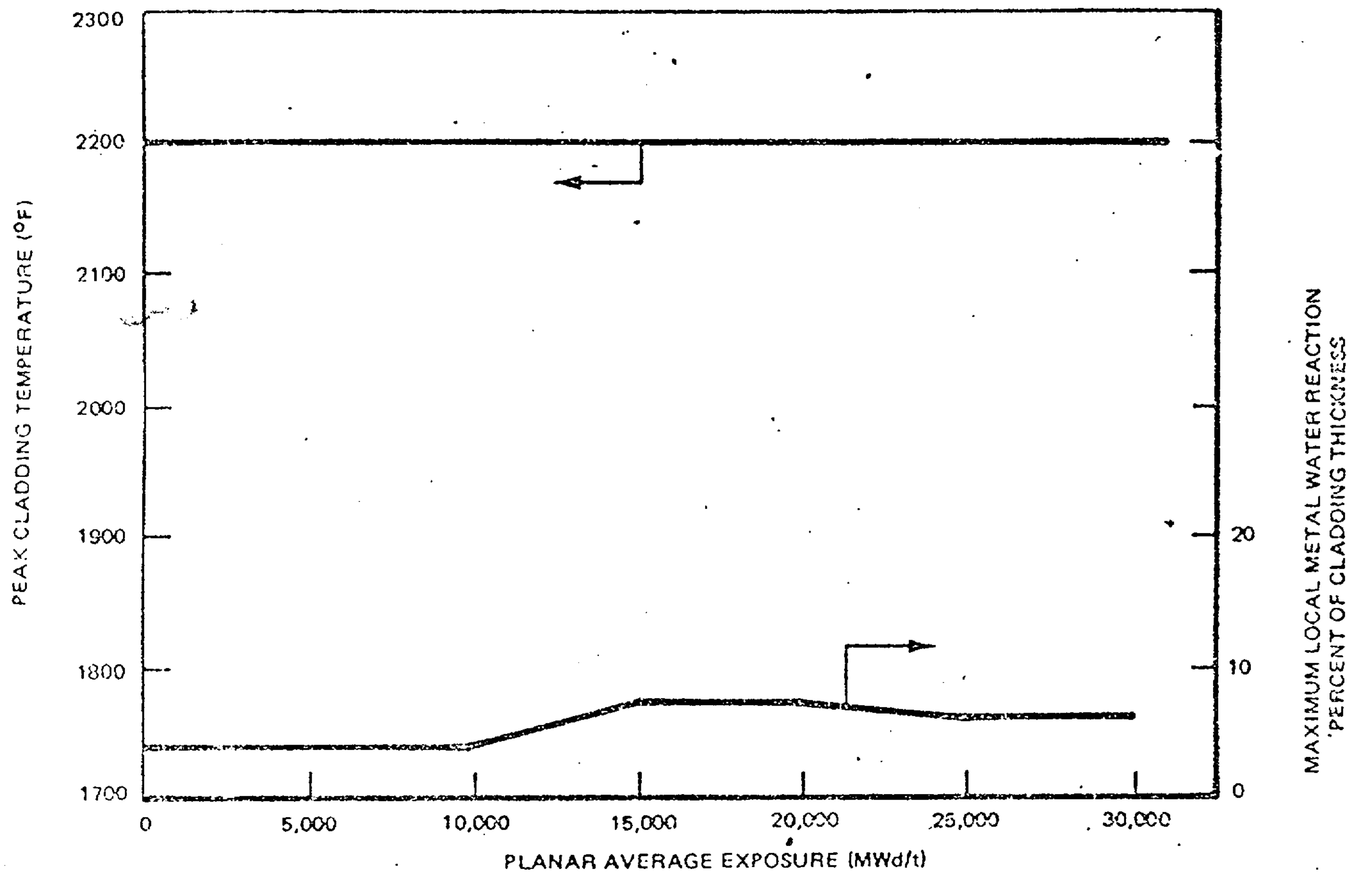
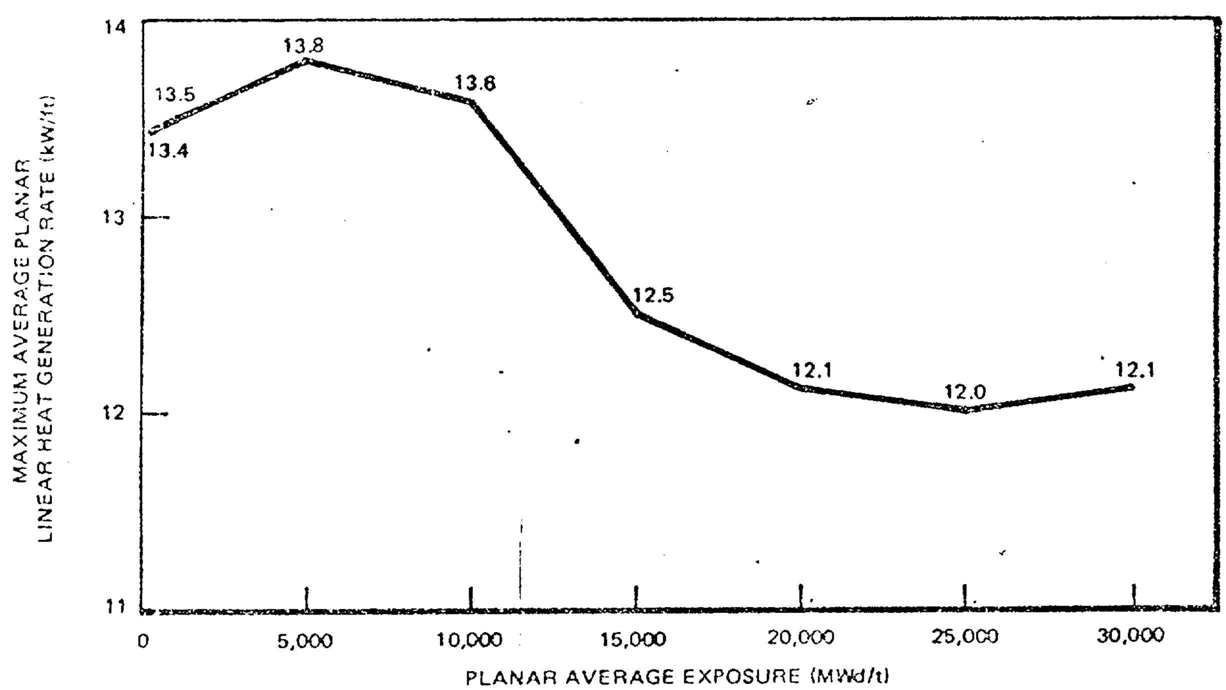


PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE

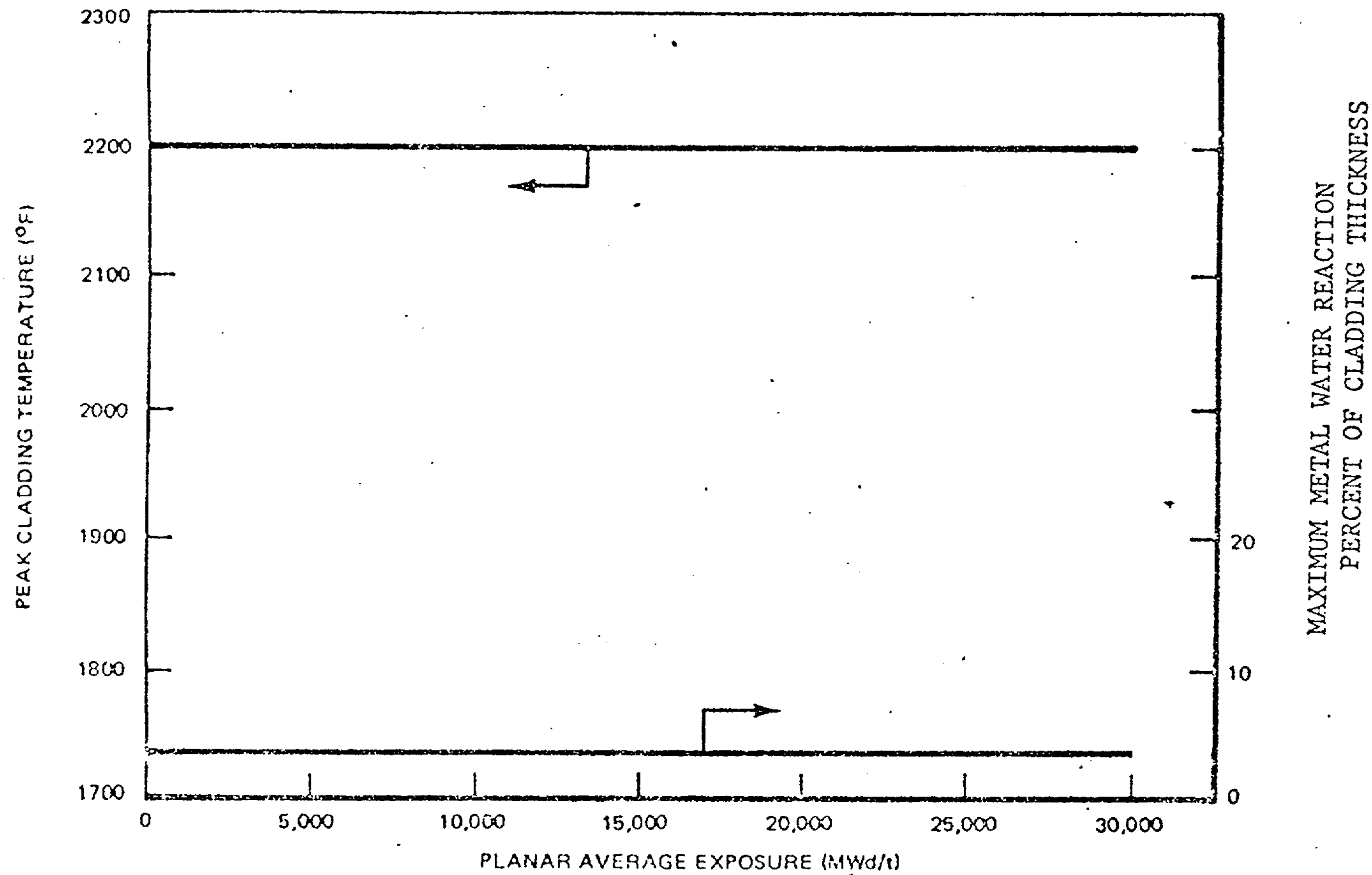
29



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

Vermont Yankee Bypass Flow Holes Plugged, 7x7 Generic B Fuel

Figure 3.11-1A



PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE

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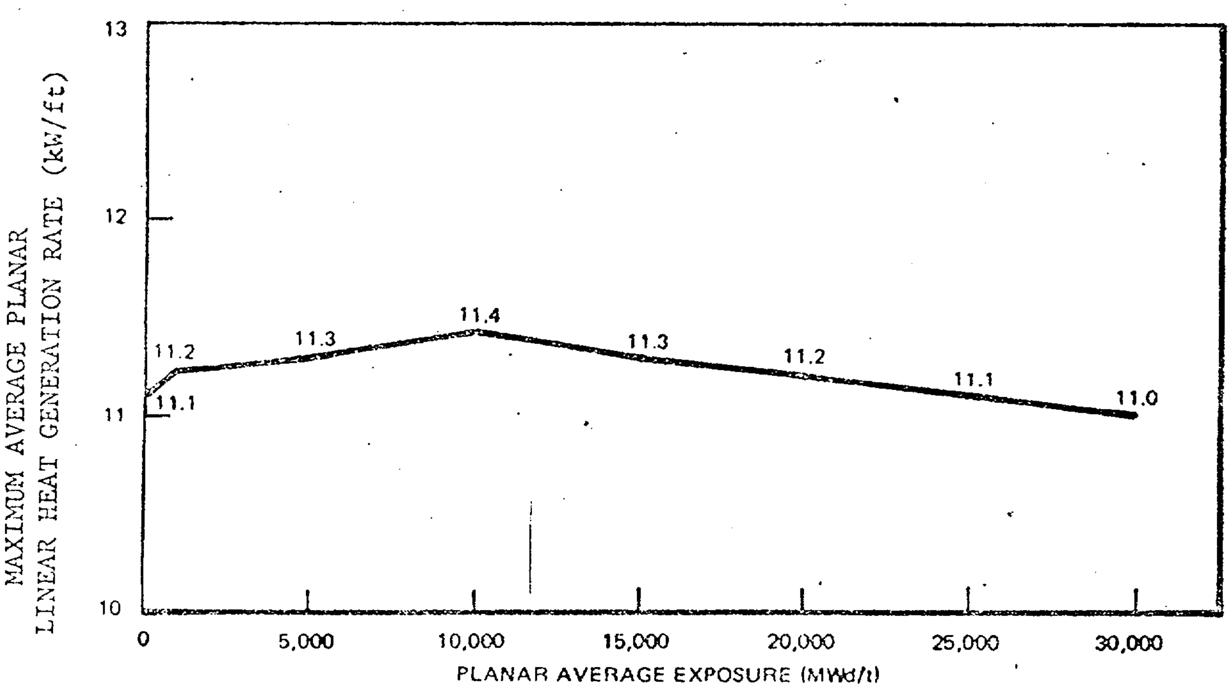
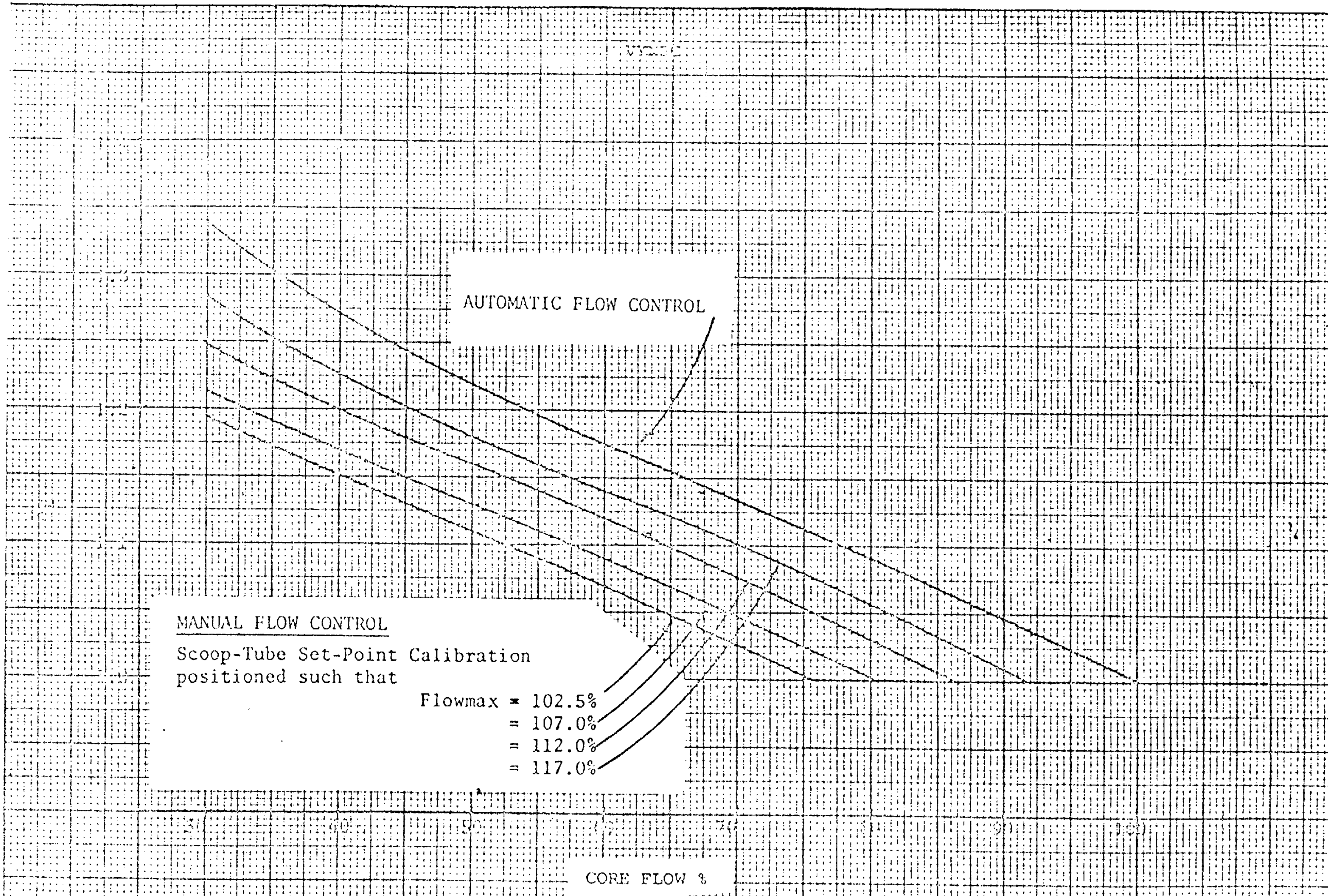


MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUR PLANAR AVERAGE EXPOSURE

Vermont Yankee Bypass Flow Holes Plugged, 8 x 8 8D219 Fuel

Figure 3.11-1B



AUTOMATIC FLOW CONTROL
MANUAL FLOW CONTROL
Scoop-Tube Set-Point Calibration
positioned such that
Flowmax = 102.5%
= 107.0%
= 112.0%
= 117.0%
CORE FLOW %

3.11-2

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by 29
the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

4.11C <u>Minimum Critical Power Ratio (MCPR) - Surveillance Requirement</u>

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit. 29

References

1. Vermont Yankee Safety Analysis with Bypass Flow Holes Plugged, Attachment A to License Amendment Submittal, July 1975, (NEDO-20967).

2. General Electric BWR Generic Reload Application for 8 x 8 fuel, NEDO-20360, Revision 1, November, 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 29
10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

29 3.12 REFUELING

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure core reactivity is within capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks

29 The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks, listed below, shall be operable except as specified in Specifications 3.12.D and 3.12.E.

1. Control Rod Blocks

a. Mode switch in Startup/Hot Standby and refueling platform over the reactor.

4.12 REFUELING 29

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective:

To verify the operability of instrumentation and interlocks used in refueling.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the Head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

29 3.12 LIMITING CONDITION FOR OPERATION

b. Fuel on any refueling hoist and refueling platform over the reactor.

c. Mode switch in Refuel with one control rod withdrawal permit.

2. Refueling Platform Reverse Motion (toward reactor vessel) Block

a. Mode switch in Startup/Hot Standby.

b. Any control rod out and fuel on any refueling hoist.

3. Refueling Platform Hoists Blocks

a. Any control rod out and fuel on any refueling hoist over the vessel.

b. Hoist overload.

c. High position limitation.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being

4.12 SURVEILLANCE REQUIREMENT 29

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response.

moved and one in an adjacent quadrant. For an SRM to be considered operable the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected into the proper circuitry which contain the required rod blocks).

2. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of at least 36 feet.

Thereafter, the SRM's shall be checked daily for response.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

D. Control Rod and Control Rod Drive Maintenance

A maximum of two non-adjacent control rods separated by more than two control cells in any direction, may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

2. Specification 3.3.A.1 shall be met, or the control rod directional control valves for a minimum of eight control rods surrounding each drive out of service for maintenance shall be disarmed electrically and sufficient margin to criticality demonstrated.

3. SRMs shall be operable (a) in each core quadrant containing a control rod on which maintenance is being performed, and (b) in a quadrant adjacent to one of the quadrants specified in
29 Specification 3.12.D.3.(a) above. Requirements for an SRM to be considered operable
29 are given in Specification 3.12.B.

E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

D. Control Rod and Control Rod Drive Maintenance

1. Sufficient control rods shall be withdrawn prior to performing this maintenance to demonstrate with a margin of 0.25 percent Δk that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

2. Alternately, if a minimum of eight control rods surrounding each control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the 0.25 percent Δk margin shall be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.

E. Extended Core Maintenance

Prior to control rod withdrawal for extended core maintenance, that control rods control cell shall be verified to contain no fuel assemblies.

29 3.12 LIMITING CONDITION FOR OPERATION

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification
29 3.12.B.

F. Fuel Movement

Fuel shall not be moved or handled in the reactor core for 24 hours following reactor shutdown to cold shutdown conditions.

4.12 SURVEILLANCE REQUIREMENT

1. This surveillance requirement is the same as that given in Specification 4.12.A. 29

2. This surveillance requirement is the same as that given in Specification 4.12.B. 29

F. Fuel Movement

Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been in the cold shutdown condition for a minimum of 24 hours.

Bases:

29 3.12 & 4.12 REFUELING

A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.

Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

B. The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assured adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored.

C. To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. This minimum water level of 36 feet is established because it would be a significant change from the normal level, well above a level to assure adequate cooling (just above active fuel).

29 3.12 & 4.12 (Continued)

D. During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. This specification provides assurance that inadvertent criticality does not occur during such maintenance.

The maintenance is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling operations as explained in Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in service insures that inadvertent criticality cannot occur during this maintenance. The shutdown margin is verified by demonstrating that the core is shut down even if the strongest control rod remaining in service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in the Bases for Specification 3.12.A. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. 29

F. The intent of this specification is to assure that the reactor core has been in the cold shutdown condition for at least 24 hours following power operation and prior to fuel handling or movement. The safety analysis for the postulated refueling accident assumed that the reactor had been shut-down for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly.

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGE TO THE

APPENDIX A TECHNICAL SPECIFICATIONS OF LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

The U. S. Nuclear Regulatory Commission (the Commission) has reviewed the licensee's proposed change to the Appendix A Technical Specifications of Facility Operating License DPR-28. This change would authorize the Vermont Yankee Nuclear Power Corporation to operate the Vermont Yankee Nuclear Power Station with certain revisions to the present limiting conditions for operation specified in Appendix A of the referenced license. These revisions result from the implementation of the Acceptance Criteria For the Emergency Core Cooling System for Light Water Nuclear Power Reactors (ECCS) as specified in Section 50.46 of Part 50 CFR. No revisions to the Environmental Technical Specifications, (Appendix B) were required as a result of this proposed change.

The Commission's Division of Reactor Licensing has prepared an environmental impact appraisal for the proposed change to the Appendix A Technical Specifications, for Facility Operating License DPR-28.

On the basis of the environmental impact appraisal

we have concluded that an environmental impact statement for this particular action is not warranted because, pursuant to the Commission's regulations in 10 CFR 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6, the Commission has determined that this proposed change to the Appendix A Technical Specifications is not a major federal action significantly affecting the quality of the human environment. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. 20555, and at the Brooks Memorial Library, 224 Main Street Brattleboro, Vermont 05301.

Dated at Rockville, Maryland, this 12th day of November 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Environmental Projects Branch No. 1
Division of Reactor Licensing

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING: AMENDMENT NO. 18 TO LICENSE NO. DPR-28

CHANGE NO. 29 TO THE TECHNICAL SPECIFICATIONS

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

I. Description of Proposed Action

By letters dated April 14, 1975, July 8 and 30, 1975, September 15 and 22, 1975, the Vermont Yankee Nuclear Power Corporation (the licensee) provided information and supportive analysis relative to a proposed change in the Appendix A Technical Specifications of Facility License No. DPR-28. The proposed change concerns revisions to the limiting conditions for operation to the Vermont Yankee Nuclear Power Station as a result of the implementation of the Acceptance Criteria for the Emergency Core Cooling System (ECCS) The implementation of the ECCS Acceptance Criteria will permit operation of the Vermont Yankee Nuclear Power Station at a power level previously evaluated in the Final Environmental Statement (FES) issued in July 1972. The FES concluded that based upon an evaluation of the proposed operating conditions, an operating license should be issued for the Vermont Yankee Nuclear Power Station.

II. Environmental Impacts of Proposed Action.

The NRC has evaluated the potential environmental impacts associated with this proposed license amendment as required by the NEPA and Section 51.7 of Part 51 CFR.

The potential NEPA concerns associated with the implementation of the ECCS Criteria for the Vermont Yankee Nuclear Power Station can be defined as:

1. Changes in benefits accruing from plant operation due to revisions to reactor power limits.
2. Variation in environmental impacts resulting from changes in non-radiological effluent releases.
3. Variation in environmental impacts resulting from changes in radiological effluent releases.

This NRC evaluation has concluded that operating power will be as previously evaluated and presented in the FES of July 1972. As such, no resultant chnages in these (3) criteria are expected. Since this change will not result in modified power levels, no changes in the Cost/Benefit balance and environmental impacts (other than expressed in the FES) are predicted. The FES evaluation of the Vermont Yankee Nuclear Station cooling water flow, thermal effluents, chemical effluents, radiological source term and effluents during operation and postulated accident conditions need not be revised as a result of the implementation of the ECCS Acceptance Criteria.

III. Conclusions and Basis for Negative Declaration

On the basis of the NRC evaluation and information supplied by the licensee, it is concluded that the implementation of the ECCS Acceptance Criteria for the Vermont Yankee Nuclear Power Station will produce no discernible environmental impacts other than those previously addressed in the FES of July 1972. The issuance of this change to the Appendix A Technical Specifications will permit operation at a power level previously evaluated in the FES and will not affect the Cost/Benefit balance, nor the evaluation of the radiological and non-radiological effluents as presented in the FES. This amendment will not require changes to the Environmental Technical Specifications (Appendix B).

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for the proposed license amendment and that a Negative Declaration shall be issued to this effect.

Clifford A. Haupt

Clifford A. Haupt, Project Engineer
Environmental Projects Branch No. 1
Division of Reactor Licensing

George W. Knighton

George W. Knighton, Chief
Environmental Projects Branch No. 1
Division of Reactor Licensing

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. DPR-28

(CHANGE NO. 29 TO THE TECHNICAL SPECIFICATIONS)

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

The Vermont Yankee Nuclear Power Corporation (the licensee) has proposed to operate the Vermont Yankee Nuclear Power Station (VYNPS):

(1) Using operating limits based on the General Electric Thermal Analysis Basis (GETAB), as requested in their application dated July 30, 1975(1), and supplements dated September 15, 1975(2), September 22, 1975(3), with references to an earlier submittal of May 28, 1975(4).

(2) Using modified operating limits based on an acceptable emergency core cooling system evaluation model that conforms with section 50.46 of 10 CFR Part 50, as requested in their application dated May 28, 1975 and revised July 30, 1975.

2.0 GENERAL ELECTRIC THERMAL ANALYSIS (GETAB)

2.1 DISCUSSION

By letter dated May 28, 1975, the licensee proposed changes to the Technical Specifications, Appendix A to Facility Operating License No. DPR-28, for VYNPS, which incorporate operating limits based on the General Electric Thermal Analysis Basis (GETAB) described in General Electric (GE) Company report NEDO-10958(5).

The proposed changes involve the adoption of a new transition boiling correlation termed GEXL which would replace the Hench-Levy critical heat flux correlation as the basis for determining the thermal-hydaulic conditions which would result in a departure from nucleate boiling.

One of the safety requirements for light water cooled nuclear reactors is prevention of damage to the fuel cladding. To prevent damage to the fuel cladding, light water cooled reactors must be designed and operated such that during normal operation and anticipated transients the heat transfer rate from the fuel cladding to the coolant is sufficient to prevent overheating of the fuel cladding. Although transition boiling would not necessarily result in damage to boiling water reactor (BWR) fuel rods, historically it has been used as a fuel damage limit because of the large reduction in heat transfer rate when film boiling occurs. A critical power ratio (CPR) is defined as the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel assembly power at which boiling transition would be predicted to occur, using the GEXL correlation, is termed the critical power. The GEXL transition boiling correlation is more recent than the previously used Hench-Levy critical heat flux correlation and is based on an extensive data base. The methods for applying the GEXL correlation to determine thermal limits have been termed the General Electric Thermal Analysis Basis (GETAB). We have accepted the GEXL correlation and the GETAB methods in a previous report(6) as a basis for establishing the safety limit and limiting conditions for operation related to prevention of fuel damage for General Electric BWR 8x8 and 7x7 fuel.

The analyses submitted by the licensee are based on the present core loading (cycle 3) of the VYNPS reactor containing a combination of 7x7 fuel and 8x8 fuel, and properly considering the effects due to plugging the core bypass flow holes in the core support plate.(7),(8) The plugging represents an interim solution to the "channel box wear problem". Some penalty in maximum average planar linear heat generation rate (MAPLHGR) is caused due to delay in calculated core flooding following a postulated loss of coolant accident (LOCA) with the holes plugged. The licensee has not installed the "permanent" fix, consisting of plugging the facility's core support plate bypass holes and in addition drilling new bypass holes in the lower tie plate.

To apply GETAB to the Technical Specifications involves (1) establishing the fuel damage safety limit, (2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and (3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied.

2.2 EVALUATION

We have evaluated the VYNPS developed thermal margins based on the NEDO-10958 report[5] and plant specific input information provided by the licensee in NEDO-20967[1] and a proprietary Supplement B to that document.

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR for the 7x7 and 8x8 fuel is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation, Table 5-1 of the licensee submittal,[1] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated lists of uncertainties for VYNPS are the same as, or conservative with respect to, those reported in NEDO-10958[5] and NEDO-20340[9] which are acceptable. The largest difference is in the reported 8.7% uncertainty in TIP readings to account for effects of the reload core, and the addition of a correction (3.95 to 4.53%) to account for additional bypass region void uncertainty due to the bypass hole plugging. These values are acceptable for VYNPS cycle 3.

The reactor core selected for the GETAB statistical analyses that incorporate the operating parameters, fuel design (R factor*), and GEXL correlation uncertainties is in the same reactor class as the VYNPS reactor. Thus, the GETAB analysis results provide a fuel cladding integrity safety limit MCPR of 1.06 which is conservatively applied to the VYNPS reactor. Comparison of the licensee submittal bundle power distributions[1] used for the GETAB application and that for the actual operation of the VYNPS reactor assume more high power bundles in the GETAB analysis which result in a conservative value of 99.9% statistical limit MCPR.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable to prevent fuel damage for VYNPS's current fuel cycle.

2.2.2 Operating Limit MCPR

Various transient events will reduce the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one

*The R factor is a parameter which characterizes the local peaking pattern with respect to the most limiting rod.

results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the results of the transient analyses which contribute a significant decrease in MCPR. Types of transients evaluated were loss of flow, pressure and power increase, and coolant temperature decrease. The most limiting transients in the stated categories were 2-pump trip, load rejection without bypass, and loss of feedwater heating. Of these three, the most limiting transient was load rejection without bypass resulting in a ΔMCPR of 0.22. Addition of this ΔMCPR to the safety limit MCPR gives the minimum operating limit MCPR required to avoid violation of the safety limit, should this limiting transient occur. The loss of flow (2-pump trip) event was not specifically analyzed for VYNPS, but was stated to be less severe than load rejection without bypass. This reasoning is based on results of similar plant analyses, and on this same basis we find it acceptable.

The transient analyses were evaluated with the end-of-cycle 3 scram reactivity insertion rates that include a design conservatism factor of 0.80. The licensee's submitted initial condition parameters used for the worst operational transient analyses are acceptable. The initial MCPR assumed in the transient analyses (1.30) was equal to or greater than the established operating limit MCPR of 1.28, which results in conservative (high) prediction of resulting ΔMCPR values.

Conservatism was applied in determining the required operating limit MCPR because the axial and local flux peaking were assumed to take place at the beginning of the fuel cycle and the peak of the axial power shape was assumed to occur in the midplane (node 12; APF of 1.40). This is the worst consistent set of parameters and is supported by a GE study[5] which has shown the required operating MCPR to be a function of the location of axial flux peak. The required MCPR's are essentially independent of peak location for axial flux peaking in the middle and upper portions of the core, whereas for bottom peaked axial fluxes the required MCPR is reduced.

The applied R factor of 1.084 for 8x8 fuel is taken at the beginning of cycle to reasonably bound the expected operating conditions. During the cycle the local peaking and therefore the R factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

2.2.3 Proper Inclusion of the Correct Void Coefficient in Calculation of the Operating Limit MCPR

The required minimum operating limit MCPR of 1.28 was based on the addition of the largest ΔMCPR (caused by the load rejection without bypass transient) to the safety limit MCPR of 1.06, which we found to be acceptable. The calculations took proper account of a recent change in method of calculating void reactivity coefficients (Neutron Effective Voids (NEV))--where the new method provides better agreement between the calculated and plant instrument power distributions. The correct, revised values of NEV were used throughout.

2.2.4 Rod Withdrawal Error Transients

The licensee discussed the rod withdrawal error transient in terms of worst case conditions.(1) The analysis shows that the local power range monitor subsystem (LPRM's) will detect high local powers and result in an alarm.

However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal while the critical power ratio is still greater than the 1.06 MCPR safety limit, and the cladding strain is under the one percent plastic strain limit. We conclude that the consequences of this localized transient are acceptable.

2.2.5 Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to his Technical Specification which requires maintaining the operating MCPR greater than 1.28 times K_f factor for core flows less than rated. The K_f factor curves were generically derived which assures that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06.

We conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.2.6 Overpressure Analysis

The licensee submitted an overpressure analysis in order to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves. The analysis was performed at 104.5% power assuming the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this cycle, no credit for relief valves, and one safety valve failing to operate. The peak pressure at the vessel bottom was calculated to be 1306 psig yielding a 69 psi margin below the code allowable; which we conclude is acceptable.

2.3 CONCLUSION

Based on the above, we conclude that the analyses and operating limits based on the use of the General Electric Thermal Analysis Basis are acceptable. The associated proposed changes to the Technical Specifications which we also conclude to be acceptable are itemized in Section 4.0.

3.0 ECCS APPENDIX K ANALYSIS

3.1 DISCUSSION

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Power Reactors." One of the requirements of the Order was that "...the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On July 30, 1975, the licensee submitted an evaluation of the ECCS performance(1) for the design basis piping break for Vermont Yankee. An amendment requesting changes to the Technical Specifications for Vermont Yankee to implement the results of the evaluation was submitted on September 15, 1975.(2)* The licensee incorporated further information and corrections relating to the Technical Specifications by letter dated September 22, 1975.(3) The above referenced submittals show compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50 for the present cycle 3 core with core support plate bypass flow holes plugged. Proper penalty in MAPLHGR was taken to account for considerably increased reflood times with bypass flow holes plugged.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of VYNPS dated December 27, 1974.

*This submittal was in the form of corrections to an earlier set of Technical Specifications submitted May 28, 1974(4), making that earlier submittal correct for application to the core with plugged bypass holes.

The background of the staff review of the General Electric (GE) ECCS models and their application to VYNPS is described in the staff Safety Evaluation Report (SER) for that facility dated December 27, 1974 (the December 27, 1974 SER) issued in connection with the Order. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model. Together, the December 27, 1974 SER and the earlier Status Report and its Supplement referenced in the December 27, 1974 SER describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The VYNPS evaluation which is covered by this SER properly conforms to the accepted model.

3.2 EVALUATION

With respect to reflood and refill computations, the VYNPS analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. These are described in the December 27, 1974 SER. The VYNPS evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974.

We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses were performed for a similar plant (Brunswick 2) and were referenced in the VYNPS ECCS submittal[1]. These analyses supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the VYNPS design. The limiting break which is the design basis accident is the complete severence of the recirculation discharge line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by the licensee for VYNPS and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Operation of the reactor will meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures 8-12a and 8-12b of NEDO-20967[1], and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain additional changes must be made to the proposed Technical Specifications to ensure that reactor operation conforms with the evaluation of ECCS performance.

(a) The largest recirculation break area assumed in the evaluation was 4.43 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. A Technical Specification (3.6.H.1) has been added to require the valve in the equalizer line to be closed during reactor operation.

(b) Technical Specifications (3.11 A, B, C and D) have been revised to report as reportable occurrences operation in excess of the limiting MAPLHGR, LHGR, and MCPR values even if corrective action was taken upon discovery.

(c) An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, a Technical Specification (3.6.G.1) was added to limit reactor operation under those conditions to 24 hours until such time that the necessary analyses are performed, evaluated, and determined acceptable.

(d) The VYNPS LOCA analysis assumed all ADS valves operated in the event of small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, a Technical Specification (3.5.f.2) has been revised so as not to allow continuous operation for more than 7 days with any ADS valve out of service.

The above mentioned changes to the proposed Technical Specifications were discussed with, and concurred in by the licensee.

3.3 CONCLUSION

We conclude that operation of the reactor in accordance with the Technical Specification changes discussed above will meet the requirements of 10 CFR Part 50, section 50.46.

4.0 TECHNICAL SPECIFICATION CHANGES

Section 1 Definitions

The Subsection which defines Minimum Critical Heat Flux Ratio will be deleted and replaced by a definition of Minimum Critical Power Ratio.

The subsection which defines peaking factor in terms of fuel rod surface heat fluxes will be replaced by a new subsection which defines a total peaking factor in terms of power profile.

These changes are needed to assure consistency with the revised format of the GETAB analysis.

Section 1.1 Fuel Cladding Integrity Safety Limits

Subsection 1.1.A for operation with reactor pressure greater than 800 psig or core flow greater than or equal to 10% of rated would be revised to state a MCPR safety limit. Subsection 1.1.B would be revised to limit core thermal power to 25% or less of rated thermal power when reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated. These changes are consistent with the GETAB analyses discussed earlier in this safety evaluation.

A new paragraph has been included entitled "Power Transient" which updates the Technical Specifications to conform to currently accepted means for determining the violation of Technical Specifications 1.1.A and 1.1.B.

Figure 1.1.1 has been deleted and replaced with Figure 2.1.1. Adoption of GETAB/GEXL requires this Figure replacement.

Section 2.1 Fuel Cladding Integrity Limiting Safety System Settings

Subsections A.1 concerning APRM neutron flux scram settings and subsection B concerning APRM rod block settings would express the settings in terms of the new definitions of peaking factors rather than in terms of heat flux, and base the required settings on the design linear heat generation rates of 18.5 and 13.4 kw/ft for 7x7 and 8x8 fuel respectively.

Section 3.1 Reactor Protection System

A new paragraph 3.1.B has been added to correspond to Technical Specification 2.1.A.1 and 2.1.B.

Specification 3.5.f.2 has been revised to restrict operation to 7 days with one of the Automatic Depressurization valves out of service.

Specification 3.6.G.1 has been added to restrict single loop operation to 24 hours.

Specification 3.6.H.1 has been added to require main equalizer values to be closed during reactor operation.

Specifications 3.11 A, B, C and D have been revised to include an action requirement when limits are violated.

GETAB Bases

The bases would also be changed to discuss the justification for the revised specifications itemized above. We would modify the proposed GETAB related bases and tables to provide what we consider to be a clearer justification for the limits.

Changes have been made throughout the Technical Specifications incorporating the terminology of GETAB/GEXL limits and to ensure consistency with other Technical Specifications.

5.0 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.

Date: November 12, 1975

REFERENCES

1. Vermont Yankee Safety Analysis with Bypass Flow Holes Plugged, NEDO-20967, July 1975, Attachment A to July 30, 1975 letter.

2. Proposed Change No. 30, Supplement 3, letter dated September 15, 1975.

3. Proposed Change No. 30, Supplement 4, letter dated September 22, 1975.

4. Proposed Change No. 30, Supplement 1, letter dated May 25, 1975.

5. "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958.

6. "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWR's", Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September 1974.

7. "Order for Modification of License", issued by the Commission, August 15, 1975, for the VYNPS Providing Authorization to install Bypass Hole Plugs in the Lower Core Plate.

8. "Order for Modification of License", issued by the Commission, August 22, 1975, for the VYNPS providing authorization for the facility to operate with plugged bypass holes.

9. General Electric "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June 1974 and December 1974, respectively.